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SUBJECT: Forwards Rev 9 to NMP USAR & annual SE Summary Rept per 10CFR50.71(e) & 10CFR50.59.

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RICHARD B. ABBOTT
Vice President and
General Manager – Nuclear

May 2, 1997
NMP2L 1703

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 9 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***

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 9 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 9, 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 8 in November 1995. 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 UFSAR (Updated) Revision 14, dated June 1996, 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.

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.

A053

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None of the Safety Evaluations 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 and General Manager - Nuclear

RBA/JJL/lmc
Enclosures

pc: Mr. H. J. Miller, Regional Administrator, Region I
Mr. D. S. Hood, Senior Project Manager, NRR
Mr. S. S. Bajwa, Acting Director, Project Directorate I-1, NRR
Mr. B. S. Norris, Senior Resident Inspector
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 1)	)	

CERTIFICATION

Richard B. Abbott, being duly sworn, states that he is Vice President and General Manager - Nuclear 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 and General Manager - Nuclear

Subscribed and sworn to before me
this 2 day of May, 1997.

Beverly W. Ripka
Notary Public

BEVERLY W. RIPKA
Notary Public State of New York
Qual. in Oswego Co. No. 4644879
My Commission Exp. ~~Mar. 30,19...~~ 2/28/98

ENCLOSURE A

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 2 USAR 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, third paragraph	Changed "Executive Vice President Nuclear" to "Chief Nuclear Officer"	Reorganization.	Reorganization approved by the NRC via Unit 1 License Amendment 157 and Unit 2 License Amendment 71, dated February 20, 1996.
Page B.1-1, Section B.1.1, first paragraph	Replaced "contractors and consultants" with "suppliers"	Editorial. NMPC uses the term "suppliers" as a synonym of "contractors" and "consultants," and prefers the term "suppliers."	The use of an all-encompassing term (i.e., using "suppliers" to include or describe contractors, consultants, or vendors) does not affect compliance with 10CFR50 Appendix B.
Page B.1-1, Section B.1.1, second paragraph	Changed "Each organizational department, including Nuclear Generation, Nuclear Engineering, and Nuclear Safety Assessment and Support (NSAS), is responsible for the quality of its own work." to read "Each organizational department is responsible for the quality of its own work."	Editorial.	Editorial. N/A
Page B.1-1, Section B.1.2.1	a. Changed "is delegated by the President to corporate officers, as described herein" to read "is delegated by the President to corporate officers and the Manager Quality Assurance, as described herein" b. Changed "Figure 13.1-1a" to read "Figure 13.1-1"	a. Editorial. To reflect that the authority and responsibility of the Manager Quality Assurance is also described. b. Editorial.	a. Editorial. N/A b. Editorial. N/A

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. Changed "Executive Vice President Nuclear" to "Chief Nuclear Officer" b. Changed "...including all functions performed by Nuclear Generation, Nuclear Engineering, Nuclear Safety Assessment and Support, Nuclear Controller, ..." to read "...including the Plant Generation and Engineering Functions under the Vice President and General Manager - Nuclear, Nuclear Safety Assessment and Support (NSAS), Business Management,..."	a. Reorganization. b. Reorganization.	a. Reorganization approved by NRC via Unit 1 License Amendment 157 and Unit 2 License Amendment 71, dated February 20, 1996. b. Same as Item a.
Page B.1-2, Section B.1.2.1.1, second paragraph	Changed "Controller Nuclear Division" to "General Manager Business Management"	Reorganization. Title change.	Position title change to reflect management of business computers and finance/accounting activities under the position of General Manager Business Management. This is an administrative management position which does not perform QA related activities.

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, Item 1	Changed "The Vice President Nuclear Generation reports to the Executive Vice President Nuclear, and is responsible for safe and efficient operation, maintenance, and modification of the Station in compliance with Station licenses, applicable regulations, and the QA Program. The Vice President Nuclear Generation delegates to the Plant Managers and other appropriate personnel authority for performance in accordance with the QA Program. See Table B-1 for QA Program element responsibilities. Activities performed under the responsibility of the Vice President Nuclear Generation include:" to read "The Vice President and General Manager - Nuclear reports to the Chief Nuclear Officer, and has the overall divisional responsibility for plant operation and engineering. The Vice President Nuclear Engineering, Plant Managers, and the General Supervisor Labor Relations report directly to this Vice President. See Table B-1 for QA Program element responsibilities. Activities performed under the responsibility of the Vice President and General Manager - Nuclear include:"	Corporate management reorganization.	Reorganization approved by NRC via Unit 1 License Amendment 157 and Unit 2 License Amendment 71, dated February 20, 1996.

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.1-3, Item 2	Changed Item 2 to read: "Responsibilities and duties of the Vice President Nuclear Engineering and the Nuclear Engineering organization are described in Unit 1 UFSAR Section XIII.A.1 and Unit 2 USAR Section 13.1.1. See Table B-1 for QA Program element responsibilities."	Corporate management reorganization.	Reorganization approved by NRC via Unit 1 License Amendment 157 and Unit 2 License Amendment 71, dated February 20, 1996.
Page B.1-3, Item 3	Changed Item 3 to read: "The Vice President Nuclear Safety Assessment and Support reports to the Chief Nuclear Officer and is responsible for Quality Assurance, Licensing, Training/Emergency Preparedness, Security, and the Unit 2 Independent Safety Engineering Group (ISEG). See Table B-1 for QA Program element responsibilities. Nuclear Safety Assessment and Support responsibilities are described in Unit 1 UFSAR Section XIII and Unit 2 USAR Section 13."	Corporate management reorganization.	Reorganization approved by NRC via Unit 1 License Amendment 157 and Unit 2 License Amendment 71, dated February 20, 1996.
Page B.1-3, Item 4, first paragraph	Changed "Executive Vice President Nuclear" to read "Chief Nuclear Officer"	Corporate management reorganization.	Reorganization approved by NRC via Unit 1 License Amendment 157 and Unit 2 License Amendment 71, dated February 20, 1996.

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.1-3, Item 4, second paragraph	a. Changed "Tasks performed to fulfill these responsibilities include" to read "Tasks performed to fulfill these responsibilities are delineated in site procedures and include"	a. Editorial.	a. Editorial. N/A
	b. Combined Inspections and NDE Examinations as one task and removed the following identified tasks: • Coordinating and Reporting Internal and External QA Assessments • Operations Experience Assessment • Administering the Evaluation and Corrective Action Program for Deviation Event Reports (DERs) • DER Trend Analysis • Preparing and Processing QA Organization Documents	b. Editorial. Many of these tasks are also described under the responsibilities of supervisors or in other sections of the QATR. Administering the Evaluation and Corrective Action Program for Deviation/Event Reports (DERs) is the responsibility of the Plant Managers.	b. Editorial. N/A
	c. Added the following tasks: • Records Management • Document Control	c. Reorganization.	c. Reorganization approved by NRC via letter dated July 13, 1995.

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.1-4, Item 4.b	Changed "...determining applicability of industry and in-plant experience" to read "...assessments determining applicability of industry and in-plant operating experience"	Editorial clarification. Incorporates the term "assessments" associated with operations experience assessments into supervisor responsibilities.	Editorial. N/A
Page B.1-4, Item 4.d	Changed "...performing source surveillances of selected procurements" to read "...performing supplier evaluations and source surveillances of selected procurements"	Editorial clarification. Qualifies the type of activities that are done by the Procurement Quality Assurance Group.	Editorial. N/A
Page B.1-4, Item 4.e	Added Item 4.e to describe the responsibilities of the General Supervisor Quality Services	Reorganization established new position of Supervisor Quality Services which was later changed to General Supervisor Quality Services.	Reorganization approved by NRC via letter dated July 13, 1995. Title change is administrative in nature and does not affect position functions or responsibilities.
Page B.2-2, Section B.2.2.3, first paragraph	Changed "Executive Vice President Nuclear" to "Chief Nuclear Officer"	Reorganization title change.	Reorganization reviewed and approved by NRC via Unit 1 License Amendment 157 and Unit 2 License Amendment 71, dated February 20, 1996.

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, Item 1	Changed wording from "The Manager Quality Assurance is responsible for reporting on the status, adequacy and effectiveness of the NMPC QA Program through the Nuclear Division Internal SALP Type Assessment Reports" to read "The Chief Nuclear Officer is responsible for reporting on the status, adequacy and effectiveness of the NMPC QA Program".	Clarification and reorganization title change. Although the Manager Quality Assurance is responsible for reporting on the status, adequacy and effectiveness of the QA Program to the Chief Nuclear Officer, it is the Chief Nuclear Officer that reports to the President or Chief Executive Officer (CEO).	The Chief Nuclear Officer reports to the President of NMPC as described in Unit 1 License Amendment 157 and Unit 2 License Amendment 71, dated February 20, 1996.
Page B.2-5, Section B.2.2.15, Item 2	Changed "Executive Vice President Nuclear" to "Chief Nuclear Officer"	Reorganization title change.	Reorganization of corporate management approved by NRC via Unit 1 License Amendment 157 and Unit 2 License Amendment 71, dated February 20, 1996.
Page B.2-6, Section B.2.2.16	Changed wording from "The SRAB is a standing committee chaired by the Vice President Nuclear Engineering and reports to the Executive Vice President Nuclear regarding designated QA functions at the Nine Mile Point Nuclear Station" to read "The SRAB is a standing committee reporting to the Chief Nuclear Officer regarding designated QA functions at the Nine Mile Point Nuclear Station"	Clarification. To more clearly reflect Plant Administrative Technical Specifications.	The change more clearly reflects Plant Technical Specifications, Administrative Controls Section, and also reflects corporate management position title changes associated with Unit 1 License Amendment 157 and Unit 2 License Amendment 71.

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.2.2.17	Changed wording from "The SORC is an independent review committee responsible to the Vice President Nuclear Generation and transmits reports to the SRAB" to read "The SORC is an independent review committee responsible to the Plant Managers and transmits reports to the SRAB"	Clarification. To more clearly reflect Plant Administrative Technical Specifications.	The change more clearly reflects Plant Technical Specifications, Administrative Controls Section, and also reflects corporate management position title changes associated with Unit 1 License Amendment 157 and Unit 2 License Amendment 71.
Page B.2-6, Section B.2.2.18	Changed wording from "...and actions are verified by Q1P personnel prior to closeout" to read "...and the actions are verified prior to closeout"	Clarification. While Q1P personnel verify the overall closure of all items, other groups may be used to do some of the actual technical verifications for completeness.	The overall independence and confidentiality of Q1P have not changed. The technical ability of other departments is used to review some of the concerns.
Page B.4-2, Section B.4.2.7	Changed wording from "NQA or Procurement personnel other than the person who generated the procurement document, but qualified in QA,..." to read "Personnel other than the person who generated the procurement document, but with adequate understanding of the requirements and intent of the procurement documents,..."	Clarification.	There is no specific requirement for any particular group to perform these reviews, only that the individuals doing the review adequately understand the requirements and intent of the procurement documents. This is in accordance with NQA-1, 4S-1, section 3, which is our stated program for meeting 10CFR50 Appendix B. This does not constitute a reduction of commitment since whoever does the review function is required to be qualified. This qualification is accomplished through training.

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.5-1, Section B.5.2.6	Added Section B.5.2.6 to describe procedure review process	As an alternative to performing procedure reviews no less frequently than every two years to determine if changes are necessary or desirable (ANS-3.2). Niagara Mohawk has programmatic controls in place to continually identify procedure revisions which may be needed to ensure procedures are appropriate for the circumstances and are maintained current.	NRC approval per 10CFR50.54 granted via letter dated January 30, 1996.
Page B.7-1, Section B.7.2.2	Changed wording from "When contractors perform work under their own QA programs..." to read "When suppliers perform work under their own QA programs"	Editorial. NMPC uses the term "suppliers" as a synonym of "contractors," and prefers the term "suppliers."	The use of an all-encompassing term (i.e., using "suppliers" to include or describe contractors, consultants, or vendors) does not affect compliance with 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.7-1, Section B.7.2.3, Item 1	a. Changed wording from "...result in the supplier being placed on the Qualified Contractor List Database (QCLD) as a qualified vendor" to "...result in the supplier being placed on the Qualified Supplier List Database (QSLD)"	a. Editorial clarification. To reflect use of the term "supplier" rather than "contractor."	a. Editorial. N/A
	b. Changed wording from "...by virtue of this ability" to read "...by virtue of their ability"	b. Editorial.	b. Editorial. N/A
	c. Changed wording from "...characteristics identified by Nuclear Engineering and NQA" to read "...Characteristics identified by Nuclear Engineering"	c. Clarification. To fully reflect Nuclear Engineering responsibilities.	c. Nuclear Engineering is responsible for maintaining the design basis of systems, structures, and components and translates design requirements to suppliers which are deemed critical for a particular item/service. The identification of critical manufacturing and functional processes and characteristics by Nuclear Engineering continues to satisfy 10CFR50 Appendix B, Criterion 7.
	d. Changed wording from "...methods have been identified and documented by which NQA will verify conformance to these requirements" to read "...methods have been identified and documented which will verify conformance to these requirements"	d. Clarification.	d. NQA is involved with verification of supplier programs with attention to critical processes/characteristics selected, unless they can be verified onsite via test and/or inspection. These responsibilities have not changed and, therefore, continue to satisfy 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.7-2, Section B.7.2.3, Item 2	a. Changed wording from "NMPC-qualified suppliers involved in active procurement are surveyed every 3 yr to maintain..." to read "NMPC-qualified suppliers involved in active procurement are surveyed every 3 years* to maintain..." "*With a tolerance of one quarter of a year" b. Changed wording from "Supplier 3-yr surveys..." to read "Supplier 3-year surveys"	a. The change from 3 yr to 3 years is editorial. The addition of a note to reflect a tolerance of one quarter of a year is also editorial as this reflects Regulatory Guide 1.28, paragraph 3.2, as described in QATR Table B-3, sheet 1 of 8. b. Editorial.	a. Editorial. N/A b. Editorial. N/A
Page B.7-2, Section B.7.2.3, Item 3	Added Item 3 to identify suppliers/organizations that are not required to be evaluated or listed on the Qualified Supplier List Database (QSLD).	Clarification.	These statements clarify the use of the National Institute of Standards and Technology and other NRC licensed utilities that meet the requirements of 10CFR50 Appendix B.
Page B.7-3, Section B.7.2.6	Changed wording from "...purchased in accordance with Nuclear Engineering Procedures that provide..." to read "purchased in accordance with procedures that provide..."	Clarification. These controls are not limited to Nuclear Engineering procedures. Several types of procedures are used to make sure that design criteria is included in purchase requirements.	Although Engineering procedures provide controls to assure that items satisfy design requirements, these controls may also be found in Nuclear Interface Procedures or department procedures other than Engineering. These procedural controls continue to satisfy 10CFR50 Appendix B Criterion 7.
Page B.9-2, Section B.9.2.9	Changed wording from "...kept by vendors and/or forwarded to NMPC" to read "...kept by suppliers and/or forwarded to NMPC"	Editorial. NMPC uses the term "suppliers" as a synonym of "vendors," and prefers the term "suppliers."	The use of an all-encompassing term (i.e., using "suppliers" to include or describe contractors, consultants, or vendors) does not affect compliance with 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.10-1, Section B.10.1	Replaced policy statement with wording from NQA-1	Editorial clarification to reflect wording provided in NQA-1.	The restructuring of the paragraph to reflect NQA-1 is consistent with 10CFR50 Appendix B, Criterion 10. All areas continue to be reviewed except for the deletion of witness points. Witness points have either been upgraded to hold points or deleted because they were not needed.
Page B.10-1, Section B.10.2.2, Item 4	Changed from "Hold points and/or witness points" to read "Hold points"	Witness points are no longer used at Nine Mile Point. All witness points have been converted to hold points or deleted from NMPC procedures.	The removal of witness points continues to satisfy Appendix B, Criterion 10, since those witness points that were required have been upgraded to hold points.
Page B.10-2, Section B.10.2.5 B.10.2.6 B.10.2.7 B.10.2.8	Deleted previous Section B.10.2.5, which stated "Witness points require sufficient notification of the specifying organization prior to performance of the specified activity" and renumbered remaining sections accordingly.	Witness points are no longer used at Nine Mile Point. All witness points have been converted to hold points or deleted from NMPC procedures.	The removal of witness points continues to satisfy Appendix B, Criterion 10, since those witness points that were required have been upgraded to hold points.
Page B.10-3, Section B.10.2.9	Changed wording from "A program for inspection and surveillance of activities affecting fire protection is established..." to read "A program for inspection and surveillance, as required, for activities affecting fire protection is established..."	Editorial clarification for ease of reading and sentence structure.	Editorial. N/A
Page B.11-2, Section B.11.2.3, Item 5	Changed wording from "Any witness and hold points" to read "Any hold points"	Witness points are no longer used at Nine Mile Point. All witness points have been converted to hold points or deleted from NMPC procedures.	The removal of witness points continues to satisfy Appendix B, Criterion 10, since those witness points that were required have been upgraded to hold points.

