Niagara Mohawk

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October 20, 2000
NMP2L 1989

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

10 C.F.R. §50.71(e)
10 C.F.R. §50.59(b)

RE: Nine Mile Point Unit 2
Docket No. 50-410
NPF-69

Subject: ***Submittal of Revision 13 to the Nine Mile Point Nuclear Station Unit 2 Updated Safety Analysis Report and the 10 C.F.R. §50.59 Safety Evaluation Summary Report (TAC No. MB0304)***

Gentlemen:

Pursuant to the requirements of 10 C.F.R. §50.71(e) and 10 C.F.R. §50.59(b), Niagara Mohawk Power Corporation hereby submits Revision 13 to the Nine Mile Point Nuclear Station Unit 2 Updated Safety Analysis Report (USAR) and the annual Safety Evaluation Summary Report.

One (1) signed original and ten (10) copies of the USAR, Revision 13, are enclosed. Copies are also being sent directly to the Regional Administrator, Region I, and the NRC Resident Inspector at Nine Mile Point. The USAR revision contains changes made since the submittal of Revision 10 in November 1998. USAR Revision 11, dated February 16, 2000, and Revision 12, dated August 16, 2000, were administrative changes and were not subject to the requirements of 10 C.F.R. §50.71(e). The revision reflects all changes up to and including April 20, 2000. In addition, various USAR sections have been edited to eliminate blank and partial pages. The elimination of blank and partial pages is editorial in nature and does not update or change substantive information previously described in the USAR. Changes to the Niagara Mohawk Quality Assurance Topical Report (NMPC-QATR-1) that were previously submitted with Unit 1 FSAR (Updated) Revision 16, dated November 1999, have been incorporated in Unit 2 USAR Appendix B. Enclosure A provides the identification, reason, and basis for each change to the quality assurance program description in accordance with 10 C.F.R. §50.54(a)(3)(ii). The certification required by 10 C.F.R. §50.71(e) is attached.

A053 1/1

The enclosed annual Safety Evaluation Summary Report (Enclosure B) contains a brief description of changes, tests, and experiments, and includes a summary of the safety evaluation of each.

None of the changes, tests, or experiments involved an unreviewed safety question as defined in 10 C.F.R. §50.59(a)(2).

Very truly yours,

Richard B. Abbott

Richard B. Abbott
Vice President Nuclear Engineering

RBA/LWB/cld
Enclosures

xc: Mr. H. J. Miller, NRC Regional Administrator, Region I
Ms. M.K. Gamberoni, Section Chief PD-I, Section 1, NRR
Mr. G. K. Hunegs, NRC Senior Resident Inspector
Mr. P. S. Tam, Senior Project Manager, NRR
Records Management

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Niagara Mohawk Power Corporation	)	Docket No. 50-410
	)	
(Nine Mile Point Nuclear Station Unit 2)	)	

CERTIFICATION

Richard B. Abbott, being duly sworn, states that he is Vice President Nuclear Engineering of Niagara Mohawk Power Corporation; that he is authorized on the part of said Company to sign and file with the Nuclear Regulatory Commission this certification; and that, in accordance with 10 C.F.R. §50.71(e)(2), the information contained in the attached letter and updated Final Safety Analysis Report accurately presents changes made since the previous submittal necessary to reflect information and analyses submitted to the Commission or prepared pursuant to Commission requirements and contains an identification of changes made under the provisions of §50.59 but not previously submitted to the Commission.

By: Richard B Abbott
Richard B. Abbott
Vice President Nuclear Engineering

Subscribed and sworn to before me
this 20th day of October, 2000

Sandra A. Oswald 10/20/00

SANDRA A. OSWALD
Notary Public, State of New York
No. 01OS6032276
Qualified in Oswego County
Commission Expires 10/25/01

Notary Public in and for

Oswego County, New York

My Commission Expires:

10/25/01

Enclosure A to
NMP2L 1989

IDENTIFICATION OF CHANGES, REASONS AND BASES FOR NMPC-QATR-1 (USAR 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.

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.1.2.1.1.1	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.

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.1.2.1.1.1 (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.

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.2-11, 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. "I" 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
NMP2L 1989

NINE MILE POINT – UNIT 2

SAFETY EVALUATION SUMMARY REPORT

2000

Docket No. 50-410
License No. NPF-69

Safety Evaluation No.: 92-075

Implementation Document No.: Simple Design Change SC2-0318-91

USAR Affected Pages: Figures 9.2-1h, 9.2-1p

System: Service Water (SWP)

Title of Change: Replace SWP Rad Monitor Root Valves

Description of Change:

This simple design change replaced the carbon steel root valves for the sample lines to and from radiation monitoring cabinet 2SWP*CAB23B with stainless steel root valves, enlarged the take-off from the large bore service water to 2", and capped the original 3/4" taps. In addition, a 3/4" test/injection tap was added to allow chemical injection and/or sampling at a later date.

Note: The affected USAR pages were updated in USAR Revision 10, dated November 1998. This safety evaluation was inadvertently omitted from the Safety Evaluation Summary Report that accompanied USAR Revision 10. Accordingly, the description of change and safety evaluation summary are included in this subsequent report.

Safety Evaluation Summary:

This simple design change will not impact the safe operation or shutdown of the plant. The SWP system is designed with suitable redundancy to provide a reliable supply of cooling water to safety-related and essential components and systems. Installation utilizing the "hot tap" method will allow addition of the new connections and root valves without a SWP system outage.

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

Safety Evaluation No.: 93-102 Rev. 3

Implementation Document No.: Simple Design Change SC2-0158-93

USAR Affected Pages: Table 9B.8-1 Sh 1,4,27

System: Residual Heat Removal (RHS)

Title of Change: Remove Internals of 2RHS*V7, V8 and V9

Description of Change:

Check valves 2RHS*V7, V8 and V9, which were originally installed in the minimum flow bypass lines of pumps 2RHS*P1A, B, and C, had been susceptible to the failures associated with the internals of the valves.

This simple design change removed the internals of these check valves, eliminating the open/close function of the valves. This function is not required for system performance. This change may create a condition where standby RHS pump rotates in reverse direction. The reverse rotation of pump 2RHS*P1B, up to 2% of the nominal rpm, was observed during the system test. This condition was evaluated by the vendor and an acceptable limit for the reverse rotation was established. This limit is 10% of the rated rpm.

Safety Evaluation Summary:

This change shall not compromise safety-related qualification of the components, nor shall it compromise integrity and performance of the systems and components important to safety. The change shall not adversely impact performance of the motors and motor protection, and shall not adversely impact the diesel generator loading sequence. The change shall not compromise the system capability to mitigate the effects of fire. The change shall satisfy the system performance requirements.

The analysis performed indicates that the change complies with all applicable criteria and, therefore, does not adversely affect any component, system, or structure important to safety.

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

Safety Evaluation No.: 93-117

Implementation Document No.: Procedures S-RTP-11, S-RTP-15, S-RTP-16, S-RTP-17, S-RTP-46, S-RTP-51, S-RTP-52, S-RTP-52B, S-RTP-53, S-RTP-55, S-RTP-56, S-RTP-71, S-RTP-75, S-RTP-80, S-RTP-82, S-RTP-128, S-RTP-146, S-RTP-155, S-RTP-162, S-RTP-163

USAR Affected Pages: 12.5-3, 12.5-4

System: N/A

Title of Change: Radiation Protection Instrumentation Calibration Frequency Change

Description of Change:

This safety evaluation evaluated a reduction in the frequency of calibration of certain radiation protection instrumentation from quarterly to semi-annually or, for some equipment, at least annually based on documented instrument reliability.

Note: The affected USAR pages were updated in USAR Revision 10, dated November 1998. This safety evaluation was inadvertently omitted from the Safety Evaluation Summary Report that accompanied USAR Revision 10. Accordingly, the description of change and safety evaluation summary are included in this subsequent report.

Safety Evaluation Summary:

The change in the calibration frequency of radiation protection instrumentation involves equipment presently utilized at NMP2. The reduction in survey instrumentation and radiation protection equipment calibration frequency will still provide a high degree of equipment reliability (based on historical calibration records), complies with industry standards (ANSI N323-1978, Regulatory Guide 8.25), is in compliance with regulatory requirements (NUREG-0800, Section 12.5), is consistent with industry practices (INPO 91-014), and maintains the objective of the radiation protection program.

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

Safety Evaluation No.: 95-049 Rev. 2

Implementation Document No.: Procedure N2-OP-52

USAR Affected Pages: N/A

System: Standby Gas Treatment (GTS)

Title of Change: 1-Hour Drawdown Analysis

Description of Change:

Secondary containment integrity requirements, as defined in Technical Specification 3/4.6.5, are applicable during plant operational conditions 1, 2, 3, and *. A separate analysis based on low heat loads of mode * is not available. However, more stringent requirements of modes 1, 2, and 3 interfere during mode * with various actions needed to support outage activities. This revision evaluates and relaxes, for mode * only, the restrictions 2, 3, and 4 listed under the NMP2 Technical Interpretation #25 heading contained in this safety evaluation. This change provides a means of keeping a GTS train operable even when cooling is lost to the associated emergency core cooling system (ECCS) heat exchanger room. Since the unit cooler and the system cannot operate, there will essentially be no heat load in the heat exchanger room. Therefore, loss of cooling will not adversely affect the drawdown analysis results. This revision also incorporated changes as a result of Disposition 00F to Calculation ES-276 for drawdown analysis.

Safety Evaluation Summary:

Since no drawdown analysis based on low heat loads during mode * is available, the drawdown requirements of modes 1, 2, and 3 are imposed for mode *. This revision evaluates and provides alternate actions in the event cooling to the ECCS pump, ECCS heat exchanger, high-pressure core spray (HPCS) pump or reactor core isolation cooling (RCIC) room unit coolers is lost. These alternate actions ensure that no heat is introduced to these rooms when cooling to associated unit coolers is lost, thus ensuring compliance with the drawdown analysis bases.

These actions ensure that the GTS system train(s) can perform the intended safety function under mode *, under postulated accidents, even when cooling to the referenced room unit coolers is lost. This keeps the GTS train(s) operational.

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

Safety Evaluation No.:	95-072 Rev. 0 & 1
Implementation Document No.:	Plant Change Request PC2-0100-97 DDC 2E11339
USAR Affected Pages:	Table 9A.3-18 Sh 2
System:	Fire Detection (FPM)
Title of Change:	Retirement of Detection in Phase Separator Tank Room, Fire Zone 255SW

Description of Change:

This change removed nine fire detectors on Reactor Building El. 300'-0" in fire zone 255SW. The combustible loading in the phase separator tank room was insufficient to warrant fire detection.

Safety Evaluation Summary:

The combustible loading in the south phase separator tank room is insufficient to warrant detection and has been documented in a fire protection engineering evaluation in accordance with NRC Generic Letter 86-10. The Safe Shutdown Analysis and Fire Hazards Analysis are unaffected by this change.

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

Safety Evaluation No.: 95-128

Implementation Document No.: N/A

USAR Affected Pages: N/A

System: N/A

Title of Change: Aerial Radiation Survey of NMP Site

Description of Change:

Section 2.2.3.1.7 of the Unit 2 USAR discusses aircraft crashes, but the discussion is only in reference to airplane flights associated with nearby airports and helicopter flights to and from the site. The NRC contracted EG&G to perform an aerial radiation survey of the Nine Mile Point site. This survey involved helicopter flights directly over the site and, as stated above, the Unit 2 USAR only evaluated flights to and from the site.

Safety Evaluation Summary:

The helicopter flights directly over the site, for the purpose of performing an aerial radiation survey, were evaluated and found to present an insignificant risk of an aircraft crash on site. The helicopter accident rates, a previous Stone & Webster Engineering Corp. calculation, an Argonne National Laboratory Study of Aircraft Crash Hazards, and the NRC Standard Review Plan (SRP) were used to assess the risk associated with the Survey Plan described by the EG&G pilot. The assessment resulted in a crash probability between 8.4E-8 and 7E-7. Per the SRP, this probability is sufficiently low enough that crashes need not be considered as design basis events.

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

Safety Evaluation No.: 95-138

Implementation Document No.: DDC 2M10847

USAR Affected Pages: Figure 11.3-1b

System: Offgas (OFG)

Title of Change: Freezeout Dryer Flushing Connections

Description of Change:

This change installed a tee at the air exhaust and the water drain in the freezer section of freezeout dryer 2OFG-DRY1B, just downstream of 2OFG-TE61B, approximately midpoint on the dryer. This allows for inspection and flushing of the freezeout dryer as required to prevent clogging in the freezer section.

Safety Evaluation Summary:

This change will not adversely affect the function of the OFG system or the dryer. The installation of this tee will allow for inspection and flushing as necessary to clean the internal coils of the dryer. Installation conforms with all applicable design criteria. There is no adverse effect on the system or nuclear safety by the installation of this tee.

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

Safety Evaluation No.:	96-085
Implementation Document No.:	DDC 2F01547
USAR Affected Pages:	Figures 9.2-1a, 9.2-1b
System:	Service Water (SWP)
Title of Change:	Removal of Service Water Pump Vibration Probes

Description of Change:

Vibration probes (two for each service water pump, one at each bearing end) were originally installed to collect the vibration data for analysis to determine the service water pump's readiness in accordance with the Inservice Testing Plan requirements. The probes have been abandoned in place for many years because they do not meet the vibration monitoring code requirements. For this reason, they have been removed from the service water pumps.

Note: The affected USAR pages were updated in USAR Revision 10, dated November 1998. This safety evaluation was inadvertently omitted from the Safety Evaluation Summary Report that accompanied USAR Revision 10. Accordingly, the description of change and safety evaluation summary are included in this subsequent report.

Safety Evaluation Summary:

The vibration probes were not part of the SWP system accident analysis. Their removal does not require entry into the SWP system or require disassembly of pump or other components necessary for the operation of the system. The change has no effect on the design and licensing basis of the SWP system, nor does it directly or indirectly affect equipment important to safety. There are no risks associated with this action which may render the SWP system or its components inoperable. The licensing basis will be maintained during the implementation of this change.

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

Safety Evaluation No.: 96-133

Implementation Document No.: Simple Design Change SC2-0014-94

USAR Affected Pages: 7.5-3, 9.3-13, 9.3-18; Table 7.5-1 Sh 2,3,4,8,9; Figures 5.4-2a, 5.4-13a, 5.4-13b, 5.4-13c, 5.4-13f, 5.4-13g, 6.2-38 Sh 7, 6.2-71a, 6.2-71b, 6.2-72a, 6.2-72k Sh 1, 6.2-73a, 9.2-3a, 9.2-3b, 9.2-3c, 9.2-3d, 9.2-4 Sh 9, 9.3-1g, 9.3-6 Sh 2, 9.3-9f, 9.3-13 Sh 4,8, 9.3-17a, 9.3-20b, 9.4-8k, 9.4-9 Sh 24, 10.1-3e, 10.1-3f, 10.1-6b, 10.4-11 Sh 5,6

System: Off-Normal Status Display (SCI)

Title of Change: Abandon in Place the Off-Normal Status Display System

Description of Change:

This change abandoned in place the off-normal status display system. The system monitored containment isolation by providing position indication for access hatch door, tip tube, and valve positions. The system was marked up since 1986 and has been controlled by hold-out tags since 1990. System power was permanently removed and the system display, located at Control Room panel 2CEC*PNL602, has a permanent label installed to designate the system as abandoned in place.

Safety Evaluation Summary:

The off-normal status display system is only one of several methods of providing information to the Control Room operators. Abandoning the system in place does not affect these methods. The information required for use by the operators will continue to be available by the other methods listed in USAR Sections 7.5 and 9.3.1.

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

Safety Evaluation No.: 97-045

Implementation Document No.: DDC 2F01620

USAR Affected Pages: Figures 9.2-17c, 11.2-1e

System: Condensate Makeup and Drawoff (CNS), Liquid Radwaste (LWS)

Title of Change: Hold-outs for the Liquid Radwaste System

Description of Change:

Valves 2CNS-V556 and 2CNS-V561 were normally open valves in the CNS system to the floor drain filter body feed tank (2LWS-TK12) and the floor drain body feed pump (2LWS-P28). Automatic valves 2LWS-SOV251 and 2LWS-SOV234 are normally closed valves for the lines that provide a flow path for makeup condensate for the above-mentioned equipment. There was concern of leakage across the seat of 2LWS-SOV251 and 2LWS-SOV234 which could raise the level in 2LWS-TK12. Therefore, it was desirable to have a second isolation point for these lines.

The flat bed filter portion of the LWS system was not being used and the flat bed filter had been removed. In order to provide condensate isolation for equipment still installed in the flat bed filter section of LWS, it was desirable to have the normal operating position of the above valves as closed. This safety evaluation evaluated changing the normal operating position of valves 2CNS-V556 and 2CNS-V56 from open to closed and allowing the hold-outs for these valves to remain in place until final design paperwork was issued and Operations Accepted.

Safety Evaluation Summary:

No safety concerns exist with these design enhancements. Potential accidents and malfunctions associated with these changes are bounded by those previously reviewed by the NRC, thus NRC approval is not required.

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

Safety Evaluation No.:	97-050
Implementation Document No.:	DDC 2F01600
USAR Affected Pages:	Figures 9.2-5a, 9.2-5c, 9.2-5d, 9.2-5e, 9.2-6a
System:	Makeup Water (MWS), Water Treatment (WTS)
Title of Change:	As-Built and Hold-Outs for the Makeup Water Treatment System

Description of Change:

The design of the WTS system, as defined in USAR Section 9.2.3, is to remove dissolved and suspended solids from raw lake water. The system is designed to produce demineralized water to the water quality as described in USAR Table 9.2-5.

Major portions of the WTS system have not been used in recent years. Based on economics, an administratively controlled trailer, which is currently controlled under Procedure N2-OP-15 (Safety Evaluation 93-32), is available to provide demineralized water. This safety evaluation evaluated various hold-out and as-built discrepancies within the WTS system.

This change isolated the air supply to valves 2WTS-AOV231 and 2WTS-AOV230 to ensure they would remain in the closed position. Valve 2WTS-V309 was replaced with a spool piece. Also, the normal configuration for valves 2WTS-V407, 2WTS-V1333, 2WTS-V450, 2WTS-PCV177, 2WTS-V295, 2WTS-V237, 2MWS-V89, 2MWS-V90, 2WTS-V156, and 2WTS-V266 was changed from open to normally closed.

Safety Evaluation Summary:

Based on the reviews performed, it has been determined that these changes conform to the applicable criteria. The potential accidents and malfunctions associated with these changes are bounded by those previously reviewed by the NRC, and thus NRC approval is not required.

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

Safety Evaluation No.:	97-081
Implementation Document No.:	Plant Change Requests PC2-0091-95, PC2-0081-97
USAR Affected Pages:	3C-22, 10.4-17; Table 3C.4-6; Figures 10.4-7a, 10.4-7b, 10.4-7d, 10.4-7e, 10.4-7h
System:	Circulating Water (CWS), Condensate Demineralizer (CND), Instrument Air Service (IAS), Liquid Radwaste (LWS), Makeup Water (MWS), Service Water Chemical Treatment (SCT), Water Treatment (WTS), Acid Chemical Feed (WTA), Waste Solidification (WSS), Service Water (SWP), Water Treatment Hypochlorite (WTH)
Title of Change:	Abandonment of Screenwell Building Acid System and Hypochlorite Chemical Feed System

Description of Change:

Portions of the WTA and WTH systems have been previously abandoned in place. This safety evaluation addresses the entire WTH system, and specifically defines the system boundaries between the functional and abandoned system/equipment. This boundary definition is required to maintain plant configuration, since the abandoned equipment will no longer be functionally maintained or its configuration controlled in the plant's design database. In addition, the WTH system in USAR Table 3C.4-6 has been deleted since it has no contribution to NMP2 flooding analysis in the Screenwell Building.

Safety Evaluation Summary:

The abandoned equipment (WTA and WTH) is not required for safe reactor shutdown or accident mitigation. The change will not adversely affect the design function of any structure, system, or component required for safe plant operation or shutdown. Based on analysis, the change does not increase the consequences of accidents previously evaluated in the USAR, and does not affect the safe operation or shutdown of the plant. Implementation of the change is in compliance with NRC standards and does not require a Technical Specification change. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.: 97-085 Rev. 0, 1 & 2

Implementation Document No.: Mod. N2-97-048

USAR Affected Pages: 6.5-8, 6.5-9, 9A.3-36; Tables 3.2-1 Sh 13, 34, 3.9A-4 Sh 4, 5, 3.9A-12 Sh 3, 13, 3.10A-1 Sh 10, 8.3-2 Sh 9, 8.3-4 Sh 11, 12, 9A.3-16, 9B.8-3 Sh 3; Figures 6.5-1 Sh 1, 1A, 2, 2A, 4, 4A, 5, 5A, 8, 8A, 9.3-20b, 9.4-8L, 9.4-8m, 9A.3-24

System: Standby Gas Treatment (GTS), Instrument Air Service (IAS)

Title of Change: GTS Actuator Replacements

Description of Change:

The standby gas treatment system (GTS) consists of two independent filter trains. Valves 2GTS*MOV2A/B are the valves which open and close to provide inlet air to their respective filter trains. 2GTS*MOV3A/B open and close to provide effluent discharge to the main stack. 2GTS*MOV28A/B open and close to provide cross connection of the two trains as required by certain operational modes. Valves 2GTS*PV5A/B modulate to provide recirculation for their respective trains as well as to maintain the required -1/4" W.G. vacuum for secondary containment integrity. Due to the excessive maintenance requirements of the Borg Warner valve actuators, they were replaced with pneumatic actuators. The pneumatic actuators are also provided with a means of manual actuation. Each air supply line originates from a tap connection into an IAS line above El. 261' in the Standby Gas Treatment Building. The instrument air pressure is boosted to the required operating pressure via in-line nonsafety-related Haskell air amplifiers. This change included the addition of two safety-related accumulators, one for each GTS train to be placed in the respective rooms of the Standby Gas Treatment Building, El. 261'. Each accumulator has its own air supply line. Air is supplied to the accumulators via the air amplifier at a pressure required to maintain valve operability for a five-day post-accident period. Lines from the accumulators feed a header (one per train) which supply branch lines to each of the four new actuators per train. Each supply header has a pressure regulator. The pressure regulator regulates the supply pressure to maintain the required pressure to the new AOVs. Control Room annunciation and computer points were provided for low pressure conditions for the accumulators and high and low pressure switches on the supply headers. A local pressure indicator accompanied each pressure switch. Accumulator tank sizing is adequate for non-design basis accident scenarios to provide a five-day supply of air to the valves should a loss-of-air supply occur.

Safety Evaluation No.: 97-085 Rev. 0, 1 & 2 (cont'd.)

Description of Change: (cont'd.)

After five days, provisions are made to use existing safety-related, missile-protected, emergency fill connections to refill the accumulators. The accumulator tanks and refill connections serve to implement the 100-day post-accident requirements.

A 4-psid door was installed at the entrance to the Standby Gas Treatment Building from the reactor track bay, door SG261-3. This door improves the environmental qualification conditions seen after a high-energy line break. The improved environment conditions are required for qualification of the new positioner that controls 2GTS*PV5A/B as well as existing equipment. This safety evaluation was revised to address the USAR changes documented in LDCR Nos. 2-97-UFS-154 Rev. 1 and 2-98-UFS-042. LDCR 2-97-UFS-154 was revised to reflect the installation of Train B, without modifying Train A. The second train is currently scheduled for completion in the fall of 2000. The installation date for this train exceeds the time limit for updating the USAR to reflect Train B. As such, an interim revision is required. The LDCR will be revised following the installation of Train A to reflect the common configuration. LDCR 2-98-UFS-042 updated the USAR to reflect a new penetration installed in the secondary containment. The penetration was intended to provide a supply of nitrogen to the GTS accumulators. The penetrations were abandoned when the system design was modified to use instrument air as the motive force in lieu of nitrogen. LDCR 2-97-UFS-156 remains unchanged.

Safety Evaluation Summary:

This change replaces the electro-hydraulic actuators on the GTS valves with pneumatic actuators. This change only affects the motive force by which the valves are moved; the requirements of the valves (stroke time, train operation, modulation, etc.) shall meet or exceed the existing requirements. The proposed change does not have any adverse effects on the operation of the GTS system trains. Based on this analysis, the proposed change does not increase the probability or consequences of an accident or malfunction previously evaluated in the USAR and does not create an accident or malfunction of a different type. The change does not reduce the margin of safety as defined in the basis for any Technical Specification and has no adverse 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.:	97-086 Rev. 0, 1, 2, 3, 4, 5, 6 & 8
Implementation Document No.:	Mod. N2-89-076
USAR Affected Pages:	8.3-37, 8.3-38, 10.4-27, 10.4-39, 11.3-6; Table 3.2-1 Sh 18a; Figures 10.1-5b, 10.1-6b, 10.1-10a, 10.4-2a, 10.4-10 Sh 19, 11.3-1a, 11.3-1b
System:	Hydrogen Water Chemistry (HWC), Condensate (CNM), Feedwater (FWS), Offgas (OFG), Oxygen Feedwater Injection (OFI), Condensate Air Removal (ARC), Instrument Air Service (IAS)
Title of Change:	Hydrogen Water Chemistry Addition (HWC) System

Description of Change:

EPRI Special Report NP-5283-SR-A provides guidelines for the addition of a permanent HWC system to control intergranular stress corrosion cracking (IGSCC). The primary reactor coolant is demineralized water containing 100 to 300 ppb dissolved oxygen from the radiolytic decomposition of water. Injecting hydrogen into the feedwater suppresses the radiolytic formation of oxygen in the reactor core. Reduction of IGSCC is achieved from the reduction of oxygen and other oxidizing species in the reactor coolant. The reduction of oxygen in the reactor coolant results in a reduction of the electrochemical corrosion potential (ECP), the measurement scale that is commonly used to predict initiation and growth of IGSCC. When the ECP is below -230 mV SHE, IGSCC essentially stops. At ECP values greater than -230 mV, but less than normal water chemistry (without HWC) values of +150 mV to 200 mV SHE, IGSCC is reduced. For boiling water reactors with high density cores similar to NMP2, injection of 1.0 to 1.6 ppm of hydrogen in the feedwater will result in ECP values less than or equal to -230 mV SHE in the recirculation piping and lower plenum of the reactor pressure vessel. Lower hydrogen injection rates (e.g., approximately 0.3 ppm) are required to achieve ECP value <-230 mV in plants that have been treated with NobleChem.

This change added a permanent HWC system at NMP2. The purpose of the HWC program is to reduce rates of IGSCC in the recirculation piping and reactor vessel internals. The HWC system includes the flow monitoring and control equipment for both hydrogen and oxygen, a system control panel, and an offgas oxygen monitor panel. The Control Room interface includes a shutdown switch, annunciation and status lights.

Safety Evaluation No.: 97-086 Rev. 0, 1, 2, 3, 4, 5, 6 & 8 (cont'd.)

Description of Change: (cont'd.)

After modification functional testing performed at completion of installation of the HWC injection system during RF06, logic testing was performed on HWC injection panels 100, 200, 300, 400, and 500 to demonstrate the functionality of the control, annunciation, and shutdown of the HWC injection system.

During the tuning and ramping test, disabling of the trip function of the offgas pretreatment radiation monitors (2OFG-RE13A and B) was procedurally required resulting in entry into Action (b) of Current Technical Specification 3.3.7.10 (Action b of Offsite Dose Calculation Manual). Disabling of the offgas radiation monitor ensures that isolation of the OFG system will not occur as a result of potentially high radiation resulting from an increase in the radiation monitor background as a result of hydrogen injection. After the testing, the offgas pretreatment radiation monitors were returned to service.

To prevent the trip of the OFG system during hydrogen injection during this test, and for future hydrogen injection, offgas monitor trip function of the offgas high-high hydrogen concentration monitors (2OFG-AT16A/B) has been permanently removed. Since the OFG system is designed to contain the effects of a hydrogen detonation, removal of trip function conforms to RG 1.143 and the NRC Standard Review Plan.

Within 24 hours prior to the start of hydrogen injection, the normal full-power radiation background level and associated trip and alarm setpoints were changed based on a calculated value of the radiation level. Within 24 hours after completion of the ramp test, the background radiation level was determined and associated trip and alarm setpoints were reset.

Safety Evaluation Summary:

This change includes the installation of the HWC system piping and components and their tie-in to the plant. The system tie-in to the OFG system was designed to meet the explosion parameters specified in the original design. All installations at interface points meet or exceed existing system requirements.

The results of this analysis determined that the installation of the HWC piping and, purging a portion of the oxygen injection piping through the OFG system, performing logic testing on HWC injection panels including extended reliability testing of offgas panel 300, and tuning/ramping that involves injection of

Safety Evaluation No.: 97-086 Rev. 0, 1, 2, 3, 4, 5, 6 & 8 (cont'd.)

Safety Evaluation Summary: (cont'd.)

hydrogen and oxygen does not a) increase the probability of occurrence or consequences of an accident previously evaluated in the USAR, b) increase the probability of occurrence or consequences of a malfunction of equipment important to safety evaluated previously in the USAR, c) create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR, or d) 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.:	97-090
Implementation Document No.:	Procedure N2-FSP-FPP-R001
USAR Affected Pages:	9A.3-32, 9A.3-34, 9A.3-35; Figure 9A.3-2
System:	N/A
Title of Change:	Modification of the Installation and Visual Surveillance Requirements for Internal Conduit Seals

Description of Change:

USAR Sections 9A.3.5.1.1 and 9A.3.5.1.2 document the design basis, the surveillance requirements, and the operability action statements for fire barriers and fire barrier penetration seals. This safety evaluation evaluated modification of 1) the penetration seal program to eliminate internal conduit seals for closed conduits and configurations of conduits 2 inches in diameter and smaller having specific installation parameters; 2) removal of the requirement that internal conduit seals be fire rated; and 3) elimination of regular visual surveillance inspections for internal conduit seals.

Note: The affected USAR pages were updated in USAR Revision 10, dated November 1998. This safety evaluation was inadvertently omitted from the Safety Evaluation Summary Report that accompanied USAR Revision 10. Accordingly, the description of change and safety evaluation summary are included in this subsequent report.

Safety Evaluation Summary:

The information documented in this safety evaluation demonstrates that the changes to the penetration seal program will not degrade the operability of the fire barriers installed in the plant. The fire test findings contained in the Conduit Fire Protection Research Program conclusively demonstrate that internal conduit seals are unnecessary for closed conduit configurations; conduits that are less than 1 inch in diameter, extending 1 foot on the other side of the barrier; and conduits between 1 inch and 2 inches in diameter having specified cable fills and open end termination distances from the fire barrier. The evaluation also demonstrates that routine visual surveillance (every 18 months) of internal conduit seals is unnecessary. The surveillance of these seals is not necessary because silicone-based fire barrier seals are age independent and are not susceptible to thermally accelerated aging. Internal seals are also fully enclosed and inaccessible and,

Safety Evaluation No.: 97-090 (cont'd.)

Safety Evaluation Summary: (cont'd.)

therefore, are immune to the normal wear and tear type deterioration non-internal conduit seals are susceptible to.

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

Safety Evaluation No.:	97-093 Rev. 0, 1 & 2
Implementation Document No.:	Mod. N2-97-015
USAR Affected Pages:	6.2-73; Tables 3.9A-12 Sh 1,3, 6.2-56 Sh 5,6, 6.2-65; Figures 9.2-3c, 9.2-3d, 9.3-9e, 9.3-9f
System:	Reactor Building Closed Loop Cooling Water (CCP), Drywell Drains (DER), Reactor Building Drains (DFR)
Title of Change:	Thermal Overpressure Protection for Primary Containment Penetrations

Description of Change:

This modification installed spring-loaded relief valves between the isolation valves of six primary containment penetrations to provide thermal overpressure protection. The penetrations are 2CCP*Z33A, *Z34A, *Z46A and *Z47, and 2DFR*Z39 and 2DER*Z40. The potential for overpressurization was identified by Generic Letter 96-06, "Assurance of Equipment Operability and Containment," where post-accident containment temperatures may heat and expand the entrapped fluid between the containment isolation valves (CIV). Analysis of these six penetrations has shown that overpressurization may occur without relief of the expanded fluid.

The relief valves are located inside primary containment. In addition to their relief function, the relief valves also serve a primary containment isolation function.

The modifications to the DER and DFR penetrations required a Technical Specification (TS) change to revise the leak rate limits specified in Table 3.6.1.2-1. The TS change was issued as Amendment 88. Safety Evaluation 97-093 supplements the amendment request in addressing the detailed design aspects of the change. The scope of Safety Evaluation 97-093 is limited to the installation of the DER and DFR modifications during plant shutdown when primary containment integrity is not required.

Safety Evaluation No.: 97-093 Rev. 0, 1, & 2 (cont'd.)

Safety Evaluation Summary:

The results of this analysis determined that the installation of relief valves in between the isolation valves of the subject CCP, DER and DFR systems will satisfy ASME Section III requirements and maintain the integrity of primary containment. Thus, the change does not a) increase the probability of occurrence or consequences of an accident previously evaluated in the USAR, b) increase the probability of occurrence or consequences of a malfunction of equipment important to safety previously evaluated in the USAR, c) create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR, or d) 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.: 97-097

Implementation Document No.: Mod. N2-97-065

USAR Affected Pages: Table 3.9A-10 Sh 1 thru 4

System: Control Building Chilled Water (HVK)
Service Water (SWP)

Title of Change: Addition of Shear Blocks for Chilled Water Pump Motor Hold-Down Bolts and USAR Table 3.9A-10 Update

Description of Change:

This modification installed a shear block on 2HVK*P1A and B motor skids to alleviate the shear load from the ASTM A307 motor hold-down bolts to assure the long-term function of the joint. Also, while performing the analysis for this design change, it was noted that the stress and deflection values for chilled water pumps 2HVK*P1A and B and service water pumps 2SWP*P1A through F needed to be updated to agree with the latest analysis.

Note: The affected USAR pages were updated in USAR Revision 10, dated November 1998. This safety evaluation was inadvertently omitted from the Safety Evaluation Summary Report that accompanied USAR Revision 10. Accordingly, the description of change and safety evaluation summary are included in this subsequent report.

Safety Evaluation Summary:

Pumps 2HVK*P1A and B are properly secured and the addition of the shear blocks will provide additional long-term function of the motor hold-down bolts. There is no impact on the performance and safety function of the pumps, thus the function of the HVK system remains unchanged. The stresses as shown on revised USAR Table 3.9A-10 for the chilled water pumps and service water pumps are all less than their corresponding allowable limits. These changes do not affect any initiators and precursors of an accident as discussed in Chapter 15 of the USAR. Therefore, the probability of occurrence of an accident previously evaluated in the USAR will not be increased.

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.: Fire Protection Engineering Evaluation FPEE 0-98-001

USAR Affected Pages: 9A.3-42

System: N/A

Title of Change: Sealed-Beam Portable Hand Lights

Description of Change:

This change revised USAR Section 9A.3.5.7.2 to eliminate the reference to "sealed-beam" battery-powered hand lights. Fire Brigade and Operations personnel use non-sealed beam type battery-powered portable hand lights when attending to emergency conditions.

Safety Evaluation Summary:

Portable hand lights are used to supplement the emergency and 8-hour battery pack lighting systems. Only the emergency and 8-hour battery pack lighting systems are required to achieve and maintain safe shutdown.

The non-sealed 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, 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 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.:	97-122
Implementation Document No.:	Procedures GAP-POL-01, NEP-POL-01
USAR Affected Pages:	13.1-3, 13.1-4, 13.1-7, 13.1-12; Figures 13.1-2, 13.1-3
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 associated with the plant process computers 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 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 organization change is analyzed against Technical Specification requirements for organization lines of authority, responsibility, and staff qualifications. Evaluation of the 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.: 97-155

Implementation Document No.: DDC 2M11515

USAR Affected Pages: Figure 9.5-1b

System: Fire Protection Water (FPW)

Title of Change: Resolution of ANI Recommendation FS95-02

Description of Change:

This change installed two new wall post-indicating valves (PIV) to replace two curb-type fire protection valves that were buried and inaccessible. The USAR has been revised to add these new wall PIVs and to document that curb valves 2FPW-V133 and 2FPW-V438 are abandoned in the open position, buried and inaccessible.

Safety Evaluation Summary:

This change leaves these curb valves buried and inaccessible on the FPW system and replaces OS&Y valves with wall PIVs as a substitute for the curb valves. These valve changes do not impact or adversely affect the probability of occurrence of a fire in the areas containing systems, structures or components important to safety or safe shutdown. Furthermore, it does not impact the fire protection systems that protect areas containing systems, structures or components important to safety or safe shutdown. This change does not impact the Fire Hazards Analysis or Safe Shutdown Analysis.

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

Safety Evaluation No.:	97-157 Rev. 0 & 1
Implementation Document No.:	Procedure GAP-POL-01
USAR Affected Pages:	13.1-11, 13.1-12; Figure 13.1-2
System:	N/A
Title of Change:	Reorganization of Maintenance Branch in Accordance with Procedure GAP-POL-01, Rev. 18

Description of Change:

GAP-POL-01, "Composition and Responsibility of the Nuclear Generation Organization," has been revised to show the reorganized Maintenance Branch. This revision changes the functional areas within the Maintenance Branch and also shows that the number of functional areas may vary. The functional areas of FIN, Valves, and Outage are being added to the existing areas of Electrical, Mechanical, I&C Maintenance, and Maintenance Support.

Safety Evaluation Summary:

The procedure changes describe the reorganized Maintenance Branch. The organization satisfies the Standard Review Plan criteria for Operating Organizations. The change does not impact the requirements of Technical Specification 6.2 (Onsite and Offsite Organizations). The changes do not impact accident initiation or consequences.

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

Safety Evaluation No.:	97-161
Implementation Document No.:	LDCR 2-97-UFS-153
USAR Affected Pages:	9A.3-52; Table 9A.3-15 Sh 1 thru 4
System:	Fire Protection Water (FPW)
Title of Change:	Update and Revision of USAR Table 9A.3-15

Description of Change:

USAR Table 9A.3-15 is a tabulation of the sectional and isolation valves in the FPW system supervised to ensure an operable flow path to all FPW systems and equipment protecting safety-related equipment. USAR Table 9A.3-15 was revised and updated to incorporate previously approved design changes made to the system, and to limit the table to valves that serve USAR required FPW systems and equipment. USAR Section 9A.3.6.3.2 was revised to more specifically state that Table 9A.3-15 includes only FPW system sectional and isolation valves that serve safety-related equipment. This change reconciles USAR Table 9A.3-15 with USAR Figure 9.5-1 and the current design configuration of the plant.

Safety Evaluation Summary:

A review of USAR Table 9A.3-15 revealed that the table included most valves in the FPW system and was not current with the plant design. An evaluation was performed and it was determined that many FPW valves listed in the table serve only balance of plant (BOP) FPW systems and equipment, and their valve position does not affect FPW systems and equipment designated to protect safety-related equipment. The evaluation also found editorial discrepancies between the table and the actual "as-built" condition of the FPW system. LDCR 2-97-UFS-153 corrects Table 9A.3-15 so it accurately depicts only the valves required to be supervised to assure an operable flow path to all FPW systems that protect safety-related equipment. The LDCR also clarifies USAR Section 9A.3.6.3.2, which references which valves are required to be in Table 9A.3-15. This evaluation does not change the system design bases or as-built configuration of the FPW system.

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

Safety Evaluation No.: 98-012 Rev. 1 & 2

Implementation Document No.: Mod. N1-94-007

USAR Affected Pages: Figure 1.2-1

System: Fire Protection-Water Foam, Fuel Oil Handling & Storage

Title of Change: Remove and Replace the Diesel Fire Pump Underground Storage Tank and Remove the Vehicle Fueling Station Underground Storage Tanks

Description of Change:

The NMP1 diesel fire pump underground storage tank and the two vehicle fuel storage tanks are required to meet the requirements of 40CFR280, "Technical Standards and Corrective Action Requirements for Owners and Operators of Underground Storage Tanks (UST);" 40CFR112, "Oil Pollution Prevention;" 6NYCRR613, "Handling and Storage of Petroleum;" and 6NYCRR614, "Standards for New and Substantially Modified Petroleum Storage Facilities," all of which govern underground oil storage tanks. 40CFR280 states that all underground tanks must be upgraded to new tank standards or replaced by new tanks no later than December 22, 1998. The work performed under this safety evaluation to bring NMP1 into compliance with the EPA and DEC regulations included: 1) removal and replacement of the diesel fire pump underground storage tank and the associated underground piping, and 2) removal of the two vehicle fuel underground storage tanks and the associated piping and pumps.

The 2000-gallon diesel fire pump underground storage tank was removed and replaced with a 2500-gallon double-wall fiberglass underground storage tank. The underground bare steel piping was replaced with fiberglass underground piping and a pipe chamber at the tank. A liquid level gauging system to monitor the tank level was installed with the tank for leak detection. A 56 foot by 16 foot by 8 inch high concrete spill containment pad was also installed for containment of a spill that could occur during the filling of fuel into the tank. The old 2000-gallon vehicle fuel underground storage tanks, the two vehicle dispensing pumps, and the concrete island have been removed.

Safety Evaluation Summary:

A temporary modification will supply fuel oil to the day tank through the use of a 2710-gallon diesel truck, which will ensure that fuel oil is available to replace the removed underground tank at all times. The removal and replacement of the diesel

Safety Evaluation No.: 98-012 Rev. 1 & 2 (cont'd.)

Safety Evaluation Summary: (cont'd.)

fire pump underground tank and piping will be accomplished after the temporary modification is implemented; therefore, the function of the diesel fire pump and the fire water system will be unaffected. The vehicle fueling station underground tanks and equipment are outside the restricted area and are not connected to any plant systems. The analysis of the NMP2 probable maximum precipitation (PMP) flooding impact on the NMP site has been reviewed and it has been determined that the installation of the concrete containment pad will have no impact on the NMP2 PMP analysis. Cutting of the tanks will be performed outside the protected area in an area away from overhead lines and safety-related buildings. Precautions will be implemented to ensure personnel safety and to eliminate the possibility of an explosion accident.

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

USAR Affected Pages: Figure 1.2-1

System: N/A

Title of Change: Demolition of Steel Fab Shop

Description of Change:

This safety evaluation addressed the demolition of the Steel Fabrication Shop located east of the Unit 2 plant structures.

This building was built for use as a temporary building during the construction of Unit 2.

Safety Evaluation Summary:

Changes are required to USAR figures that depict structures at Nine Mile Point. LDCRs have been generated to change these figures. The building to be 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 being demolished has no impact on the previously calculated X/Q values. The design margins for the Control Room fresh air intakes are not compromised. 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.:	98-027
Implementation Document No.:	Procedures N2-CTP-CWS-W801, N2-CTP-CWS-D808, N2-CSP-2D and N2-OP-10A
USAR Affected Pages:	2.3-23, 10.4-16, 10.4-17, 10.4-25, 10.4-26; Table 10.4-2
System:	Circulating Water (CWS), Condensate (CNM)
Title of Change:	Revision of the Circulating Water System Chemical Composition and Use of DeposiTrol BL5323

Description of Change:

In a continued effort to a) reduce the potential for scaling, b) reduce costs, and c) minimize environmental effects, use of a new Cooling Tower scaling inhibitor, BetzDearborn DeposiTrol BL5323, has been evaluated. This chemical is a blend of a sulfonated copolymer (HPS1) and HEDP (hydroxyethylidene diphosphonic acid) and, therefore, will serve as both a scale inhibitor and as a dispersant. Use of this product will permit operation of the CWS at cycles of concentration (COC) as high as 4 (i.e., conductivity as high as 1594 mS/cm) and at pH levels as high as 9.0, the maximum pH allowed by the New York State SPDES Permit, without the potential for scaling of the condenser tubes.

Safety Evaluation Summary:

Assuming a "conservative" CWS concentration matrix which corresponds to current, cycled up CWS system conditions, the condensate demineralizers will maintain design effluent water quality both during normal operation and during the 88 gpm design base tube leak event. Minimum resin capacities of 0.13 meq/ml are needed to meet these design requirements. The USAR requirement that polisher resin beds be replaced when anion resin hydroxyl sites fall below 0.4 meq/ml (66.7% of total capacity utilized) will be retained. This requirement provides a reserve capacity that is more than three times the reserve required (10.1%) for an 88 gpm postulated leak rate during a six-hour orderly shutdown.

Monitoring of ion exchange resin capacity will continue using a validated computer tracking program. Use of the computer tracking program for resin tracking obviates the need for resin sampling described in Regulatory Guide 1.56. Use of this program meets the design assumptions regarding methodology for monitoring ionic loading on condensate demineralizer beds.

Safety Evaluation No.: 98-027 (cont'd.)

Safety Evaluation Summary: (cont'd.)

The potential impact of BL5323 on condensate/feedwater water quality and on nuclear fuel is lower for DeposiTrol BL5323 than for Powerline 3450. Compared to the approximate 300 ppm sulfate present in the tower from lake contribution and acid addition, the sulfate contribution from BL5323 is small. Acid functional groups that were not present in the Powerline 3450 will serve to improve the BL5323's relative affinity for anion exchange resins. Thus, the potential impact of BL5323 on condensate/feedwater water quality and on nuclear fuel is lower for BL5323 than for Powerline 3450.

The increase in salt deposition from Cooling Tower drift with increase of CWS COC to 4 times is considered insignificant considering other relative errors in the assumptions used in the drift model in Appendix 2D of the USAR. With respect to the impact of drift from the addition of Cooling Tower chemical treatment additives including BL5323, it can be seen that deposition rates and airborne concentration rates resulting from Cooling Tower drift are insignificant in comparison to associated limits.

Based on the review conducted, increasing the CWS COC to 4 times (conductivity as high as 1594 mS/cm and pH levels as high as 9.0) and addition of BL5323 at rates permitted by the New York State SPDES Permit do not involve an unreviewed safety question.

Safety Evaluation No.:	98-028
Implementation Document No.:	Mod. N2-89-076
USAR Affected Pages:	9.5-87, 10.4-30; Figure 10.1-6e
System:	Zinc Injection Passivation (ZIP)
Title of Change:	Installation of Passive Depleted Zinc Oxide Skid

Description of Change:

This modification replaced the injection of natural zinc addition with a zinc oxide depleted in the Zn-64 isotope. Also, the existing skid that had active components was replaced with a new skid with only passive components.

Safety Evaluation Summary:

The ZIP system is not required to effect or support safe shutdown of the reactor or to perform in the operation of any reactor safety features. The addition of zinc is a water chemistry management system and is not considered in any of the USAR transients or accident analyses. The two proposed changes will not alter the function or classification of this system.

Replacement of the natural zinc with a zinc oxide depleted in the Zn-64 isotope will reduce high-energy gamma radiation. Replacing the existing skid with a new skid with only passive components simplifies the system and increases reliability and reduces maintenance on the system. Neither change affects the function or concentration of zinc in the reactor recirculation water.

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

Safety Evaluation No.:	98-030
Implementation Document No.:	Calculation Dispositions PR-C-24-G-05B, PR-C-24-D-02B
USAR Affected Pages:	Table 12.3-3 Sh 1 & 2
System:	Post-Accident Sampling (SSP) (PASS)
Title of Change:	Revision of Post-Accident Access Doses and LOCA Doses

Description of Change:

During the NMP2 power uprate project, it was found that a non-conservative reactor water volume was used to calculate PASS sample concentrations, which resulted in non-conservative post-accident sampling and analysis doses being calculated and provided in USAR Table 12.3-3. While in the process of revising the applicable calculations and USAR sections, it was found that corrections to other vital area doses were required.

As a result of the changes made to design basis calculations, personnel doses for PASS sampling, transport, and analysis, and post-accident vital area access have been increased in USAR Table 12.3-3.

Safety Evaluation Summary:

This change is restricted to the recalculation of post-LOCA (loss-of-coolant accident) sampling/analysis doses and the correction of Control Room LOCA doses. None of the initiators or precursors to any USAR Chapter 15 accidents are changed and, therefore, the probability of the occurrence of a LOCA or any USAR Chapter 15 accident is not increased. All recalculated doses remain within GDC 19 dose limits as described in NUREG-0737 and the USAR.

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

Safety Evaluation No.: 98-032

Implementation Document No.: Second Ten-Year ASME Section XI Program Plans

USAR Affected Pages: 1.10-41, 1.12-28, 1.12-29, 3.9A-23, 3.9A-25, 3.9A-37, 3.9A-38, 3.9A-39, 3.9B-43, 5.2-15, 5.2-27, 5.2-29, 5.2-30, 5.2-31, 5.3-2, 5.3-10, 5.3-16, 5.3-17, 5.4-11, 6.2-59, 6.2-60, 6.2-87, 6.2-88, 6.6-1, 6.6-2, 6.6-3, 9.2-3, 9.3-6, 9.3-8, 10.3-4, 17.1-24; Tables 1.3-9 Sh 8, 1.8-1 Sh 47, 1.9-1 Sh 28,33, 1.11-1 Sh 5,9, 6.2-55a Sh 2, 6.2-55b Sh 2, 6.2-55c Sh 2, 6.2-55d Sh 2

System: N/A

Title of Change: USAR Changes to Support Second 10-Year Interval for ASME Section XI Programs

Description of Change:

This change updated and corrected the Regulatory, Code, and Program Plan references to the ASME Section XI programs (ISI, ISPT, IST) in the USAR. Changes to the USAR text clarify the Regulatory and the Code basis for the ASME Section XI programs. The changes identify the Code references that pertain to the preservice inspection and test program plan, the Code references to the first 10-year ASME Section XI program plans, and the regulatory basis for the second and successive 10-year ASME Section XI program plan updates.

Safety Evaluation Summary:

The proposed changes define the commitment of the ISI, ISPT, and IST program plans to the requirements of 10CFR50.55a. This commitment neither increases, changes, nor decreases the level of commitment of these program plans to the regulations or to the ASME Code incorporated by reference into the regulations. Although this proposed change affects the USAR, these changes do not increase the probability of occurrence of a previously analyzed accident, either singly or in combination with the other changes.

The level of testing, the frequency of testing, and the test acceptance criteria remain at least as conservative as the requirements of the regulation and the Code. This level and frequency of testing has been shown by local and industry experience not to increase the probability of an analyzed accident. Since the level

Safety Evaluation No.: 98-032 (cont'd.)

Safety Evaluation Summary: (cont'd.)

of inservice inspection and testing remains at or above that required by 10CFR50.55a, this change does not increase the probability of occurrence of an accident previously evaluated in the USAR.

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

Safety Evaluation No.:	98-041
Implementation Document No.:	Mod. N2-97-086
USAR Affected Pages:	N/A
System:	Secondary Containment
Title of Change:	Secondary Containment Penetrations to Support Chemical Decontamination

Description of Change:

This modification installed two permanent penetrations through the Reactor Building wall on El. 261', Azimuth 240. At that location there are unused, sealed conduit sleeves which were reopened and modified. The results of the change were two new empty but sealed penetrations in secondary containment. These new penetrations have been added to support any future use such as the passage of power cables or other needs. Currently, these penetrations are planned to be used for power cables to support reactor coolant system chemical decontamination. This use, as well as any other future uses, will be addressed in separate evaluations. The penetrations were filled with safety-related silicone elastomer sealant which serves as the secondary containment barrier when the penetrations are not in use. The penetrations were filled on the inside and outside with a minimum of six inches of silicone elastomer sealant in accordance with approved site procedures.

Safety Evaluation Summary:

The new penetrations are designed and qualified for their intended purpose as secondary containment isolation, including consideration of seismic qualification and missile protection. It is sealed with safety-related sealant in accordance with approved qualified site procedures and specifications. The installation sequence will include the necessary steps and verification to ensure that installation while the plant is at power does not adversely impact secondary containment requirements. Based on this analysis, the proposed change does not increase the probability or consequences of an accident or malfunction previously evaluated in the USAR and does not create an accident or malfunction of a different type. The proposed change does not reduce the margin of safety as defined in the basis for any Technical Specification and has no adverse 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.: 98-042

Implementation Document No.: Calculation A10.1-0-93

USAR Affected Pages: 9.2-12, 9.2-13, 9.2-14, 9.2-14a; Table 9.2-3

System: Reactor Building Closed Loop Cooling (CCP)

Title of Change: CCP USAR Updates

Description of Change:

The CCP system and auxiliary equipment cooled by the CCP system are designed for 95°F water temperature. In reality, the CCP system is generally operated at a lower temperature due to the most limiting components cooled by the CCP system, drywell unit coolers. The maximum operating CCP temperature limit is imposed to maintain the drywell average temperature to no greater than 135°F. Thus, at least 15°F margin will normally exist with respect to the Technical Specification LCO (for the average drywell temperature). This change updated the USAR section which describes the CCP system.

Safety Evaluation Summary:

This change will update the CCP system in the USAR to be consistent with its design documents and current operating practices. The proposed changes will not result in any physical modification to the plant.

As stated in USAR Section 9.2.2.2, the CCP system is not required to mitigate any accident or transient evaluated in the USAR. The CCP system is not required to operate during an emergency or accident condition. However, during an emergency condition, portions of the CCP system serve as a seismic Category 1 pressure boundary for backup cooling provided from the service water system. These portions of the CCP system 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.:	98-052
Implementation Document No.:	Calculation FPW-028 Disposition 8C
USAR Affected Pages:	Tables 9A.3-4 Sh 1, 9A.3-18 Sh 2, 9B.6-1 Sh 1, 9B.6-3 Sh 1, 9B.8-1 Sh 13, 9B.8-2 Sh 5; Figures 9.5-1e, 9A.3-3
System:	Fire Protection Water (FPW)
Title of Change:	Fire Hazards Analysis Update of Fire Zone 306NW & USAR Table 9A.3-4

Description of Change:

This change revised USAR Table 9A.3-4 to remove the cable tray water suppression reference for fire zone 306NW and to update the fire loading to include the combustibles located in the Operations Records Storage Room. This change also revises the fire zone designation from 306NW to 306NZ.

Safety Evaluation Summary:

This evaluation concluded that water spray or sprinkler protection is not necessary to protect the cabling or components in this fire zone. The NMP2 Safe Shutdown Analysis credits the loss of all equipment in this fire zone (as well as all other fire zones in fire area 16) and 3-hour fire separation of redundant trains to achieve the objective of maintaining one train free from fire damage. The evaluation also determined that the increase in the fire loading in this fire zone does not increase the postulated consequences of a fire. This change is acceptable because it does not adversely affect any postulated consequence of a fire on any structure, system, or component important to safety; has no effect on the ability of the plant to achieve and maintain safe shutdown; and satisfies the 10CFR50.48 requirements.

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

Safety Evaluation No.:	98-058
Implementation Document No.:	Mod. N2-98-080
USAR Affected Pages:	Table 3.9A-12 Sh 4,14; Figure 9.4-1a
System:	Control Building Chilled Water (HVK)
Title of Change:	The HVK Makeup Water Supply Pressure Control Valve Upgrade and Pressure Relief Valve Addition

Description of Change:

This modification replaced the existing nonsafety-related pressure control valve set at 35 psig with a safety-related pressure control valve set at 25 psig (to maintain the expansion tank level). This change also installed a safety-related pressure relief valve set at 50 psig in each train of the HVK system makeup line downstream of the safety-related pressure regulating valve.

Safety Evaluation Summary:

The HVK system supplies chilled water during normal operation, plant shutdown, and design basis accident conditions to air conditioning units serving the Control Room, Relay Room, Remote Shutdown Room, and Computer Room. The HVK is a supporting system for the Control Building environment and does not act as an initiator to any accidents. This modification installed a safety-related pressure relief valve and a pressure control valve in each train of the HVK system. The pressure relief valve is sized to maintain the HVK system design pressure below 100 psig in case of the pressure control valve failure or seat leakage. The pressure relief valve relief capacity meets the requirements of ASME Code Subsections ND-3612.4 and ND-7412. The changes will be installed in the nonsafety-related portion of the HVK system. The HVK system nonsafety-related makeup water supply is not relied upon to be available during an event. During accident conditions, the HVK system makeup water can be manually supplied from the SWP system. The changes will not result in operating the HVK system outside of its design basis. The modification will not impact any system interface in a way that would increase the likelihood of an accident. The pressure relief valve is sized to protect the system from overpressurization should the safety-related pressure control valve fail. In case the pressure relief valve should not re-seat, the HVK system ASME pressure boundary check valve (2HVK*V95 or 2HVK*V327) will maintain the system pressure integrity. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.:	98-077
Implementation Document No.:	DDC 2M11298
USAR Affected Pages:	Figure 9.2-8a
System:	Domestic Water (DWS)
Title of Change:	Removal of 2DWS-V381 from Documentation

Description of Change:

This change removed valve 2DWS-V381 from documentation. The valve was never physically installed.

Safety Evaluation Summary:

No physical plant work shall be performed as a result of these changes, only documentation corrections. The change involves the removal of valve 2DWS-V381 from USAR Figure 9.2-8a. The valve would have been responsible for the isolation of domestic water from both the supply header and the pressurized backup tank. Valve 2DWS-V381 would have been a locked open valve used to isolate flow to a safety shower during maintenance. This isolation can still be accomplished with existing valves, which removes the need for 2DWS-V381. The portion of the DWS system affected by this change is nonsafety related and is only responsible for the delivery of water to a safety shower. The DWS system does not have a role in any aspect of safe plant shutdown. This change does not increase the likelihood for failure of the DWS system as the current configuration meets specification requirements. If, however, failure of DWS in this area did occur, it would not impact the operability of equipment important to safety because none exists in this area.

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

Safety Evaluation No.: 98-079

Implementation Document No.: Temporary Mod. 98-023

USAR Affected Pages: N/A

System: Main Steam (MSS)

Title of Change: Disable Main Steam Line Low Point Level Alarms from 2MSS-LS1A, B, C, D

Description of Change:

This change disabled the MSS level alarms originating from 2MSS-LS1A,B,C,D. The alarms from these switches have been activated during much of the time since startup after RF06 for no apparent reason, and are considered to be nuisance alarms. These level switches monitor for the buildup of condensation in the low point drain trap in each of the four large main steam lines as those lines pass through the main steam tunnel at El. 250'. The subject level switches recently were installed to replace similar Magnetrol float switches that had a lower temperature and pressure rating than the new switches. This change temporarily disabled alarm window 602217 and process computer points MSSLC01 through MSSLC04.