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.18-1, Section B.18.1	Changed wording from "...including those elements of the program implemented by suppliers and contractors" to read "including those elements of the program implemented by suppliers"	Editorial. NMPC uses the term "suppliers" as a synonym of "contractors," and prefers the term "suppliers."	The use of an all-encompassing term (i.e., using "suppliers" to include or describe contractors, consultants, or vendors) does not affect compliance with 10CFR50 Appendix B.
Page B.18-1, Section B.18.2.3	Changed wording from "once every 2 yr" to read "once every 2 years"	Editorial.	Editorial. N/A

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, sheet 1 of 2	a. Under Procedures column, identified Quality Assurance (QA), Nuclear Licensing (NL), and Nuclear Training (NT) under the Vice President Nuclear Safety Assessment and Support (VP-NSAS), and identified Nuclear Engineering (NE) and Nuclear Generation (NG) under the Vice President and General Manager - Nuclear (VPGM-N).	a. Editorial, to reflect corporate management restructuring.	a. Editorial. N/A
	b. Removed Nuclear Procurement (NP) from the NSAS Procedures column to reflect transfer of the Nuclear Procurement function to Nuclear Engineering. Identified Nuclear Engineering as responsible for QA Program elements associated with Criterion IV, accordingly.	b. Reorganization.	b. The Nuclear Procurement organization was transferred from Nuclear Safety Assessment and Support (NSAS) to Nuclear Engineering. The duties, functions, and responsibilities of Nuclear Procurement have not been altered.
	c. Removed Technical Services (TS) and Information Management (IM) from the NSAS Procedures column.	c. Reorganization.	c. The duties, responsibilities, and functions performed by Technical Services (TS) and Information Management (IM) have been reassigned to other branches, as appropriate. The QA Program elements once implemented by TS and IM have been integrated into the appropriate branch and are identified on the responsibility matrix.

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, sheet 1 of 2 (cont'd.)	d. Identified Quality Assurance responsibility for QA Program elements associated with Criteria VI (Document Control) to reflect transfer of responsibility from Nuclear Engineering.	d. Reorganization.	d. Reorganization approved by NRC via letter dated July 13, 1995.
Table B-1, sheet 2 of 2	a. Removed Nuclear Procurement (NP), Technical Services (TS), and Information Management (IM), from under NSAS Procedures column and from listing of NMPC organizations.	a. Reorganization.	a. The function of Nuclear Procurement was transferred from the Nuclear Safety Assessment and Support organization to Nuclear Engineering. This transfer does not affect duties or functional responsibilities and, therefore, continues to satisfy 10CFR50 Appendix B criteria. The duties, responsibilities and functions of Technical Services and Information Management have been transferred to other organizations as appropriate.
	b. Identified Quality Assurance responsibility for QA Program elements associated with Criteria XVII (Quality Assurance Records) to reflect transfer of responsibility from Nuclear Engineering.	b. Reorganization.	b. Reorganization approved by NRC via letter dated July 13, 1995.
Table B-3, sheet 2 of 8	Changed Document column row "d" from Para. 4 to read "Section 4"	Editorial.	Editorial. N/A

NIAGARA MOHAWK
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RICHARD B. ABBOTT
Vice President and
General Manager – Nuclear

May 2, 1997
NMP2L 1703

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 9 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***

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Very truly yours,

Richard B. Abbott

Richard B. Abbott
Vice President and General Manager - Nuclear

RBA/JJL/lmc
Enclosures

pc: Mr. H. J. Miller, Regional Administrator, Region I
Mr. D. S. Hood, Senior Project Manager, NRR
Mr. S. S. Bajwa, Acting Director, Project Directorate I-1, NRR
Mr. B. S. Norris, Senior Resident Inspector
Records Management

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Niagara Mohawk Power Corporation	)	Docket No. 50-410
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(Nine Mile Point Nuclear Station Unit 1)	)	

CERTIFICATION

Richard B. Abbott, being duly sworn, states that he is Vice President and General Manager - Nuclear 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 and General Manager - Nuclear

Subscribed and sworn to before me
this 2 day of May, 1997.

Beverly W. Ripka
Notary Public

BEVERLY W. RIPKA
Notary Public State of New York
Qual. in Oswego Co. No. 4644879
My Commission Exp. ~~Mar. 30 19~~ 2/28/98

ENCLOSURE A

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 2 USAR 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, third paragraph	Changed "Executive Vice President Nuclear" to "Chief Nuclear Officer"	Reorganization.	Reorganization approved by the NRC via Unit 1 License Amendment 157 and Unit 2 License Amendment 71, dated February 20, 1996.
Page B.1-1, Section B.1.1, first paragraph	Replaced "contractors and consultants" with "suppliers"	Editorial. NMPC uses the term "suppliers" as a synonym of "contractors" and "consultants," and prefers the term "suppliers."	The use of an all-encompassing term (i.e., using "suppliers" to include or describe contractors, consultants, or vendors) does not affect compliance with 10CFR50 Appendix B.
Page B.1-1, Section B.1.1, second paragraph	Changed "Each organizational department, including Nuclear Generation, Nuclear Engineering, and Nuclear Safety Assessment and Support (NSAS), is responsible for the quality of its own work." to read "Each organizational department is responsible for the quality of its own work."	Editorial.	Editorial. N/A
Page B.1-1, Section B.1.2.1	a. Changed "is delegated by the President to corporate officers, as described herein" to read "is delegated by the President to corporate officers and the Manager Quality Assurance, as described herein" b. Changed "Figure 13.1-1a" to read "Figure 13.1-1"	a. Editorial. To reflect that the authority and responsibility of the Manager Quality Assurance is also described. b. Editorial.	a. Editorial. N/A b. Editorial. N/A

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. Changed "Executive Vice President Nuclear" to "Chief Nuclear Officer" b. Changed "...including all functions performed by Nuclear Generation, Nuclear Engineering, Nuclear Safety Assessment and Support, Nuclear Controller, ..." to read "...including the Plant Generation and Engineering Functions under the Vice President and General Manager - Nuclear, Nuclear Safety Assessment and Support (NSAS), Business Management,..."	a. Reorganization. b. Reorganization.	a. Reorganization approved by NRC via Unit 1 License Amendment 157 and Unit 2 License Amendment 71, dated February 20, 1996. b. Same as Item a.
Page B.1-2, Section B.1.2.1.1, second paragraph	Changed "Controller Nuclear Division" to "General Manager Business Management"	Reorganization. Title change.	Position title change to reflect management of business computers and finance/accounting activities under the position of General Manager Business Management. This is an administrative management position which does not perform QA related activities.

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, Item 1	Changed "The Vice President Nuclear Generation reports to the Executive Vice President Nuclear, and is responsible for safe and efficient operation, maintenance, and modification of the Station in compliance with Station licenses, applicable regulations, and the QA Program. The Vice President Nuclear Generation delegates to the Plant Managers and other appropriate personnel authority for performance in accordance with the QA Program. See Table B-1 for QA Program element responsibilities. Activities performed under the responsibility of the Vice President Nuclear Generation include:" to read "The Vice President and General Manager - Nuclear reports to the Chief Nuclear Officer, and has the overall divisional responsibility for plant operation and engineering. The Vice President Nuclear Engineering, Plant Managers, and the General Supervisor Labor Relations report directly to this Vice President. See Table B-1 for QA Program element responsibilities. Activities performed under the responsibility of the Vice President and General Manager - Nuclear include:"	Corporate management reorganization.	Reorganization approved by NRC via Unit 1 License Amendment 157 and Unit 2 License Amendment 71, dated February 20, 1996.

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.1-3, Item 2	Changed Item 2 to read: "Responsibilities and duties of the Vice President Nuclear Engineering and the Nuclear Engineering organization are described in Unit 1 UFSAR Section XIII.A.1 and Unit 2 USAR Section 13.1.1. See Table B-1 for QA Program element responsibilities."	Corporate management reorganization.	Reorganization approved by NRC via Unit 1 License Amendment 157 and Unit 2 License Amendment 71, dated February 20, 1996.
Page B.1-3, Item 3	Changed Item 3 to read: "The Vice President Nuclear Safety Assessment and Support reports to the Chief Nuclear Officer and is responsible for Quality Assurance, Licensing, Training/Emergency Preparedness, Security, and the Unit 2 Independent Safety Engineering Group (ISEG). See Table B-1 for QA Program element responsibilities. Nuclear Safety Assessment and Support responsibilities are described in Unit 1 UFSAR Section XIII and Unit 2 USAR Section 13."	Corporate management reorganization.	Reorganization approved by NRC via Unit 1 License Amendment 157 and Unit 2 License Amendment 71, dated February 20, 1996.
Page B.1-3, Item 4, first paragraph	Changed "Executive Vice President Nuclear" to read "Chief Nuclear Officer"	Corporate management reorganization.	Reorganization approved by NRC via Unit 1 License Amendment 157 and Unit 2 License Amendment 71, dated February 20, 1996.

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.1-3, Item 4, second paragraph	a. Changed "Tasks performed to fulfill these responsibilities include" to read "Tasks performed to fulfill these responsibilities are delineated in site procedures and include"	a. Editorial.	a. Editorial. N/A
	b. Combined Inspections and NDE Examinations as one task and removed the following identified tasks: • Coordinating and Reporting Internal and External QA Assessments • Operations Experience Assessment • Administering the Evaluation and Corrective Action Program for Deviation Event Reports (DERs) • DER Trend Analysis • Preparing and Processing QA Organization Documents	b. Editorial. Many of these tasks are also described under the responsibilities of supervisors or in other sections of the QATR. Administering the Evaluation and Corrective Action Program for Deviation/Event Reports (DERs) is the responsibility of the Plant Managers.	b. Editorial. N/A
	c. Added the following tasks: • Records Management • Document Control	c. Reorganization.	c. Reorganization approved by NRC via letter dated July 13, 1995.

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.1-4, Item 4.b	Changed "...determining applicability of industry and in-plant experience" to read "...assessments determining applicability of industry and in-plant operating experience"	Editorial clarification. Incorporates the term "assessments" associated with operations experience assessments into supervisor responsibilities.	Editorial. N/A
Page B.1-4, Item 4.d	Changed "...performing source surveillances of selected procurements" to read "...performing supplier evaluations and source surveillances of selected procurements"	Editorial clarification. Qualifies the type of activities that are done by the Procurement Quality Assurance Group.	Editorial. N/A
Page B.1-4, Item 4.e	Added Item 4.e to describe the responsibilities of the General Supervisor Quality Services	Reorganization established new position of Supervisor Quality Services which was later changed to General Supervisor Quality Services.	Reorganization approved by NRC via letter dated July 13, 1995. Title change is administrative in nature and does not affect position functions or responsibilities.
Page B.2-2, Section B.2.2.3, first paragraph	Changed "Executive Vice President Nuclear" to "Chief Nuclear Officer"	Reorganization title change.	Reorganization reviewed and approved by NRC via Unit 1 License Amendment 157 and Unit 2 License Amendment 71, dated February 20, 1996.

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, Item 1	Changed wording from "The Manager Quality Assurance is responsible for reporting on the status, adequacy and effectiveness of the NMPC QA Program through the Nuclear Division Internal SALP Type Assessment Reports" to read "The Chief Nuclear Officer is responsible for reporting on the status, adequacy and effectiveness of the NMPC QA Program"	Clarification and reorganization title change. Although the Manager Quality Assurance is responsible for reporting on the status, adequacy and effectiveness of the QA Program to the Chief Nuclear Officer, it is the Chief Nuclear Officer that reports to the President or Chief Executive Officer (CEO).	The Chief Nuclear Officer reports to the President of NMPC as described in Unit 1 License Amendment 157 and Unit 2 License Amendment 71, dated February 20, 1996.
Page B.2-5, Section B.2.2.15, Item 2	Changed "Executive Vice President Nuclear" to "Chief Nuclear Officer"	Reorganization title change.	Reorganization of corporate management approved by NRC via Unit 1 License Amendment 157 and Unit 2 License Amendment 71, dated February 20, 1996.
Page B.2-6, Section B.2.2.16	Changed wording from "The SRAB is a standing committee chaired by the Vice President Nuclear Engineering and reports to the Executive Vice President Nuclear regarding designated QA functions at the Nine Mile Point Nuclear Station" to read "The SRAB is a standing committee reporting to the Chief Nuclear Officer regarding designated QA functions at the Nine Mile Point Nuclear Station"	Clarification. To more clearly reflect Plant Administrative Technical Specifications.	The change more clearly reflects Plant Technical Specifications, Administrative Controls Section, and also reflects corporate management position title changes associated with Unit 1 License Amendment 157 and Unit 2 License Amendment 71.

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.2.2.17	Changed wording from "The SORC is an independent review committee responsible to the Vice President Nuclear Generation and transmits reports to the SRAB" to read "The SORC is an independent review committee responsible to the Plant Managers and transmits reports to the SRAB"	Clarification. To more clearly reflect Plant Administrative Technical Specifications.	The change more clearly reflects Plant Technical Specifications, Administrative Controls Section, and also reflects corporate management position title changes associated with Unit 1 License Amendment 157 and Unit 2 License Amendment 71.
Page B.2-6, Section B.2.2.18	Changed wording from "...and actions are verified by Q1P personnel prior to closeout" to read "...and the actions are verified prior to closeout"	Clarification. While Q1P personnel verify the overall closure of all items, other groups may be used to do some of the actual technical verifications for completeness.	The overall independence and confidentiality of Q1P have not changed. The technical ability of other departments is used to review some of the concerns.
Page B.4-2, Section B.4.2.7	Changed wording from "NQA or Procurement personnel other than the person who generated the procurement document, but qualified in QA,..." to read "Personnel other than the person who generated the procurement document, but with adequate understanding of the requirements and intent of the procurement documents,..."	Clarification.	There is no specific requirement for any particular group to perform these reviews, only that the individuals doing the review adequately understand the requirements and intent of the procurement documents. This is in accordance with NQA-1, 4S-1, section 3, which is our stated program for meeting 10CFR50 Appendix B. This does not constitute a reduction of commitment since whoever does the review function is required to be qualified. This qualification is accomplished through training.

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.5-1, Section B.5.2.6	Added Section B.5.2.6 to describe procedure review process	As an alternative to performing procedure reviews no less frequently than every two years to determine if changes are necessary or desirable (ANS-3.2). Niagara Mohawk has programmatic controls in place to continually identify procedure revisions which may be needed to ensure procedures are appropriate for the circumstances and are maintained current.	NRC approval per 10CFR50.54 granted via letter dated January 30, 1996.
Page B.7-1, Section B.7.2.2	Changed wording from "When contractors perform work under their own QA programs..." to read "When suppliers perform work under their own QA programs"	Editorial. NMPC uses the term "suppliers" as a synonym of "contractors," and prefers the term "suppliers."	The use of an all-encompassing term (i.e., using "suppliers" to include or describe contractors, consultants, or vendors) does not affect compliance with 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.7-1, Section B.7.2.3, Item 1	a. Changed wording from "...result in the supplier being placed on the Qualified Contractor List Database (QCLD) as a qualified vendor" to "...result in the supplier being placed on the Qualified Supplier List Database (QSLD)"	a. Editorial clarification. To reflect use of the term "supplier" rather than "contractor."	a. Editorial. N/A
	b. Changed wording from "...by virtue of this ability" to read "...by virtue of their ability"	b. Editorial.	b. Editorial. N/A
	c. Changed wording from "...characteristics identified by Nuclear Engineering and NQA" to read "...Characteristics identified by Nuclear Engineering"	c. Clarification. To fully reflect Nuclear Engineering responsibilities.	c. Nuclear Engineering is responsible for maintaining the design basis of systems, structures, and components and translates design requirements to suppliers which are deemed critical for a particular item/service. The identification of critical manufacturing and functional processes and characteristics by Nuclear Engineering continues to satisfy 10CFR50 Appendix B, Criterion 7.
	d. Changed wording from "...methods have been identified and documented by which NQA will verify conformance to these requirements" to read "...methods have been identified and documented which will verify conformance to these requirements"	d. Clarification.	d. NQA is involved with verification of supplier programs with attention to critical processes/characteristics selected, unless they can be verified onsite via test and/or inspection. These responsibilities have not changed and, therefore, continue to satisfy 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.7-2, Section B.7.2.3, Item 2	a. Changed wording from "NMPC-qualified suppliers involved in active procurement are surveyed every 3 yr to maintain..." to read "NMPC-qualified suppliers involved in active procurement are surveyed every 3 years* to maintain..." "*With a tolerance of one quarter of a year" b. Changed wording from "Supplier 3-yr surveys..." to read "Supplier 3-year surveys"	a. The change from 3 yr to 3 years is editorial. The addition of a note to reflect a tolerance of one quarter of a year is also editorial as this reflects Regulatory Guide 1.28, paragraph 3.2, as described in QATR Table B-3, sheet 1 of 8. b. Editorial.	a. Editorial. N/A b. Editorial. N/A
Page B.7-2, Section B.7.2.3, Item 3	Added Item 3 to identify suppliers/organizations that are not required to be evaluated or listed on the Qualified Supplier List Database (QSLD).	Clarification.	These statements clarify the use of the National Institute of Standards and Technology and other NRC licensed utilities that meet the requirements of 10CFR50 Appendix B.
Page B.7-3, Section B.7.2.6	Changed wording from "...purchased in accordance with Nuclear Engineering Procedures that provide..." to read "purchased in accordance with procedures that provide..."	Clarification. These controls are not limited to Nuclear Engineering procedures. Several types of procedures are used to make sure that design criteria is included in purchase requirements.	Although Engineering procedures provide controls to assure that items satisfy design requirements, these controls may also be found in Nuclear Interface Procedures or department procedures other than Engineering. These procedural controls continue to satisfy 10CFR50 Appendix B Criterion 7.
Page B.9-2, Section B.9.2.9	Changed wording from "...kept by vendors and/or forwarded to NMPC" to read "...kept by suppliers and/or forwarded to NMPC"	Editorial. NMPC uses the term "suppliers" as a synonym of "vendors," and prefers the term "suppliers."	The use of an all-encompassing term (i.e., using "suppliers" to include or describe contractors, consultants, or vendors) does not affect compliance with 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.10-1, Section B.10.1	Replaced policy statement with wording from NQA-1	Editorial clarification to reflect wording provided in NQA-1.	The restructuring of the paragraph to reflect NQA-1 is consistent with 10CFR50 Appendix B, Criterion 10. All areas continue to be reviewed except for the deletion of witness points. Witness points have either been upgraded to hold points or deleted because they were not needed.
Page B.10-1, Section B.10.2.2, Item 4	Changed from "Hold points and/or witness points" to read "Hold points"	Witness points are no longer used at Nine Mile Point. All witness points have been converted to hold points or deleted from NMPC procedures.	The removal of witness points continues to satisfy Appendix B, Criterion 10, since those witness points that were required have been upgraded to hold points.
Page B.10-2, Section B.10.2.5 B.10.2.6 B.10.2.7 B.10.2.8	Deleted previous Section B.10.2.5, which stated "Witness points require sufficient notification of the specifying organization prior to performance of the specified activity" and renumbered remaining sections accordingly.	Witness points are no longer used at Nine Mile Point. All witness points have been converted to hold points or deleted from NMPC procedures.	The removal of witness points continues to satisfy Appendix B, Criterion 10, since those witness points that were required have been upgraded to hold points.
Page B.10-3, Section B.10.2.9	Changed wording from "A program for inspection and surveillance of activities affecting fire protection is established..." to read "A program for inspection and surveillance, as required, for activities affecting fire protection is established..."	Editorial clarification for ease of reading and sentence structure.	Editorial. N/A
Page B.11-2, Section B.11.2.3, Item 5	Changed wording from "Any witness and hold points" to read "Any hold points"	Witness points are no longer used at Nine Mile Point. All witness points have been converted to hold points or deleted from NMPC procedures.	The removal of witness points continues to satisfy Appendix B, Criterion 10, since those witness points that were required have been upgraded to hold points.

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.18-1, Section B.18.1	Changed wording from "...including those elements of the program implemented by suppliers and contractors" to read "including those elements of the program implemented by suppliers"	Editorial. NMPC uses the term "suppliers" as a synonym of "contractors," and prefers the term "suppliers."	The use of an all-encompassing term (i.e., using "suppliers" to include or describe contractors, consultants, or vendors) does not affect compliance with 10CFR50 Appendix B.
Page B.18-1, Section B.18.2.3	Changed wording from "once every 2 yr" to read "once every 2 years"	Editorial.	Editorial. N/A

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, sheet 1 of 2	a. Under Procedures column, identified Quality Assurance (QA), Nuclear Licensing (NL), and Nuclear Training (NT) under the Vice President Nuclear Safety Assessment and Support (VP-NSAS), and identified Nuclear Engineering (NE) and Nuclear Generation (NG) under the Vice President and General Manager - Nuclear (VPGM-N).	a. Editorial, to reflect corporate management restructuring.	a. Editorial. N/A
	b. Removed Nuclear Procurement (NP) from the NSAS Procedures column to reflect transfer of the Nuclear Procurement function to Nuclear Engineering. Identified Nuclear Engineering as responsible for QA Program elements associated with Criterion IV, accordingly.	b. Reorganization.	b. The Nuclear Procurement organization was transferred from Nuclear Safety Assessment and Support (NSAS) to Nuclear Engineering. The duties, functions, and responsibilities of Nuclear Procurement have not been altered.
	c. Removed Technical Services (TS) and Information Management (IM) from the NSAS Procedures column.	c. Reorganization.	c. The duties, responsibilities, and functions performed by Technical Services (TS) and Information Management (IM) have been reassigned to other branches, as appropriate. The QA Program elements once implemented by TS and IM have been integrated into the appropriate branch and are identified on the responsibility matrix.

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, sheet 1 of 2 (cont'd.)	d. Identified Quality Assurance responsibility for QA Program elements associated with Criteria VI (Document Control) to reflect transfer of responsibility from Nuclear Engineering.	d. Reorganization.	d. Reorganization approved by NRC via letter dated July 13, 1995.
Table B-1, sheet 2 of 2	a. Removed Nuclear Procurement (NP), Technical Services (TS), and Information Management (IM), from under NSAS Procedures column and from listing of NMPC organizations. b. Identified Quality Assurance responsibility for QA Program elements associated with Criteria XVII (Quality Assurance Records) to reflect transfer of responsibility from Nuclear Engineering.	a. Reorganization. b. Reorganization.	a. The function of Nuclear Procurement was transferred from the Nuclear Safety Assessment and Support organization to Nuclear Engineering. This transfer does not affect duties or functional responsibilities and, therefore, continues to satisfy 10CFR50 Appendix B criteria. The duties, responsibilities and functions of Technical Services and Information Management have been transferred to other organizations as appropriate. b. Reorganization approved by NRC via letter dated July 13, 1995.
Table B-3, sheet 2 of 8	Changed Document column row "d" from Para. 4 to read "Section 4"	Editorial.	Editorial. N/A

Enclosure B to
NMP2L 1703

NINE MILE POINT - UNIT 2

SAFETY EVALUATION SUMMARY REPORT

1997

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

Safety Evaluation No.: 89-024 Rev. 2

Implementation Document No.: Procedure N2-OP-55

USAR Affected Pages: 9.4-41, 9.4-42

System: Turbine Building Ventilation (HVT)

Title of Change: Operating with the Turbine Building Roof Vents Open

Description of Change:

Under Revision 1 of Safety Evaluation 89-024, operation with the Turbine Building roof vents open was not permitted unless the main steam tunnel lead enclosure temperature approached setpoint limits. This revision of the safety evaluation evaluates operation of the roof vents seasonally to provide general Turbine Building cooling.

Safety Evaluation Summary:

Opening the Turbine Building roof vents will provide additional building cooling which will have the effect of relieving the stresses of elevated temperature on both personnel and equipment. Providing an environment that ensures habitability within the temperature limits shown in USAR Table 9.4-1 will be enhanced. Because the roof vents will be opened to provide cooling, operation with the vents opened will not adversely impact Turbine Building equipment by exposure to severe winter conditions.

Opening the Turbine Building roof vents will not impact the direction of ventilation flow within ducts, and so the movement of air from clean areas to areas of progressively greater potential contamination prior to final exhaust will be maintained.

The Turbine Building supply air subsystem is balanced to supply slightly less air than is exhausted, thereby maintaining a subatmospheric pressure and inhibiting the exfiltration of air from the building. Because the overall building is maintained at subatmospheric pressures, when the roof vents are opened, cooler outside air will be drawn in and through Turbine Building ventilation system. The roof vents located over the hottest Turbine Building areas may have some outflow.

The Turbine Building ventilation system has no safety-related function, and the failure or malfunction of the system will not compromise any safety-related system or component or prevent safe reactor shutdown. The Turbine Building is not

Safety Evaluation No.: 89-024 Rev. 2 (cont'd.)

Safety Evaluation Summary: (cont'd.)

classified as a containment structure; however, the ventilation system is designed to exhaust more air from the building than is being supplied, thereby maintaining subatmospheric pressure to inhibit exfiltration. These criteria are not changed by operating with the roof vents open.

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

Safety Evaluation No.: 90-311 Rev. 0 & 1

Implementation Document No.: Mod. PN2Y88MX133

USAR Affected Pages: Figure 11.4-1h

System: Material Handling - Radwaste

Title of Change: Radwaste CCTV Camera Replacement

Description of Change:

The original CCTV system installed in the radwaste building required extensive repairs and maintenance. The original equipment became obsolete and spare/repair parts were not available without special tooling by the manufacturer. The original CCTV system was replaced with a new system. Revision 0 also evaluated the addition of a boom-mounted camera/channel to be installed in the truck bay. However, after the original camera system was replaced, it was determined that the additional boom-mounted camera was not needed. Revision 1 deletes the evaluation for an additional camera.

Safety Evaluation Summary:

The CCTV system in the radwaste building is passive in nature. This system allows remote monitoring of process handling from a central location. The radwaste CCTV system is nonsafety related and there are no seismic or environmental requirements for the installation. The new equipment is installed to the same standards as the original equipment.

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

Safety Evaluation No.: 92-075

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

USAR Affected Pages: Figure 9.2-1c

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*CAB23A 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.

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.: 93-025 Rev. 2

Implementation Document No.: Calculation HVC-074 Rev. 1

USAR Affected Pages: N/A

System: N/A

Title of Change: Resolution of DER 2-93-0032

Description of Change:

When an air conditioning unit (Control Room or Relay Room) is put back into service (to declare the unit operable) after undergoing preplanned maintenance or repair, the operating air conditioning unit is put in Pull-to-Lock (PTL) and the other unit is allowed to auto start. During this process, the unit which has undergone maintenance or repair is declared inoperable. When the operating unit is in the PTL mode, it is procedurally declared inoperable per Operating Procedure N2-OP-53A. The net result is an entry into Limiting Condition of Operation (LCO) 3.0.3 (one hour to commence reactor shutdown), as described in DER 2-93-0032.

The purpose of this safety evaluation was: 1) to show that placing an air conditioning unit in PTL does not necessarily make the unit inoperable as long as an Operator is standing by to remove the control switch from PTL, and 2) to stipulate that when placing a running unit in PTL to start the standby unit or post-maintenance test (PMT), the unit under maintenance or repair is acceptable provided an air conditioning unit is restored to operating mode prior to exceeding the design temperature.

Revision 1 of this safety evaluation determined that 10 minutes was sufficient to restore the PTL air conditioning unit to operating mode prior to exceeding the design temperature. However, Calculation HVC-074 was revised (Revision 1) to show that the time interval to restore the PTL air conditioning unit is 5 minutes instead of 10 minutes. Therefore, this safety evaluation was revised to reflect the time interval of 5 minutes to restore the PTL air conditioning unit prior to exceeding the maximum design temperature.