Safety Evaluation Summary:

This change only affects the drain trap level switches and alarms as described above. The only potential damage-causing scenario is possible long-term erosion of the turbine rotors. This is bounded by analysis of larger-scale MSS piping and system failures. Therefore, there is no effect on nuclear safety in a way not previously evaluated. The constructability aspects of this change have been reviewed, and appropriate work sequencing instructions will be included with the applicable work and design documents. Based on the foregoing analysis, the change does not increase the consequences of accidents previously evaluated in the USAR and does not adversely affect 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.:	98-083
Implementation Document No.:	Procedure S-MMP-GEN-014
USAR Affected Pages:	N/A
System:	Low-Pressure Feedwater Heater Drains (HDL)
Title of Change:	Freeze Seal for Isolation of Continuous Vent Line From Fourth Point Feedwater Heater, 2CNM-E4C

Description of Change:

During Operating Cycle 7, heater drain pump 2HDL-P1C tripped on motor overcurrent. To remove the pump for inspection and maintenance, it was necessary to remove a piping interference in a portion of the continuous vent line piping which is routed from the heater drain pump casing to the shell side of fourth point feedwater heater 2CNM-E4C.

To remove the piping interference, a freeze seal was required in the piping to provide a closure boundary between the feedwater heater and the location where the piping was to be cut. Once the freeze seal was added to the continuous vent line, the piping was drained downstream of the freeze seal, the piping was cut, and an isolation valve with break flanges and blind flange were added to the piping system. The new isolation and blind flange valve provided a pressure-retaining closure and maintained system structural integrity once the freeze seal was thawed.

Safety Evaluation Summary:

A Contingency Plan and Recovery Plan Criteria are addressed on Freeze Seal Data Sheet FSN 98-010, and describe actions which are to be taken to minimize or preclude those conditions which may result in the loss of the freeze seal, protection of personnel and plant equipment, or damage to the piping system after the freeze seal is defrosted. Dimensional measurements of the piping and liquid penetrant examinations before and after freeze seal activities ensure that piping pressure-retaining and structural integrity are maintained. In addition, because of the initial fluid conditions in the piping, an ultrasonic test of the piping will be conducted to confirm that the pipe is full of water prior to applying the freeze seal. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.: 98-084

Implementation Document No.: DDC 2M11441

USAR Affected Pages: Figure 10.1-7n

System: Feedwater Heater Drains (HDL)

Title of Change: Addition of a Maintenance Valve in Line 2-HDL-002-446-4

Description of Change:

Heater drain pump 2HDL-P1C recently tripped on motor overcurrent. In order to provide ease of removal of the pump for inspection and maintenance, a portion of the continuous vent line piping which is routed from the heater drain pump casing to the shell side of the fourth point feedwater heater (2CNM-E4C) needed to be removed. This line serves to vent gases from the HDL pump back to the fourth point feedwater heater.

In order to provide efficient removal of this line for future maintenance activities, permanent addition of an isolation valve, a vent valve, and a pair of break-out flanges were added to this line (2-HDL-002-446-4). The valves and flanges were added to allow removal of a portion of the line that interferes with pump removal.

Safety Evaluation Summary:

The change will install a maintenance valve, a vent valve and a set of break-out flanges to aid in the maintenance of a heater drain pump located in a heater bay off the Turbine Building. The maintenance valve will normally be in the open position providing a vent for the HDL pump. The addition of these piping components does not affect the HDL system function. The new installation is consistent with the existing design requirements. The operation of these new components will be controlled by approved site procedures.

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

Safety Evaluation No.:	98-085
Implementation Document No.:	Mod. N2-97-067
USAR Affected Pages:	5.4-30, 6.1-4, 6.1-5, 6.2-49, 6.2-50, 6.2-53, 6.3-7, 6.3-8, 6.3-9, 6.3-10, 6.3-11a, 6.3-16; Table 1.8-1 Sh 29; Figure 5.4-14 Sh 2
System:	Residual Heat Removal (RHS), Low-Pressure Core Spray (CSL), High-Pressure Core Spray (CSH)
Title of Change:	ECCS Suction Strainers-Change from 50% Clogged to Debris-Based Loading

Description of Change:

This modification changed the 50% clogged emergency core cooling system (ECCS) suction strainers to debris-based loaded strainers in accordance with Regulatory Guide (RG) 1.82, Revision 2, and NRC Bulletin 96-03.

Safety Evaluation Summary:

The probability of an accident previously evaluated in the USAR will remain unchanged with the proposed change from 50% clogged to a debris-based strainer loading. The new larger strainers were installed to ensure operability of the ECCS pumps and maintain long-term recirculation cooling capability during post-LOCA conditions. The strainers are passive and are not accident initiators. Strainers in the ECCS prevent debris from fouling the ECCS pumps and plugging the spray nozzles/orifices. The new strainers were designed to provide adequate net positive suction head to all ECCS pumps, even when clogged with a plant-specific debris loading meeting the requirements of RG 1.82, Revision 2.

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

Safety Evaluation No.:	98-086
Implementation Document No.:	DDC 2E11545
USAR Affected Pages:	Figure 9.2-2 Sh 14
System:	Service Water (SWP) Control Building Chilled Water (HVK)
Title of Change:	Clarification of the Fail Position of the HVK Chiller Service Water Temperature Control Valves

Description of Change:

This change revised USAR Figure 9.2-2 to indicate that valves 2SWP*TV35A and 2SWP*TV35B fail open upon loss of electrical power. The actuator is equipped with an accumulator which holds the hydraulic fluid at approximately 2000 psi to drive the valve to its fail-safe position should the electrical power to the actuator be lost.

Safety Evaluation Summary:

This change does not require any physical work in the plant and is intended only for document reconciliation maintaining full compliance to all codes and standards. Since the system is designed with 100% redundancy, the valves' fail position is not critical. The HVK chillers and the associated service water are a support system used to ensure Control Room habitability is maintained and do not act as precursors to an accident. The proposed change will not affect the normal function or single failure capability of the HVK system.

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

Safety Evaluation No.: 98-087

Implementation Document No.: Calculation EHV-7

USAR Affected Pages: 9.4-1, 9.4-2, 9.4-18, 9.4-31, 9.4-38, 9.4-45, 9.4-48, 9.4-51, 9.4-57, 9.4-60, 9.4-63, 9.4-66, 9.4-68; Table 6.2-54, 9.4-1 Sh 1 thru 3, 9.4-2 Sh 1 thru 14, 9.4-3 Sh 1 thru 10, 9.4-4 Sh 1 thru 8, 9.4-5 Sh 1 thru 20, 9.4-6 Sh 1 thru 3, 9.4-7 Sh 1 & 2, 9.4-8 Sh 1 thru 6, 9.4-9, 9.4-10 Sh 1 & 2, 9.4-11 Sh 1 & 2, 9.4-12 Sh 1 thru 5

System: Control Building Ventilation (HVC), Drywell Cooling (DRS), Reactor Building Ventilation (HVR), Turbine Building Ventilation (HVT), Radwaste Building Ventilation (HVW), Service Building Access Passageway Ventilation (HVE), Auxiliary Service Building HVAC (HVL), Diesel Generator Building Ventilation (HVP), Yard Structures Ventilation (HVY), Miscellaneous Ventilation (HVI), Control Building Chilled Water Ventilation (HVK), Normal Switchgear Building Ventilation (HVN), Hot Water & Glycol Heating (HVG), Hot Water Heating (HVH)

Title of Change: Clarification of USAR Table 9.4-1 and Labeling Tables 9.4-2 through 9.4-12 as Historical

Description of Change:

USAR Table 9.4-1 has been revised to provide a range of temperatures and relative humidities that can be expected for various areas of the plant during the normal plant operation. Tables 9.4-2 through 9.4-12 have been revised to be labeled "Historical."

Safety Evaluation Summary:

This change provides temperature and relative humidity ranges within the Environmental Design Criteria for various areas in the plant. This change to Table 9.4-1 does not violate any operating parameters and does not provide for a different manner by which this equipment operates. This proposal also labels

Safety Evaluation No.: 98-087 (cont'd.)

Safety Evaluation Summary: (cont'd.)

Tables 9.4-2 through 9.4-12 as "Historical." This safety evaluation determines that these tables include redundant data (also shown on USAR figures). The environmental parameters controlled by the described equipment are all listed in Table 9.4-1.

The safety evaluation has concluded that the changes do not increase the probability of occurrence or the consequences of any accidents or malfunctions of equipment important to safety previously evaluated in the USAR, and they 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 USAR. In addition, it was concluded that the proposal does 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.:	98-088
Implementation Document No.:	Procedure N2-CTP-CWS-807
USAR Affected Pages:	10.4-17
System:	Circulating Water (CWS), Water Treatment Hypochlorites (WTH)
Title of Change:	Sodium Hypochlorite Addition to Circulating Water System

Description of Change:

This safety evaluation evaluated the current method of chlorine (sodium hypochlorite) addition as a permanent method. This method was previously considered as a temporary measure under Safety Evaluation 92-077 (reported November 30, 1998).

The addition of sodium hypochlorite to the CWS system, by direct pumping from barrels into the Cooling Tower flumes, allows continued control of the biological growth within the CWS system. The reason for this change is that the WTH system is inoperative and portions of the system have been abandoned in place. The method of adding sodium hypochlorite to the CWS system does not impact the continued operation of the system, nor does it impact the New York State SPDES Permit requirements, or adversely impact Control Room habitability requirements.

Safety Evaluation Summary:

The change is an alternative method for chlorine addition to control the biological growth in the condenser. Since the original WTH system is inoperative, there is no system impact. The change will not result in any physical modification to the plant. Furthermore, based on engineering judgment, the probability of sodium hypochlorite and sulfuric acid coming into direct contact (in large quantities) is remote. This is based on the administrative controls in procedure N2-CTP-CWS-807 and equipment design features discussed in the evaluation.

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

Safety Evaluation No.:	98-090
Implementation Document No.:	NMP2 Emergency Operating Procedures, Severe Accident Procedures, Plant-Specific Technical Guidelines and Plant-Specific Severe Accident Guidelines
USAR Affected Pages:	4.6-19, 13.5-5; Tables 7.5-2 Sh 6, 18.2-1, 18.2-2 Sh 6,8; Figure 18.2-6 Sh 2
System:	Various
Title of Change:	Incorporation of the BWROG EPG/SAG into a New Issue of PSTG/PSSAG/EOPs/SAPs

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 Boiling Water Reactor 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 its Appendices as the basis documents for general revision of the NMP2 Plant-Specific Technical Guidelines and Severe Accident Guidelines (PSTG/PSSAG), 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 is applied to the NMP2 EOPs/SAPs.

Safety Evaluation No.: 98-090 (cont'd.)

Safety Evaluation Summary: (cont'd.)

It is concluded that the exceptions taken from Revision 1 of the EPGs/SAGs, as depicted in the NMP2 PSTGs/PSSAGs and EOPs/SAPs, are appropriate and warranted.

The proposed changes do not increase the probability of occurrence or consequences of any accident previously evaluated in the USAR. It was 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 USAR. 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 USAR, 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 these changes do not involve an unreviewed safety question.

Safety Evaluation No.:	98-091
Implementation Document No.:	LDCR 2-98-UFS-108
USAR Affected Pages:	Table 9A.3-18 Sh 1 thru 5
System:	Fire Protection Halon Suppression (FPG), Fire Protection Fire Detection (FPM)
Title of Change:	Update and Revision of USAR Table 9A.3-18

Description of Change:

USAR Table 9A.3-18 is a tabulation of the fire detectors by zone and type protecting safety-related equipment. The table has been revised to incorporate previously approved design changes made to the system. This change reconciles USAR Table 9A.3-18 with the current design configuration of the plant.

Safety Evaluation Summary:

This safety evaluation addresses a change to USAR Table 9A.3-18. This USAR change reconciles Table 9A.3-18 to reflect the current design, types and numbers of fire detection systems designated to protect safety-related systems, structures and components. This change does not affect any analytical assumptions made in the FPG and/or FPM system design bases or USAR accidents or transients. Additionally, there is no impact to the basis 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.: 98-092

Implementation Document No.: Calculations A10.1-N-91, A10.1-N-96, A10.1-N-97, A10.1-N-98, A10.1-N-99, A10.1-N-112, A10.1-N-340, A10.1-N-341

USAR Affected Pages: Table 3.9A-12 Sh 11

System: Service Water (SWP), Control Building Chilled Water (HVK)

Title of Change: Active Function of Check Valves 2SWP*V219A and 2SWP*V219B

Description of Change:

Service water check valves 2SWP*V219A and 2SWP*V219B are installed in the inlet line to condensers for Control Building chillers 2HVK*CHL1A and 2HVK*CHL1B, respectively. USAR Table 3.9A-12 identified the function of these valves as preventing reverse flow, while the NMP2 Inservice Testing Program only identifies the forward direction as requiring testing. This safety evaluation documents the lack of need for check valves in this application and, therefore, the acceptability of not testing the subject valves in the reverse direction. Concurrently, this safety evaluation proposes an update to USAR Table 3.9A-12 to reflect its active function as flow in the forward direction.

Safety Evaluation Summary:

Based on a review of the design basis hydraulic calculations, and the maximum head of the HVK chiller recirculating pumps, the pumps cannot produce enough head to cause reverse flow in the SWP system. This, combined with continued forward flow testing of these valves, ensures that the function of the SWP system and the associated supported components (HVK chillers, emergency diesel generators, HVAC unit coolers) will not be impacted by the change in active safety function classification. As such, this change will not 1) cause an increase in the probability of occurrence or consequences of any accidents or malfunctions of equipment important to safety previously evaluated; 2) create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated; or 3) 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.: 98-094

Implementation Document No.: LDCR 2-98-UFS-113

USAR Affected Pages: Table 9A.3-3 Sh 1

System: Standby Gas Treatment (GTS)

Title of Change: Update and Revision of USAR Table 9A.3-3

Description of Change:

Table 9A.3-3 is a tabular listing of Fire Hazards Analysis for the Standby Gas Treatment Building. The table lists the fire zones, combustible materials present, the total energy content in those combustible materials, and the total energy content in each fire zone within the building. One of the combustible materials present in the Standby Gas Treatment Building is charcoal, used as a filter media in the GTS filters. The quantity of charcoal assumed in the Fire Hazards Analysis was based on the design minimum number of pounds mass required to meet radiological performance objectives. The system design requires a minimum of 1360 lbm of charcoal. The quantity of charcoal actually present in the filters is 1400 lbm. This change reconciled the Fire Hazards Analysis with the actual installed mass of charcoal.

This change revised USAR Section 9A.3.1.2.5.1 to remove the general description of the combustible loading in the Standby Gas Treatment Building and insert in its place a reference to the combustible loading information in Table 9A.3-3.

Safety Evaluation Summary:

A review of USAR Table 9A.3-3 revealed that the table was not current with as-built plant conditions. The changes to USAR Table 9A.3-3 only update the USAR to the actual design configuration of the GTS system. No physical design or configuration changes are proposed. The increased fire loading, calculated as a result of the higher actual charcoal mass used (approximately one minute), is insignificant when compared to the overall fire loading in the area (30 minutes), and the installed fire protection features (the GTS filters have a manual water deluge suppression system and the redundant trains of GTS filters are separated by a 3-hour rated fire barrier). This change will have no adverse impact on the conclusions of the Fire Hazards Analysis, nor does it affect the Appendix R Safe Shutdown Analysis. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.: 98-095

Implementation Document No.: GE Stress Report No. G 471-6 125.04-07 Rev. 08, GE Stress Report Supplement No. 20/23182 Rev. 00

USAR Affected Pages: Table 3.9B-2j Sh 1 thru 5

System: Main Steam (MSS)

Title of Change: USAR Table 3.9B-2j Corrections

Description of Change:

The nuclear pressure relief system consists of 18 main steam safety relief valves (SRV). These valves are part of the reactor coolant system and provide three main protection functions: 1) Overpressure Relief Operation, 2) Overpressure Safety Operation, and 3) Depressurization Operation.

To fulfill the safety functions, the SRVs underwent a series of testing and analysis. The analysis was performed to the requirements of ASME Section III, Subsection NB. USAR Table 3.9B-2j was revised to provide the summary of this analysis.

Safety Evaluation Summary:

This safety evaluation demonstrates that correcting Table 3.9B-2j to agree with the latest ASME analysis will not impact the SRVs' structural and pressure boundary integrities. All revised stresses and fatigue factors are still within their corresponding ASME limits.

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

Safety Evaluation No.:	98-096
Implementation Document No.:	Calculation Disposition PR-C-13-G-04C
USAR Affected Pages:	Table 12.2-6 Sh 1 thru 3
System:	Reactor Core Isolation Cooling (ICS) (RCIC)
Title of Change:	Reactor Core Isolation Cooling Design Activities

Description of Change:

As a result of correcting errors which supported USAR Table 12.2-6, and incorporating the most radiologically conservative RCIC test condition into the system activity analysis, changes were made to the isotopic activities, in units of μCi/cc, shown in USAR Table 12.2-6.

Safety Evaluation Summary:

There is sufficient margin in existing RCIC turbine and piping shielding such that, although design basis RCIC system activity increased, there are no changes to the radiation zones in the plant. Equipment qualification doses remain within the qualified doses for all impacted equipment. The change to the USAR did not result in changes to existing 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.: 98-097

Implementation Document No.: DDC 2M11611, Calculations A10.1-E-142, A10.1-F-034, A10.1-G-050, 085-3131-01, Appendix 1 to NEDC-31830P (and other supporting dispositions), Procedures N2-OSP-CSH-Q@002, N2-OSP-CSL-Q@002, N2-OSP-RHS-Q@004, N2-OSP-RHS-Q@005,N2-OSP-RHS-Q@006, N2-OP-31, N2-OP-32, N2-OP-33

USAR Affected Pages: 5.4-29 thru 5.4-32, 5.4-36, 6.2-18 thru 6.2-20, 6.2-34, 6.2-35, 6.3-10, 6.3-14, 6.3-17, 6.3-37, 15.8-1, 15.8-5; Tables 6.2-4, 6.2-6 Sh 1,2, 6.3-1 Sh 1 thru 3; Figures 5.4-14 Sh 2, 5.4-15, 6.2-24 thru 6.2-27, 6.2-47 thru 6.2-50, 6.3-1 Sh 1, 6.3-2, 6.3-3a, 6.3-3b, 6.3-4a, 6.3-4b, 6.3-5a, 6.3-5b

System: High-Pressure Core Spray (CSH), Low-Pressure Core Spray (CSL), Residual Heat Removal (RHS)

Title of Change: ECCS Pump Performance Reconciliation

Description of Change:

This safety evaluation did not address any physical changes to the plant. Rather, it addressed the impact of differences in Technical Specification worst-case delivered flows from the CSH, CSL and RHS/low-pressure coolant injection (LPCI) systems relative to the performance or functional requirements, as extracted from the supporting plant analyses and the associated Technical Specifications and USAR sections. Through discussions and reference to various calculations and analyses and other supporting dispositions, this safety evaluation demonstrated that surveillance testing, with the acceptance criteria of Technical Specification 4.5.1, will assure acceptable performance for all functions of the associated systems. As an enhancement to the Surveillance Test Program, instrument uncertainties have also been quantified which can be added to the surveillance test procedures. As a result of this reconciliation effort and the various analyses performed, pertinent sections of the USAR were clarified to identify that the system performance requirements are determined by the applicable plant analyses, rather than the original sizing data presented in the process diagrams. In addition, various discussions, figures and tables that detail inputs or results of various analyses are annotated to identify that, while the specific shape of a curve or

Safety Evaluation No.: 98-097 (cont'd.)

Description of Change: (cont'd.)

exact numerical value may be impacted by assuming the fully degraded pump, the bounding critical parameter is not impacted.

Safety Evaluation Summary:

Hydraulic calculations were developed to establish bounding pump curve for the RHS, CSL, and CSH systems by translating the manufacturer's characteristic curve through the surveillance test point of Technical Specification 4.5.1. In addition, the total developed head, required to meet the flow rate requirements imposed on these systems in the applicable lineup by various analyses (adjusted accordingly for friction losses, elevation head, emergency diesel generator frequency variation, and minimum flow valve position), was determined. In comparing the pump curve versus performance requirements, it has been shown that the required flow will be delivered by these worst-case available curves, or an acceptable reconciliation was performed. Of primary importance is that a reconciliation of the SAFER/GESTR LOCA Analysis demonstrated that the peak cladding temperature of the fuel will not increase, even with fully degraded pumps. Similarly, it has been demonstrated that the peak pressure for the large break accident analysis will not increase above the 39.75 specified in several Technical Specifications under 3/4.6. For transient conditions, the performance of the CSH system as a backup to the reactor core isolation cooling (RCIC) system, and as a reactor coolant makeup source for anticipated transient without scram, has been shown to be acceptable. Acceptable performance has also been demonstrated for the suppression pool cooling, shutdown cooling, containment spray, and the off-normal modes of the RHS system (alternate shutdown cooling in the event of a loss of normal shutdown cooling, and pseudo-LPCI in the event of a failure of RCIC for Appendix R fire scenarios). Through this effort, it has been demonstrated that surveillance testing with the acceptance criteria of Technical Specification 4.5.1 will assure acceptable performance for all functions of the associated systems. As such, the operability of all equipment associated with Technical Specifications 3/4.5.1 (ECCS), 3/4.6.2.2 (Containment Spray), 3/4.6.2.3 (Suppression Pool Cooling), 3/4.9.11 (Residual Heat Removal), as well as various subsections of 3/4.6 which address containment leak testing, is assured. As an enhancement to the Surveillance Test Program, this effort also quantified instrument uncertainties which can be added to the surveillance test procedures. Based on these considerations, it is concluded that the proposed changes to the USAR will not a) increase the probability of occurrence or consequences of accidents previously evaluated in the USAR, b) increase the probability of occurrence or consequences of a malfunction of equipment important to safety previously evaluated in the USAR, c) create the possibility of an accident or malfunction of equipment important to safety of a

Safety Evaluation No.: 98-097 (cont'd.)

Safety Evaluation Summary: (cont'd.)

different type than previously evaluated in the USAR, or d) 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.:	98-098
Implementation Document No.:	DDC 2S11142
USAR Affected Pages:	4.6-24, 4.6-25, 4.6-35, 4.6-36
System:	Main Steam (MSS)
Title of Change:	Control Rod Drive Housing Support Gap Measurement

Description of Change:

The control rod drive (CRD) housing support is designed to prevent any significant nuclear transients in the event a CRD housing breaks or separates from the bottom of the reactor vessel. This change revised the acceptable temperature range for the CRD housing support (shoot-out steel) gap measurement, and the criteria/tolerance for the clearance specified between the CRD ring flange cap screw and the grid clamp on the CRD housing support.

Safety Evaluation Summary:

Based on analysis, the revision of the installation gap specified for the CRD housing support from 1.00 ± 0.12 inches to 1.00 +0.50/-0.25 inches is acceptable. This gap relaxation will not result in any unacceptable reactivity discharge.

The original CRD system design basis, which indicated that there is no stress due to thermal expansion and no contact between the CRD housings and the supports during normal operation, remains valid for the minimum gap tolerance of -0.25 inches. It is also expected that there will not be any contact even for the worst-case expected conditions. In addition, since the AISC (Steel Construction Manual, 7th Edition) requirements for stress allowables are met for all components, the maximum tolerance of +0.50 inches is also acceptable for safe reactor operation. The new recommended gap tolerance will also allow the CRD housing temperature to be as high as 110°F during reassembly, due to the insignificant amount of thermal expansion within the steel members.

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

Safety Evaluation No.: 98-099

Implementation Document No.: Procedures N2-OP-61A, N2-OP-61B, N2-OP-52

USAR Affected Pages: 6.5-6, 6.5-7

System: Standby Gas Treatment (SGTS)

Title of Change: Allow Closure of GTS Filter Recirculation Line

Description of Change:

This safety evaluation demonstrates the acceptability of operating the SGTS system with either valve 2GTS*V52 or valve 2GTS*V51 in the closed position. It also allows the closure of valve 2GTS*PV5A in lieu of or in addition to *V52, and it allows the closure of *PV5B in lieu of or in addition to *V51. This system alignment (pressure modulation defeated) represents an off-normal system operation. Applicable operating procedures and the Equipment Status Log reflect the revised system alignment.

Safety Evaluation Summary:

This change allows operation of the SGTS filter trains with the modulation feature defeated. This change will not prevent the safety function of GTS to properly drawdown the Reactor Building after an accident and release the drawn air through the plant's main stack. The impact of this change is that if an accident occurs while in this off-normal system alignment, the Reactor Building would be maintained at a greater vacuum than it would have been with the modulation feature functioning. This assures that the proper control of radioactive releases will be maintained within regulatory limits.

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.: Procedures GAP-POL-01, NIP-ECA-01

USAR Affected Pages: 13.1-3, 13.1-4, 13.1-6, 13.1-12, 13.1-13, 13.1-14, 13.4-2; Figure 13.1-2

System: N/A

Title of Change: Reorganization

Description of Change:

This safety evaluation evaluated: 1) 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; 2) transfer of responsibility for implementation of the Unit 1 inservice testing (IST) program from the Manager Unit 1 Maintenance to the Manager Unit 1 Technical Support; and 3) alignment of the Unit 1 Work Control/Outage Management branch organizational structure with the Unit 2 Work Control/Outage Management branch organizational structure.

Safety Evaluation Summary:

The 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.: 99-011

Implementation Document No.: NEP-POL-01

USAR Affected Pages: Figure 13.1-3

System: N/A

Title of Change: Nuclear Engineering Organization Changes

Description of Change:

A new group, "ASME Section XI Programs," has been created within the Engineering Services Branch of the Engineering Department.

Safety Evaluation Summary:

The new organization changes do not affect plant structures, systems and components, do not affect accidents or malfunctions previously evaluated in the USAR, do not increase any consequences of previously evaluated accidents or malfunctions, do not create new accidents or malfunctions, and do not reduce the margin of safety in any Technical Specification basis. The organization will continue to perform the functions described in the USAR. The intent is to provide greater focus and control of ASME Section XI programs while creating greater programmatic consistency and administrative oversight.

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

Safety Evaluation No.: 99-017

Implementation Document No.: DDC 1M00729

USAR Affected Pages: 12.5-2

System: Radwaste

Title of Change: Shower Facility

Description of Change:

This change retired the NMP1 laundry and converted the area into a shower facility that can be utilized by workers during outages. All of the old laundry equipment was retired, with the exception of the hot water heater. The laundry room utilities (water supply, drains, and HVAC) will be reused with some slight modifications. The previous shower facility did not comply with OSHA 29CFR1926.1101, which states that "one shower shall be provided for each ten employees of each sex or numerical fraction thereof." That shower had insufficient water pressure to supply more than one shower at a time and the cold water was rusty. Upon completion of the new shower, the old shower trailer was removed from site.

Safety Evaluation Summary:

The abandoned NMP1 laundry possesses all of the necessary utilities to support the installation of a shower facility. A hot water supply is available, along with the drain system that feeds through a filter to the radwaste system. The exhaust ventilation system has a high-efficiency particulate air (HEPA) filtration system. This will assure that any stray asbestos fibers will be caught in the filter before they could exit the room. Furthermore, the room has not been operated as a laundry facility for over a decade, and this will allow the equipment to be retired properly while making good use of the space and the existing utilities.

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

Safety Evaluation No.: 99-030

Implementation Document No.: DDC 1S00424

USAR Affected Pages: Figure 9.2-21

System: Sewage Treatment

Title of Change: Sewage Treatment Plant Sludge Drying Beds

Description of Change:

Two 10 ft. by 64 ft. sludge drying beds were constructed under a nonpressure-treated truss roof that covers both adjacent drying beds. The beds were located north of the existing Wastewater Treatment Plant. The sludge drying beds consist of the following design features:

- asphalt paved beds with one 3-foot wide sand drain area;
- covered beds utilizing a nonpressure-treated wood truss system to support translucent fiberglass panels;
- pressure-treated columns for the trusses, built-up by 8 inch by 8 inch posts;
- 18-inch high concrete perimeter walls to allow liquid sludge containment and to act as push walls for the skid-steer loader for sludge removal; and
- a polyethylene liner under the beds with perforated piping pitched to a drainage sump for collecting drainage water.

The design is such that water is drained from the sludge via the sand drain and the sun evaporates water from the sludge. The remaining product is a 70% to 80% dried sludge by weight. Its consistency is similar to sand and is suitable for shipping. The water will be pumped from the sump back to the Wastewater Treatment Plant. The installation of the drying bed facility has no effect or impact on the probable maximum precipitation flooding analysis.

Safety Evaluation Summary:

The addition of sludge drying beds at the Wastewater Treatment Plant does not require a Technical Specification change for either Unit 1 or Unit 2. The possible effects of the sludge drying beds on equipment important to safety are no different than those of the Wastewater Treatment Plant. The sludge drying beds have no connections with any plant system or component.

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

Safety Evaluation No.: 99-039

Implementation Document No.: Procedures NSAS-POL-01, GAP-POL-01, NEP-POL-01

USAR Affected Pages: 13.1-1, 13.1-3, 13.1-4, 13.1-5, 13.1-6, 13.1-13, 13.1-14, 13.2-1, 13.2-20; Figures 13.1-1, 13.1-2, 13.1-3, 13.1-5

System: N/A

Title of Change: Reorganization of Nuclear Safety Assessment and Support (NSAS) Functions

Description of Change:

Procedure revisions were made to assign Nuclear Safety Assessment and Support functions, except those of the Unit 2 Independent Safety Engineering Group, under the Vice President Nuclear Generation, Vice President Nuclear Engineering, or under the new corporate officer position of Vice President Quality Assurance - Nuclear. The changes now have Licensing and Security Branches reporting to the Vice President Nuclear Engineering, the Training Branch reporting to the Vice President Nuclear Generation, and Quality Assurance reporting to the newly-appointed Vice President Quality Assurance - Nuclear position.

Safety Evaluation Summary:

The procedure changes establish and define the lines of authority, responsibility, and communication in conformance with plant Technical Specifications and the applicable acceptance criteria of SRP 13.1.1 and SRP 13.1.2-13.1.3.

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

Safety Evaluation No.: 99-043

Implementation Document No.: NEP-POL-01

USAR Affected Pages: 13.1-4, 13.1-5; Figure 13.1-3

System: N/A

Title of Change: Nuclear Engineering Organization Change

Description of Change:

Procedure revisions have been made to integrate the Unit 1 and Unit 2 Project Management sections into a new Engineering Branch called Project Management. The procedure also integrates the five Procurement sections into three sections, and eliminates four supervisor positions in the Engineering Department.

Safety Evaluation Summary:

The organization changes remain within the acceptance criteria utilized for the basis of the current Engineering organization, including NUREG-0800 (SRP). Implementation of the proposed organization change requires revision to existing policies and procedures that govern the applicable activities. Primary areas affected by this change are the administrative procedures that describe functional responsibility and activities. These changes do not alter the requirements that govern the procedures or the activities described in the USAR. 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.: 99-051

Implementation Document No.: DDC 2M11468

USAR Affected Pages: Figures 5.4-13d, 5.4-13e

System: Residual Heat Removal (RHS)

Title of Change: Incorrect Signal Identified for 2RHS*SOV35B & 2RHS*SOV36A

Description of Change:

The origin of the actuation signal for 2RHS*SOV35B was incorrectly shown on USAR Figure 5.4-13e as coming from Division 2 NSSS. The correct origin of the actuation signal is from Division 1 NSSS. Likewise, the origin of the actuation signal for 2RHS*SOV36A is shown coming from Division 1 NSSS when the correct origin of the actuation signal is Division 2 NSSS. The USAR has been revised to reflect these changes.

Safety Evaluation Summary:

This change will correctly list the origin of the actuation signal for two sampling valves in the RHS system. The valves are installed and connected per design requirements and this change only corrects a drawing error. There is no effect on any previously analyzed accidents or transients, as this change does not make a physical change to the plant, does not change the method of operation of any system, and does not result in operating the RHS system outside its design basis.

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

Safety Evaluation No.: 99-054

Implementation Document No.: DDC 2M11436

USAR Affected Pages: Figures 9.4-10d, 9.4-10e

System: Radwaste Building Ventilation (HVW)

Title of Change: Correction of USAR Figure 9.4-10e

Description of Change:

Three discrepancies between the design drawings, as-built conditions, and USAR Figures 9.4-10d and 9.4-10e were corrected as follows:

1. USAR Figure 9.4-10d was updated to show volume damper 2HVW-DMPV63.

2. USAR Figure 9.4-10e was updated to show volume damper 2HVW-DMPV83 upstream of damper 2HVW-DMP109.

3. USAR Figure 9.4-10e was updated to show duct branch and volume damper 2HVW-DMPV151 inside the Electrical Equipment Room.

Safety Evaluation Summary:

These corrections will not result in any change in physical configuration, function, or operation of the plant.

The changes do not increase the probability of occurrence or the consequences of any accidents or malfunctions of equipment important to safety previously evaluated in the USAR. It also concluded that these 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 USAR. In addition, it was concluded that the proposal does 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.: 99-055

Implementation Document No.: Mod. N2-99-012

USAR Affected Pages: Table 6.2-56 Sh 2

System: Residual Heat Removal (RHS)

Title of Change: Gear Set Changes for 2RHS*MOV15A/B

Description of Change:

This modification changed the overall gear ratio for valves 2RHS*MOV15A/B from 46.66 to 52.57. The calculated valve opening and closing stroke times for each valve increases from 76.58 to 86.29 seconds.

The NMP2 valve sizing calculation for valves 2RHS*MOV15A/B was revised for the required application factor of 0.9. The use of an application factor of 0.9 resulted in the need to change gears to meet reduced voltage requirements and to ensure the valves are capable of performing their safety functions within the current design and licensing bases of the facility.

Safety Evaluation Summary:

Replacement of the gears for valves 2RHS*MOV1A/B will provide an acceptable torque switch setting thrust/torque range to allow the valves to respond to design basis accidents and transients as required. This new range will also accommodate the use of the VOTES diagnostic test equipment and allow for actuator degradation and margin. Qualification for the new Limitorque gears has been performed to ensure continued structural integrity and operability of the modified valve assemblies. The valves are primary containment isolation valves and are leak tested to verify isolation integrity. The leak test methods will not be 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.:	99-056 Rev. 0 & 1
Implementation Document No.:	Mod. N2-97-083
USAR Affected Pages:	9.1-38, 9.1-39; Tables 9.1-3, 9.1-4 Sh 2; Figures 1.2-7 Sh 2, 12.3-7, 12.3-40
System:	Control Rod Drive (RDS) (CRD)
Title of Change:	Deletion of the CRD Handling Equipment "Rocket Launcher"

Description of Change:

This modification removed the CRD handling equipment, better known as the "Rocket Launcher," on a permanent basis from the Drywell and allows for the usage of a CRD handling tool for the removal and installation process of the CRDs. On an "as required basis," the CRD handling tool will be installed during refueling outages and removed prior to startup, upon completion of the CRD scope of work.

Safety Evaluation Summary:

The appropriate site procedures have been changed to state that the CRD handling tool has adequate brakes or gearing to prevent uncontrollable movements upon loss of power or component failure. Also, the appropriate site procedures note that the CRD handling tool cannot exceed 3,000 lbs., unless approved by Engineering, and must be removed from the Drywell prior to startup from a refueling outage.

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

Safety Evaluation No.: 99-058

Implementation Document No.: Mod. N2-99-007

USAR Affected Pages: 7.7-33; Figure 1.2-15 Sh 2

System: Common Electrical Equipment-Control Room Complex (CEC)

Title of Change: Remediate 3-D Monicore Software for Y2K Compliance

Description of Change:

The previous 3D Monicore primary computer was a DEC Micro VAX 3800 and associated peripherals running General Electric Company's (GE) Baseline 94 3D Monicore software. The primary computer has been replaced with a DEC AlphaStation and associated peripherals running GE's Baseline 96 3D Monicore software.

The previous 3D Monicore backup computer was a DEC VAXstation 3100 and associated peripherals running GE's Baseline 94 3D Monicore software. The backup computer has been replaced with a DEC AlphaStation and associated peripherals running GE's Baseline 96 3D Monicore software.

The previous 3D Monicore Control Room workstation is a DEC VAXstation 3100 and associated peripherals running GE's Baseline 94 3D Monicore software. The previous 3DWINR Processor was a commercial grade computer and associated peripherals running GE's 3DWINR and MVD client software. The Control Room workstation and the 3DWINR Processor have been replaced with a single commercial grade computer that will continue to provide required display/interaction functions described in GE specifications 23A5008, 23A5030, and 23A5066.

The previous 3D Monicore reactor engineering workstation was a DEC VAXstation 3100 and associated peripherals running GE's Baseline 94 3D Monicore software. This reactor engineering workstation has been eliminated. Remote access to the new 3D Monicore primary computer from existing commercial grade computers on the site network in the reactor engineering area has been established to provide required display/interaction functions described in GE specification 23A5008.

Safety Evaluation Summary:

This change is a replacement of the existing 3D Monicore system with newer hardware and software that is Y2K ready. The 3D Monicore system function,

Safety Evaluation No.: 99-058 (cont'd.)

Safety Evaluation Summary: (cont'd.)

method of performing the function, and accuracy are not being altered by this change. The 3D Monicore system is a nonsafety-related monitoring system and is not a transient or accident initiator. As such, this change does not increase the probability of occurrence of an accident previously evaluated in the USAR.

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

Safety Evaluation No.: 99-059

Implementation Document No.: DDC 2F01987

USAR Affected Pages: Figures 9.5-5 Sh 1, 9.5-20 Sh 2

System: COJ - Communications (Maintenance)

Title of Change: Add Maintenance/Calibration Jacks

Description of Change:

The maintenance/calibration system provides for voice communication in areas of the plant requiring communication for testing, instrument calibration, and maintenance activities. Maintenance/calibration jacks are conveniently located throughout the plant. This change provided the design required to allow the installation of four additional maintenance/calibration jacks in the North and South Auxiliary Bays, and four additional jacks in the Division I and II Switchgear Rooms. The jacks were added at Maintenance's request to increase the availability of jacks for valve testing during outages. The addition of the subject jacks has no effect on the operation of the system. The new jacks will function in the same manner as existing jacks.

Safety Evaluation Summary:

The installation of additional maintenance/calibration jacks does not increase the probability of occurrence or the consequences of an accident or a malfunction previously evaluated in the USAR. The change does not create the possibility of an accident or malfunction of a different type than already analyzed in the USAR. There is no decrease in the margin of safety and no adverse 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.:	99-060
Implementation Document No.:	Calculation ES-135
USAR Affected Pages:	6.2-93; Figure 6.2-72I
System:	Hydrogen Recombiner (HCS)
Title of Change:	Deletion of the Requirement to Inert the Hydrogen Recombiner

Description of Change:

This change removed a requirement to inert the portions of the HCS system outside of and normally isolated from the primary containment during normal plant operations. There are no procedures in place to verify that an inert atmosphere inside of the recombiners is maintained.

Safety Evaluation Summary:

Removal of the requirement to inert the portions of the HCS system outside of and normally isolated from primary containment during normal operation does not affect the ability of the HCS system to perform its intended design function. There are no vendor or EQ requirements to maintain the HCS system inerted during normal operation.

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

Safety Evaluation No.: 99-061

Implementation Document No.: LDCR 2-98-UFS-111

USAR Affected Pages: 5.4-11

System: Main Steam (MSS)

Title of Change: Delete Augmented Inspections on MSIV 2MSS*AOV7A Valve-to-Pipe Weld

Description of Change:

The USAR commitment to examine weld no. 2MSS-01-13-FW021 "during the next three inspection periods" has been deleted.

The USAR contained a commitment to examine weld no. 2MSS-01-13-FW021 during each of three consecutive inservice inspection (ISI) periods, beginning with the first inspection period of the first 10-year ISI interval. An ISI inspection period is approximately 1/3 of the 10-year inspection interval. The first 10-year interval and this commitment expired on April 4, 1998. The expiration of this commitment means that the inspection frequency for weld no. 2MSS-01-13-FW021 returns to once every 10 years. The technical basis for not renewing the commitment to examine this weld on an increased frequency is that the ultrasonic testing examination in 1998 showed no growth in these indications over the 10 years between examinations. Therefore, an increased examination frequency is not required.

Safety Evaluation Summary:

This change to the USAR deletes the expired requirement to examine weld no. 2MSS-01-13-FW021 each inspection period. This change returns the examination requirements for the subject weld to the ASME Section XI Code requirement and is consistent with Technical Specification 4.0.5, the regulatory requirements for ISI in 10CFR50.55a(g), and ASME Boiler and Pressure Vessel Code, Section XI.

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

Safety Evaluation No.:	99-062 Rev. 0 & 1
Implementation Document No.:	Mod. N2-98-017
USAR Affected Pages:	Tables 9A.3-1 Sh 9, 9B.5-1; Figure 1.2-10 Sh 1
System:	Reactor Water Cleanup (WCS), Instrument Air Service (IAS)
Title of Change:	WCS Appendix R High-/Low-Pressure Interface

Description of Change:

This modification added electrical and mechanical isolation devices to the subject high-/low-pressure interface valves to preclude the spurious actuation of the valves, resulting in a loss-of-coolant accident (LOCA) condition due to the single Appendix R fire in a single fire area.

Safety Evaluation Summary:

The design, material and construction standards for installing the transfer switches and manual isolation valves will be performed to the original criteria, specifications and standards. The IAS portion of the design will not introduce any changes to the system design such that the loss of instrument air transient "remote" frequency is changed. The high-/low-pressure interface valves will continue to operate as before except to eliminate spurious actuation due to a fire. The loss of this WCS coolant pressure boundary remains unchanged and bounded from a LOCA analysis standpoint. The performance of the systems will not be changed and will operate with the same functional capability as before; therefore, the change or activity will not increase the probability of occurrence of any transient or accident previously evaluated in the USAR.

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

Safety Evaluation No.: 99-063

Implementation Document No.: Procedure N2-OP-37

USAR Affected Pages: 5.4-45

System: Reactor Water Cleanup (WCS)

Title of Change: Update USAR to Allow the WCS Filter Demineralizer to Use a Pure Resin Mix

Description of Change:

The plant now uses a resin mix supplied by Epicore (Part No. PD-23H). This mix is a premix precoat filter material comprised of 2-part cation and 3-part anion, which has no filter aid material. The change of resin was part of the WCS system improvements, which resulted in more reliable system operation and improved water chemistry. This change is reflective of industry trends to enhance and improve boiling water reactor (BWR) water chemistry by improving the WCS system performance.

Safety Evaluation Summary:

Based upon this analysis, the pure resin mix aids in maintaining or improving reactor water quality within the limits imposed by the plant's administrative procedure. The plant's administrative requirements impose a more stringent water chemistry parameter than that stated in the plant's license documents. The change in resin mix precoat material will not impact or change any BWR water chemistry operating limit that was previously imposed, and the margin of safety required in the Technical Specifications and Regulatory Guide 1.56 remains unchanged. Further, the proposed change does not alter in any way the design, configuration or operation of the WCS.

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

Safety Evaluation No.:	99-065
Implementation Document No.:	Procedure N2-REI-07
USAR Affected Pages:	3.1-30, 4.6-22, 4.6-34, 4.6-34a
System:	Control Rod Drive (RDS) (CRD)
Title of Change:	USAR Update for Control Rod Withdrawal Speed

Description of Change:

This safety evaluation changed the USAR to provide a cycle generic maximum control rod withdrawal rate. This change allows operation with a control rod withdraw speed up to 6.0 inches per second for Cycle 7, corresponding to a 24-second stroke time for total control rod stroke (144 inches), and 5.0 inches per second for all remaining cycles at NMP2. An analysis provided by Generic Electric concludes that the assumptions used in the rod withdrawal error (RWE) analysis and the Cycle 7 operating limit minimum critical power ratio bound such operation.

Safety Evaluation Summary:

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, representing fast withdrawal speed.

The original design and function of the CRD hydraulic system is unchanged. The ability of the CRD to function as described in the USAR is not affected. 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.:	99-066
Implementation Document No.:	Calculation A10.1-P-022
USAR Affected Pages:	9.3-11
System:	Instrument Air Service (IAS), Automatic Depressurization (ADS)
Title of Change:	Incorrect Analytical Value Reported in the USAR

Description of Change:

The upper analytical value for the pneumatic supply for the ADS system was recalculated in 1985 to reflect instrument error. The USAR has been revised to reflect a value of 186 psig, as supported by system calculations.

Safety Evaluation Summary:

The change will correctly list the upper analytical limit of the pneumatic supply to the ADS in the USAR. This limit is already reflected in the system calculations. There is no effect on any previously analyzed accidents or transients, as this change does not make a physical change to the plant, does not change the method of operation of any system, and does not result in operating the ADS system outside the design basis. Therefore, this change does not increase 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.:	99-067 Rev. 0 & 1
Implementation Document No.:	Mods. N2-99-026, N2-99-027
USAR Affected Pages:	5.4-39; Table 3.9A-12 Sh 14; Figure 5.4-13e
System:	Residual Heat Removal (RHS)
Title of Change:	Provide Bonnet Pressure Relief for 2RHS*MOV115 and 2RHS*MOV116

Description of Change:

Valves 2RHS*MOV115 and 2RHS*MOV116 provide a connection between the service water (SWP) system and Train B of the RHS system. This allows the primary containment flooding capability using service water in post-accident conditions, if required. These valves are normally closed and the piping section between these valves is drained via normally open solenoid valve 2RHS*SOV126.

The modifications provide bonnet pressure relief by drilling a vent hole in the disc for valve 2RHS*MOV115 (on the disc toward the system high pressure), and installing bonnet pressure relief line from the bonnet to the system high pressure side for valve 2RHS*MOV116. This allows venting of the bonnet and eliminating the potential pressure locking conditions.

Safety Evaluation Summary:

Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," was issued to identify the power-operated gate valves pressure locking or thermal binding as a result of operational configurations.

Niagara Mohawk evaluation of the power-operated safety-related gate valves has determined that valves 2RHS*MOV115 and 2RHS*MOV116 are susceptible to pressure locking following an accident, but they are not susceptible to thermal binding.

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

Safety Evaluation No.:	99-068
Implementation Document No.:	Mod. N2-99-022, N2-99-024, N2-99-025
USAR Affected Pages:	5.4-39; Table 3.9A-12 Sh 14
System:	Residual Heat Removal (RHS)
Title of Change:	Provide Bonnet Pressure Relief for 2RHS*MOV4A/B/C

Description of Change:

Valves 2RHS*MOV4A/B/C are located in the RHS pump minimum flow lines. They are normally open valves (with RHS pumps not running) and they will close once the pump has reached a set flow rate. These valves are required to open at the RHS pumps low flow rate to protect the pumps from overheating.

The modifications provide bonnet pressure relief by drilling a vent hole in the disc on the upstream side of the valves (inlet). This will allow venting of the bonnet should a pressure locking environment exist.

Safety Evaluation Summary:

Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," was issued to identify the power-operated gate valves pressure locking or thermal binding as a result of operational configurations.

Niagara Mohawk evaluation of the power-operated safety-related gate valves has determined that valves 2RHS*MOV4A/B/C are susceptible to pressure locking as a result of convective heat transfer from the RHS pump discharge as the limiting case involving post-accident recovery, but they are not susceptible to thermal binding.

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

Safety Evaluation No.:	99-069
Implementation Document No.:	Mod. N2-99-023
USAR Affected Pages:	Table 3.9A-12 Sh 13
System:	Low-Pressure Core Spray (CSL) (LPCS)
Title of Change:	Provide Bonnet Pressure Relief for 2CSL*MOV107

Description of Change:

Valve 2CSL*MOV107 is located in the LPCS pump minimum flow line. It is a normally open valve (with LPCS pump not running) and it will close once the pump has reached a set flow rate. This valve is required to open at the LPCS pump low flow rate to protect the pumps from overheating.

The modification provides bonnet pressure relief by drilling a vent hole in the disc on the upstream side of the valve (inlet). This will allow venting of the bonnet should a pressure locking environment exist.

Safety Evaluation Summary:

Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," was issued to identify the power-operated gate valves pressure locking or thermal binding as a result of operational configurations.

Niagara Mohawk evaluation of the power-operated safety-related gate valves has determined that valve 2CSL*MOV107 is susceptible to pressure locking as a result of convective heat transfer from the LPCS pump discharge following a design basis accident, but is not susceptible to thermal binding.

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

Safety Evaluation No.: 99-071

Implementation Document No.: DDC 2M11173, DDC 2M11174, Procedure N2-OP-19

USAR Affected Pages: Figures 9.2-9a, 9.3-1b, 9.3-1g, 9.3-1m

System: Instrument Air Service (IAS), Sanitary Waste Treatment (PBS)

Title of Change: Update to IAS and PBS USAR Figures

Description of Change:

Pressure indicator 2PBS-PI116 and root valve 2PBS-V15 were removed from USAR Figure 9.2-9a. This equipment was depicted in the drawing, but was not installed in the field.

In addition, discrepancies in the IAS system drawings (USAR Figures 9.3-1b, 1g and 1m), regarding valve positions which were contrary to the plant's operating procedure, were resolved by reflecting the plant's as-built valve position lineup.

Safety Evaluation Summary:

Based upon this analysis, the IAS and PBS system USAR figure changes are within the requirements of the applicable regulations of 10CFR50, Appendix A (GDC 5 and 60). Therefore, this change does not alter in any way the design, configuration, or operation of the IAS or PBS systems as previously evaluated by the NRC.

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

Safety Evaluation No.:	99-072
Implementation Document No.:	Mod. N2-99-016
USAR Affected Pages:	Table 9B.8-3 Sh 12; Figures 9.2-1h, 9.2-2 Sh 18
System:	Service Water (SWP), Motor Control Center Emergency System (EHS), Common Electrical System (CES)
Title of Change:	Isolate Bar Rack Heater Control Circuits from Control/Relay Room Fire Area

Description of Change:

The control circuits of the bar rack heating system from the control/relay room fire area have been isolated by providing disconnect switches outside the main control/relay room fire area to ensure the availability of these control circuits for the remote shutdown operation of the plant.

Safety Evaluation Summary:

The design, material and construction standards for installing devices such as relays and fuses will be performed to the original criteria, specification and standards. The bar rack heating system will continue to perform as before except that the system will automatically be activated by the actuation of disconnect switches during a control/relay room evacuation. The performance of the systems will not be changed and will operate with the same functional capability as before.

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

Safety Evaluation No.: 99-073

Implementation Document No.: Mod. N2-99-017

USAR Affected Pages: Table 3.9A-12 Sh 5

System: Reactor Core Isolation Cooling (ICS)

Title of Change: Convert the Actuator of 2ICS*MOV124 from SMB to SB - Change the Gear Set Ratio

Description of Change:

This modification revised the existing gear set ratio of 2ICS*MOV124 from 67.5 to 77.0. This change increases the output torque and meets the torque design basis requirements for this valve. This torque increase results in a reduced motor speed and a longer time required to open and close the valve. This change also modified the actuator model from SMB-00-10 to SB-00-10 to accommodate higher motor torque output while the valve is not fully closed.

Safety Evaluation Summary:

This safety evaluation indicates that 2ICS*MOV124 is capable of meeting its thrust requirements as revised by the more restrictive valve application factor. The change to the actuator will accommodate the operator's capability of exerting more torque while the valve is not fully closed. Although the amount of water injected into the reactor will be somewhat reduced because of a slower closing time of the test return valve, the total amount of water delivered by the system into the reactor will be greater than assumed in accident analyses.

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

Safety Evaluation No.:	99-075
Implementation Document No.:	Procedure N2-OP-35
USAR Affected Pages:	5.4-19
System:	Reactor Core Isolation Cooling (ICS) (RCIC)
Title of Change:	Manual Initiation and Subsequent Automatic System Operation for the Observation of Proper System Responses

Description of Change:

Operating Procedure N2-OP-35 has been revised to provide for the demonstration of proper system response.

The activity consisted of manual initiation and subsequent automatic system operation for the observation of proper system responses. This activity was performed with the reactor in Mode 1, at approximately 10-15% power, with the main turbine shut down and tripped.

Safety Evaluation Summary:

Operating Procedure N2-OP-35 is issued to provide for manual initiation of the RCIC system and subsequent automatic operation. This will allow demonstration of system response.

The safety evaluation has considered all the possible impacts of this activity of system response on plant operation in general and on other particular systems and components. It was determined that this activity will not cause any system or component to operate outside its safety parameters.

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

Safety Evaluation No.: 99-076

Implementation Document No.: Temporary Mod. 99-011

USAR Affected Pages: N/A

System: Circulating Water (CWS)

Title of Change: Alternate Sulfuric Acid Supply for the Circulating Water System

Description of Change:

This temporary modification provided an alternate method of sulfuric acid dispersant into the CWS system. The typical amount of acid dispersed is between 1/4 and 1 gallon per minute. The temporary equipment provided had the ability to control the amount of sulfuric acid injected into CWS within this amount.

Safety Evaluation Summary:

No safety concerns exist with the temporary modification. All the potential accidents and malfunctions associated with these changes are bounded by those previously reviewed by the NRC, and thus NRC approval is not required.

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

Safety Evaluation No.:	99-078 Rev. 0 & 1
Implementation Document No.:	DDC 2E11917
USAR Affected Pages:	Figures 10.4-7e, 10.4-7h
System:	Service Water Chemical Treatment (SCT)
Title of Change:	Service Water Chemical Treatment Fill Station Tank Level Indication

Description of Change:

This change added a level indicator and level alarm for the sodium bisulfite tank and the hypochlorite tank located in the Screenwell Building. The indicators and alarms were mounted on the outside wall of the building at truck fill stations so that tank levels may be monitored during filling.

The added level indication will be repeated from existing level indication provided at SCT control panel 2SCT-PNL100, located within the Screenwell Building.

USAR Figures 10.4-7e and 10.4-7h have been revised to show the new level indicators and alarms being added to the existing level indication loops for tanks 2SCT-TK1 and 2SCT-TK2. Also, additional "document only" changes were made to the figures to show existing high-/low-level switches which are internal to existing level indicators which are mounted in 2SCT-PNL100.

Safety Evaluation Summary:

Based on the foregoing analysis, changes are required to the USAR figures depicting the SCT system chemical storage tanks and associated instrumentation. Any impact on the licensing basis is limited to these figure changes. Implementation of the change will not increase the consequences of accidents previously evaluated in the USAR and will not affect 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.: 99-081

Implementation Document No.: Mod. N2-99-037

USAR Affected Pages: Figure 10.4-7j

System: Circulating Water (CWS), Chemical Injection Sulfuric Acid (WTA)

Title of Change: Replacement of the Sulfuric Acid Storage Tank

Description of Change:

This modification replaced tank 2WTA-TK3 in the WTA system for the CWS system. The new tank is slightly smaller and constructed of a different material, stainless steel ASTM A240-316L, versus high-density cross-linked polyethylene. USAR Figure 10.4-7j was revised to reflect the new tank size.

Safety Evaluation Summary:

This change will replace acid storage tank 2WTA-TK3 and associated components which are used for chemistry control in the CWS system. There is no effect on any previously analyzed accidents or transients, as this change does not change the method of operation of any system and does not result in operating the WTA of CWS system outside their design basis. Neither CWS nor WTA are accident precursors or are relied upon to prevent any accidents. A line 5 outage is governed by Technical Specification 3/4 3.8.1 and has previously been evaluated in the context of the Technical Specification LCO. A Line 4 outage does not affect NMP2. A line 4 outage has been evaluated in the context of Unit 1 Technical Specification 3.6.3.b and USAR Section 1.2. Therefore, this change does not increase 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.:	99-082
Implementation Document No.:	Mod. N2-99-043
USAR Affected Pages:	Figures 11.3-1b, 11.5-2, 11.5-2a, 11.5-2b
System:	Offgas (OFG), Condensate Air Removal (ARC), Condensate (CNM)
Title of Change:	Offgas Radiation Monitor Sample Line Delay Volume

Description of Change:

The volume and length of a portion of the offgas radiation monitor sampling line was increased to provide a 3-minute delay for decay of short-lived activation products.

The delay pipe allows sufficient time for decay of transient activation products. This change reduces the risk of tripping the plant, due to a spurious transient in either offgas flow or short-lived radioisotopes resulting from activation, and improves the accuracy and reliability of measurement of noble gases in the OFG system. Increased offgas flow reduces the time available for decay of radionuclides. The reduction of O19 and N16 is most sensitive to the normal flow variations in the OFG system. The O19 has a half-life of 27 seconds, and N16 has a half-life of 7 seconds. With the delay pipe, the radiation monitors will be less sensitive to flow variations and more accurately measure the true level of noble gases in the OFG system.

Two 12-inch diameter stainless steel flanged pipes, approximately 11 feet long, were installed, one in each of the two sample lines. This provides the necessary volume to delay the offgas sample to the radiation monitors. The design of the delay pipe is consistent with codes committed to in the USAR. The installation was in a nonseismic area of the plant.

Sample lines were rerouted and lengthened to accommodate the delay pipe. Valves were added to isolate and bypass the delay pipe. Particulate and iodine filters were relocated and installed upstream of the delay pipe, along with their associated valves and differential pressure switches. This will not affect their capability of being removed for laboratory analysis. The additional sample line tubing will not affect sampling system isokinetics.

Safety Evaluation No.: 99-082 (cont'd.)

Safety Evaluation Summary:

Each of the offgas radiation monitor sampling lines will have a delay pipe installed to provide at least a 3-minute delay for decay of short-lived activation products. The delay pipe allows sufficient time for decay of transient activation products. This change increases the accuracy and reliability of the measurement of noble gases in the OFG system without increasing the consequences and probability of design bases accidents, nor affecting the function or failure modes of equipment important to safety. No other functions of the offgas radiation monitors or offgas system are 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.:	99-086
Implementation Document No.:	Calculations A10.1-R-018, A10.1-R-020
USAR Affected Pages:	3.9A-1, 3.9A-3, 3.9A-21; Tables 3.8-6 Sh 1 & 2, 3.9A-2 Sh 3, 4, 7
System:	High-Pressure Core Spray (CSH), Low-Pressure Core Spray (CSL), Reactor Core Isolation Cooling (ICS), Reactor Coolant (RCS), Residual Heat Removal (RHS), Standby Liquid Control (SLS)
Title of Change:	Generic Letter 96-06 Thermal Overpressurization - Appendix F Analysis

Description of Change:

NMPC has completed a thermal overpressurization assessment of potentially vulnerable piping segments at NMP2 and developed an action plan to correct the affected installations. All of the segments that could not be addressed by administrative control, with the exception of the primary containment penetrations for the RCS system flow control valve hydraulic piping and miscellaneous double-valve vent, drain and test connections, have been fitted with thermal relief devices addressed under a separate safety evaluation. Engineering's review of the RCS hydraulic penetrations and miscellaneous vent, drain and test connections determined that neither administrative controls nor hardware modifications were appropriate for these applications.