Safety Evaluation Summary:

An air conditioning unit is not considered inoperable when in PTL as long as an Operator is standing by to remove the control switch from PTL. With an Operator standing by, the unit can be immediately returned to service to provide its intended safety function.

Safety Evaluation No.: 93-025 Rev. 2 (cont'd.)

Safety Evaluation Summary: (cont'd.)

During normal plant operation, the Control and Relay Rooms are maintained at 75°F. The design temperature of the components in these rooms is 90°F or higher. Calculation HVC-074 calculates the time interval required to reach 90°F with both air conditioning units out-of-service. This time interval will allow starting of the standby air conditioning unit or PMT of the air conditioning unit under maintenance/repair without entering LCO 3.0.3.

Calculation HVC-074 Rev. 1 determines that it takes 5 minutes for room air temperature to reach 90°F, assuming initial temperature is 75°F. If cooling is restored within 5 minutes after putting an operating air conditioning unit in PTL, the operability of any component in the Control or Relay Room is not affected.

Therefore, it is concluded that putting an otherwise operable air conditioning unit in PTL does not make the unit inoperable as long as an Operator is standing by to place the unit out of PTL. Also, both air conditioning units may be placed in PTL as long as two conditions are met: 1) the cooling can be restored within 5 minutes, and 2) Control or Relay Room temperature prior to this event is 75°F or less.

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

Safety Evaluation No.:	94-001
Implementation Document No.:	Simple Design Change SC2-0255-91 (EDC 2F01090)
USAR Affected Pages:	Figure 9.2-5a
System:	Makeup Water Treating (WTS)
Title of Change:	Ecolochem Filtered and Purge Water Connections

Description of Change:

This simple design change added permanent connections to the existing WTS system piping to enable continued use of the Ecolochem portable demineralized trailer. Previously, EDC 2F00915 provided a connection for the purge water from the Ecolochem to the makeup waste neutralizing tank (2WTS-TK1). This change made that connection permanent as installed. In addition, a new connection was installed from the water treating filter drain line, 2-WTS-002-134-4, to supply the Ecolochem trailer. Makeup water from the Ecolochem demineralized trailer is controlled in accordance with procedure N2-OP-15.

Safety Evaluation Summary:

This change is specific to the WTS system; therefore, no other systems and/or interlocks are affected. Installing additional connections to facilitate the Ecolochem demineralized water process will improve the system performance without causing any safety or operability issues. All work involved with this change will take place in the Screenwell Building and will not involve radioactive components nor the potential for high radiation.

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

Safety Evaluation No.:	94-006
Implementation Document No.:	EDC 2M10658A
USAR Affected Pages:	Figures 9.2-17c, 9.2-19d, 9.3-1b, 10.1-8c, 11.2-1a through 11.2-1h, 11.2-1L, 11.4-1a through 11.4-1h
System:	Solid Radwaste
Title of Change:	Abandonment In-Place of Asphalt Solidification Equipment

Description of Change:

This change abandoned in-place selected portions of the original asphalt-based solid radwaste processing system.

Safety Evaluation Summary:

The original plant design for radwaste solidification (i.e., removal of free water from miscellaneous wet wastes) utilized the Werner & Pfleiderer (WasteChem) asphalt volume reduction system addressed by Topical Reports WPC-VRS-001 and WPC-VRS-002. Due to various deficiencies, process problems, and offsite disposal facility burial criteria associated with the use of this system, the original asphalt-based solidification system was "abandoned in-place." The abandonment in-place of the asphalt-based solidification system will have minimal impact on radwaste processing, since a radwaste dewatering process providing an acceptable method of volume reduction utilizing methodology and equipment addressed in Chem Nuclear Systems, Inc., Topical Report RDS-25506-01-P/NP (reviewed and approved by the NRC) will be utilized. Abandonment in-place was accomplished in such a manner to assure proper pressure boundary confinement of all process applications.

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

Safety Evaluation No.: 94-019

Implementation Document No.: Temporary Mod. 94-021

USAR Affected Pages: N/A

System: Heater Drain (HDL)

Title of Change: Heater Drain Pump Mechanical Seal

Description of Change:

This temporary modification isolated the turbine building closed loop cooling system (CCS) from the jacket cooler of pump 2HDL-P1A. Due to an apparent leakage problem with the cooler, some of the heater drain water was mixed with CCS water causing radiological concerns.

The function of the jacket cooler is to provide intermediate cooling of the heater drains (via CCS) before it reaches the pump's mechanical seal. This feature was provided with the original mechanical seal design to protect the mechanical seal from potential degradation due to continuous exposure to high temperature fluid. The original mechanical seal was upgraded and the upgraded parts, including the elastomers, have higher temperature resistance. The implementation of this change removed this intermediate cooling feature by isolating the cooling medium (CCS).

Safety Evaluation Summary:

The new mechanical seal will continue to perform its intended function without the intermediate cooling feature. The highest temperature of the heater drains that may be experienced by the mechanical seal is approximately 330°F. All of the mechanical seal parts are qualified for this temperature. As a prudent measure, the mechanical seal will be monitored for temperature (<330°F) and leakage through pump startup. This change will have no effects on any equipment important to safe shutdown of the plant or maintaining the plant in a safe shutdown condition. This temporary change will eliminate the potential radiological contamination of the CCS water until the next available opportunity to replace the jacket cooler or to make this change permanent.

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

Safety Evaluation No.: 94-025 Rev. 3

Implementation Document No.: Simple Design Changes SC2-0276-91, SC2-0057-92

USAR Affected Pages: N/A

System: Reactor Water Cleanup (WCS)

Title of Change: WCS Filter/Demins. System Improvements and Addition of Operator Interface Workstation

Description of Change:

Simple Design Change (SDC) SC2-0276-91 replaced the existing programmable controller software that controls the operation of the four WCS filter/demineralizers with an updated, improved version. The basic system logic remains the same as before.

SDC SC2-0057-92 added a graphical Operator interface workstation near the existing WCS control panels that is connected to the four filters' programmable logic controllers through a "data highway" communications loop.

Safety Evaluation Summary:

The proposed changes to the control of the WCS filter/demineralizers will enhance system operation by allowing the Operator to better see and control filter backwash and precoat operations. The graphic display gives a live representation of the status of all valves and pumps and flow paths. Also, special functions such as system fill/vent or boron injection can be controlled from the Operator interface with proper password access. The software refinements are designed to keep the basic logic the same as before, but give more reliability and Operator flexibility. The software verification/validation has been performed in depth by Finetech, Inc., and NMPC personnel. Upon installing the changes in the field, the system will be tested first with the outputs disabled prior to actually operating the system components.

These changes are nonsafety related and will have no impact on the safe operation or shutdown of the plant. No changes to reactor water chemistry will occur. Installation and testing will be performed on a non-outage basis. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.:	94-038
Implementation Document No.:	Temporary Mod. 94-036
USAR Affected Pages:	N/A
System:	Condensate Demineralizer (CND)
Title of Change:	Defeat Discharge Pressure Switch for Low Conductivity Tank Waste Pump

Description of Change:

The discharge header pressure switch 2CND-PS331, which was wired in the control logic of the low conductivity waste pump, was removed electrically by this temporary modification. The pressure switch had two switches which were wired to a local junction instrument rack. The leads from the switches were lifted and taped. The waste pump is available for operation and continues to provide auto trip function on the low waste tank level signal.

Safety Evaluation Summary:

This temporary modification will disable the pressure switch located in the discharge header of the low conductivity waste pump. The manual mode of the waste pump operation and the auto mode from waste tank level switch will remain unaffected. This temporary modification will ensure that the CND system is available for operation in order to maintain the proper water chemistry limits as required by the Technical Specifications.

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

Safety Evaluation No.:	94-052 Rev. 0 & 1
Implementation Document No.:	Simple Design Change SC2-0053-93
USAR Affected Pages:	Figures 9.2-5e, 9.2-6a, 9.4-22b
System:	Hot Water and Glycol (HVH, HVG), Condenser Air Removal (ARC), Circulating Water (CWS), Control Building Chilled Water (HVK)
Title of Change:	Demineralized Water Makeup as Replacement for Raw Water

Description of Change:

The water treatment system (WTS) supplies filtered raw water as makeup to the following systems:

ARC system for the air removal pump separators
CWS pump seals
CWS system for the vacuum pump skid
HVH system for hot water heating
HVG system for glycol heating in the Turbine, Reactor and Radwaste Buildings
HVK system for Control Building chilled water

Piping in these systems, except for CWS pump seals, experienced blockage due to the formation of hard calcium scale caused by hardness in the raw water supply, and by buildup of corrosion product (tubercles) due to microbiologically influenced corrosion (MIC). Also, high organic carbons in WTS can have a detrimental effect on reactor water since some of this WTS water is processed by radwaste from ARC and CWS systems, which is then pumped to the condensate storage tank (CST) which is used for condensate makeup.

CWS pump seals piping was previously upgraded with stainless steel which inhibits tubercles, due to MIC and WTS makeup to the CWS pump seals which were not changed.

This simple design change supplied demineralized water from the makeup water (MWS) system to the systems listed above, except for CWS pump seals. This change is nonsafety related.

Safety Evaluation No.: 94-052 Rev. 0 & 1 (cont'd.)

Safety Evaluation Summary:

Raw water being used for the systems listed above contains amounts of calcium carbonate (CaCO3) that tends to plate out on the HVH system piping and heat exchanger tubes due to high temperatures, resulting in scaling which has made the systems difficult to maintain by reducing pressure and flow, and has made the heat exchanger tubes prone to elevated rates of corrosion.

This raw water also provides the environment for MIC that results in the formation of tubercles in the piping that reduces the flow area and, as such, reduces system pressure and flow. MIC can also produce sulfuric acid beneath the tubercles that can lead to through-wall leaks.

In addition, raw water contains undesirable levels of organic carbons that break down into organic acids in the presence of the neutron flux in the reactor vessel, which causes more acidic reactor water (lower pH). Radwaste supplies water to the CST, which is makeup for condensate water, and has a goal of reducing organic carbons to the CST.

Using demineralized water instead of raw water reduces the amount of calcium carbonate significantly, thus mitigating the prospect of scaling and underdeposit corrosion and prevents further tubercles growth, allowing required pressures and flows. This will provide enhanced makeup water quality and system performance, and also extend the service life of the piping.

Demineralized water also has significantly lower levels of organic carbons that will lower the level of acidity in the reactor water due to the breakdown of the organic carbons into organic acids by the neutron flux.

The function of the MWS system as described is nonsafety related. Using the MWS system as an alternate makeup source is consistent with the original design.

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

Safety Evaluation No.:	94-081 Rev. 0 & 1
Implementation Document No.:	Mod. PN2Y94MX0016
USAR Affected Pages:	Figures 1.2-19 Sh 2, 10.1-6b, 12.3-14, 12.3-47
System:	Feedwater (FWS), Plant Process Computer (CEC)
Title of Change:	Use of LEFM Feedwater Flow Indication For Calorimetric

Description of Change:

This safety evaluation assessed the impact of adding a leading edge flow meter (LEFM) to the plant computer that permits adjustment of the feedwater flow measurement used by the plant computer in the calculation of core thermal power. The adjustment is made by manually entering a correction factor that accounts for biases in the feedwater flow measurement instrument loops. These biases include those present due to venturi fouling, as well as those introduced during routine instrument loop calibration or resulting from instrument loop drift. The correction factor is determined by comparing the feedwater flow rate as indicated by the plant computer to the feedwater flow rate as indicated by the LEFM, a state of the art ultrasonic flow measurement system manufactured by Caldon.

By tuning the feedwater flow instrumentation to the LEFM, the plant computer may be able to more accurately determine feedwater flow; thus it will be able to more accurately determine core thermal power. This allows the plant to operate closer to its thermal power rating and potentially recover electrical output that has been lost due to degraded feedwater flow measurement instrumentation.

Safety Evaluation Summary:

Section 15.0.3.3.3 of the USAR assumes a 2% margin for total calorimetric error in determining the reactor core thermal power. GE Report NEDO-20340, "Process Computer Performance Evaluation Accuracy," calculates the venturi-based feedwater flow measurement uncertainty contribution to determine total core thermal power at ±1.76 percent. The uncertainty for determining feedwater flow by tuning to the LEFM has been calculated at ±1.72 percent. Therefore, feedwater flow accuracy is improved; thus the accuracy with which thermal power

Safety Evaluation No.: 94-081 Rev. 0 & 1 (cont'd.)

Safety Evaluation Summary: (cont'd.)

is determined is also improved and remains less than the 2% value required by Regulatory Guide 1.49 and considered in the NMP2 accident analysis.

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

Safety Evaluation No.: 94-084

Implementation Document No.: Procedure S-RAP-RPP-0402

USAR Affected Pages: 12.5-14

System: N/A

Title of Change: Respirator Issuance Based on TEDE

Description of Change:

This change updated the USAR to agree with the 10CFR20 requirement to consider Total Effective Dose Equivalent (TEDE) when determining whether respiratory devices should be worn. Since TEDE is the sum of exposure from both internally deposited radioactivity and external whole body exposure, the decision whether or not to wear a respirator in an area must now consider not only the airborne radioactivity present in an area (DAC) but also the whole body exposure rate present in that area.

Safety Evaluation Summary:

Inclusion of whole body exposure rate, when determining the necessity for wearing respiratory protection, does not increase the probability of occurrence of any accident previously evaluated in the USAR since the wearing of respiratory protective devices is simply a method to mitigate individual exposure.

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

Safety Evaluation No.:	94-085
Implementation Document No.:	EDC 2E10934
USAR Affected Pages:	Figures 5.4-13a, 6.3-6a, 6.3-7a
System:	Low Pressure Core Spray (CSL), High Pressure Core Spray (CSH), Residual Heat Removal (RHS)
Title of Change:	Change of Safety Classification of Air Test Solenoid Valves and Associated Limit Switches

Description of Change:

Appendix B Determinations 91-006, 91-007, and 91-008 re-evaluated the safety classification of air test valves (SOVs) 2CSL*SOV101 and 2CSH*SOV108 from safety related (SR) to nonsafety related (NSR); 2RHS*SOV16A, B, C and 2RHS*SOV39A, B from SR to Q4. The associated position limit switches are also reclassified from SR to Q4.

The purpose of the SOVs is to test and assure operability of isolation valves (AOVs) 2CSL*AOV101; 2RHS*AOV16A, B, C; 2RHS*AOV39A, B; and 2CSH*AOV108. The purpose of the position limit switches is to provide information as to the position of the AOVs to the main control room. The AOVs perform a safety-related function to isolate the containment in case of a line break during system operation. The safety classification of the AOVs is not affected by the above Appendix B determinations and remains SR.

Since the reclassified SOVs and limit switches are located in the primary containment, the tagging of these components will be verified during the outage. If the affected components are tagged, a retagging will be performed to reflect new NSR identification of these components.