In lieu of a hardware fix, the RCS hydraulic penetrations and double-valve vent, drain and test connections were analyzed using ASME Section III, Appendix F acceptance criteria. Generic Letter 96-06, Supplement 1, dated November 13, 1997, states that analytical solutions employing the permanent use of the acceptance criteria contained in Appendix F can be used where appropriate, justified and evaluated in accordance with NRC requirements such as 10CFR50.59, as applicable. NMP2 is committed to the 1974 Edition of the ASME Boiler and Pressure Vessel Code, Section III, as discussed in Section 3.9A.1.4.2 of the USAR. Early editions of the ASME Code discussed the use of Appendix F for Class 1 applications only. However, the NMP2 licensing basis endorses the use of Appendix F for Class 1, 2, and 3 applications, as indicated in USAR Section 3.9A.1.4.2 and Table 3.9A-8. The RCS hydraulic piping and miscellaneous vents, drains and test connections are Class 2. This safety evaluation evaluated the acceptability of using Appendix F analysis as a permanent resolution to the RCS hydraulic and miscellaneous vent and drain applications.

Safety Evaluation No.: 99-086 (cont'd.)

Safety Evaluation Summary:

The revised analyses demonstrate that the RCS hydraulic penetrations and associated piping and valves, as well as the affected miscellaneous vent, drain and test connections, meet the applicable stress limits specified in either ASME Section III, Subsection NC, or Appendix F (1974, Summer 1974 or Summer 1983 Addenda, with the appropriate reconciliation). Based on these considerations, it is concluded that the use of the revised analysis will not a) increase the probability of occurrence or consequences of accidents previously evaluated in the USAR, b) increase the probability of occurrence or consequences of a malfunction of equipment important to safety previously evaluated in the USAR, c) create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR, or d) 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.: 99-087

Implementation Document No.: DDC 2M11596

USAR Affected Pages: Figure 9.4-12a

System: Turbine Building Ventilation (HVT)

Title of Change: Correction of USAR Figure 9.4-12a

Description of Change:

This change corrects USAR Figure 9.4-12a, regarding the location of ventilation dampers, to match the design drawing and field configuration. These corrections are editorial in nature and did not result in any change in physical configuration, function, or operation of plant.

Safety Evaluation Summary:

These dampers are part of the original design of the system and are included in the design calculations.

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

Safety Evaluation No.:	99-091
Implementation Document No.:	Calculations EGF-016, EGF-017
USAR Affected Pages:	9.5-28
System:	Diesel Generator Fuel Oil (EGF)
Title of Change:	Revise as Described in the USAR Minimum Required 7-Day Supply of Fuel Oil for Each of the Standby Diesel Generator Fuel Oil Storage Tanks (LDCR 2-99-UFS-108)

Description of Change:

USAR Section 9.5.4.2 has been revised to read as follows:

"Each storage tank is sized to store sufficient fuel oil for continuous operation of its respective diesel engine for 7 days. (Minimum required 7-day supply of fuel oil is 47,824 gallons for each of the standby diesel generator fuel oil storage tanks, and 35,342 gallons for the high-pressure core spray (HPCS) diesel generator fuel oil storage tank)."

Safety Evaluation Summary:

It has been determined, based on the latest disposition of Calculations EGF-016 and EGF-017, that the minimum required 7-day supply of fuel oil in the tank is 47,824 gallons for each of the standby diesel generator fuel oil storage tanks, and 35,342 gallons for the HPCS diesel generator fuel oil storage tank. Fuel oil storage levels are maintained per operations procedures, which are higher than the minimum quantities stated above. Quantities of fuel oil stored in the standby diesel generator fuel oil storage tanks are higher than the minimum required. USAR Section 9.5.4.1 criteria for storage capacity of each standby diesel generator fuel oil storage tank and the recommendations of Regulatory Guide 1.137 are met.

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

Safety Evaluation No.: 99-092

Implementation Document No.: Calculation AX-076A

USAR Affected Pages: Table 6A.9-4

System: Reactor Core Isolation Cooling (ICS) (RCIC)

Title of Change: Revise Stress Analysis Results (CUF only) for AX-076A DER #2-99-2236

Description of Change:

In original analysis, the RCIC piping was qualified for 30 injection stress cycles. Due to qualification of piping for an increased number of cycles (70), the values for CUF have changed.

Safety Evaluation Summary:

RCIC piping was stress analyzed for 30 injection stress cycles. Due to an increased number of cycles, the CUF has been reevaluated. The reevaluation indicates that for a revised number of 70 stress cycles, CUF is within Code allowable. Stress intensities due to this change remain the same. The consequences of a design basis accident are determined not to have increased. Furthermore, this change does not affect the probability of an accident or malfunction of equipment or other systems.

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

Safety Evaluation No.: 99-093

Implementation Document No.: Procedure N2-OP-11, NER-2M-026

USAR Affected Pages: N/A

System: Service Water (SWP)

Title of Change: Service Water System Valve Lineup During Unit Outages

Description of Change:

A minimum header pressure criterion was established (i.e., 63.5 psig) to supplement the operational criteria provided in the SWP system shutdown specification.

Operating restrictions for the use of the manual cross-tie valves during shutdown conditions was also clarified. Due to the operational requirements of the system, major service water maintenance can only be accommodated during plant shutdown conditions or outages. During these periods, the SWP system is placed in various operational lineups to accommodate divisional outages. The outage lineups are established to ensure compliance with Technical Specification requirements for operation during Modes 4 and 5. Although not a requirement, it is desirable to maintain the supply of cooling water to the nonessential portions of the system (i.e., Reactor Building nonessential loads, CCP HXs, CCS HXs, HVT, etc.) during these outages. This allows the continued use of various plant support systems such as instrument air, radwaste, normal building ventilation, etc. Due to the configuration of the system, automatic isolation of the nonessential piping can only be accomplished via the normal supply lines off Division I. Normally closed manual cross-tie valves 2SWP*V17 and 2SWP*V32 have been provided to supply these loads off of Division II, with Division I isolated. These manual valves bypass the automatic isolation valves that normally isolate the system to conserve flow. A manual cross-tie valve (2SWP-V8) is also provided between the nondivisional portions of the return headers. This valve is also used to support the operation of various equipment during system outages.

In addition to using the manual valves, it is sometimes necessary to temporarily defeat the SWP logic for the safety-to-nonsafety interface valves during service water outages to accommodate the desired system lineups. The isolation logic for these valves (i.e., 2SWP*MOV19A, *MOV19B, *MOV93A, *MOV93B, *MOV3A, *MOV3B and *MOV599) initiates their closure on a complete or partial loss of offsite power (LOOP), or the loss of all pumps in the adjacent division. Divisional SWP outages routinely require that all pumps in one division be secured. This initiates an auto-closure signal that isolates the flow paths, limiting the operational

Safety Evaluation No.: 99-093 (cont'd.)

Description of Change: (cont'd.)

flexibility. Flow control during unit outages with offsite power available is maintained by administrative action. Additionally, the loss of pump signal can be defeated without impacting the LOOP (partial and complete) isolation features. This ensures that the required safety function is maintained.

Safety Evaluation Summary:

The objectives of this safety evaluation are to a) provide operational guidelines for the application of the minimum header pressure criteria during Modes 4 and 5, b) clarify the operating restrictions for the use of the manual cross-tie valves during shutdown conditions, and c) evaluate the acceptability of temporary alterations to equipment interlocks and protective logic features during shutdown conditions.

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

Safety Evaluation No.: 99-095

Implementation Document No.: N/A

USAR Affected Pages: 3.9B-22, 3.9B-30, 3.9B-71, 6A.9-15, 9.1-10, 15.2-2, 15.2-32, A.0-1, A.4.3-1, A.4.4-3, A.5.2-4, A.6-2, A.15.0-2, A.15.0-7, A.15.1-8, A.15.2-5, A.15.2-6, A.15.2-9, A.15.4-7, A.15B-1, A.15D-1; Tables 3.9B-2o, A.15.0-4 Sh 1, 2

System: Various

Title of Change: Operation of NMP2 Reload 7/Cycle 8

Description of Change:

This change added new fuel bundles and established a new core loading pattern for Reload 7/Cycle 8 operation of NMP2. Two hundred and forty-eight (248) 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 are documented in the Core Operating Limits Report.

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 7, 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 has been included. Limits on plant operation were established to assure that applicable fuel and reactor coolant system safety limits are not exceeded.

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

Safety Evaluation No.: 00-026

Implementation Document No.: Procedures N2-CTP-RMS-@306, N2-CSP-RMS-W310, and N2-ECP-214

USAR Affected Pages: 7.5-6, 9.4-34, 11.5-5, 11.5-6, 11.5-8, 11.5-13, 12.3-14, 12.3-31; Table 11.5-1 Sh 1, 12.3-2 Sh 2, 3; Figure 11.5-5

System: Radiation Monitoring (RMS)

Title of Change: Gaseous Effluent Monitoring System Iodine and Particulate Monitoring

Description of Change:

This safety evaluation evaluated changing the USAR to reflect an acceptable alternate method of operation of the gaseous effluent monitoring system (GEMS) which has been used at NMP2 since plant startup. Previously, the USAR described on-line isotopic iodine and particulate radioactive gaseous effluent monitors in both the combined Reactor/Radwaste Building vent and the main stack GEMs cabinets. However, it has been shown that using the alternate method of laboratory counting of manually collected GEMs particulate and iodine filters is acceptable.

Safety Evaluation Summary:

The alternate method of quantifying radioactive gaseous iodine and particulate releases evaluated in this safety evaluation is used for both normal operations and post-accident monitoring conditions. Calculations have shown that following a design basis loss-of-coolant accident, plant personnel can retrieve and analyze charcoal and particulate filters in the laboratory without exceeding General Design Criteria dose criteria as implemented by NUREG 0737. Compliance with Technical Specification iodine and particulate dose and dose rate is per the direction given in the Offsite Dose Calculation Manual, which does not provide any discussion or setpoint methodology for iodine or particulate radioactive gaseous effluent monitors. The manual collection of GEMs iodine and particulate filters for subsequent laboratory analysis is neither an initiator nor precursor to any accident. Also, this method of monitoring radioactive gaseous iodine and particulate does not increase the probability or consequences of any accident. Finally, this method of monitoring does not affect the safety of operations or the health and safety of the public. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.: 00-028 Rev. 0 & 1

Implementation Document No.: Mod. N2-099-038

USAR Affected Pages: 9.3-10, 9.3-11, 9.3-14, 9.3-15; Table 3.9A-12 Sh 3, 12; Figures 9.3-1d, 9.3-20b

System: Nitrogen (GSN), Instrument Air Service (IAS)

Title of Change: NMP2 Nitrogen System (GSN) Upgrade

Description of Change:

This modification reduced the leakage of gaseous nitrogen (N2) from automatic depressurization system (ADS) receiving tanks 2IAS*TK4 and 2IAS*TK5 and restored the automatic fill capacity of the GSN system. The scope involved the replacement of poor performing metal-seated valves with soft-seated valves, and the elimination of potential leak paths through abandoned connections to other equipment.

The affected valves were manual valves 2GSN*V72A, 2GSN*V72B, 2GSN*V73A and 2GSN*V73B (located in the missile-protected yard area), and check valves 2GSN*V70A and 2GSN*V70B (located in the Reactor Building). New test connections were added to the GSN system to allow leakage testing of the ADS tank pressure boundary at 2GSN*V70A and 2GSN*V70B and exercise testing of these check valves.

The flanged piping connections leading to previously removed safety valves 2GSN*SV34A and 2GSN*SV34B were eliminated from the piping run and replaced with pieces of piping that do not contain the connections. The lines from ADS receiving tanks 2IAS*TK4 and 2IAS*TK5 to abandoned ADS compressor C2 were physically isolated and end caps were installed on each end to eliminate the potential of leakage through the abandoned equipment.

This modification also replaced pressure control valves (PCV) 2GSN-PCV24A and 2GSN-PCV24B. These valves, located in the yard area, reduce the pressure of the gaseous nitrogen flowing from the nitrogen tank farm for use inside the plant.

The operating position of manual valves 2GSN*V73A and 2GSN*V73B in the missile-protected yard area, and yard manual valve 2GSN-V119 was changed from normally closed to normally open to allow the PCVs to perform their automatic function. The operating position of yard manual valve 2GSN-V120 was changed from normally open to normally closed to maintain the manual isolation of

Safety Evaluation No.: 00-028 Rev 0 &1 (cont'd.)

Description of Change: (cont'd.)

the backup line to the instrument nitrogen system at this location. Valve 2GSN-V119 was opened by this modification.

Safety Evaluation Summary:

The valves and piping replaced by this modification meet or exceed the requirements stated in the applicable ASME or ANSI Codes and the NMP2 site-specific design basis documentation. The safety-related ASME valves will be capable of performing their intended function during design basis accidents, including seismic events, a loss-of-coolant accident, or a main steam line break. These valves and associated piping will maintain their structural and pressure integrity during and following design basis events.

Any postulated nitrogen leakage from the GSN or IAS systems, after this modification is installed, would not affect any scenario postulated in the loss-of-instrument-air accident in USAR Chapter 15 or any other accident described in Chapter 15. No environmental conditions in any safety-related structures or habitability of any areas are affected by this modification. The ability of the valves and piping to distribute nitrogen through the system during a seismic event or the postulated maximum environmental conditions will not be changed by this modification. The emergency fill nitrogen connection in the missile-protected yard area would still be available if any conditions or postulated failures of the nonsafety-related yard valves prevent the yard nitrogen tank farm from being used.

Based on the evaluation performed, it is concluded that these changes do not involve an unreviewed safety question.

Safety Evaluation No.: 00-030

Implementation Document No.: LDCR 2-00-UFS-012

USAR Affected Pages: Table 3.9A-12 Sh 9 & 10

System: Standby Liquid Control (SLS)

Title of Change: Reverse Flow Prevention of 2SLS*V12 & *V14 is an Active Function

Description of Change:

This change revised USAR Table 3.9A-12. A note has been added to signify that valves 2SLS*V12 and *V14 have an active safety function to prevent reverse flow.

Safety Evaluation Summary:

This activity consists of adding a note to USAR Table 3.9A-12 to show that the reverse flow prevention of check valves 2SLS*V12 and *V14 is an active function. The change assures that the closure function of these valves is classified as an active safety function and appropriate actions are taken commensurate with their importance to safety.

The safety evaluation has determined that this activity will not cause any system or component to operate outside its safety parameters.

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

Safety Evaluation No.: 00-031

Implementation Document No.: Temporary Mod. 2000-002

USAR Affected Pages: N/A

System: Service Water (SWP)

Title of Change: SWP to 2CCP-E2B During RFO7

Description of Change:

This Safety Evaluation evaluated the acceptability of cooling 2CCP-E2B with service water during RFO7.

To ensure this source of cooling could be implemented during RFO7, Engineering initiated a configuration change, DDC 2F02128, which incorporated tee fittings, valves, and flanges which have been installed in the inlet and outlet piping for the tubeside flow connections for 2CCP-E2B. This change was implemented prior to RFO7 to minimize the work during the outage and provide physical tie-in locations for connecting the SWP system during the outage. The flow path for SWP to and from 2CCP-E2B will be provided by the supply and return piping to unit cooler 2HVT-UC226, which is located above El. 250 ft. in the Turbine Building. Permanent plant existing piping, valves, and flange connections are provided in supply and return lines for unit cooler 2HVT-UC226. SWP flow will be diverted from 2HVT-UC226 and valved into 2CCP-E2B. Temporary Modification 2000-002 incorporated the design issued by DDC 2F02130, which routed temporary hose and piping to and from the unit cooler flange connections to the piping and flange connections provided by DDC 2F02128.

Engineering Calculation A10.1-0-128 Rev. 00 evaluated the available SWP flow and the resultant heat removal capacity provided by the temporary modification. The analysis concludes that any number of air compressors may be placed in service and that sufficient SWP flow and heat removal capacity exists to sustain continued air compressor operation until SWP flow is restored to 2CCP-E1A, E1B, and E1C.

Safety Evaluation Summary:

Implementing Temporary Modification 2000-002 creates no safety concerns.

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

Safety Evaluation No.: 00-032

Implementation Document No.: Mod. N2-00-003

USAR Affected Pages: 4.6-14, 4.6-15; Figure 9.3-9b

System: Control Rod Drive (RDS), Reactor Building Equipment Drains (DER), Reactor Building Ventilation (HVR)

Title of Change: SDV Vent Piping Reroute

Description of Change:

The vent from the scram discharge volume (SDV), line number 2-RDS-001-104-4, originally tied into a vent line in the DER system, line number 2-DER-002-211-4. That line in turn exhausted to the HVR system. In that configuration, when the SDV was drained following a reset from a scram, hot fluid from the SDV flashed causing two-phase flow in the vent line. The two-phase flow condensed and collected in the ventilation duct. The contaminated liquid then exited the duct via the nearest ventilation duct registers and contaminated the floor at Reactor Building El. 175'-0".

This modification rerouted the SDV vent line from its original destination directly to a tank (2-DER-TK2B). This arrangement accommodates phase separation in the tank and does not introduce liquid into the ventilation duct, thus eliminating the contamination problem.

An existing section of piping was removed and disposed of, and the existing connection to 2-DER-002-211-4 was capped.

Entry to the tank was through an existing connection that was previously installed for a spray nozzle. This spray nozzle was no longer used and there was no pipe connected to this tank connection. An opening was cut into the shielding cover at the man-way to accommodate the new line.

Safety Evaluation Summary:

Based on a review of the evaluated accidents described in USAR Chapter 15, none of the accidents or their probability of occurrence are affected by this change. The portions of the systems involved in the modification have no role in any of the accidents discussed in the USAR. The modification meets the original design, material and construction standards of these systems, and analysis shows that no design ratings are exceeded. The ability to perform a reactor scram is unaffected by the proposed change because it changes piping downstream of the SDV vent

Safety Evaluation No.: 00-032 (cont'd.)

Safety Evaluation Summary: (cont'd.)

isolation valves. Furthermore, no new accident initiators or precursors will be added as a result of the proposed modification. As such, the probability of occurrence of an accident previously evaluated in the USAR is not increased.

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

Safety Evaluation No.: 00-035

Implementation Document No.: Procedure N2-PM-@044

USAR Affected Pages: N/A

System: Unit 1 Substation (NJS), Lighting AC Turbine Area (LAT), Material Handling Turbine Area (MHT)

Title of Change: Temporary Power Supplies to 2NJS-PNL302, 2LAT-PNL300 and 2MHT-CRN1

Description of Change:

The 600-V ac load center 2NJS-US3 must be periodically de-energized during unit outages for the purpose of performing preventive maintenance on the A, B, and C buses. To prevent interfering with other outage activities, 2LAT-PNL300, 2NJS-PNL302 and 2MHT-CRN1 may be supplied temporary power using new cables of comparable size and characteristics as the original cables. The combined A and C or B and C buses of 2NJS-US3 were aligned such that the A and B buses were not de-energized at the same time during the maintenance activities. This minimized impact to loads supplied by the A and B buses.

The 600-V ac, 3-phase nonsafety-related normal distribution panel, 2LAT-PNL300, is normally supplied power from normal 600-V ac load center 2NJS-US3 Bus B. This panel provides power to normal Turbine Building circuits, which provide lighting and outlet receptacles for west Turbine Building El. 250', 277' and 306'. Also supplied from this panel is the normal ac supply for 2VBB-UPS1C, which supplies Turbine Building Gaitronics (communications paging system). A temporary source of power was provided from 2NJS-US3 Bus A.

The 600-V ac, 3-phase nonsafety-related normal distribution panel, 2NJS-PNL302, is normally supplied power from normal 600-V ac load center 2NJS-US3 Bus C. This panel provides power to the following site cafeteria loads: hot water heater, 480-V ac panel, 208/120-V ac panel, HVAC Unit 1, HVAC Unit 2, and makeup air unit heater. A source of temporary power was supplied from a temporary 13.8-kV/575-V ac transformer located in proximity to 2NJS-PNL302 and the NMP2 power loop 13-kV power pack (construction power).

Turbine Building crane 2MHT-CRN1 is normally supplied power from 600-V ac load center 2NJS-US3 Bus C. A temporary source of power was provided from a temporary 575-V ac/480-V ac transformer (200-kVA or greater) via power loop

Safety Evaluation No.: 00-035 (cont'd.)

Description of Change: (cont'd.)

480-V ac power pack (construction power) and 2MHT-SWS01 (Turbine Building crane disconnect switch).

Safety Evaluation Summary:

The facility power loop is currently loaded to less than 50% of its capacity. Adding the additional loads from 2NJS-PNL302 and 2MHT-CRN1 does not affect the ability of this power loop to perform its intended function.

The installation of the proposed temporary power supplies will be performed prior to the start of the plant outage and prior to de-energizing 2NJS-US3 for maintenance. The installation will be performed in accordance with Operations Preventive Maintenance Procedure N2-PM-@044.

The temporary power supplies will be removed and connections to normal power supplies reconnected to ensure normal permanent design configuration is restored upon completion of outage activities. This function will also be controlled by Procedure N2-PM-@044.

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

Safety Evaluation No.: 00-036 Rev. 0 & 1

Implementation Document No.: Mod. N2-00-018

USAR Affected Pages: 5.4-16, 5.4-17, 5.4-18, 5.4-28; Tables 3.9A-12 Sh 5, 3.10B-1 Sh 4, 6.2-56 Sh 4, 7.2-2 Sh 13; Figures 5.4-9c, 5.4-13b, 7.3-10, 7.4-1 Sh 4

System: Reactor Core Isolation Cooling (ICS) (RCIC)

Title of Change: Provide a Time Delay for the RCIC Low Suction Pressure Trip, Install a Manual Keep Fill Capability and Pressure Indicating Instrumentation for the Discharge Piping, and Verify the Adequacy of 60 Seconds to Start Injection

Description of Change:

The RCIC low suction pressure trip was equipped with a 3.5-second time delay to prevent a trip, resulting from either a spurious low pressure or suction trip as a result of a water hammer. A normally closed, manual keep fill line bypassing the RCIC injection valve, and a pressure instrumentation line penetrating primary containment, was installed. This change is supported by an analysis performed by General Electric Company confirming that the change does not impact critical parameters (reactor pressure vessel water level lower than the top of active fuel).

Safety Evaluation Summary:

This safety evaluation has determined that these activities will enhance the reliability of the RCIC system and will not cause the system or any of its components to operate outside of their safety parameters.

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

Safety Evaluation No.: 00-037

Implementation Document No.: Mod. N2-98-009

USAR Affected Pages: N/A

System: Feedwater (FWS)

Title of Change: FWS Weld Overlay Construction Activity Evaluation

Description of Change:

This safety evaulation evaluated the construction activities for installing a weld overlay on the feedwater nozzle N4D of Modification N2-98-009.

This safety evaluation only addresses the impact on nuclear safety from the installation/construction activities to be performed during RF07 on this modification. The installation will take place with NMP2 in Mode 5 (vessel head off, flooded up with the spent fuel pool gates out, and fuel moves may or may not be occurring). The feedwater-nozzle-to-safe-end transition in question is required to be capable of maintaining its pressure boundary function during the repair activity in order to support inventory control.

The weld overlay method, using code cases N-504-1 and N-638 with modification, was found acceptable by the NRC as documented in their letter to NMPC dated March 30, 2000.

Safety Evaluation Summary:

Engineering Analysis NMPC-17Q-304 concludes that the welding activities will not degrade the original nozzle-to-safe-end (pipe) pressure boundary beyond limit load allowables evaluated as a net section collapse under ASME Section XI-1989, Appendix C. The analysis conservatively assumes the nozzle-to-safe-end transition is further evaluated to be degraded by 0.5 inches of wall thickness due to loss of strength during the welding process. Also considered in the evaluation is a leak quantity in the unlikely event that a blow-through or flaw propagation were to cause a throughwall crack resulting in a potential leak path.

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

Safety Evaluation No.: 00-038

Implementation Document No.: Mod. N2-98-009

USAR Affected Pages: N/A

System: Feedwater (FWS)

Title of Change: FWS Weld Overlay Construction Activity Evaluation Mode 4 N4D Nozzle

Description of Change:

This modification placed a weld overlay on the welded transition between the reactor vessel feedwater nozzle N4D and feedwater safe end as a means of repair for the flaw detected by inservice inspection (ISI) examination. Rejectable indications found during ISI examinations were unacceptable for continuous operation for the next operating cycle.

This safety evaluation only addresses the impact on nuclear safety from the installation/construction activities to be performed during RF07 on this modification. The final installation that started in Mode 5 may not be complete before entering into Mode 4. Full structural integrity capability cannot be taken for the weld overlay, since the actual welding or repair exams may not have been completed. This safety evaluation evaluates the acceptability for entry into Mode 4 to allow the welding and/or exams to be completed while in Mode 4 if needed (vessel head on, flooded up to cover the FWS nozzle or higher). The feedwater-nozzle-to-safe-end transition in question is required to be capable of maintaining its pressure boundary function during the repair activity in order to support reactor decay heat removal, reactivity control, and shutdown safety criteria.

Safety Evaluation Summary:

Engineering Analysis NMPC-17Q-304 concludes that the welding activities will not degrade the original nozzle-to-safe-end (pipe) pressure boundary beyond limit load allowables evaluated as a net section collapse under ASME Section XI-1989, Appendix C. The analysis conservatively assumes the nozzle-to-safe-end transition is degraded by 0.5 inches of wall thickness due to loss of strength during the welding process. Also considered in evaluation NMPC-17Q-304 is a leak quantity in the unlikely event that a blow-through or flaw propagation were to cause a throughwall crack resulting in a potential leak path. As the weld is completed, the degraded conditions imposed on the original pipe wall become more conservative and virtually go away after 3 layers are applied. During Mode 4, the weld overlay may be in a semi-completed state. The conditions considered for the Mode 5 Safety Evaluation 2000-037 are modified slightly for Mode 4 (the reactor pressure

Safety Evaluation No.: 00-038 (cont'd.)

Safety Evaluation Summary: (cont'd.)

vessel head is installed and the studs are tensioned, vessel water level lowered to top of vessel flange, and the vessel may be pressurized for noncritical hydro). The results of this safety evaluation conclude that welding and NDE can proceed into Mode 4 as long as the following conditions are maintained during Mode 4, until completion of the weld and acceptable NDE.

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

Safety Evaluation No.:	00-039
Implementation Document No.:	Procedure NTP-TQS-101
USAR Affected Pages:	13.2-6; Table 1.9-1 Sh 52
System:	N/A
Title of Change:	Licensed Operator Candidate Training Program Update

Description of Change:

This change removed the outdated prescriptive requirements for the Licensed Operator Candidate Training Program in the NMP2 USAR and replaced them with a description of a systems approach to training (SAT)-based Licensed Operator Candidate Training Program. The SAT-based program allows flexibility in addressing identified weaknesses and current issues, while satisfying required training specified in 10CFR55.

Safety Evaluation Summary:

The Niagara Mohawk Licensed Operator Candidate Training Program has been developed using a systems approach to training as stated in 10CFR55, and is accredited by the National Nuclear Accrediting Board. Based on this accreditation and certification to the Nuclear Regulatory Commission in NMPC letter NMP1L 0749, dated March 24, 1993, this change satisfies 10CFR55 requirements for Licensed Operator Candidate Training.

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

Safety Evaluation No.: 00-040

Implementation Document No.: DDC 2E12086

USAR Affected Pages: Table 8.3-4 Sh 6

System: Reactor Core Isolation Cooling (ICS)(RCIC)

Title of Change: Actuator Motor Changeout for 2ICS*MOV121

Description of Change:

The RCIC system turbine steam supply isolation outboard valve 2ICS*MOV121 actuator motor was inspected, found to be leaking grease from the sealed bearing, and was replaced with a new motor. The new actuator motor is slightly longer and heavier than the previous motor. In addition, the motor full load current (FLA) and locked rotor current (LRA) are different than the ones addressed in USAR Table 8.3-4. The USAR Table has been revised to reflect that the replacement motor FLA and LRA are 9.10A and 76.3A, respectively.

Safety Evaluation Summary:

Calculations EC-151 Revision 1, Disposition 1L, EC-154 Revision 4, Disposition 4G, and A10.1-H-059 Revision 00, Disposition 00T, indicate that the FLA and LRA difference between the installed motor and USAR Table 8.3-4 is insignificant. The motor LRA insignificant increase (2A) will not impact Division I Diesel Generator loading and capacity. Therefore, operability of the valve is not a concern.

The revised actuator motor FLA and LRA of the new motor will have no impact on the valve stroke time and/or performance. This change does not impact the valve capability to perform its safety function. The RCIC system operation and functions utilized to help mitigate the effects of a design basis accident remain the same. The RCIC valve 2ICS*MOV121 is used as containment isolation to mitigate accident consequences, and is neither an accident precursor nor an initiator. This slight increase in the valve actuator motor LRA will not impact Division I Diesel Generator loading and capacity. Therefore, the proposed change will not increase the probability of occurrence of an accident previously evaluated in the USAR.