The SOVs are used periodically to test the AOVs to assure that the AOVs are available to perform their safety-related function; i.e., to close in case of line break outside the containment during the system operation and, therefore, to prevent loss of coolant outside the containment.

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

Safety Evaluation No.: 94-086

Implementation Document No.: EDC 2E00749

USAR Affected Pages: Figures 5.1-2a, 5.1-2b, 5.1-2c, 5.4-16a

System: Nuclear Boiler Instrumentation (ISC), Reactor Water Cleanup (WCS), (EGS)

Title of Change: Revision of Component Identifiers for Components Whose Safety Classification has Changed

Description of Change:

Some components were identified which had their safety classification revised through the Appendix B Determination process, but the resulting changes were not incorporated into all of the associated documentation. The applicable documents and the Master Equipment List database have been updated accordingly. Also, the identification tags on the components in the field were changed.

The following components were affected: 2EGS-PY1051A, B changed from Q4 to SR (now 2EGS*PY1051A, B); 2WCS-PDS115 changed from Q5 to SR (now 2WCS*PDS115); 2ISC*TE27A through *TE27D changed from SR to Q4 (now 2ISC-TE27A through -TE27D).

Safety Evaluation Summary:

This change will revise the component identifiers for certain plant components whose safety classifications were changed by prior Appendix B Determinations. There are no changes in system equipment or logic.

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

Safety Evaluation No.: 94-088 Rev. 1

Implementation Document No.: Simple Design Change SC2-0167-94

USAR Affected Pages: 7.7-20; Figure 7.7-6 Sh 3

System: Reactor Coolant (RCS)

Title of Change: Recirculation Flow Control Valve Minimum Position Change

Description of Change:

The reactor recirculation flow control valves can become stuck at minimum position due to the differential pressure across the valve after the respective pump is transferred to high speed. This change increased the valve position to a maximum of 22% open (hot indicated), with the valve limit switch bypassed while the first pump is upshifted, and a maximum of 20% open (hot indicated) while the second pump is upshifted.

Safety Evaluation Summary:

The recirculation flow control valve position switch is a permissive to the recirculation pump upshift circuit. The purpose of the switch is to minimize the neutron flux spike which results from the increased core flow when the pump is upshifted. Protection to assure that fuel design limits are not exceeded is provided by the average power range monitor high flux scram. The peak neutron flux that will result from the increased flow when the recirculation pumps are upshifted is conservatively below the high flux scram.

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

Safety Evaluation No.: 94-091

Implementation Document No.: Procedure N2-OP-31

USAR Affected Pages: Figure 15.2-11 Notes Sh 2, Figure 15.2-13a

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

Title of Change: Add Alternate Shutdown Cooling Mode Using CSL

Description of Change:

A mode of alternate shutdown cooling was added that is bounded by the analysis contained in the USAR. This mode, use of low-pressure core spray (LPCS) as an injection path and residual heat removal (RHR) A in suppression pool cooling, will adequately remove decay heat and is an acceptable alternate shutdown cooling method.

Safety Evaluation Summary:

The LPCS injection with RHR A in suppression pool cooling method is bounded by the original analysis for alternate shutdown cooling. This method was not originally described in the USAR, but is included to verify its acceptability as an acceptable mode of operation.

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

Safety Evaluation No.:	95-021
Implementation Document No.:	Procedures GAP-POL-01, NSAS-POL-01
USAR Affected Pages:	13.1-4, 13.1-5, 13.1-6, 13.1-13, B.1-2, B.1-3; Table B-1 Sh 1 & 2; Figures 13.1-1, 13.1-2, 13.1-5
System:	N/A
Title of Change:	Reorganization; Changes to GAP-POL-01 & NSAS-POL-01 to Establish Nuclear Business Management Organization

Description of Change:

Procedures GAP-POL-01 and NSAS-POL-01 have been revised to reorganize the functions of Finance, Computer Software Development, Business Planning, and Nuclear Procurement under a new organization titled, "Nuclear Business Management," reporting to the Vice President Nuclear Generation.

Safety Evaluation Summary:

These procedure changes establish departmental responsibilities and lines of authority, responsibility, and communication for the Nuclear Business Management organization. The proposed organization structure satisfies the criteria of SRP 13.1.1 and conforms with the requirements of Section 6.1.2 of the Plant Technical Specifications. The proposed changes do not impact accident or malfunction initiation or consequences.

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

Safety Evaluation No.:	95-022
Implementation Document No.:	Procedures GAP-POL-01, NSAS-POL-01
USAR Affected Pages:	13.1-5
System:	N/A
Title of Change:	Reorganization; Changes to GAP-POL-01 & NSAS-POL-01 to Transfer Management and Operational Responsibility for the Site Sewage Treatment Facility from Technical Services (Environmental) to Unit 1 Chemistry

Description of Change:

Procedures GAP-POL-01 and NSAS-POL-01 have been revised to transfer management and operational responsibility for the Site Sewage Treatment Facility from the Technical Services Branch (Environmental) of the Nuclear Safety Assessment & Support Department to the Unit 1 Chemistry Branch of Nuclear Generation.

Safety Evaluation Summary:

These procedure changes establish departmental responsibilities and lines of authority, responsibility, and communication for management and operation of the Site Sewage Treatment Facility. The proposed organizational structure satisfies the criteria of SRP 13.1.1 and conforms with the requirements of Section 6.1.2 of the Plant Technical Specifications. The proposed changes do not impact accident or malfunction initiation or consequences.

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

Safety Evaluation No.:	95-023
Implementation Document No.:	Procedure NSAS-POL-01
USAR Affected Pages:	13.1-5; Table 13.5-1 Sh 1; Figure 13.1-5
System:	N/A
Title of Change:	Reorganization; Changes to NSAS-POL-01 to Transfer Procedure Program Coordination from Technical Services to Quality Assurance

Description of Change:

Procedure NSAS-POL-01 has been revised to delete the responsibility assigned to the Manager Technical Services to manage implementation of the procedure program including publication. The "managing" function assigned to the Manager Technical Services was to provide overall coordination of the procedure program. Responsibility for overall coordination of the procedure program has been transferred to the Manager Quality Assurance as a Quality Assurance administrative service function.

Safety Evaluation Summary:

These procedure changes establish departmental responsibilities and lines of authority, responsibility, and communication for implementation of the procedure program including publication. The proposed organizational structure satisfies the criteria of SRP 13.1.1 and conforms with the requirements of Section 6.1.2 of the Plant Technical Specifications. The proposed changes do not impact accident or malfunction initiation or consequences.

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

Safety Evaluation No.:	95-025
Implementation Document No.:	Procedure NSAS-POL-01
USAR Affected Pages:	13.1-5, 13.1-6; Figure 13.1-5
System:	N/A
Title of Change:	Reorganization; Changes to NSAS-POL-01 to Transfer Environmental Protection Functions from Technical Services to Licensing and Emergency Preparedness

Description of Change:

Procedure NSAS-POL-01 has been revised to reorganize (transfer) responsibility for the functional areas of environmental monitoring (including control of hazardous and industrial wastes, and assessing effects of radioactive effluent) from the Manager Technical Services to the Manager Licensing, and meteorological monitoring from the Manager Technical Services to the Director Emergency Preparedness.

Safety Evaluation Summary:

These procedure changes establish departmental responsibilities and lines of authority, responsibility, and communication for implementation of the procedure program including publication. The proposed organizational structure satisfies the criteria of Standard Review Plan 13.1.1 and conforms with the requirements of Section 6.1.2 of the Plant Technical Specifications. The proposed changes do not impact accident or malfunction initiation or consequences.

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

Safety Evaluation No.: 95-027

Implementation Document No.: Simple Design Change SC2-0016-94

USAR Affected Pages: 9.4-3; Figures 1.2-15 Sh 1, 9.4-1c, 9.4-5 Sh 6

System: Control Building Ventilation (HVC)

Title of Change: Control Room Smoke Removal

Description of Change:

This change installed two redundant safety-related electrical disconnect switches in the power supply to the Control Building smoke removal makeup unit 2HVC-HVU1. The electrical leads to this unit had previously been de-termed and taped at the circuit breaker for the unit. This prevented inadvertent actuation of the unit, and the potential for introduction of unfiltered, potentially-contaminated air into the Control Room envelope, but severely hampered alignment of the Control Building HVAC system for smoke removal. The necessity to assure that 2HVC-HVU1 does not operate after an accident was previously addressed in Safety Evaluation 89-006.

Safety Evaluation Summary:

The redundant disconnect switches will satisfy the single failure criterion, and assure that 2HVC-HVU1 does not operate after an accident and introduce unfiltered, potentially-contaminated air into the Control Room envelope. This installation conforms to the requirements of 10CFR50, Appendix A, General Design Criteria 2 and 19, and appropriate guidance provided in NUREG-0800 for the Control Room Area Ventilation System.

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

Safety Evaluation No.: 95-028

Implementation Document No.: EDC 2M00461

USAR Affected Pages: Figure 10.4-2a

System: Condenser Air Removal (ARC)

Title of Change: 2ARC-SP1A/1B Flow Indicators Detectors in Fire Zone 252SW

Description of Change:

Flow indicator piping is installed on hogger separator 2ARC-SP1A with one flow indicator installed and the other flow indicator in storage with its piping connection plugged. However, the installation of the flow indicator piping did not appear to be a permanent installation. Flow indicators 2ARC-FI1B/2B and the piping were not installed on 2ARC-SP1B.

This change installed the piping and supports necessary for the flow indicators and shows the installation of only one pair of flow indicators with the other separator flow indicator piping connections being blanked off. Depending on which hogger is used to measure leakage, the flow indicators can be switched to the appropriate hogger as necessary.

Safety Evaluation Summary:

The proposed change installs flow indicators on the hogger separators to measure leakage discharged from the vacuum pumps. The proposed change is consistent with the design of the hoggers.

The flow indicators are used to measure condenser air in-leakage during startup. One of each pair of flow indicators has a high range to measure gross leakage and the other one has a low range to measure small leakage. Only one hogger is operated at a time during startup to measure leakage and, therefore, only one pair of flow indicators is required. The hoggers do not perform a safety function and are not required for safe shutdown of the reactor. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.: 95-029

Implementation Document No.: Mod. PN2Y94MX012

USAR Affected Pages: Figures 5.4-2b, 5.4-2c

System: Reactor Coolant (RCS)

Title of Change: (RCS) Vibration Monitoring

Description of Change:

This modification replaced the existing vibration monitoring system (GEROM) on 2RCS-PNL107, and 2 chart recorders on 2RCS-PNL100, with a new, upgraded system from Bruel & Kjaer called COMPASS. The COMPASS system provides an online diagnostic system with continuous monitoring, trending reporting, and analysis to provide early warning shaft cracking.

Safety Evaluation Summary:

The proposed change will utilize existing signals to provide remote access to the vibration data without impacting the configuration or performance of the existing local panel. The new system will provide enhanced data processing and analysis capabilities.

Based on the evaluation performed, it is concluded that this change will enhance vibration monitoring for shaft cracking without creating an unreviewed safety question.

Safety Evaluation No.:	95-033
Implementation Document No.:	Mod. PN2Y94MX003 (EDC 2S00098)
USAR Affected Pages:	Figures 1.2-8 Sh 1, 9A.3-5, 12.3-1, 12.3-8, 12.3-34, 12.3-41, 12.3-69
System:	Domestic Water (DWS), Sanitary Plumbing (PBS), Auxiliary Service Building HVAC (HVL), Paging System (COP)
Title of Change:	Auxiliary Service Building Renovation, RFO-4 Scope

Description of Change:

This change renovated the Auxiliary Service Building elevation 261'-0" to allow use as a controlled personnel ingress and egress to/from the Turbine and Reactor Buildings via the linkway during RFO-4. This change involved making an opening in the 13 line wall at elevation 261' near the entrance to the south electrical tunnel stairwell, installation of an additional 1.5-hour fire-rated door (ET262-6) for stairway isolation, removal of lockers, removal of the drinking fountain and wash basins, the capping of floor drains in the temporary access passageway, and the removal of door AS261-7 for improved access. Also, door ET262-4 was removed while the area was being used only for access and egress during RFO-4. The temporary access passageway was created by installing painted Gypsum wallboard partitions. The ceiling tile grid and associated services were revised in the area of the passageway. These changes are partial scope for this modification.

Safety Evaluation Summary:

The changes being made to the south electrical tunnel stairwell were addressed for conformance with General Design Criterion 2 and no adverse impact was created. Potential impact to adjacent safety-related areas and conformance to General Design Criterion 3 and 10CFR50 Appendix R were evaluated and conformance was maintained. Electrical and mechanical services were revised and did not impact any operation of equipment important to safety.

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

Safety Evaluation No.: 95-039

Implementation Document No.: Mod. PN2Y94MX005

USAR Affected Pages: Figures 1.7-1c, 10.1-3h, 10.1-4d, 10.1-8b, 10.1-8f

System: Main Steam (MSS), Gland Steam Supply (TME), Auxiliary Steam (ASS), Moisture Separator Reheater Vent and Drain (DSR)

Title of Change: High Pressure Instrument Air to Actuators

Description of Change:

This modification replaced the shuttle block and solenoid valves on 11 process valves with redesigned block and solenoid valves. This eliminated excessive maintenance on the air booster pumps and related components associated with the high-pressure instrument air supply to the 11 process valves. The 11 process valves are 2MSS-AOV92A/B, 2TME-AOV121, 2ASS-AOV145, 2DSR-AOV81A/B/C, 2DSR-AOV82A/B and 2DSR-AOV83A/B.

Safety Evaluation Summary:

All of the process valves on which the block and solenoid valves are mounted are nonsafety related. These process valves are not required to function during or after an accident. The subject MSS and TME block and solenoid valves are seismically designed and mounted to the actuator and will have no adverse impact on the surrounding safety-related equipment. The replacement block and solenoid valves will perform the same function as the original shuttle block and solenoid valves.

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

Safety Evaluation No.: 95-040

Implementation Document No.: Temporary Mod. 95-002

USAR Affected Pages: N/A

System: Condenser Air Removal (ARC), Makeup Water (MWS), Water Treating (WTS)

Title of Change: Temporary Modification 95-02

Description of Change:

WTS is designed to produce demineralized water quality for various plant uses, including separators of the mechanical vacuum pumps 2ARC-P1A and P1B. During startup from a forced outage on 2/2/95, pump 2ARC-P1A was started to draw vacuum in the main condenser. However, vacuum was lost due to the loss of separator water.. The WTS supply was subsequently determined to have very little pressure and flow makeup capability.

An alternate source of cooling from the MWS was provided by connecting a hose at 2MWS-V95 to 2ARC-P1A just upstream of 2ARC-SOV18A. This necessitated removal of strainer 2ARC-STR3A to facilitate the hose installation. In addition, isolation valve 2ARC-V8A was closed to isolate the WTS supply from 2ARC-P1A.

Safety Evaluation Summary:

This temporary modification will have no significant effects on the design bases of the systems involved. The needed makeup water for the ARC pumps will be supplied via MWS using this temporary modification. The hardware changes to implement the change will conform to the original installation specification or engineering-approved equivalent. No safety-related functions are being added to the systems, and no system interactions important to safety are being created.

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

Safety Evaluation No.: 95-041

Implementation Document No.: Simple Design Change SC2-0162-94

USAR Affected Pages: 10.2-3

System: Turbine Main Alarms and Trips (TMA)

Title of Change: Exhaust Hood Temperature Trips

Description of Change:

Turbine exhaust hood thermostats operate on increasing temperature to trip the turbine in the event of excessive temperature. The thermostats, one at each low-pressure turbine, are connected in parallel to provide a one-out-of-three taken once logic. This logic is not single failure proof and, as such, makes the plant susceptible to unwanted turbine trips and a reactor SCRAM upon the single failure of thermostats.

This simple design change eliminated turbine trips as a result of high-high exhaust hood temperature. Administrative controls, control room indication, and control room alarms are relied upon to manually maintain turbine exhaust hood temperature below the previous trip value. The condenser low vacuum trip provides automated temperature protection for the hood without the added assurance of a redundant hood trip.

Safety Evaluation Summary:

Sufficient plant features exist to provide the operator with adequate warning and a means of controlling hood temperature. The condenser low vacuum trip provides more conservative and reliable protection than that being eliminated. The effects of the proposed change are limited to the turbine; therefore, it is concluded that all accidents or malfunctions associated with this change are bounded by previously evaluated scenarios.

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

Safety Evaluation No.: 95-042

Implementation Document No.: Temporary Mod. 95-003

USAR Affected Pages: N/A

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

Title of Change: Service Water Makeup to HVK

Description of Change:

Makeup water supply to the closed loop HVK was via WTS. However, the WTS makeup water supply lines became clogged, limiting the flow of makeup water to HVK to an unacceptable level. This temporary modification enabled SWP to be used to supply makeup water to HVK. For train A, the makeup water supply hose was routed from valve 2HVK*V239 to valve 2HVK-V272. For train B, the makeup water supply hose was routed from valve 2HVK*V256 to valve 2HVK-V271. A mechanical filter was installed for each of the makeup water supply lines along with the necessary fittings and supporting hardware.

Safety Evaluation Summary:

The proposed temporary design change of providing makeup water to HVK from SWP does not adversely affect the system to perform its safety functions because the SWP water design parameters (flow, pressure, temperature) have been reviewed and found to be acceptable to interface with HVK. This temporary change will have no significant effects on SWP because the amount of water to be diverted from SWP for makeup to HVK is approximately 13 US gallons at a time, on an intermittent basis. This amount is insignificant relative to the total SWP flow and capacity. In the event of a hose failure during the makeup water transfer, the makeup water will be isolated by closing the supply valves. This will be done by an Operator who will be present in the Chiller Room while the makeup water is being transferred. Therefore, if a hose failure occurs, the loss of service water and the potential for flooding the rooms through these lines will be minimal. This change does not affect the WTS system in any adverse way because the change will only isolate the normal makeup water supply of WTS to HVK.

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

Safety Evaluation No.:	95-046
Implementation Document No.:	Simple Design Change SC2-0109-94
USAR Affected Pages:	6.2-23, 6.2.24, 6.2.25, 6.2-26, 6.2-27, 6.2-28, 6.2-29, 6.2-34, 6.2-35, 6.2-51, 6.2-52, 6.2-53, 6.2-54, 6.2-55, 6.2-56; Tables 6.2-4a (Deleted), 6.2-9, 6.2-27A, 6.2-52, 6.2-53, 6A.4-1; Figures 6.2-28, 6.2-28A, 6.2-28B, 6.2-28C, 6.2-45, 6.2-46
System:	Containment Monitoring (CMS)
Title of Change:	Suppression Chamber Air Temperature Change

Description of Change:

This simple design change increased the suppression chamber air normal operating temperature from 110°F to 122°F. The previous limit of 110°F was an arbitrary selection based on expected temperature at the time, even though the maximum allowable temperature could be higher than 110°F.

Safety Evaluation Summary:

The consequences of the wetwell air temperature increase for accident, transient events, equipment qualification, station blackout, emergency operating procedures, drawdown analysis, HVAC analysis and structural analyses are either recalculated or evaluated against the existing design limits of structures and/or components. In all cases, the design limits have been met.

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

Safety Evaluation No.: 95-048

Implementation Document No.: Temporary Mod. 94-025

USAR Affected Pages: N/A

System: Turbine Building Ventilation (HVT)

Title of Change: Three Fan Operation of the Turbine Building Exhaust

Description of Change:

Normal operation of the Turbine Building ventilation system has two 50% capacity exhaust fans running with a third fan in standby. Electrical interlocks prevent concurrent operation of all three fans. This temporary modification defeated the interlocks, allowing simultaneous operation of all three fans in an effort to provide enhanced pressure and temperature control.

Safety Evaluation Summary:

Parallel operation of three exhaust fans will increase the ventilation system capacity, ensuring the Turbine Building is maintained at subatmospheric pressure with temperatures that range from 50°F to 130°F. A review of bus 2NJS-US8 loading reveals sufficient capacity for all three fans to be run without overloading the bus. Turbine Building exhaust flow analysis with three fans running indicates a system flow increase of 17,500 cfm (80,000 to 97,500) is expected for three fan operation. This flow increase will cause a corresponding increase in fan total pressure from 10.74 in. W.G. to 14.2 in. W.G., which is within the stable operating range of the fans. Pressure and flow measurements will be obtained during initial three fan operation to verify these values. The steam tunnel temperature and differential temperature will be monitored during initial three fan operation to verify that the potential change in exhaust flow from the main steam tunnel does not have a detrimental effect on the temperatures sensed by the LDS trip units. Running the third fan will not significantly impact the air balance for the steam tunnel. Therefore, this will not impact the function of the leak detection instruments.

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

Safety Evaluation No.:	95-055 Rev. 2
Implementation Document No.:	Simple Design Change SC2-0107-94
USAR Affected Pages:	9.2-14; Table 3.9A-12 Sh 1, 13, 15; Figure 9.2-3c
System:	Reactor Building Closed Loop Cooling Water (CCP)
Title of Change:	Alternate Drywell Cooling

Description of Change:

This simple design change added two piping penetrations through the southeast quadrant of the Reactor Building wall. In addition, a new 4" hose connection was added on the CCP supply and return headers.

During outages, a chiller (located in the yard) is connected to the Reactor Building penetrations. Hoses are now routed from the Reactor Building penetrations through emergency air lock to the CCP connections in the drywell.

Safety Evaluation Summary:

DER 2-95-3092 identified that the cooler skid does not conform to the requirements of Standard Review Plan 3.5.3 for a missile barrier. Analysis provides justification for not providing missile protection during Modes 4 and 5 when the alternate drywall cooling system is in operation. Revision 2 of this safety evaluation removes the requirement to park the cooler skid in front of the penetration. The permanent changes are designed in accordance with design criteria for CCP. The Reactor Building penetrations are designed to ASME III NC-3600 requirements and include redundant spring-loaded check valves/blind flanges to assure that secondary containment integrity is maintained when alternate drywell cooling is operating/secured. The hoses will be routed so as to prevent physical interaction with safety-related items in the event of connector failure. All potentially affected essential equipment or systems are designed for flood or spray. The implementation of this change will ensure that drywell temperature is controlled during an outage such that personnel stay times are maximized. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.:	95-058
Implementation Document No.:	Procedure GAP-POL-01
USAR Affected Pages:	13.1-12; Figure 13.1-2
System:	N/A
Title of Change:	Dissolution of the Unit 2 Technical Support Branch Support Section

Description of Change:

The Nuclear Division is organized into departments, with departments being subdivided into branches, and branches being subdivided into sections. Sections compose the lowest organizational tier of the Nuclear Division. This safety evaluation analyzed changes to the Technical Support Branch as a result of dissolving the Support Section.

The Unit 2 Technical Support Branch's Support Section was dissolved by:

1. Eliminating the position of Lead Support Engineer.

2. Converting the Administrative Technician position to a supervisory position with the title of Supervisor Administrative Support.

3. Having the Supervisor Administrative Support report directly to the Manager Technical Support and to be responsible for:

    a. Administration of Station Operations Review Committee (SORC).

    b. Administration of the plant technical review program.

    c. Supervision of the Branch's clerical staff.

4. Redistributing the remaining personnel and functions (including coordination of plant modifications) of the Support Section to the System Engineering Sections under the supervision of the Lead System Engineers.

Safety Evaluation Summary:

Dissolution of the Unit 2 Technical Support Branch's Support Section does not involve a change to the established responsibilities of the Technical Support

Safety Evaluation No.: 95-058 (cont'd.)

Safety Evaluation Summary: (cont'd.)

Branch as described in the USAR; only the reporting structure within the branch is being affected. The organization continues to provide for the integrated management of activities that support the operation of the facility and maintains clear management control and effective lines of authority and communication.

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

Safety Evaluation No.:	95-065 Rev. 2
Implementation Document No.:	BWROG EPGs, Rev. 4
USAR Affected Pages:	N/A
System:	Various
Title of Change:	Revision 6 of the NMP2 Emergency Operating Procedures

Description of Change:

Revision of the Emergency Operating Procedures (EOPs) changed some operating limits as a result of uprated power at Unit 2, as well as the new General Electric fuel. The limits which were revised are as follows:

- High reactor pressure vessel (RPV) pressure setpoint
- Boron injection initiation temperature
- RPV pressure at which all turbine bypass valves are fully open
- Heat capacity temperature limit
- Suppression chamber spray initiation pressure
- Heat capacity level limit
- Pressure suppression pressure
- Minimum zero injection RPV water level
- Minimum RPV flooding pressure
- Minimum steam cooling RPV water level

Safety Evaluation Summary:

Although some changes have been made to the EOPs by Revision 6, it was verified that:

- The operator actions prescribed in this new revision are in accordance with the Boiling Water Reactor Owners' Group (BWROG) Emergency Procedure Guidelines (EPGs), and
- When applied to licensing basis accidents and transients, the EOPs will not increase the consequences of these events as depicted in the USAR.

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

Safety Evaluation No.:	95-067
Implementation Document No.:	Calc. H21C-045, Procedures EPIP-EPP-23; N2-OP-53A, N2-EOP-6, N2-ISP-MSS-R001, N2-ISP-MSS-R002
USAR Affected Pages:	6.2-61, 15.6-12, 15.6-14, 15.6-16; Tables 6.2-55b Sh 1 & 2, 6.2-55d Sh 1 & 2, 15.6-13 Sh 1 & 13, 15.6-16b
System:	Main Steam (MSS), Control Building Ventilation (HVC)
Title of Change:	MSIV Leakage Rate Below Which Control Room Air Intakes Are 100% Redundant

Description of Change:

This change establishes a main steam isolation valve (MSIV) leak rate for all 4 main steam lines below which operating restrictions on the Control Room outside air intakes are not required. This value has been determined to be 15.0 scfh per MSIV for all 4 lines, which is below the Technical Specification allowable leak rate of 24.0 scfh per line. In addition, individual line leak rates above 15.0 scfh, but below 24.0 scfh, are permitted as long as other lines are sufficiently below 15.0 scfh to provide compensating dose reduction in the Control Room.

Safety Evaluation Summary:

An MSIV leak rate equivalent to 15.0 scfh for each of the 4 lines can be applied without requiring post-LOCA isolation of the more contaminated Control Room air intake, or the availability of the less contaminated air intake, to maintain doses within applicable limits. The two (east and west) Control Room air intakes would be 100% redundant without distinction. In addition, MSIV leak rates above 15.0 scfh, and up to the Technical Specification limit of 24.0 scfh, can be justified for individual line(s) if the MSIVs in the remaining lines have leak rates sufficiently low so as to provide compensating dose reductions. A change to the leak rate limit for the MSIVs does not affect either the manual or automatic actions that would close the MSIVs. Therefore, the proposed change to the allowable MSIV leak rate cannot affect the probability of the closure of one or more of the MSIVs. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.: 95-071 Rev. 1

Implementation Document No.: N/A

USAR Affected Pages: A.4.4-1, A.4.4-3, A.5.2-4, A.6-2, A.15.0-7, A.15.1-8, A.15.2-9, A.15.4-7, A.15B-1, A.15D-1; Table A.15.0-4 Sh 1

System: Various

Title of Change: Operation of NMP2 Reload 4/Cycle 5

Description of Change:

This change added new fuel bundles and established a new core loading pattern for Reload 4/Cycle 5 operation of Unit 2. Two hundred forty eight (248) new fuel bundles of the GE11 design were loaded. Also, 32 twice-burned GE6B bundles that were discharged at the end of Reload 1/Cycle 2 were re-inserted. All 124 of the GE6B bundles from Reload 3/Cycle 4, and 156 of 196 GE9B bundles (P8CWB299), were discharged to the spent fuel pool. Various evaluations and analyses were performed to establish appropriate operating limits for the reload core. These cycle-specific limits were documented in the Core Operating Limits Report.

Revision 1 of this Safety Evaluation incorporated the necessary changes to the operating limit and safety limit as a result of the revised Supplemental Reload Licensing Report prepared by General Electric (GE). In May 1996, GE notified the NRC of a reportable condition involving the generic safety limit calculational methodology. As a result of this notification, GE performed a cycle-specific safety limit calculation for all affected plants. The new safety limit minimum critical power ratio (SLMCPR) at Unit 2 is 1.10 for two-loop operation and 1.12 for single-loop operation.

Safety Evaluation Summary:

The reload analyses and evaluations are performed based on the General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A-10 and NEDE-24011-P-A-10-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 4, the evaluations included transients and accidents likely to limit operation because of MCPR

Safety Evaluation No.: 95-071 Rev. 1 (cont'd.)

Safety Evaluation Summary: (cont'd.)

considerations; overpressurization events; loss-of-coolant accident; and stability analysis. Appropriate consideration of equipment out of service was included. Limits on plant operation were established to assure that 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.: 95-073

Implementation Document No.: Simple Design Change SC2-0039-95

USAR Affected Pages: Figure 9.1-19a

System: FNR

Title of Change: Refuel Floor Rigging Improvements

Description of Change:

This simple design change implemented the use of 25-ton capacity chain hoists and shackles as an alternate to turnbuckles in the lifting arrangements for the fuel transfer shielding bridge and the refueling canal plug lifts.

Safety Evaluation Summary:

The use of 25-ton capacity chain hoists and shackles as an alternate to turnbuckles in the lifting arrangements for the fuel transfer shielding bridge and refueling canal plug lifts will improve personnel safety during rigging activities for these lifts. The implementation of this alternate rigging was performed in accordance with the single-failure-proof requirements of NUREG-0612.