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

Safety Evaluation No.: 00-041 Rev. 0 & 1

Implementation Document No.: Mod. N2-00-020

USAR Affected Pages: Figure 9.2-2 Sh 6, 27

System: Service Water (SWP)

Title of Change: Add a Confirmatory Signal to the SWP Nonessential Isolation Valve Auto-Close Logic

Description of Change:

This modification restored the logic to single failure proof to ensure isolation when required (a full loss of offsite power [LOOP]), but eliminated the potential for inadvertent isolation of all nonessential flow paths simultaneously. Specifically, a confirmatory signal was added to the no-flow circuit in series, in each division, from the associated divisions' LOOP monitoring circuit. This ensures that a single failure will not inadvertently close the valves when there is no true loss of power to the bus, while at the same time retain a single failure proof logic to ensure isolation of the SWP Category I/II nonessential isolation valves on a full LOOP. In the event of a partial LOOP, the circuit will also isolate the nonessential valves of the respective division, but it need not be single failure proof to ensure isolation under these conditions.

Safety Evaluation Summary:

After the proposed logic change is incorporated, the SWP system will continue to remove the design basis heat load and reject it to the ultimate heat sink under all conditions, in conformance with 10CFR50 Appendix A, GDC-44, USAR Section 9.2.1, and the licensing basis of NMP2. With the administrative controls in place to address the SWP system status under non-design basis operational events, the pumps and system as a whole are not expected to be exposed to any detrimental operating conditions that would result in equipment degradation. Finally, this change will not subject the nonessential portion of the system to unacceptable fluid transient conditions.

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

Safety Evaluation No.:	00-049
Implementation Document No.:	Procedure GAP-CHE-01
USAR Affected Pages:	9.2-41
System:	Condensate Makeup and Drawoff (CNS)
Title of Change:	Deletion of Redundant/Irrelevant Water Quality Parameters from Condensate Storage Facilities Power Generation Design Bases, USAR Section 9.2.6.1.2

Description of Change:

This safety evaluation evaluated removing total dissolved solids (as $CaCO_3$) and pH from design objective water quality parameters for condensate makeup. These objectives have been eliminated from consideration as a requirement for routine sampling, as other parameters have become industry standards for monitoring makeup water.

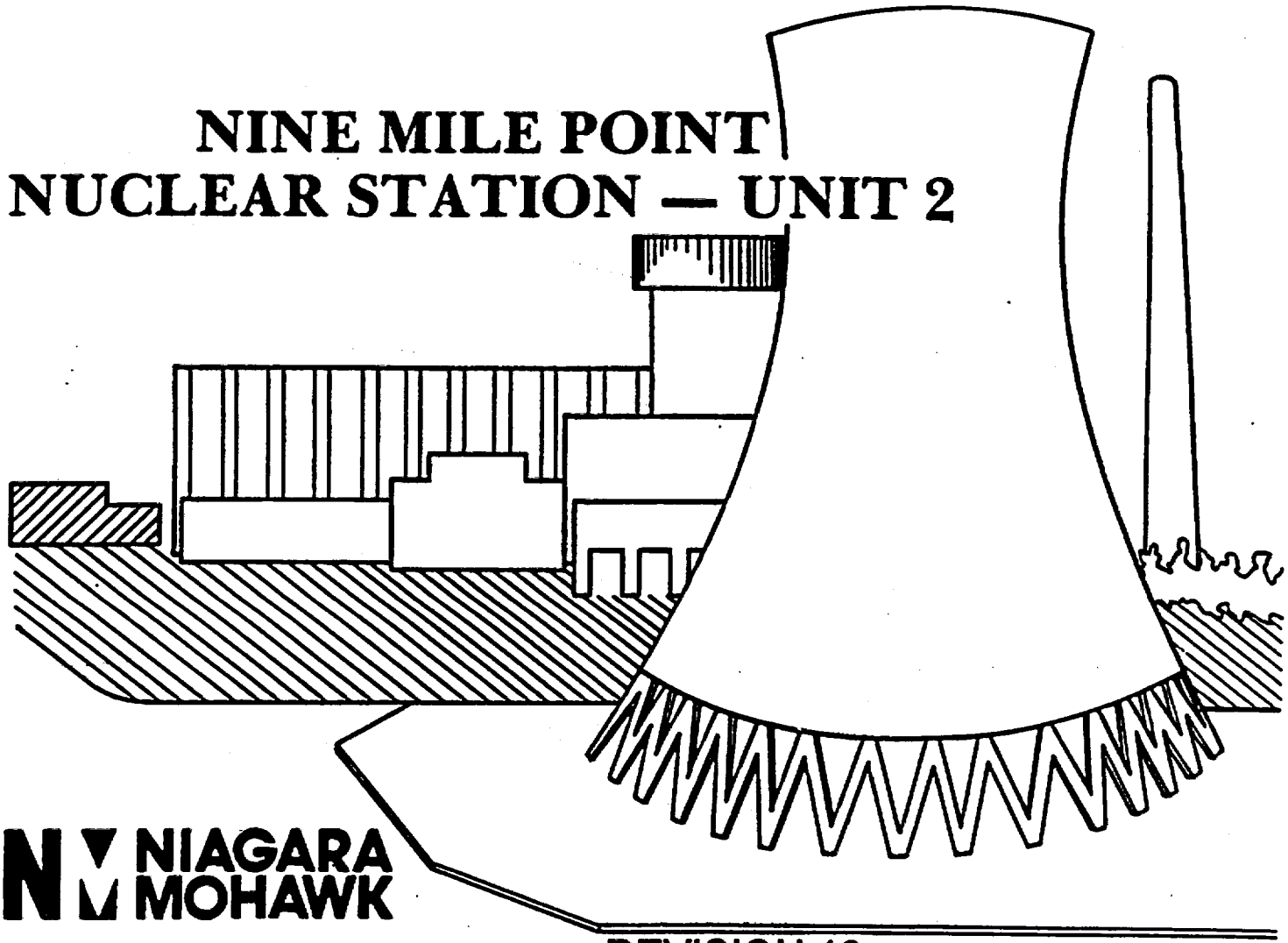
Safety Evaluation Summary:

Although water quality could potentially affect the reactor coolant system pressure boundary integrity, this change to water quality objectives for stored water will not impact the monitoring of reactor water conductivity and chloride, which are the parameters affecting reactor coolant system integrity. Therefore, this change will not increase the probability of occurrence of an accident evaluated in the USAR, which depends on the reactor coolant system pressure boundary.

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

UPDATED SAFETY ANALYSIS REPORT

NINE MILE POINT NUCLEAR STATION — UNIT 2



NIAGARA MOHAWK

REVISION 13

INSERTION INSTRUCTIONS

The following instructions are for the insertion of the current revision into the Nine Mile Point Unit 2 USAR.

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

INSERTION INSTRUCTIONS

LISTS OF EFFECTIVE PAGES

REMOVE	INSERT
EP i	EP i
EP 1-1 through EP 1-4	EP 1-1 through EP 1-4
EP 2-1 through EP 2-20	EP 2-1 through EP 2-20
EP 3-1 through EP 3-11	EP 3-1 through EP 3-11
EP 4-1	EP 4-1 through EP-4-2
EP 5-1 through EP 5-2	EP 5-1 through EP 5-2
EP 6-1 through EP 6-5	EP 6-1 through EP 6-5
EP 7-1 through EP 7-4	EP 7-1 through EP 7-4
EP 8-1 through EP 8-3	EP 8-1 through EP 8-3
EP 9-1 through EP 9-10	EP 9-1 through EP 9-10
EP 10-1 through EP 10-2	EP 10-1 through EP 10-2
EP 11-1 through EP 11-2	EP 11-1 through EP 11-2
EP 12-1 through EP 12-2	EP 12-1 through EP 12-2
EP 13-1	EP 13-1
EP 14-1 through EP 14-4	EP 14-1 through EP 14-4
EP 15-1 through EP 15-5	EP 15-1 through EP 15-5
EP 16-1	EP 16-1
EP 17-1	EP 17-1
EP 18-1	EP 18-1
EP A-1	EP A-1
EP B-1	EP B-1
DAR-1 through DAR-3	DAR-1 through DAR-3
SSA-1 through SSA-2	SSA-1 through SSA-2

INSERTION INSTRUCTIONS

VOLUME 1

REMOVE	INSERT
F 1.2-1	F 1.2-1
F 1.2-7 Sh 2	F 1.2-7 Sh 2
F 1.2-10 Sh 1	F 1.2-10 Sh 1
F 1.2-15 Sh 2	F 1.2-15 Sh 2
T 1.3-9 Sh 8	T 1.3-9 Sh 8

INSERTION INSTRUCTIONS

VOLUME 2

REMOVE	INSERT
1.8-1/-	1.8-1/-
T 1.8-1 Sh 4 & 5	T 1.8-1 Sh 4 & 5
T 1.8-1 Sh 21	T 1.8-1 Sh 21
T 1.8-1 Sh 27	T 1.8-1 Sh 27
T 1.8-1 Sh 29	T 1.8-1 Sh 29
---	T 1.8-1 Sh 29a
T 1.8-1 Sh 41	T 1.8-1 Sh 41
T 1.8-1 Sh 43	T 1.8-1 Sh 43
T 1.8-1 Sh 45	T 1.8-1 Sh 45
---	T 1.8-1 Sh 45a
T 1.8-1 Sh 47	T 1.8-1 Sh 47
T 1.8-2 Sh 1	T 1.8-2 Sh 1
T 1.9-1 Sh 1 thru 53/53a	T 1.9-1 Sh 1 thru 52
1.10-41/42	1.10-41/41a
---	1.10-41b/42
1.10-103/104	1.10-103/104
1.10-105/106	1.10-105/106
T 1.11-1 Sh 3 thru 12	T 1.11-1 Sh 3 thru 12
1.12-27/28 thru 29/30	1.12-27/28 thru 29b/30

INSERTION INSTRUCTIONS

VOLUME 3

REMOVE	INSERT
2.3-23/24	2.3-23/23a
---	2.3-23b/24
2.3-31/32 thru 2.3-33/34	2.3-31/32 thru 2.3-33/34
2.3-47/48	2.3-47/48
2.3-59/60	2.3-59/60
T 2.3-7/T 2.3-8	T 2.3-7/T 2.3-8

INSERTION INSTRUCTIONS

VOLUME 4

REMOVE	INSERT
NO ACTION REQUIRED	

INSERTION INSTRUCTIONS

VOLUME 5

REMOVE	INSERT
NO ACTION REQUIRED	

INSERTION INSTRUCTIONS

VOLUME 6

REMOVE	INSERT
NO ACTION REQUIRED	

INSERTION INSTRUCTIONS

VOLUME 7

REMOVE	INSERT
NO ACTION REQUIRED	

INSERTION INSTRUCTIONS

VOLUME 8

REMOVE	INSERT
3-v/vi	3-v/vi
3-xiii/xiv thru 3-xv/xvi	3-xiii/xiv thru 3-xv/xvi
3.1-29/30	3.1-29/30
---	3.1-30a/30b
T 3.2-1 Sh 13	T 3.2-1 Sh 13
T 3.2-1 Sh 18a	T 3.2-1 Sh 18a
T 3.2-1 Sh 33/34	T 3.2-1 Sh 33/34

INSERTION INSTRUCTIONS

VOLUME 9

REMOVE	INSERT
3.6A-1/2 thru 3.6A-35/36	3.6A-1/2 thru 3.6A-35/36
3.6B-1/2 thru 3.6B-19/-	3.6B-1/2 thru 3.6B-19/-

INSERTION INSTRUCTIONS

VOLUME 10

REMOVE	INSERT
T 3.7A-10	T 3.7A-10
T 3.8-3 Sh 1	T 3.8-3 Sh 1
T 3.8-6 Sh 1 & 2	T 3.8-6 Sh 1 & 2
T 3.8-13	T 3.8-13

INSERTION INSTRUCTIONS

VOLUME 11

REMOVE	INSERT
3.9A-1/2 thru 3.9A-39/40	3.9A-1/2 thru 3.9A-39/40
T 3.9A-2 Sh 3 & 4	T 3.9A-2 Sh 3 & 4
T 3.9A-2 Sh 7	T 3.9A-2 Sh 7
T 3.9A-3	T 3.9A-3
T 3.9A-4 Sh 4 & 5	T 3.9A-4 Sh 4 & 5
T 3.9A-10 Sh 1 thru 4	T 3.9A-10 Sh 1 thru 4
T 3.9A-12 Sh 1 thru 5	T 3.9A-12 Sh 1 thru 5
T 3.9A-12 Sh 7 thru 14	T 3.9A-12 Sh 7 thru 14
3.9B-15/16	3.9B-15/15a
---	3.9B-15b/16
3.9B-19/20 thru 3.9B-21/22	3.9B-19/19a thru 3.9B-21/22
3.9B-29/30 thru 3.9B-31/32	3.9B-29/30 thru 3.9B-31/32
3.9B-43/44	3.9B-43/44
3.9B-71/-	3.9B-71/-
T 3.9B-1 Sh 1/2	T 3.9B-1 Sh 1/2
T 3.9B-2j Sh 1 thru 5	T 3.9B-2j Sh 1 thru 5
T 3.9B-2o	T 3.9B-2o
T 3.9B-2r Sh 1 thru 4	T 3.9B-2r Sh 1 thru 4
T 3.9B-2s Sh 1 & 2	T 3.9B-2s Sh 1 & 2
T 3.9B-2t Sh 1 thru 3	T 3.9B-2t Sh 1 thru 3
T 3.9B-2u Sh 1 thru 3	T 3.9B-2u Sh 1 & 2
T 3.9B-2v Sh 1 thru 4	T 3.9B-2v Sh 1 thru 4
T 3.9B-2y Sh 1 & 2	T 3.9B-2y Sh 1 & 2
T 3.9B-2z Sh 1 thru 3	T 3.9B-2z Sh 1 thru 3

INSERTION INSTRUCTIONS

VOLUME 12

REMOVE	INSERT
3.10A-1/2 thru 3.10A-5/-	3.10A-1/2 thru 3.10A-5/-
T 3.10A-1 Sh 10 thru 17	T 3.10A-1 Sh 10 thru 17
T 3.10B-1 Sh 4	T 3.10B-1 Sh 4
3A-ix/x	3A-ix/x
3A-1/2	3A-1/2
3A.15-1/2	3A.15-1/2
3B-7/8	3B-7/8
3C-21/22	3C-21/22
T 3C.4-6	T 3C.4-6
4.3-1/2	4.3-1/2
4.6-13/14 thru 4.6-15/16	4.6-13/14 thru 4.6-15/16
4.6-19/20 thru 4.6-25/26	4.6-19/20 thru 4.6-25/26
4.6-33/34 thru 4.6-37/38	4.6-33/34 thru 4.6-37/38

INSERTION INSTRUCTIONS

VOLUME 13

REMOVE	INSERT
5-i/ii thru 5-iii/iv	5-i/ii thru 5-iii/iv
5.1-1/2	5.1-1/2
5.2-15/16	5.2-15/16
5.2-25/26 thru 5.2-31/32	5.2-25/26 thru 5.2-31/32
T 5.2-5 Sh 1	T 5.2-5 Sh 1
---	T 5.2-5 Sh 1a
F 5.2-4 Sh 1	F 5.2-4 Sh 1
5.3-1/2	5.3-1/2
---	5.3-2a/2b
5.3-7/8 thru 5.3-9/10	5.3-7/8 thru 5.3-9/10
5.3-15/16 thru 5.3-17/18	5.3-15/16 thru 5.3-17/18
T 5.3-1 Sh 1 & 2	T 5.3-1 Sh 1 & 2
T 5.3-2a	T 5.3-2a
T 5.3-2b	T 5.3-2b
5.4-11/12	5.4-11/12
5.4-15/16 thru 5.4-31/32	5.4-15/16 thru 5.4-31/32
5.4-35/36	5.4-35/36
---	5.4-36a/36b
5.4-39/40	5.4-39/39a
---	5.4-39b/40
5.4-45/46	5.4-45/46
F 5.4-2a	F 5.4-2a
F 5.4-2d	F 5.4-2d
F 5.4-9a	F 5.4-9a
F 5.4-9c	F 5.4-9c
F 5.4-9d	F 5.4-9d
F 5.4-13a thru F 5.4-13g	F 5.4-13a thru F 5.4-13g
F 5.4-14 Sh 2	F 5.4-14 Sh 2
F 5.4-15	F 5.4-15
F 5.4-16b	F 5.4-16b
F 5.4-16d	F 5.4-16d
F 5.4-16e	F 5.4-16e
T 5A-4	T 5A-4
Appendix 5B thru 5B-3/-	Appendix 5B thru 5B-3/-
6-v/vi	6-v/vi
6-xiii/xiv	6-xiii/xiv
6.1-1/2 thru 6.1-5/-	6.1-1/2 thru 6.1-5/-

INSERTION INSTRUCTIONS

VOLUME 14

REMOVE	INSERT
6.2-17/18 thru 6.2-19/20	6.2-17/18 thru 6.2-19/20
---	6.2-20a/20b
6.2-33/34 thru 6.2-35/36	6.2-33/34 thru 35b/36
6.2-49/50	6.2-49/50
6.2-53/54	6.2-53/54
6.2-59/60	6.2-59/60
---	6.2-60a/60b
6.2-73/74	6.2-73/74
6.2-87/88	6.2-87/88
---	6.2-88a/88b
6.2-93/94	6.2-93/94
6.2-97/98	6.2-97/97a
---	6.2-97b/98
T 6.2-4	T 6.2-4
T 6.2-6 Sh 1 & 2	T 6.2-6 Sh 1 & 2
T 6.2-43B	T 6.2-43B
T 6.2-44A thru T 6.2-55	T 6.2-44A thru T 6.2-55
T 6.2-55a Sh 2	T 6.2-55a Sh 2
T 6.2-55b Sh 2	T 6.2-55b Sh 2
T 6.2-55c Sh 2	T 6.2-55c Sh 2
T 6.2-55d Sh 2	T 6.2-55d Sh 2
T 6.2-56 Sh 2	T 6.2-56 Sh 2
T 6.2-56 Sh 4	T 6.2-56 Sh 4
---	T 6.2-56 Sh 4a
T 6.2-56 Sh 5	T 6.2-56 Sh 5
T 6.2-56 Sh 6	T 6.2-56 Sh 6
T 6.2-63 Sh 1 & 2	T 6.2-63 Sh 1 & 2
T 6.2-65 Sh 1/2	T 6.2-65 Sh 1/2
F 6.2-24 thru F 6.2-27	F 6.2-24 thru F 6.2-27
F 6.2-38 Sh 7	F 6.2-38 Sh 7
F 6.2-47 thru F 6.2-50	F 6.2-47 thru F 6.2-50
F 6.2-71a	F 6.2-71a
F 6.2-71b	F 6.2-71b
F 6.2-72a	F 6.2-72a
F 6.2-72i	F 6.2-72i
F 6.2-72k Sh 1	F 6.2-72k Sh 1
F 6.2-73a	F 6.2-73a

INSERTION INSTRUCTIONS

VOLUME 15

REMOVE	INSERT
6.3-7/8 thru 6.3-19/20	6.3-7/8 thru 6.3-19/20
---	6.3-20a/20b
6.3-35/36 thru 6.3-37/38	6.3-35/36 thru 6.3-37/38
T 6.3-1 Sh 1/2 thru 3/4	T 6.3-1 Sh 1/2 thru 3/4
F 6.3-1 Sh 1	F 6.3-1 Sh 1
F 6.3-2	F 6.3-2
F 6.3-3a thru F 6.3-5b	F 6.3-3a thru F 6.3-5b
6.4-1/2 thru 6.4-5/6	6.4-1/2 thru 6.4-5/6
6.5-1/2 thru 6.5-9/-	6.5-1/2 thru 6.5-9/-
F 6.5-1 Sh 1	F 6.5-1 Sh 1
---	F 6.5-1 Sh 1a
F 6.5-1 Sh 2	F 6.5-1 Sh 2
---	F 6.5-1 Sh 2a
F 6.5-1 Sh 4	F 6.5-1 Sh 4
---	F 6.5-1 Sh 4a
F 6.5-1 Sh 5	F 6.5-1 Sh 5
---	F 6.5-1 Sh 5a
F 6.5-1 Sh 8	F 6.5-1 Sh 8
---	F 6.5-1 Sh 8a
6.6-1/2 thru 6.6-3/-	6.6-1/2 thru 6.6-3/-
Appendix 6B thru 6B-7/8	Appendix 6B thru 6B-7/8
7-i/ii	7-i/ii
7.1-1/2	7.1-1/2
7.2-1/2 thru 7.2-3/4	7.2-1/2 thru 7.2-3/4
---	7.2-4a/4b
7.2-17/18	7.2-17/18
T 7.2-2 Sh 3 & 4	T 7.2-2 Sh 3 & 4
T 7.2-2 Sh 6 & 7	T 7.2-2 Sh 6 & 7
T 7.2-2 Sh 9	T 7.2-2 Sh 9
T 7.2-2 Sh 12 & 13	T 7.2-2 Sh 12 & 13
T 7.2-3 Sh 1/2	T 7.2-3 Sh 1/2
7.3-25/26	7.3-25/26
---	7.3-26a/26b
7.3-33/34	7.3-33/34
T 7.3-11 Sh 1/2	T 7.3-11 Sh 1/2
T 7.3-17 Sh 1	T 7.3-17 Sh 1
F 7.3-10 Sh 2	F 7.3-10 Sh 2

INSERTION INSTRUCTIONS

VOLUME 16

REMOVE	INSERT
7.4-21/22	7.4-21/22
F 7.4-1 Sh 4	F 7.4-1 Sh 4
F 7.4-2	F 7.4-2
7.5-1/2 thru 7.5-5/6	7.5-1/2 thru 7.5-5/6
T 7.5-1 Sh 1 thru 14	T 7.5-1 Sh 1 thru 9
T 7.5-2 Sh 6	T 7.5-2 Sh 6
7.6-7/8	7.6-7/8
7.6-13/14	7.6-13/14
7.6-19/20 thru 7.6-25/26	7.6-19/20 thru 7.6-25b/26
F 7.6-6 Sh 1	F 7.6-6 Sh 1
F 7.6-6 Sh 4	F 7.6-6 Sh 4
F 7.6-8	F 7.6-8
7.7-33/34	7.7-33/34
F 7.7-2 Sh 2	F 7.7-2 Sh 2
F 7.7-8	F 7.7-8
T 7B-1 Sh 6	T 7B-1 Sh 6
8-i/ii	8-i/ii
8.1-1/2 thru 8.1-7/-	8.1-1/2 thru 8.1-7/-
8.2-25/26 thru 8.2-27/28	8.2-25/26 thru 8.2-27b/28
8.3-1/2 thru 8.3-83/-	8.3-1/2 thru 8.3-85/-

INSERTION INSTRUCTIONS

VOLUME 17

REMOVE	INSERT
T 8.3-1 Sh 6 & 7	T 8.3-1 Sh 6 & 7
T 8.3-1 Sh 9	T 8.3-1 Sh 9
T 8.3-1 Sh 19 & 20	T 8.3-1 Sh 19 & 20
T 8.3-1 Sh 29	T 8.3-1 Sh 29
T 8.3-1 Sh 31	T 8.3-1 Sh 31
T 8.3-2 Sh 4	T 8.3-2 Sh 4
T 8.3-2 Sh 6 & 7	T 8.3-2 Sh 6 & 7
T 8.3-2 Sh 9	T 8.3-2 Sh 9
T 8.3-2 Sh 18	T 8.3-2 Sh 18
T 8.3-2 Sh 20	T 8.3-2 Sh 20
T 8.3-2 Sh 27 & 28	T 8.3-2 Sh 27 & 28
---	T 8.3-2 Sh 28a
T 8.3-2 Sh 30	T 8.3-2 Sh 30
T 8.3-4 Sh 1	T 8.3-4 Sh 1
T 8.3-4 Sh 4	T 8.3-4 Sh 4
T 8.3-4 Sh 6 & 7	T 8.3-4 Sh 6 & 7
T 8.3-4 Sh 11 & 12	T 8.3-4 Sh 11 & 12
T 8.3-4 Sh 14	T 8.3-4 Sh 14
T 8.3-5 Sh 1 thru 6	T 8.3-5 Sh 1 thru 6
T 8.3-6 Sh 1 thru 6	T 8.3-6 Sh 1 thru 6
9-iii/iv	9-iii/iv
9-xiii/xiv	9-xiii/xiv
9-xix/xx	9-xix/xx
9.1-9/10 thru 9.1-11/12	9.1-9/10 thru 9.1-11/12
9.1-19/20	9.1-19/20
---	9.1-20a/20b
9.1-33/34 thru 9.1-39/40	9.1-33/34 thru 9.1-39/40
9.1-53/54	9.1-53/54
T 9.1-3	T 9.1-3
T 9.1-4 Sh 1/2	T 9.1-4 Sh 1/2
F 9.1-26a	F 9.1-26a

INSERTION INSTRUCTIONS

VOLUME 18

REMOVE	INSERT
9.2-3/4	9.2-3/4
9.2-5/6	9.2-5/6
---	9.2-6a/6b
9.2-11/12 thru 9.2-15/16	9.2-11/12 thru 9.2-15/16
9.2-19/20	9.2-19/20
9.2-29/30	9.2-29/30
9.2-39/40 thru 9.2-41/42	9.2-39/40 thru 9.2-41/42
T 9.2-1A Sh 1 & 2	T 9.2-1A Sh 1 & 2
T 9.2-1B Sh 1 & 2	T 9.2-1B Sh 1 & 2
T 9.2-3	T 9.2-3
T 9.2-6	T 9.2-6
T 9.2-7	T 9.2-7
F 9.2-1h	F 9.2-1h
F 9.2-2 Sh 6	F 9.2-2 Sh 6
F 9.2-2 Sh 11	F 9.2-2 Sh 11
F 9.2-2 Sh 14	F 9.2-2 Sh 14
F 9.2-2 Sh 18	F 9.2-2 Sh 18
F 9.2-2 Sh 27	F 9.2-2 Sh 27
F 9.2-3a	F 9.2-3a
F 9.2-3b	F 9.2-3b
F 9.2-3c	F 9.2-3c
F 9.2-3d	F 9.2-3d
F 9.2-4 Sh 9	F 9.2-4 Sh 9
F 9.2-5a	F 9.2-5a
F 9.2-5c	F 9.2-5c
F 9.2-5d	F 9.2-5d
F 9.2-5e	F 9.2-5e
F 9.2-6a	F 9.2-6a
F 9.2-8a	F 9.2-8a
F 9.2-17c	F 9.2-17c
F 9.2-21	F 9.2-21

INSERTION INSTRUCTIONS

VOLUME 19

REMOVE	INSERT
9.3-5/6 thru 9.3-17/18	9.3-5/6 thru 9.3-17/18
9.3-29/30	9.3-29/29a
---	9.3-29b/30
9.3-37/38	9.3-37/38
F 9.3-1a	F 9.3-1a
F 9.3-1d	F 9.3-1d
F 9.3-1g	F 9.3-1g
F 9.3-5g	F 9.3-5g
F 9.3-6 Sh 2	F 9.3-6 Sh 2
F 9.3-6 Sh 4	F 9.3-6 Sh 4
F 9.3-9a thru F 9.3-9b	F 9.3-9a thru F 9.3-9b
F 9.3-9e thru F 9.3-9f	F 9.3-9e thru F 9.3-9f
F 9.3-13 Sh 4	F 9.3-13 Sh 4
F 9.3-13 Sh 8	F 9.3-13 Sh 8
F 9.3-17a	F 9.3-17a
F 9.3-20b	F 9.3-20b

INSERTION INSTRUCTIONS

VOLUME 20

REMOVE	INSERT
9.4-1/2 thru 9.4-3/4	9.4-1/2 thru 9.4-3/4
9.4-9/10	9.4-9/10
9.4-17/18	9.4-17/18
9.4-25/26	9.4-25/25a
---	9.4-25b/26
9.4-31/32 thru 9.4-33/34	9.4-31/32 thru 9.4-33/34
---	9.4-34a/34b
9.4-37/38	9.4-37/38
9.4-45/46 thru 9.4-59/60	9.4-45/46 thru 9.4-59/60
9.4-63/64 thru 9.4-67/68	9.4-63/64 thru 9.4-67/68
9.4-73/-	9.4-73/-
T 9.4-1 Sh 1 thru 3	T 9.4-1 Sh 1 thru 3
T 9.4-2 Sh 1 thru 14	T 9.4-2 Sh 1 thru 14
T 9.4-3 Sh 1 thru 10	T 9.4-3 Sh 1 thru 10
T 9.4-4 Sh 1 thru 8	T 9.4-4 Sh 1 thru 8
T 9.4-5 Sh 1 thru 20	T 9.4-5 Sh 1 thru 20
T 9.4-6 Sh 1 thru 3	T 9.4-6 Sh 1 thru 3
T 9.4-7 Sh 1/2	T 9.4-7 Sh 1/2
T 9.4-8 Sh 1 thru 6	T 9.4-8 Sh 1 thru 6
T 9.4-9	T 9.4-9
T 9.4-10 Sh 1/2	T 9.4-10 Sh 1/2
T 9.4-11 Sh 1/2	T 9.4-11 Sh 1/2
T 9.4-12 Sh 1 thru 5	T 9.4-12 Sh 1 thru 5
F 9.4-1a	F 9.4-1a
F 9.4-8k thru F 9.4-8L	F 9.4-8k thru F 9.4-8L
---	F 9.4-8m