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

Safety Evaluation No.:	95-075
Implementation Document No.:	Procedure N2-MMP-MSS-917
USAR Affected Pages:	6.3-35
System:	Main Steam (MSS)
Title of Change:	Post-Installation Testing Requirements for Non-ADS SRVs

Description of Change:

This change allowed post-installation testing of non-ADS SRVs without lifting the valve disc off the seat. Correct installation of the valves was verified by valve actuator movement. This change to the testing requirements was made in an attempt to reduce valve seat leakage, which can be aggravated by any disc seat movement.

Safety Evaluation Summary:

This change will not impact the safe operation of the plant as the valve will be restored to operational readiness before the plant is returned to service. These attributes are now to be verified by valve actuator movement, with the actuator disengaged from the valve stem. The valve movement is confirmed prior to installation. The actuator is reconnected before plant startup, restoring the relief function of the valve.

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

Safety Evaluation No.:	95-076
Implementation Document No.:	EDC 2E00867
USAR Affected Pages:	Figure 9.2-5a
System:	Makeup Water Treating (WTS)
Title of Change:	Abandon Heat Trace on Line 2WTS-008-199-4

Description of Change:

Line 2WTS-008-199-4 was used as a drain line for activated carbon filter system backwashing in the WTS system and was connected to the Unit 1 sewage treatment system. This line was heat traced. The activated carbon filter in Unit 2 was abandoned in place via temporary modification 87-2008. The USAR and its associated drawings were not updated as the changes were made under temporary modification. However, this line was used as a drain line from the waste neutralizer tank (WNT), which stores 60,000 gallons of demineralized water coming from the transient runs made on the Ecolochem system. The heat tracing has been abandoned, and the requirement for heat tracing line 2WTS-008-199-4 has been removed.

Safety Evaluation Summary:

The WNT containing demineralized water is drained every 30 to 45 days to the Unit 1 sewage treatment system for a very short time, i.e., a few seconds. The sanitary system is an open system, thus there is no possibility of backfilling these lines under any operating condition. Based on the fact that the sanitary system is an open system, and the short duration for which this pipeline is used, it is concluded that heat tracing for this line is not required.

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

Safety Evaluation No.: 95-077

Implementation Document No.: EDC 2M00573A and 2M00574

USAR Affected Pages: Table 3.9A-12 Sh 15; Figures 5.4-13d, 5.4-13e, 9.3-5g

System: Residual Heat (RHS)

Title of Change: Revise/Delete the Leak Rate Acceptance Criteria and Test Frequency for RHS Valves 2RHS*MOV142, MOV149, SOV35A/B and SOV36A/B

Description of Change:

This change revised the leak rate acceptance criteria and test frequency for valves 2RHS*MOV142, MOV149, SOV35A/B, and SOV36A/B. The leakage acceptance criteria of less than or equal to 1 gpm times the number of hydrostatically tested valves was increased to 20 gpm for 2RHS*SOV35A/B and SOV36A/B, and to 10 gpm for 2RHS*MOV142 and MOV149 at normal system operating pressure. The test frequency was revised from once every 18 months to once every 2 years. Leak testing requirements for the valves remain in the In-Service Testing program; however, changes to NIP-DES-04 (by revising the footnote "m") and supporting operations procedures were required to implement the new leakage criteria. Implementation of simple design change SC2-0046-95 to install ASME Class 2 reducers to replace the leakage control function of the solenoid-operated valves (SOVs) was determined to be an acceptable alternative to leak testing the SOVs.

Safety Evaluation Summary:

This safety evaluation has concluded that an unreviewed safety question does not result from the proposed change. This conclusion is based on the ability to demonstrate RHS system leakage boundary integrity by satisfying the functional requirements of the low-pressure coolant injection system with the increased leakage, and determining that the consequences of the increased leakage into secondary containment post-LOCA are radiologically acceptable.

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

Safety Evaluation No.: 95-082 Rev. 0 & 1

Implementation Document No.: Procedure N2-OP-36A

USAR Affected Pages: Table 6.2-56 Sh 6 & 24

System: Standby Liquid Control (SLS)

Title of Change: Safety Function of Valves 2SLS*MOV5A/B

Description of Change:

Clarification was added to the USAR to describe the position of globe stop check valves 2SLS*MOV5A/B during normal operation, and to clarify their safety function. Operator action is required after SLS system injection to close valves 2SLS*MOV5A/B in order to meet the long-term containment isolation requirements.

Safety Evaluation Summary:

An engineering review of the proposed change has been performed and included conformance to regulatory requirements for containment isolation and an anticipated transient without scram event. This change will have no effect on the safety or operability of the SLS system to perform its designed function.

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

Safety Evaluation No.:	95-083
Implementation Document No.:	Temporary Mod. 95-017
USAR Affected Pages:	N/A
System:	Vital Bus (VBB)
Title of Change:	Hardwire UPS Load to the Maintenance Supply

Description of Change:

Maintenance activities at 2VBB-UPS1G required an extended outage of the equipment during refuel outage RFO4. The maintenance performed required all the uninterruptible power supply (UPS) power sources to be de-energized, subsequently de-energizing the UPS load. This change implemented a temporary modification to shorten the duration for which the UPS load was de-energized. The UPS maintenance supply was directly connected to the UPS load by splicing the supply cable to the load cable. In this configuration the UPS, the normal ac source, and the backup dc source were completely isolated from the maintenance source supplied load.

Safety Evaluation Summary:

This temporary modification will manually provide the same result as if the unit were intentionally placed in the bypassed configuration. The reliability of multiple sources will be sacrificed for the duration of the temporary modification the same as if the unit was bypassed using the built-in transfer switches. The transfer will not be automatic and will require the unit to be temporarily de-energized, but will allow the UPS loads to be fed from one of their designated sources throughout the UPS maintenance activities.

Existing feeder cables at the UPS unit will be spliced and appropriately insulated using approved bolting hardware and Raychem insulating material. As such, no cable ampacity or circuit protection concerns will be introduced 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.:	95-085 Rev. 0 & 1
Implementation Document No.:	Simple Design Change SC2-0048-95
USAR Affected Pages:	Figure 9.3-1c
System:	Instrument Air (IAS)
Title of Change:	Reroute Drain Trap Balancing Line for 2IAS-TRP51A, B

Description of Change:

The balance line for each of these drain traps was to the upstream side of the IAS dryers. This balance line saw a slightly higher pressure than the drain line. This higher pressure caused the water and moisture to backup into the dryer and impede its operation.

The original configuration was contrary to the vendor's recommendation, which stated the line should be at a pressure equal to or slightly less than the drain line. This allows the water from the IAS dryer to flow freely into the drain trap. This change rerouted the tubing into the downstream side of the IAS dryer providing the required pressure.

Safety Evaluation Summary:

The design of the IAS dryer will provide clean dry air to a dewpoint of 35°F for the plant. This change will only insure that the dryer functions properly by providing water removal capability for the dryer. The rerouting of the drain trap balancing line will allow for the drain trap to function more efficiently and prevent moisture from backing up into the dryer.

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

Safety Evaluation No.: 95-087

Implementation Document No.: Temporary Mod. 95-028

USAR Affected Pages: N/A

System: Makeup Water Storage (MWS), Chilled Water Ventilation (HVN)

Title of Change: Temporary Makeup Water to the HVN System

Description of Change:

This temporary modification installed a temporary hose from the MWS system to the HVN system. This hose provided an alternate source of makeup water to the HVN system. The existing makeup water source emanates from the MWS system, and due to the degradation of the water treatment (WTS) piping (insufficient flow due to plugging within the pipe), the system does not meet demand. This water source was not valved in because a new hookup for the Ecolochem trailer was installed. There was no available water source for the WTS system downstream of the circulating water pump seals. It was estimated that, with this alternate source of water, up to 60 gph of makeup water was required. Demand was controlled utilizing the existing control mechanisms. A temporary hose and associated components installed were routed from a 3/4" connection in the Screenwell Building at valve 2MWS-V7 to a 3/4" connection within the Chilled Water Building at valve 2HVN-V21. Valve 2HVN-V21 is an existing connection at the chilled water expansion tank makeup line. The WTS system was isolated by closing valve 2HVN-V410. The MWS system was the single source for makeup.

Safety Evaluation Summary:

The alternate makeup from the MWS system will be sufficient through a hose of equal size as a minimum. The new source of makeup water is demineralized water in lieu of filtered water. Water quality is enhanced and supply will be adequate to meet demand. All hoses and associated components shall be rated for their intended service conditions, and will be adequately secured. The 60 gph of water from the MWS system will not affect the MWS system capacity to feed water to its originally intended systems.

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

Safety Evaluation No.: 95-088 Rev. 0 & 1

Implementation Document No.: Procedure N2-STP-PUPA-24

USAR Affected Pages: N/A

System: Electrohydraulic Control (EHC)

Title of Change: Turbine Control Valve Amplifier Output Ceiling Limit

Description of Change:

The turbine control valve amplifier output ceiling limit was adjusted during RFO4 with a number of circuit cards removed as per site calibration procedure. When the circuit cards were reinstalled, the circuit loading changed and the ceiling limit was reduced. This limited the opening of the number 4 turbine control valve to 24%. This change permits dynamic adjustment of the turbine control valve amplifier output ceiling limit potentiometer during performance of turbine valve surveillance test N2-STP-PUPA-24.

Safety Evaluation Summary:

Using administrative controls, the control valve amplifier ceiling limit will be adjusted slowly with the turbine on-line so as to not introduce any perturbations in the EHC system nor in the reactor. During this activity, there are other control signals also adding their input to limit the amount any valve can open. If the bypass valves are open for too long, feedwater heating may be reduced. To preclude adverse effects, feedwater temperature will be monitored and the test procedure will be terminated if the feedwater temperature decrease exceeds 10°F. This will maintain feedwater temperature change within the current transient analysis bounds. Also, an EHC/control valve malfunction is not an accident initiator.

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

Safety Evaluation No.:	95-090
Implementation Document No.:	G.E. Design Specification 22A2887AL
USAR Affected Pages:	5.2-8
System:	Main Steam (MSS)
Title of Change:	Safety Relief Valve Opening Time Requirement (2MSS*PSV120 through 2MSS*PSV137)

Description of Change:

This change corrected the safety relief valve (SRV) stroke time (from 0.2 seconds to 0.25 seconds) to resolve the conflict between design documents, environmental qualification (EQ) documents, USAR and In-Service Inspection/In-Service Testing (ISI/IST) documents. The previous USAR SRV opening time of 0.2 seconds (delay plus stroke time) was based upon the valve purchase specification, which conservatively identified a valve opening time of 0.2 seconds. The Nuclear Boiler System Design Specification, which is the design basis document, identifies the required design analyses opening time for the SRVs (delay plus stroke time) as 0.25 seconds. Also, the valve recertification testing per the IST Program Plan and the environmental qualification of the SRVs both require that the valve be able to open within 0.25 seconds.

Safety Evaluation Summary:

Correcting the stroke time of the SRVs from 0.2 seconds to 0.25 seconds in the USAR meets the requirements of the Nuclear Boiler System Design Specification and the SRV Environmental Qualification Report. This change does not affect the vessel overprotection analysis because relief mode of the SRVs is not credited for the pressure relief.

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

Safety Evaluation No.: 95-091

Implementation Document No.: Procedure N2-OP-48

USAR Affected Pages: Figures 9.5-52a, 9.5-52c

System: Auxiliary Boiler Feedwater (ABF), Auxiliary Boiler Chemical Feed (ABH), Auxiliary Boiler Blowdown (ABD)

Title of Change: Auxiliary Boiler Systems Normal Valve Lineup Changes

Description of Change:

The normal valve positions for components 2ABF-V176, 2ABH-V57, and 2ABD-V29 are now shown as normally closed and the positions for valves 2ABD-V37 and 2ABD-V38 are now shown as normally open.

Safety Evaluation Summary:

These changes maintain configuration control, and enhance system performance and normal/off-normal system configurations to support auxiliary boiler modes of operation. The new normal valve lineups, and throttling the valves to balance cooling water flow, will not impact or degrade the operation of the auxiliary boilers.

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

Safety Evaluation No.: 95-092

Implementation Document No.: Calc. EC-161, 162, 163, 164

USAR Affected Pages: 9B.5-1

System: Various

Title of Change: Appendix R Safe Shutdown Analysis Update

Description of Change:

The results of Calculations EC-161, 162, 163 and 164 concluded that adequate protection is provided for all but 12 of the associated circuits which are related to common power source. The primary and backup protective devices (molded case circuit breakers) of these associated circuits are not selectively coordinated in the instantaneous region of the time current characteristics curves of these breakers during a fault initiated by a fire. Therefore, during a fire event in a specific fire area, the backup breaker of the associated circuit may trip instead of the primary breaker and cause loss of power to safe shutdown related unit coolers fed from the affected 600-V distribution panel.

This change revised the USAR to include the results of Calculations EC-161, 162, 163 and 164 and to state that the breakers of the associated circuits not having selective coordination are administratively controlled during a fire event in the specific fire areas to achieve and maintain safe shutdown of the plant.

Safety Evaluation Summary:

Safe shutdown equipment that may be lost due to the associated circuit of concern are unit coolers fed from the 600-V distribution panels. Because an accident is not postulated with a fire event, the heat rise in various rooms due to the temporary loss of unit coolers is minimal. Administrative controls proposed will isolate the affected associated circuit, restore power to the unit coolers and ensure safe shutdown of the plant. The changes are also in conformance with the requirements of applicable criteria documents and regulatory guidance.

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

Safety Evaluation No.: 95-093 Rev. 0 & 1

Implementation Document No.: EDC 2M00475

USAR Affected Pages: Figures 5.4-16d, 5.4-16e

System: Reactor Water Cleanup (WCS)

Title of Change: WCS ASME Repairs and Replacements

Description of Change:

This change relocated the ASME Class breaks on valves that serve the WCS filter/ demineralizers. These valves are located in the Secondary Containment outboard of the containment isolation valves. This portion of the WCS system is designed, certified, and stamped per ASME III requirements; however, it is not safety related. The design specification for the system requires dual valves to be installed at high/low pressure interfaces to prevent inadvertent overpressurization of components by actuation of a single valve. This dual valve requirement causes additional post-repair/replacement testing that can be reduced by moving the ASME III Class break to the other side of the outboard valve.

Safety Evaluation Summary:

This change will allow the second of two valves in series, located in the Secondary Containment, to be designed, maintained, and replaced to quality group D requirements. The ANSI pressure and temperature rating of the valves will not be changed. The outboard valve will be maintained to the requirements of ANSI B31.1. This change will assure that the system will still have the same high/low pressure protection as before and serves to reduce post-maintenance requirements on the outboard valves.

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

Safety Evaluation No.:	95-094
Implementation Document No.:	DER 2-94-2475
USAR Affected Pages:	3A.34-1, 3A.34-2; Table 3.9B-2L Sh 1 & 2
System:	Recirculation Control (RCS)
Title of Change:	Update USAR Table 3.9B-2L and Appendix 3A Section 3A.34

Description of Change:

This change updated USAR Table 3.9B-2L, Recirculation Flow Control Valves, as per stress report revisions issued by General Electric Company.