INSERTION INSTRUCTIONS

VOLUME 21

REMOVE	INSERT
F 9.4-9 Sh 24	F 9.4-9 Sh 24
F 9.4-10b	F 9.4-10b
F 9.4-10d	F 9.4-10d
F 9.4-10e	F 9.4-10e
F 9.4-12a	F 9.4-12a
F 9.4-22a	F 9.4-22a
F 9.4-22b	F 9.4-22b
9.5-3/4 thru 9.5-7/8	9.5-3/4 thru 9.5-7/8
9.5-27/28	9.5-27/28
---	9.5-28a/28b
9.5-31/32	9.5-31/32
9.5-43/44	9.5-43/44
---	9.5-44a/44b
9.5-49/50 thru 9.5-56a/56b	9.5-49/50 thru 9.5-56a/56b
9.5-61/62	9.5-61/62
9.5-73/74	9.5-73/74
9.5-81/82 thru 9.5-87/88	9.5-81/82 thru 9.5-87/88
T 9.5-3 Sh 5	T 9.5-3 Sh 5

INSERTION INSTRUCTIONS

VOLUME 22

REMOVE	INSERT
F 9.5-1b	F 9.5-1b
F 9.5-1e	F 9.5-1e
F 9.5-5 Sh 1	F 9.5-5 Sh 1
F 9.5-20 Sh 2	F 9.5-20 Sh 2

INSERTION INSTRUCTIONS

VOLUME 23

REMOVE	INSERT
9A-i/ii thru 9A-vii/-	9A-i/ii thru 9A-vii/-
9A.2-1/2	9A.2-1/2
9A.3-3/4 thru 9A.3-5/6	9A.3-3/4 thru 9A.3-5/6
9A.3-9/10 thru 9A.3-11/12	9A.3-9/10 thru 9A.3-11/12
9A.3-15/16	9A.3-15/16
9A.3-19/20	9A.3-19/20
9A.3-29/30	9A.3-29/29a
---	9A.3-29b/30
9A.3-35/36 thru 9A.3-37/38	9A.3-35/36 thru 9A.3-37/38
9A.3-41/42	9A.3-41/42
9A.3-47/48	9A.3-47/48
9A.3-51/52 thru 9A.3-55/56	9A.3-51/52 thru 9A.3-55/56
9A.3-59/60	9A.3-59/60
9A.3-63/64	9A.3-63/64
T 9A.3-1 Sh 3 & 4	T 9A.3-1 Sh 3 & 4
T 9A.3-1 Sh 9	T 9A.3-1 Sh 9
T 9A.3-3 Sh 1	T 9A.3-3 Sh 1
T 9A.3-4 Sh 1	T 9A.3-4 Sh 1
T 9A.3-4 Sh 7	T 9A.3-4 Sh 7
T 9A.3-4 Sh 9	T 9A.3-4 Sh 9
T 9A.3-15 Sh 1 thru 4	T 9A.3-15 Sh 1 thru 4
T 9A.3-16	T 9A.3-16
T 9A.3-18 Sh 1 thru 6	T 9A.3-18 Sh 1 thru 5
T 9A.3-19	T 9A.3-19
F 9A.3-3 thru F 9A.3-6	F 9A.3-3 thru F 9A.3-6
F 9A.3-22	F 9A.3-22
---	F 9A.3-24
10-v/vi	10-v/vi
F 10.1-3e	F 10.1-3e
F 10.1-3f	F 10.1-3f
F 10.1-4c	F 10.1-4c
F 10.1-5b	F 10.1-5b
F 10.1-6b	F 10.1-6b
F 10.1-6e	F 10.1-6e
F 10.1-7n	F 10.1-7n
F 10.1-7w	F 10.1-7w
F 10.1-9g	F 10.1-9g
F 10.1-10a	F 10.1-10a

INSERTION INSTRUCTIONS

VOLUME 24

REMOVE	INSERT
10.3-3/4 thru 10.3-5/-	10.3-3/4 thru 10.3-5/-
10.4-3/4	10.4-3/4
10.4-15/16 thru 10.4-17/18	10.4-15/16 thru 10.4-17b/18
10.4-25/26 thru 10.4-31/32	10.4-25/26 thru 10.4-31/32
10.4-39/40	10.4-39/40
T 10.4-1/T 10.4-2	T 10.4-1/T 10.4-2
F 10.4-2a	F 10.4-2a
F 10.4-7a thru F 10.4-7b	F 10.4-7a thru F 10.4-7b
F 10.4-7d thru F 10.4-7e	F 10.4-7d thru F 10.4-7e
F 10.4-7h thru F 10.4-7j	F 10.4-7h thru F 10.4-7j
F 10.4-10 Sh 19	F 10.4-10 Sh 19
F 10.4-11 Sh 5 & 6	F 10.4-11 Sh 5 & 6
11-v/vi	11-v/vi
T 11.1-5/T 11.1-6	T 11.1-5/T 11.1-6
11.2-13/14	11.2-13/14
F 11.2-1e	F 11.2-1e
F 11.2-1L	F 11.2-1L

INSERTION INSTRUCTIONS

VOLUME 25

REMOVE	INSERT
11.3-5/6 thru 11.3-7/8	11.3-5/6 thru 11.3-7/8
F 11.3-1a thru F 11.3-1b	F 11.3-1a thru F 11.3-1b
11.4-3/4	11.4-3/4
11.5-5/6 thru 11.5-7/8	11.5-5/6 thru 11.5-7/8
---	11.5-8a/8b
11.5-11/12 thru 11.5-13/-	11.5-11/12 thru 11.5-13/-
T 11.5-1 Sh 1	T 11.5-1 Sh 1
T 11.5-2 Sh 1 thru 4	T 11.5-2 Sh 1 thru 4
F 11.5-2	F 11.5-2
F 11.5-2a	F 11.5-2a
---	F 11.5-2b
F 11.5-5	F 11.5-5
T 11A.1-10	T 11A.1-10
12-iii/iv	12-iii/iv
12.1-13/-	12.1-13/-
12.2-5/6	12.2-5/6
T 12.2-6 Sh 1 thru 3	T 12.2-6 Sh 1 thru 3
12.3-1/2 thru 12.3-7/8	12.3-1/2 thru 12.3-7/8
12.3-11/12 thru 12.3-13/14	12.3-11/12 thru 12.3-13/14
12.3-21/22 thru 12.3-26a/26b	12.3-21/22 thru 12.3-26a/26b
12.3-31/32	12.3-31/32
T 12.3-1 Sh 1	T 12.3-1 Sh 1
T 12.3-2 Sh 1 thru 3	T 12.3-2 Sh 1 thru 3
T 12.3-3 Sh 1 & 2	T 12.3-3 Sh 1 & 2
F 12.3-7	F 12.3-7
F 12.3-40	F 12.3-40

INSERTION INSTRUCTIONS

VOLUME 26

REMOVE	INSERT
12.4-1/2 thru 12.4-3/4	12.4-1/2 thru 12.4-3/4
12.5-1/2 thru 12.5-3/4 12.5-9/10 T 12.5-3	12.5-1/2 thru 12.5-3/4 12.5-9/10 T 12.5-3
13-i/ii thru 13-iii/iv	13-i/ii thru 13-iii/iv
13.1-1/2 thru 13.1-13/14 F 13.1-1 thru F 13.1-3 F 13.1-5	13.1-1/2 thru 13.1-13/14 F 13.1-1 thru F 13.1-3 F 13.1-5
13.2-1/2 13.2-5/6 thru 13.2-23/-	13.2-1/2 13.2-5/6 thru 13.2-21/-
13.3-1/2	13.3-1/2
13.4-1/2 thru 13.4-3/4	13.4-1/2 thru 13.4-3/4
13.5-5/6	13.5-5/6
13.6-1/-	13.6-1/-

INSERTION INSTRUCTIONS

VOLUME 27

REMOVE	INSERT
15.2-1/2	15.2-1/2
15.2-7/8	15.2-7/8
15.2-31/32	15.2-31/32
T 15.2-3/T 15.2-4	T 15.2-3/T 15.2-4
15.4-13/14	15.4-13/14
T 15.4-5 thru T 15.4-8	T 15.4-5 thru T 15.4-8
15.6-3/4	15.6-3/4
15.6-9/10 thru 15.6-11/12	15.6-9/10 thru 15.6-11/12
T 15.6-13 Sh 12	T 15.6-13 Sh 12
15.7-5/6	15.7-5/6
15.7-13/14 thru 15.7-14a/14b	15.7-13/14 thru 15.7-14a/14b
15.8-1/2	15.8-1/2
15.8-5/-	15.8-5/-

INSERTION INSTRUCTIONS

VOLUME 28

REMOVE	INSERT
15A-29/30	15A-29/30
T 15A-5	T 15A-5
15B.4-1/2	15B.4-1/2
F 15B.5-1	F 15B.5-1
15B.8-1/2	15B.8-1/2
15C-1/2	15C-1/2
F 15C-9	F 15C-9
15E.3-3/4	15E.3-3/4
15G-5/6	15G-5/6
15G-11/-	15G-11/-
17.1-23/24	17.1-23/24
17.1-43/44	17.1-43/44
18.1-1/2	18.1-1/2
T 18.2-1	T 18.2-1
T 18.2-2 Sh 6	T 18.2-2 Sh 6
T 18.2-2 Sh 8	T 18.2-2 Sh 8
F 18.2-6 Sh 2	F 18.2-6 Sh 2
A-v/vi	A-v/vi
A.0-1/2	A.0-1/2
A.4.3-1/-	A.4.3-1/-
A.4.4-1/2 thru A.4.4-3/-	A.4.4-1/2 thru A.4.4-3/-
A.5.2-3/4	A.5.2-3/4
A.6-1/2	A.6-1/2
A.15.0-1/2	A.15.0-1/2
A.15.0-7/-	A.15.0-7/-
T A.15.0-4 Sh 1 & 2	T A.15.0-4 Sh 1 & 2
A.15.1-7/8	A.15.1-7/8
A.15.2-5/6 thru A.15.2-9/-	A.15.2-5/6 thru A.15.2-9/-
A.15.4-5/6 thru A.15.4-7/-	A.15.4-5/6 thru A.15.4-7/-

INSERTION INSTRUCTIONS

VOLUME 28 (Cont'd.)

REMOVE	INSERT
A.15B-1/-	A.15B-1/-
A.15D-1/-	A.15D-1/-
B.0-1/-	B.0-1/2
B.1-1/2 thru B.1-3/4	B.1-1/2 thru B.1-5/-
B.2-5/6	B.2-5/6 thru B.2-11/12
B.5-1/2	B.5-1/2 thru B.5-5/-
B.17-1/-	B.17-1/2 thru B.17-3/-
B.18-1/2	B.18-1/2
T B-1 Sh 1 & 2	T B-1 Sh 1 & 2
T B-3 Sh 2	T B-3 Sh 2
T B-3 Sh 4	T B-3 Sh 4

INSERTION INSTRUCTIONS

APPENDIX 6A

REMOVE	INSERT
6A.1-1/2 thru 6A.1-3/4	6A.1-1/2 thru 6A.1-3/4
6A.2-1/2 thru 6A.2-11/12	6A.2-1/2 thru 6A.2-11/12
6A.3-1/2 thru 6A.3-13/-	6A.3-1/2 thru 6A.3-13/-
6A.9-15/- T 6A.9-4	6A.9-15/- T 6A.9-4

INSERTION INSTRUCTIONS

APPENDIX 9B

REMOVE	INSERT
9B-i/ii thru 9B-iii/iv	9B-i/ii thru 9B-iii/iv
9B.4-1/2 thru 9B.4-7/8 F 9B.4-2	9B.4-1/2 thru 9B.4-7/8 F 9B.4-2
9B.5-1/2 thru 9B.5-3/4 T 9B.5-1	9B.5-1/2 thru 9B.5-3/4 T 9B.5-1
T 9B.6-1 Sh 1 thru 6 T 9B.6-3 Sh 1 thru 9	T 9B.6-1 Sh 1 thru 3 T 9B.6-3 Sh 1 thru 4
9B.8-1/2 thru 9B.8-7/- T 9B.8-1 Sh 1 thru 38 T 9B.8-2 Sh 1 thru 15 T 9B.8-3 Sh 1 thru 12	9B.8-1/2 thru 9B.8-7/- T 9B.8-1 Sh 1 thru 38 T 9B.8-2 Sh 1 thru 15 T 9B.8-3 Sh 1 thru 12
9B.9-1/-	9B.9-1/-

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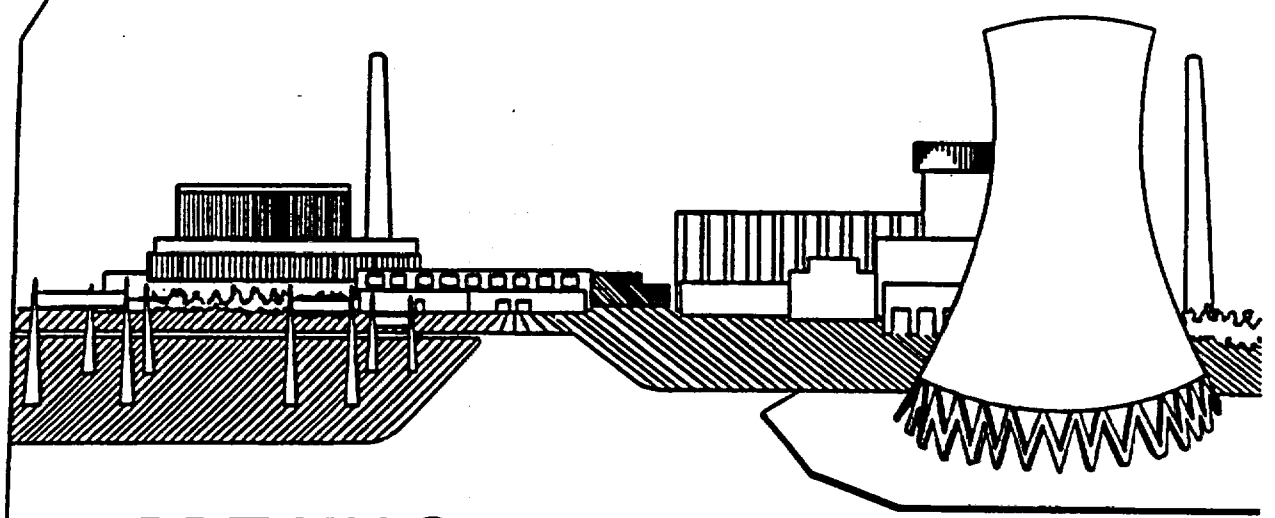

NIAGARA MOHAWK

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17.1-24	R13				
17.1-25	R09				
17.1-26	R09				
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18.2-10	R09				
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A-vi	R10	A.15.1-6	R10		
		A.15.1-7	R10		
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A.4.3-1	R13	A.15.2-7	R13		
A.4.4-1	R13	A.15.2-8	R10		
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		A.15.4-1	R10		
A.5.2-1	R10	A.15.4-2	R10		
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		B.9-1	R09		
B.1-1	R13	B.9-2	R09		
B.1-2	R13	B.9-3	R09		
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B.1-4	R13	B.10-1	R09		
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		B.10-3	R09		
B.2-1	R09				
B.2-2	R09	B.11-1	R09		
B.2-3	R09	B.11-2	R09		
B.2-4	R10				
B.2-5	R13	B.12-1	R09		
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		B.15-2	R10		
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B.5-1	R09	B.18-3	R09		
B.5-2	R13				
B.5-3	R13	T B-1 Sh 1	R13		
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6A-iv	R08	T 6A.2-1a Sh 1	R10	6A.3-9	R13
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6A-vii	R08	T 6A.2-1b Sh 2	A09	6A.3-12	R13
6A-viii	R08	T 6A.2-2 Sh 1	A23	6A.3-13	R13
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6A-xi	R08	T 6A.2-3 Sh 1	A00	T 6A.3-3	R05
6A-xii	R08	T 6A.2-3 Sh 2	A23	T 6A.3-4	R05
6A-xiii	R08	T 6A.2-3 Sh 3	A00	T 6A.3-5	R05
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6A-xv	R08	F 6A.2-2	A00	F 6A.3-1	R05
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6A-xviii	R08	F 6A.2-5	R00	F 6A.3-4	R05
6A-xix	R08	F 6A.2-6	R02	F 6A.3-5	R05
6A-xx	R08	F 6A.2-7	R02		
6A-xxi	R08	F 6A.2-8	R00	6A.4-1	R08
6A-xxii	R08	F 6A.2-9	R00	6A.4-2	R08
		F 6A.2-10	R00	6A.4-3	R08
6A.1-1	R13	F 6A.2-11	R00	6A.4-4	R08
6A.1-2	R13	F 6A.2-12	R00	6A.4-5	R08
6A.1-3	R13	F 6A.2-13	R02	6A.4-6	R08
6A.1-4	R13	F 6A.2-14	R00	6A.4-7	R08
T 6A.1-1 Sh 1	R04	F 6A.2-15	R00	6A.4-8	R08
T 6A.1-1 Sh 2	R04	F 6A.2-16	A00	6A.4-9	R08
T 6A.1-1 Sh 3	R04	F 6A.2-17	R00	6A.4-10	R08
F 6A.1-1	A17	F 6A.2-18	R00	6A.4-11	R08
F 6A.1-2	A00	F 6A.2-19	R00	6A.4-12	R08
F 6A.1-3	R02	F 6A.2-20	R00	6A.4-13	R08
		F 6A.2-21	R08	6A.4-14	R08
6A.2-1	R13	F 6A.2-22	R00	6A.4-15	R08
6A.2-2	R13	F 6A.2-23	R00	6A.4-16	R08
6A.2-3	R13	F 6A.2-24	R00	6A.4-17	R08
6A.2-4	R13	F 6A.2-25	R08	6A.4-18	R08
6A.2-5	R13	F 6A.2-26	A00	6A.4-19	R08
6A.2-6	R13			6A.4-20	R08
6A.2-7	R13	6A.3-1	R13	6A.4-21	R08
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6A.2-9	R13	6A.3-3	R13	6A.4-23	R08
6A.2-10	R13	6A.3-4	R13	6A.4-24	R08
6A.2-11	R13	6A.3-5	R13	6A.4-25	R08

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6A.4-27	R08	F 6A.4-5	R08	F 6A.4-50	A21
6A.4-28	R08	F 6A.4-6	A00	F 6A.4-51	A25
6A.4-29	R08	F 6A.4-7	R08	F 6A.4-52	A21
6A.4-30	R08	F 6A.4-8	R05	F 6A.4-53	A21
6A.4-31	R08	F 6A.4-9	R05	F 6A.4-54	A21
6A.4-32	R08	F 6A.4-10	R05	F 6A.4-55	A21
6A.4-33	R10	F 6A.4-11	R00	F 6A.4-56	A21
6A.4-33a	R10	F 6A.4-12	R05	F 6A.4-57	A21
6A.4-33b	R10	F 6A.4-13	A17		
6A.4-34	R08	F 6A.4-14	R05	6A.5-1	R08
6A.4-35	R08	F 6A.4-15	R05	6A.5-2	R08
6A.4-36	R10	F 6A.4-16	R05	6A.5-3	R08
6A.4-36a	R10	F 6A.4-17	R05	6A.5-4	R08
6A.4-36b	R10	F 6A.4-18	R05	6A.5-5	R08
6A.4-37	R08	F 6A.4-19	R05	6A.5-6	R08
6A.4-38	R08	F 6A.4-20	R05	T 6A.5-1	A26
6A.4-39	R08	F 6A.4-21	R05	T 6A.5-2 Sh 1	A26
6A.4-40	R08	F 6A.4-22	R05	T 6A.5-2 Sh 2	A26
6A.4-41	R08	F 6A.4-23	R05	T 6A.5-3	A23
T 6A.4-1	R09	F 6A.4-24	R05	T 6A.5-4 Sh 1	A24
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T 6A.4-5	A00	F 6A.4-28	R05	T 6A.5-6 Sh 2	A07
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T 6A.4-10 Sh 1	A17	F 6A.4-34	R05	F 6A.5-3	A00
T 6A.4-10 Sh 2	A27	F 6A.4-35	R05	F 6A.5-4	A00
T 6A.4-10 Sh 3	A17	F 6A.4-36	R00	F 6A.5-5	A00
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T 6A.4-12	R05	F 6A.4-38	A00	F 6A.5-7	A00
T 6A.4-13	R05	F 6A.4-39	R02	F 6A.5-8	A00
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F 6A.5-25	A00
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F 6A.5-27	A23
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F 6A.5-29	A23
F 6A.5-30	A23
F 6A.5-31	A23
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F 6A.5-35	A00
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F 6A.5-41	A23
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6A.6-1	R08
6A.6-2	R08
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T 6A.6-2 Sh 2	A23
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6A.9-6	R08
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6A.9-10	R08
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T 6A.9-2	R00
T 6A.9-2a	R00
T 6A.9-2b	R00
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T 6A.9-5	R08
T 6A.9-6	R08
T 6A.9-7	R00
F 6A.9-1	R07
F 6A.9-2	A23
F 6A.9-3	R07
6A.10-1	R08
6A.10-2	R08
6A.10-3	R08
6A.10-4	R08
6A.10-5	R08
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T 6A.10-2 Sh 1	R08
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F 6A.10-5	R00
F 6A.10-6	R00
F 6A.10-6a	R08
F 6A.10-7	R00
F 6A.10-8	R00
F 6A.10-9	R00
F 6A.10-10	R00
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T 9B.6-1 Sh 1	R13
T 9B.6-1 Sh 2	R13
T 9B.6-1 Sh 3	R13
T 9B.6-2	A11
T 9B.6-3 Sh 1	R13
T 9B.6-3 Sh 2	R13
T 9B.6-3 Sh 3	R13
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9B.7-1	R09
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9B.8-3	R13
9B.8-4	R13
9B.8-5	R13
9B.8-6	R13
9B.8-7	R13
T 9B.8-1 Sh 1	R13
T 9B.8-1 Sh 2	R13
T 9B.8-1 Sh 3	R13
T 9B.8-1 Sh 4	R13
T 9B.8-1 Sh 5	R13
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T 9B.8-1 Sh 37	R13
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T 9B.8-2 Sh 1	R13
T 9B.8-2 Sh 2	R13
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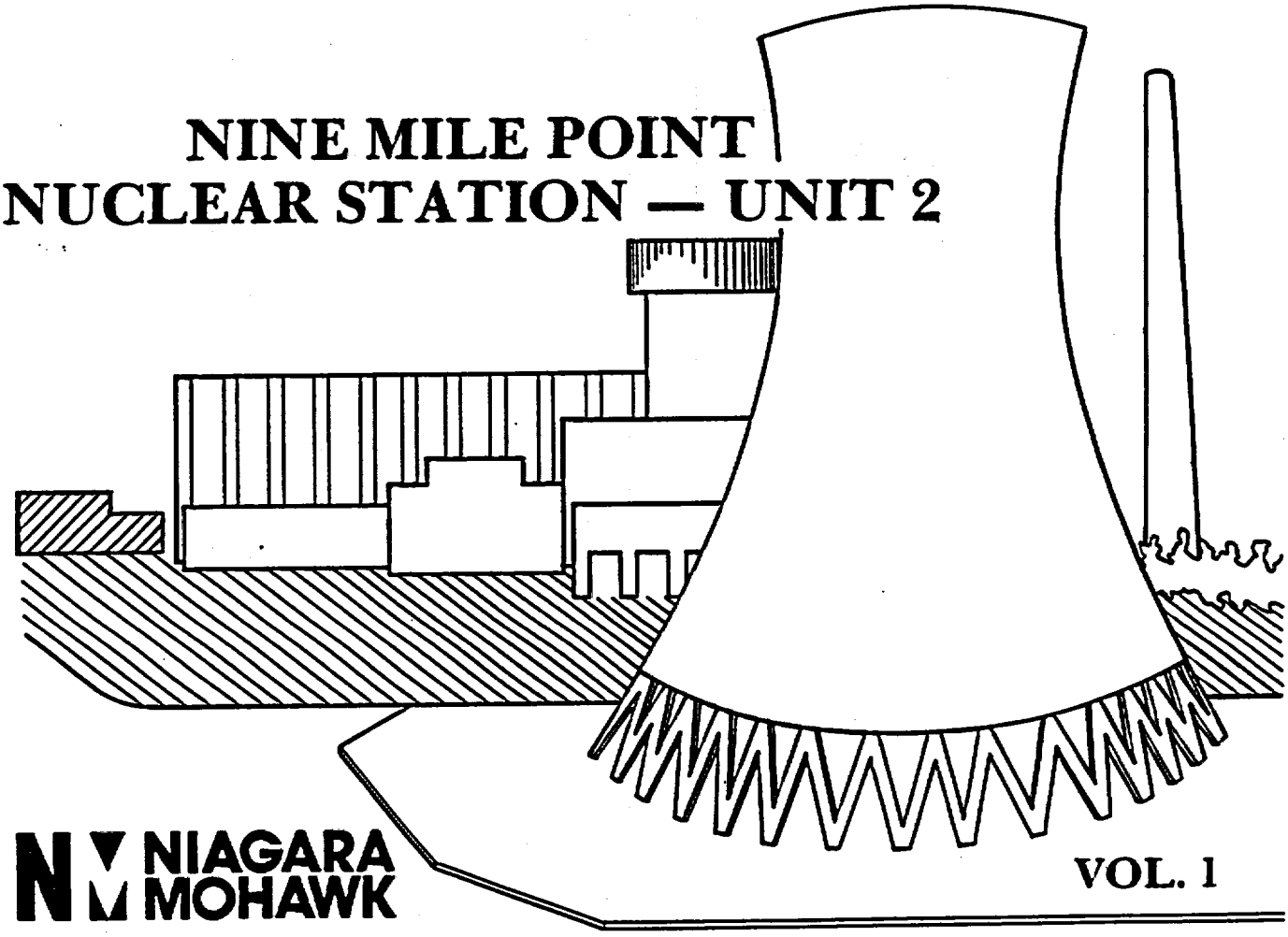
9B.12-8	R09				
9B.12-9	R09				

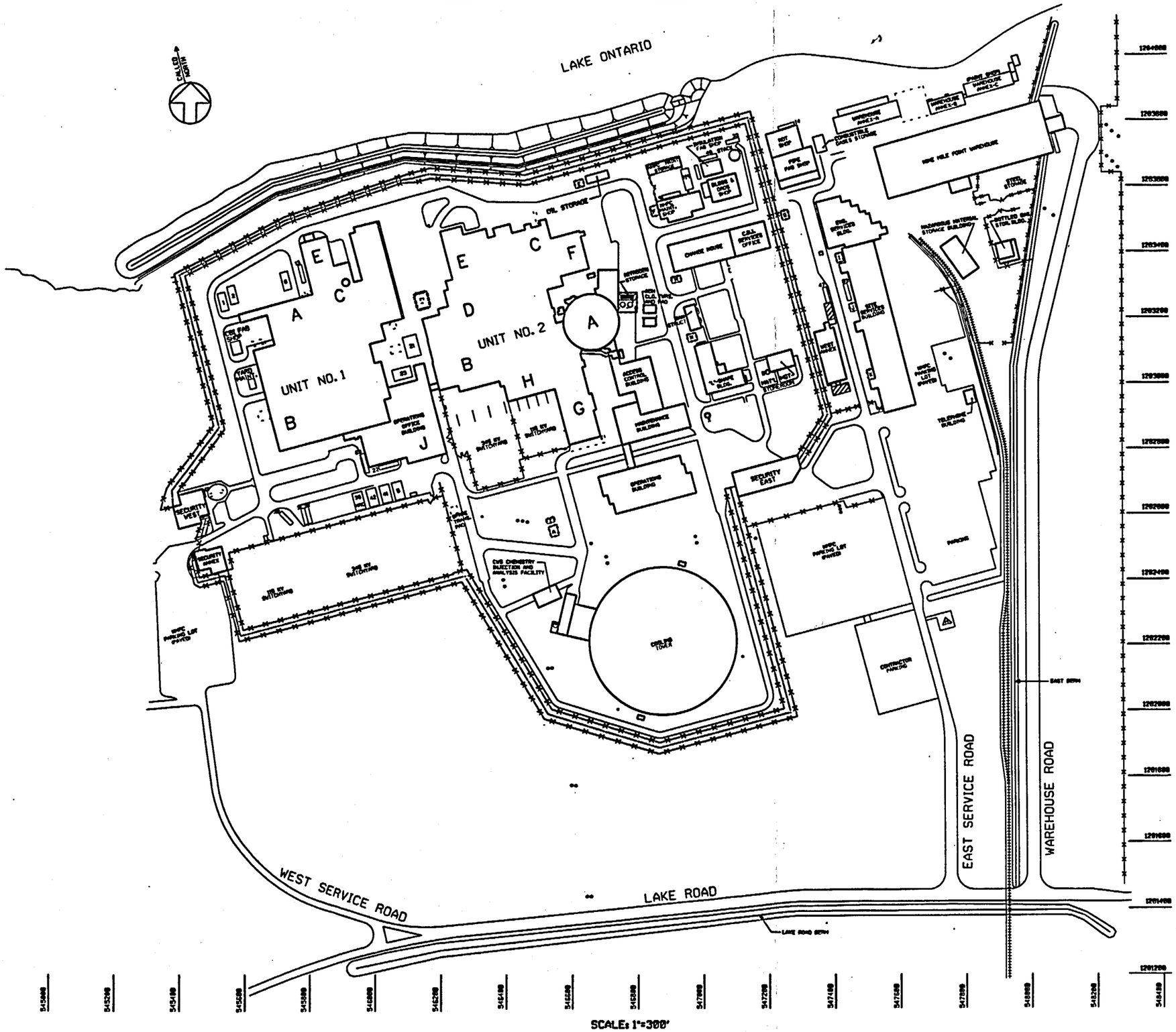
UPDATED SAFETY ANALYSIS REPORT

NINE MILE POINT NUCLEAR STATION — UNIT 2



NIAGARA MOHAWK

VOL. 1



IDENTIFICATION LEGEND

- A REACTOR BUILDING
- B TURBINE BUILDING
- C RADWASTE BUILDING
- D HEATER BAYS
- E SCREENWELL BUILDING
- F CONDENSATE STORAGE TANK BLDG
- G CONTROL BUILDING
- H NORMAL SWITCHGEAR BUILDING
- J ADMINISTRATION BUILDING

KEY

- PERMANENT STRUCTURES
- TEMPORARY STRUCTURES
- 73 OFFICE TRAILERS
- DOCUMENT STORAGE VAULTS
- S STORAGE TRAILERS
- PERMANENT FENCE
- CONSTRUCTION FENCE
- RAILROAD TRACKS
- TRANSMISSION LINE POLES
- A ELECTRIC SUBSTATION
- CONCRETE SLABS AND PADS

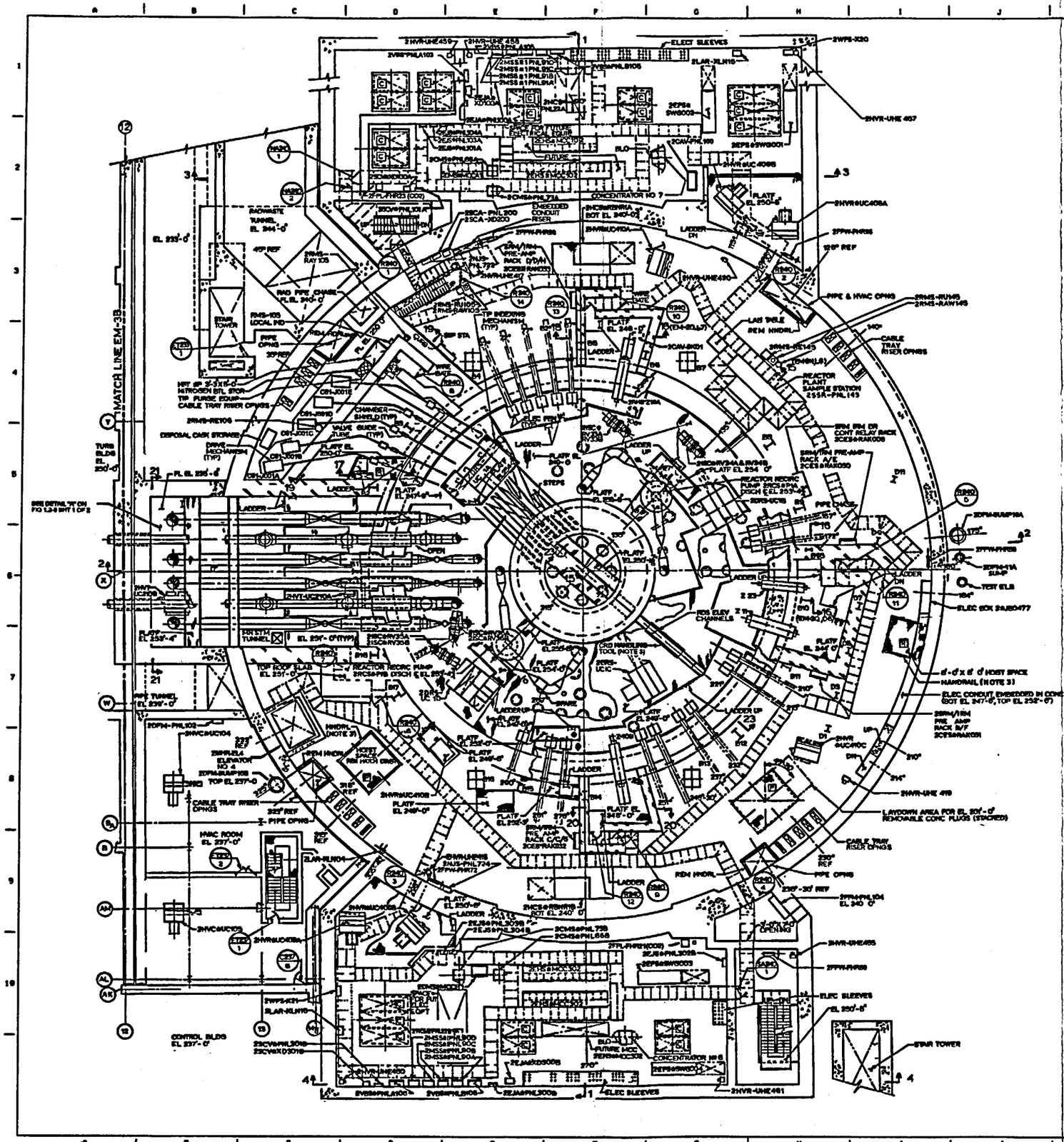
THIS DRAWING WAS PRODUCED ELECTRONICALLY, DO NOT MAKE ANY CHANGES MANUALLY.

SOURCE: EY-8S

FIGURE 1.2-1

PLOT PLAN

NIAGARA MOHAWK POWER CORPORATION
NINE MILE POINT-UNIT 2
UPDATED SAFETY ANALYSIS REPORT



NOTES:

1. BOLD: NONE
2. GENERAL NOTES & REFERENCES SEE EM-0A
3. CRD HANDLING TOOL REMOVED DURING PLANT OPERATIONS.

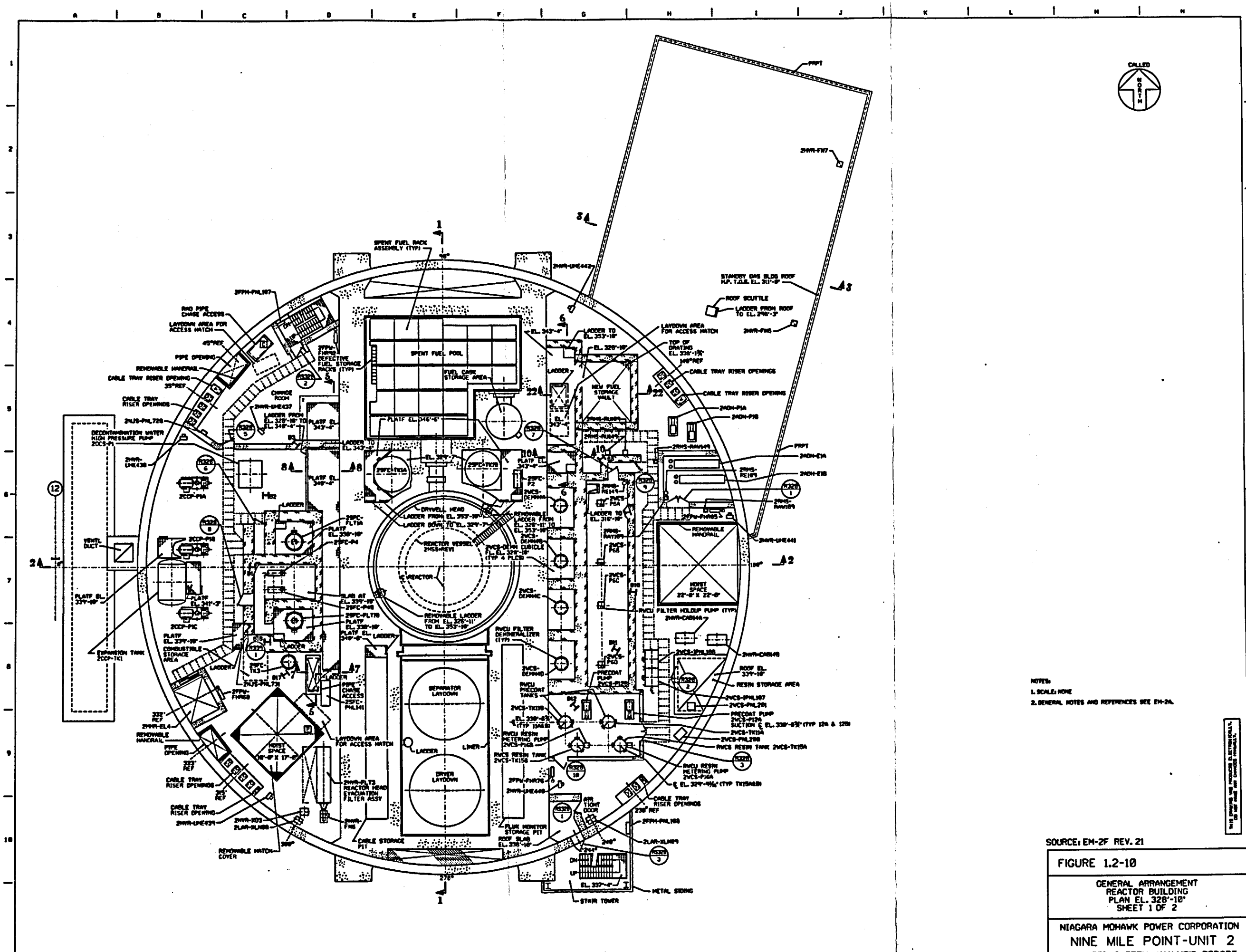
THIS DRAWING WAS PRODUCED ELECTRONICALLY. DO NOT MAKE ANY CHANGES MANUALLY.

SOURCE: EM-002C, REV. 22

FIGURE 1.2-7

GENERAL ARRANGEMENT REACTOR BUILDING PLAN EL. 240'-0" SHEET 2 OF 2

NIAGARA MOHAWK POWER CORPORATION
NINE MILE POINT-UNIT 2
UPDATED SAFETY ANALYSIS REPORT



NOTES:
1. SCALE: NONE
2. GENERAL NOTES AND REFERENCES SEE EM-2A.

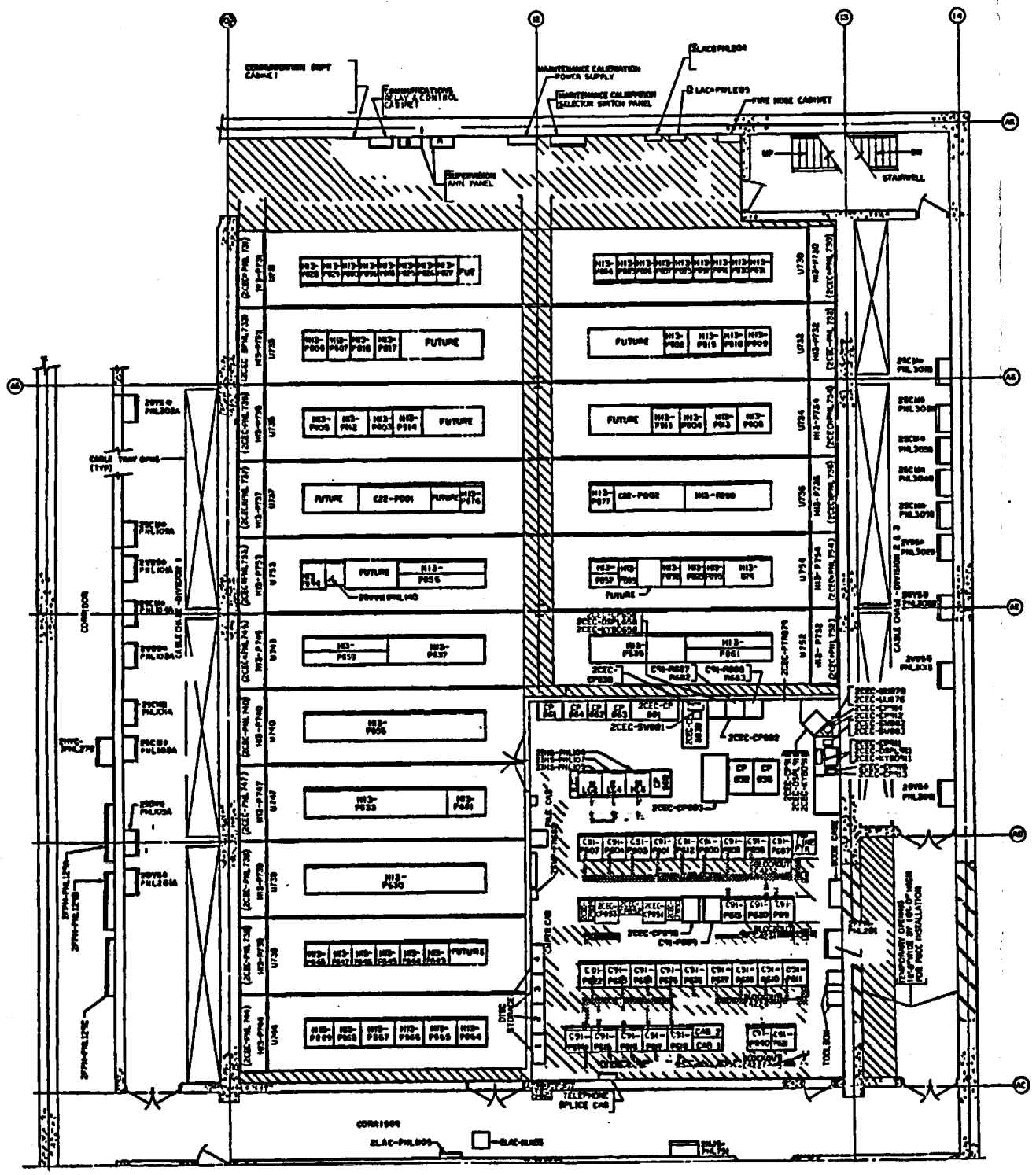
THIS DRAWING WAS PRODUCED ELECTRONICALLY. DO NOT MAKE ANY CHANGES MANUALLY.

SOURCE: EM-2F REV. 21

FIGURE 1.2-10

GENERAL ARRANGEMENT
REACTOR BUILDING
PLAN EL. 328'-10"
SHEET 1 OF 2

NIAGARA MOHAWK POWER CORPORATION
NINE MILE POINT-UNIT 2
UPDATED SAFETY ANALYSIS REPORT



STONE AND WEBSTER EQPT NUMBER	G.E. ITEM NUMBER	DESCRIPTION	EQUIPMENT CLASSIFICATION
		NSS SYSTEM PANELS	
2CEC-PNL830	H13-P830	NSS ANNUNCIATOR LOGIC & RETRANSMITTING RELAY CABINET	NE
		BOP SYSTEM PANELS	
2 CEC-PNL 998	H13-P998	LOOSE PARTS MONITORING PANEL	NE
2 CEC-PNL 990	H13-P990	SEISMIC INSTRUMENTATION PANEL	NE
2 CEC-PNL 999	H13-P999	ERF ISOLATION PANEL	CI,NE
2 CEC-PNL 991	H13-P991	RETARD PANEL	NE
2 SVS-PNL 980	———	SAFETY RELIEF VALVE INDICATION MONITORING	NE
2CEC-PNL 800	H13-P800	AUX BLR SVCS SPWR 2A05-01 BACK UP PROT	
2CEC-PNL 801	H13-P801	4.16KV BUS 2A05-2A0201 BUS PROT	
2CEC-PNL 802	H13-P802	4.16KV BUS 2A05-2A0203 BUS PROT	
2CEC-PNL 803	H13-P803	RCS STA SVCS SPWR 2A10-2S01A PRI PROT	
2CEC-PNL 805	H13-P805	RCS STA SVCS SPWR 2A10-2S01A BACK UP PROT	
2CEC-PNL 807	H13-P807	AUX SPWR 2A10-2S01 PROT	
2CEC-PNL 808	H13-P808	RCS STA SVCS SPWR 2A10-2S01B BACK UP PROT	
2CEC-PNL 809	H13-P809	RCS STA SVCS SPWR 2A10-2S01B PRI PROT	
2CEC-PNL 810	H13-P810	AUX SPWR 2A10-2S02 PROT	
2CEC-PNL 811	H13-P811	AUX BLR SVCS SPWR 2A05-01 PRI PROT	
2CEC-PNL 813	H13-P813	4.16KV BUS 2A05-2A0205 PROT 4.16KV BUS 2A05-2A0207 PROT	
2CEC-PNL 814	H13-P814	4.16KV BUS 2A05-2A0202 PROT 4.16KV BUS 2A05-2A0204 PROT	
2CEC-PNL 815	H13-P815	4.16KV BUS 2A05-2A0206 PRI BUS PROT	
2CEC-PNL 816	H13-P816	4.16KV BUS 2A05-2A0208 PROT 4.16KV BUS 2A05-2A0210 PROT	
2CEC-PNL 817	H13-P817	ENERGY MANAGEMENT SYSTEM INTERFACE EAS	
2CEC-PNL 819	H13-P819	ENERGY MANAGEMENT SYSTEM REMOTE TERMINAL UNIT	
2CEC-PNL 820	H13-P820		
2CEC-PNL 821	H13-P821		
2CEC-PNL 822	H13-P822		
2CEC-PNL 823	H13-P823		
2CEC-PNL 824	H13-P824		
2CEC-PNL 825	H13-P825		
2CEC-PNL 826	H13-P826		
2CEC-PNL 827	H13-P827		
2CEC-PNL 828	H13-P828		
2CEC-PNL 829	H13-P829		
2CEC-PNL 830	H13-P830		
2CEC-PNL 890	H13-P890	BOP INSTRUMENT PANELS	
2CEC-PNL 850	H13-P850		
2CEC-PNL 851	H13-P851		
2CEC-PNL 852	H13-P852		
2CEC-PNL 853	H13-P853		
2CEC-PNL 854	H13-P854		
2CEC-PNL 855	H13-P855		
2CEC-PNL 856	H13-P856		
2CEC-PNL 857	H13-P857		
2CEC-PNL 858	H13-P858		
2CEC-PNL 859	H13-P859		
2CEC-PNL 860	H13-P860		
2CEC-PNL 861	H13-P861	VIBRATION MONITORING PANEL	NE
2CEC-PNL 862	H13-P862	RCIC & SUPPRESSION POOL TEMP MONITORING PANEL	NE
2CEC-PNL 863	H13-P863	RHR & SUPPRESSION POOL TEMP MONITORING PANEL	NE
2CEC-PNL 864	H13-P864		
2CEC-PNL 865	H13-P865		
2CEC-PNL 866	H13-P866		
2CEC-PNL 867	H13-P867		
2CEC-PNL 868	H13-P868		
2CEC-PNL 869	H13-P869		
2CEC-PNL 870	H13-P870		
2CEC-PNL 871	H13-P871		
2CEC-PNL 872	H13-P872		
2CEC-PNL 873	H13-P873		
2CEC-PNL 874	H13-P874		
2CEC-PNL 875	H13-P875		
2CEC-PNL 876	H13-P876		
2CEC-PNL 877	H13-P877		
2CEC-PNL 878	H13-P878		
2CEC-PNL 879	H13-P879		
2CEC-PNL 880	H13-P880	BOP ANNUNCIATOR LOGIC CABINET	NE
2CEC-PNL 881	H13-P881	DIVISION 1 ISOLATION CABINET	CI,NE
2CEC-PNL 882	H13-P882	DIVISION 2 ISOLATION CABINET	CI,NE
2CEC-PNL 883	H13-P883	DIVISION 3 ISOLATION CABINET	CI,NE
2CEC-PNL 884	H13-P884		
2CEC-PNL 885 (TYP)	H13-P885 (TYP)	TURBINE EHC PANELS(A THRU F)	NE
2CEC-PNL 886 (TYP)	H13-P886 (TYP)		
2CEC-PNL 887	H13-P887	AUXILIARY RELAY PANELS	
2CEC-PNL 888	H13-P888	BOP ANNUNCIATOR LOGIC & RETRANSMITTING RELAY CAB	NE
2CEC-PNL 889	H13-P889	DIVISION 1 AUXILIARY RELAY PANELS	NE
2CEC-PNL 891	H13-P891	DIVISION 2 AUXILIARY RELAY PANELS	NE
2CEC-PNL 892 (TYP)	H13-P892 (TYP)	GENERATOR MAIN TRANSFORMER AND SIGNAL STATION SERVICE TRANSFORMER RELAY PANELS	NE
2CEC-PNL 893	H13-P893		
2CEC-PNL 894	H13-P894	345KV SWYD RELAY PANEL ALT 1	NE
2CEC-PNL 895	H13-P895	345KV SWYD RELAY PANEL ALT 2	NE
2CEC-PNL 896	H13-P896	SPARE PANEL	
2CEC-PNL 897	H13-P897	SPARE PANEL	
2CEC-CP001	CH-P001	**COMPUTER & POOL EQUIPMENT** CENTRAL SYSTEMS UNIT, NSS SYSTEM	
2CEC-CP100	CH-P100	CENTRAL SYSTEMS UNIT, BOP	
2CEC-CP207	CH-P207	COMMON CORE MEMORY	
2CEC-CP203	CH-P203	LARGE CORE STORAGE CABINET NO.1(BOP)	
2CEC-CP209	CH-P209	LARGE CORE STORAGE CABINET NO.2 (NSS)	
2CEC-CP210	CH-P210	DISC STORAGE	
2CEC-CP212	CH-P212	ANALOG INPUT CABINET, BOP NO.1	
2CEC-CP213	CH-P213	ANALOG INPUT CABINET, BOP NO.2	
2CEC-CP214	CH-P214	ANALOG INPUT CABINET, BOP NO.3	
2CEC-CP215	CH-P215	ANALOG INPUT CABINET, BOP NO.4	
2CEC-CP216	CH-P216	ANALOG INPUT CABINET, BOP NO.5	
2CEC-CP218	CH-P218	ANALOG INPUT CABINET, BOP NO.6	
2CEC-CP219	CH-P219	DIGITAL IO CABINET, BOP NO.1	
2CEC-CP220	CH-P220	DIGITAL IO CABINET, BOP NO.2	
2CEC-CP221	CH-P221	DIGITAL IO CABINET, BOP NO.3	
2CEC-CP222	CH-P222	DIGITAL IO CABINET, BOP NO.4	
2CEC-CP223	CH-P223	DIGITAL IO CABINET, BOP NO.5	
2CEC-CP224	CH-P224	DIGITAL IO CABINET, BOP NO.6	
2CEC-CP225	CH-P225	DIGITAL IO CABINET NSS SYSTEM	
2CEC-CP226	CH-P226	I/O PRINTER/TYPER NO.1	
2CEC-CP227	CH-P227	I/O PRINTER/TYPER NO.2	
2CEC-CP228	CH-P228		
2CEC-CP229	CH-P229		
2CEC-CP230	CH-P230	I/O PRINTER/TYPER NO.1	
2CEC-CP230	CH-P230A	CARD READER	
2CEC-CP231	CH-P231	DISPLAY GENERATION CABINET	
2CEC-DSPL 001	CH-P231	CALIBRATION CRT	
2CEC-DSPL 002	CH-P231A	RESULTS CENTER CRT	
2CEC-KYBD001	CH-P231B	RESULTS CENTER KEYBOARD	
2CEC-KYBD002	CH-P231C	CALIBRATION KEYBOARD	
2CEC-CP232	CH-P232	NSS COMPUTER ROOM CABINET NO.1	
2CEC-CP233	CH-P233	NSS COMPUTER ROOM CABINET NO.2	
2CEC-CP234	CH-P234	NSS COMPUTER ROOM CABINET NO.3	
2CEC-CP235	CH-P235	NSS COMPUTER ROOM CABINET NO.4	
2CEC-CP236	CH-P236	NSS COMPUTER ROOM CABINET NO.5	
2CEC-CP237	CH-P237	NSS COMPUTER ROOM CABINET NO.6	
2CEC-CP238	CH-P238	NSS TECHNICAL CONSOLE	
2CEC-CP239	CH-P239	NSS MODEM TERMINAL	
2CEC-CP240	CH-P240	NSS MODEM CONSOLE	
2CEC-CP241	CH-P241		
2CEC-CP242	CH-P242		
2CEC-CP243	CH-P243		
2CEC-CP244	CH-P244	CONNECTOR WALL	
2CEC-CP245	CH-P245	10 TYPE ECC-1	
2CEC-CP246	CH-P246	10 TYPE ECC-2	
2CEC-CP247	CH-P247	MAGNETIC TAPE DRIVE CABINET	
2CEC-CP408	CH-P408	DIGITAL I/O CABINET BOP NO.6	
2CEC-CP410	CH-P410	DIGITAL I/O CABINET BOP NO.7	
2CEC-CP411	CH-P411	DIGITAL I/O CABINET BOP NO.8	
2CEC-CP418	CH-P418	ANALOG/DIGITAL I/O CABINET NSS SYSTEM	
2CEC-CP427	CH-P427	DIGITAL I/O CABINET BOP NO.9	
2CEC-CP428	CH-P428	DIGITAL I/O CABINET BOP NO.10	
2CEC-CP436	CH-P436	ELECTRIC BAR GRAPH (NOT NO.7)	
2CEC-CP437	CH-P437	SWITCHING CABINET	
2CEC-CP440	CH-P440	DIGITAL I/O CABINET NO.8	
2CEC-CP888	———	TERMINET REMOTE INTERFACE MODULE (TRIM) COMPUTER	
2CEC-DSPL808	———	TERMINET REMOTE INTERFACE MODULE (TRIM) CRT	
2CEC-KYBD808	———	TERMINET REMOTE INTERFACE MODULE (TRIM) KEYBOARD	
2CEC-SW881	———	TERMINET REMOTE INTERFACE MODULE (TRIM) CROSSOVER SWITCH	
2CEC-MUX808	———	TERMINET REMOTE INTERFACE MODULE (TRIM) MODEM SHARING DEVICE	
2CEC-MUX51	———	TERMINET REMOTE INTERFACE MODULE (TRIM) MODEM SHARING DEVICE	
2CEC-MUX52	———	TERMINET REMOTE INTERFACE MODULE (TRIM) MODEM	
2CEC-MUX53	———	TERMINET REMOTE INTERFACE MODULE (TRIM) MODEM	

NIAGARA MOHAWK DEVICE NUMBER	DESCRIPTION OF DEVICE	EQUIPMENT CLASSIFICATION
	3-D MONICORE SYSTEM	
2CEC-UA870	RECEIVER FOR PRINTERS	NE
2CEC-UA871	TRANSCEIVER (SPARE) } LOCATED	
2CEC-UA872	TRANSCEIVER (SPARE) } IN	
2CEC-UA873	TRANSCEIVER FOR UA874 } FALSE	
2CEC-UA875	TRANSCEIVER FOR UA876 } FLOOR	
2CEC-UA878	RECEIVER FOR TELLABS	
2CEC-PTR879	COMPUTER ROOM PRINTER LGN	
2CEC-CP880	SPARE DISC DRIVE	
2CEC-CP881	DEC ALPHA WORKSTATION (APOLLO)	
2CEC-DSPL881	DISPLAY FOR CP880	
2CEC-KYBD881	KEYBOARD FOR CP880	
2CEC-CP882	TAPE BACKUP DRIVE TL299-DS FOR CP880	
2CEC-CP884	MODEM FOR CP880	
2CEC-CP885	DEC ALPHA WORKSTATION (SPARE)	
2CEC-DSPL881	DISPLAY FOR CP881	
2CEC-KYBD881	KEYBOARD FOR CP881	
2CEC-CP883	TAPE BACKUP DRIVE TL299-DS FOR CP881	
2CEC-CP886	MODEM FOR CP881	
2CEC-UA886	TRANSCEIVER FOR CP880	
2CEC-UA881	TRANSCEIVER FOR CP881	
2CEC-SW882	A/B CROSSOVER SWITCH	
2CEC-SW883	A/B CROSSOVER SWITCH	

SOURCE: EE-27B

FIGURE 1.2-15

GENERAL ARRANGEMENT
CONTROL BUILDING
RELAY & COMPUTER ROOMS EL.288'-6"
SHEET 2 OF 4

NIAGARA MOHAWK POWER CORPORATION
NINE MILE POINT-UNIT 2
UPDATED SAFETY ANALYSIS REPORT

TABLE 1.3-9 (Cont'd.)

Item	Change	Reason for Change	FSAR Reference
PSI of main steam lines	Up to turbine stop valves, including branch connection lines in main steam, main steam bypass lines 2 1/2-in diameter and larger, and up to and including the first stop valve in each line were examined in accordance with ASME Section XI, 1980 Edition, Winter 1980 Addenda, in lieu of that described in PSAR Section H.2.26.	To be consistent with overall in-service inspection (ISI) program.	5.2.4.8