In addition, the USAR has been updated to allow the use of upwardly compatible versions of the PC-based finite elements computer program IMAGES. This allows NMPC to use the latest qualified version of IMAGES.

Safety Evaluation Summary:

The revisions to USAR Table 3.9B-2L show that the actual stress values and the fatigue usage factor for the subject valves are within the corresponding ASME Code allowable limits.

The upwardly compatible versions of the PC-based finite elements program IMAGES will be subjected to the same verifications and controls as the original version.

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

Safety Evaluation No.:	95-095
Implementation Document No.:	Procedure N2-MMP-WCS-125
USAR Affected Pages:	N/A
System:	Reactor Water Cleanup (WCS)
Title of Change:	Temporary Partition Between RWCU Pump Rooms for Maintenance

Description of Change:

The heating, ventilating, and air conditioning (HVAC) air flow from one room was interfering with the other room temperature because of an open communication flow path between the WCS pump rooms. An air conditioner installed in either pump room A or B is not effective due to air circulation via this vent area between the rooms. To increase the stay-time during maintenance, a temporary partition was installed in the vent area to curtail the circulation rate between the pump rooms, thus lowering their temperature during maintenance.

Safety Evaluation Summary:

The RWCU pump rooms A and B temperatures are normally over 105°F, thus making it hard to do any maintenance work in either room for a longer duration due to potential heat exhausting. A temporary partition of insulating foam will reduce the air circulation between the pump rooms and lower the air temperature. The insulating foam will add a small quantity of additional combustible loading to the room while it is being used. This additional loading has been reviewed and determined to be within the accepted transient fireloads of the area. There is a fire alarm in each room and outside each room; therefore, there is no concern on changing the air flow pattern, which can change or impact the fire alarm behavior. The insulating foam material, if blown off into the other RWCU room, does not impede the functioning of any equipment. The RWCU rooms have been included in the high-energy line break (HELB) analysis. There is a margin of 15% between the design and the calculated pressure differential generated due to HELB. The lightweight temporary partition installed in the vent area is intended to be displaced from its location in case of a HELB. A local pressure increase of 0.25 psid will displace the temporary partition and open the vent area for free air flow between the pump rooms A and B. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.:	95-096
Implementation Document No.:	Simple Design Change SC2-0061-95
USAR Affected Pages:	Table 5.4-3; Figure 5.4-17 Sh 2
System:	Reactor Water Cleanup (WCS)
Title of Change:	One Pump 3 Filter/Demin Operation

Description of Change:

This change allows an alternate operating mode of WCS which uses one recirculating pump to supply 3 filter/demineralizers. This change allows greater water chemistry control if one WCS recirculating pump is out of service.

Safety Evaluation Summary:

The operation of WCS allowing one recirculating pump to supply 3 filter/demineralizers does not increase the probability of occurrence of a previously evaluated accident as this condition is within the bounds of the original system design capability. Total system flow is not increased and, therefore, the probability of occurrence of an accident is not increased. Administrative procedures are in place to monitor the flow velocity increase in certain portions of the WCS system. Evaluation of the flow increase indicates that adequate margin is available to contain the increased flow. Additionally, these lines with increased flow are added to the Erosion/Corrosion Program for increased monitoring.

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

Safety Evaluation No.:	95-097
Implementation Document No.:	Procedures N2-OP-29, N2-OP-101D, N2-SOP-29, N2-SOP-101D
USAR Affected Pages:	N/A
System:	Recirculation Control (RCS)
Title of Change:	Temporary Lock-up of RCS Flow Control Valve 2RCS*HYV17B

Description of Change:

This change temporarily locked up valve 2RCS*HYV17B. The position feedback signal was "noisy" and caused the valve to fluctuate intermittently. This in turn caused the reactor power to fluctuate. Testing indicated that the position element on the valve was the source of the "noise". Since the valve is in the primary containment, repairs could not be made to the position element until the plant was shut down. Therefore, the valve was temporarily locked in the open position until repairs could be made. In addition, the valve hydraulic system was periodically restored to allow valve position changes and flow adjustments.

Safety Evaluation Summary:

Locking up the recirculation flow control valve (FCV) will reduce reactor power fluctuations which are due to intermittent recirculation FCV fluctuations. The applicable transients and accidents (design basis accident [DBA] loss-of-coolant accident [LOCA], loss of feedwater heating, total loss of feedwater flow, recirculation flow control failure, anticipated transient without scram [ATWS], and single feedwater pump trip) have been reviewed for the impact of locking up a single recirculation FCV. It has been determined that the existing analyses for the DBA LOCA, loss of feedwater heating, recirculation flow control failure and ATWS bound the effect of locking up a single FCV. A locked-up FCV with a total loss of feedwater flow would result in a scram occurring slightly sooner; however, the locked-up control valve does not increase the consequences of this event in that the radiological consequences remain unchanged. In addition, operation at near full power with a single FCV locked up will increase the likelihood of a scram in response to a single feedwater pump trip. However, the radiological consequences will not increase. This temporary change will have no impact on the safe 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.:	95-099
Implementation Document No.:	EDC 2F01244
USAR Affected Pages:	N/A
System:	Recirculation Control (RCS)
Title of Change:	Temporary Revision of RCS Flow Control Valve 2RCS*HYV17B Control Circuit

Description of Change:

This temporary design change provided a means to control the repositioning of recirculation flow control valve (FCV) 2RCS*HYV17B during the period it was hydraulically locked up. The position feedback signal had been "noisy" and caused the valve to fluctuate intermittently. This in turn caused the reactor power to fluctuate. Testing indicated that the position element on the valve was the source of the "noise." Since the valve is in the primary containment, repairs could not be made to the position element until the plant was shut down. A previous safety evaluation (95-097) provided for temporarily locking the FCV in the open position until repairs were made. Implementation of this change periodically allows valve position changes and flow adjustments.

Safety Evaluation Summary:

Recirculation FCV 2RCS*HYV17B is being temporarily hydraulically locked up due to intermittent recirculation FCV fluctuations. This new temporary modification is an addition to allow periodic repositioning of the valve during the period that the valve is in this condition. The FCV control circuit will be modified to lift leads of the feedback elements that are responsible for injecting noise that is interfering with normal position control. A clean dc signal will be inserted in place which will allow the existing position controller to operate the valve in an open loop configuration. Existing valve position indication will still be operational. Additional operator action is required, because when the controller is manually operated in this way, the operator must also stop the valve manually when it reaches the desired position. Operating Procedure N2-OP-29 has also been revised to include instructions for operation in this condition.

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

Safety Evaluation No.:	95-102 Rev. 0 & 1
Implementation Document No.:	Procedures NSAS-POL-01, GAP-POL-01, NEP-POL-0101, NIP-FPP-01, NIP-TQS-01, GAP-OPS-01
USAR Affected Pages:	9A.3-1, 9A.3-2, 13.1-3, 13.1-4, 13.1-5, 13.1-6, 13.1-7, 13.1-11, 13.1-12, 13.1-13, B.1-3; Table 13.5-1 Sh 2, B-1 Sh 1; Figures 13.1-1, 13.1-2, 13.1-3, 13.1-5
System:	N/A
Title of Change:	Restructuring of Nuclear SBU in Accordance with Revised Procedures NSAS-POL-01, GAP-POL-01, NEP-POL-01, NIP-FPP-01, NIP-TQS-01, and GAP-OPS-01

Description of Change:

NSAS-POL-01, "Composition and Responsibility of the Nuclear Safety Assessment & Support Organization," GAP-POL-01, "Composition and Responsibility of the Nuclear Generation Organization," NEP-POL-01, "Nuclear Engineering Department Organization," NIP-FPP-01, "Fire Protection Program," NIP-TQS-01, "Qualification and Certification," and GAP-OPS-01, "Administration of Operations," have been revised to:

- Transfer responsibility for Office Administration activities from Nuclear Safety Assessment & Support (NSAS) Site Services to Business Management, and remove the Business Management organization from the Nuclear Generation Department (the General Manager Business Management reports directly to the Executive Vice President Nuclear).

- Transfer responsibility for Procurement and Integrated Planning functions from the Business Management Organization to the Engineering Department.

- "Unitize" the coordination of contractor maintenance/modification activities previously performed by NSAS Site Services and transfer responsibility for the functions to the Maintenance Branch at each unit.

- Transfer responsibility for administration and implementation of the Fire Protection Program from NSAS Technical Services to Unit 1 Operations.

Safety Evaluation No.: 95-102 Rev. 0 & 1 (cont'd.)

Description of Change: (cont'd.)

- Transfer responsibility for administration of Central Maintenance activities (a site function that includes M&TE calibration, security system support, material testing, and warehouse preventive maintenance) from NSAS Technical Services to Unit 2 Maintenance.
- Transfer responsibility for administration of Buildings & Grounds/Facilities Planning activities (a site function) from the NSAS Site Services to Unit 1 Maintenance.
- Abolish the positions of Manager Technical Services and Manager Site Services.
- Transfer responsibility for In-service Testing at Unit 1 from Operations to Maintenance.
- Consolidate the Unit 1 Operations Engineering and Planning Sections and combine with the Fire Protection Section (currently in NSAS Technical Services) into a new Operations Support Section to be headed by a new position, General Supervisor Operations Support.
- Assign I&C Technicians to Unit 2 Technical Support Branch-Lead Performance Engineer.
- Consolidate Unit 1 and Unit 2 Operations Training organizations into one common section under the supervision of the General Supervisor Operations Training.
- Assign responsibility for management of Site Relay & Control Testing activities (formerly a Corporate support function) to the Manager Maintenance Unit 2.
- Transfer responsibility for Maintenance Planning at Unit 2 from Maintenance to Work Control/Outage.

Safety Evaluation Summary:

These procedure changes establish departmental responsibilities and lines of authority, responsibility, and communication within the Nuclear SBU. The proposed organizational structure satisfies the criteria of SRPs 9.5.1, 13.1.1, and 13.1.2-13.1.3, and conforms with the requirements of Section 6.2 of the Plant

Safety Evaluation No.: 95-102 Rev. 0 & 1 (cont'd.)

Safety Evaluation Summary: (cont'd.)

Technical Specifications. The proposed changes do not impact accident or malfunction initiation or consequences.

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

Safety Evaluation No.:	95-106
Implementation Document No.:	N/A
USAR Affected Pages:	Figure 1.2-1
System:	N/A
Title of Change:	Demolition of Temporary Structures Inside the Protected Area, East of the Unit 2 Structures

Description of Change:

This safety evaluation addresses the demolition of the following buildings located east of the Unit 2 plant structures.

1. Carpenters' shop
2. Paint shop
3. Electric fab shop

All of these buildings were built for use as temporary buildings during the construction of Unit 2. These buildings have been demolished and activities consolidated within the remaining buildings.

Safety Evaluation Summary:

All of the buildings to be demolished are located in an area that was not used as a flow channel for the Probable Maximum Precipitation analysis. Removal of these buildings and the consequent reduction in the runoff coefficient would make the analysis more conservative. These buildings have no impact on the previously calculated X/Q values. The design margins for the Control Room fresh air intakes are not compromised. Location of 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 these changes do not involve an unreviewed safety question.

Safety Evaluation No.:	95-107
Implementation Document No.:	Procedure NTP-TQS-102
USAR Affected Pages:	13.2-9; Table 1.9-1 Sh 7 & 53
System:	N/A
Title of Change:	NTP-TQS-102, Licensed Operator Requalification Training Changes to Reflect the Requirements of the NRC Approved Systems Approach to Training Program

Description of Change:

This change more clearly defines a Systems Approach to Training (SAT)-based Licensed Operator Requalification Program. The SAT-based program allows flexibility in addressing identified weaknesses and current issues while satisfying required training specified in 10CFR55.

Safety Evaluation Summary:

Units 1 and 2 Licensed Operator training programs have been developed using a systems approach to training and are accredited by the National Nuclear Accrediting Board. Based on this certification and NRC approval, this change satisfies 10CFR55 requirements for Licensed Operator Requalification Training.

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

Safety Evaluation No.:	95-108
Implementation Document No.:	Procedure GAP-RPP-01
USAR Affected Pages:	13.2-10
System:	N/A
Title of Change:	10CFR19 Required Training for Personnel Outside the Restricted Area

Description of Change:

This change updated the USAR to agree with the present regulatory requirement to provide radiation protection training for personnel outside the site Restricted Area who would be likely to receive an occupational exposure in excess of 100 mRem/year.

Safety Evaluation Summary:

This revision to the USAR involves training for personnel outside the Restricted Area to comply with 10CFR19 requirements. As such, it has no impact on any aspect of plant equipment design, function, or operation. Additionally, the proposed training will provide at least an equivalent level of training for site personnel as currently described in the USAR.

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

Safety Evaluation No.:	95-126
Implementation Document No.:	Simple Design Change SC2-0014-95
USAR Affected Pages:	Figure 9.2-19a
System:	Turbine Building Closed Loop Cooling Water (CCS)
Title of Change:	Turbine Building Closed Loop Cooling Water Pump Stuffing Box Cooling Water Line Removal

Description of Change:

This simple design change replaced the existing packing from within the stuffing boxes of pumps 2CCS-P1B and P1C. The new self-lubricating and heat-conductive packing was recommended by NMPC Maintenance based on successful performance at other plants in similar applications. No cooling water was required for this packing, thus the stuffing box cooling water lines were removed and ports capped.

Safety Evaluation Summary:

This change will allow for the removal of the stuffing box cooling water lines and associated throttle valves. The remaining tapped ports will be plugged with an approved fitting. The change will not degrade the function of the pumps or the method in which they perform their function. This change will have no adverse impact on CCS system capabilities, normal operation, or flow path. This change is specific to the CCS system with no adverse interaction effects on other systems or components.

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

Safety Evaluation No.: 95-127

Implementation Document No.: ASTM E185

USAR Affected Pages: 5.3-7; Table 5A-4

System: N/A

Title of Change: Missing Neutron Dosimeter

Description of Change:

Unit 2 had a supplement neutron dosimeter located in the funnel of the reactor pressure vessel (RPV) surveillance specimen holder at the 3 degree azimuth. The dosimeter was to be used for fluence measurements to be made at the vessel inside diameter after the first fuel cycle. This measurement would have verified the predicted fluence at an early date in plant operation. The dosimeter was removed from the RPV to the spent fuel pool during the first refueling outage, but never sent to a vendor for fluence measurements. In April 1995, a visual inspection of the spent fuel pool area confirmed the neutron dosimeter was gone. Based on the root cause evaluation, the missing dosimeter was probably mistaken for a local power range monitor or control rod blade cutout parts and shipped along with those parts to Barnwell, SC.

ASTM E185 requires that: (a) dosimetry be included with surveillance capsules in the vessel, and (b) additional dosimetry need be included only if the capsule flux wires will saturate before the capsule is withdrawn.

The drawings on the NMP2 surveillance capsule contents show that there are two iron and two copper flux wires contained with the Charpy specimens. Therefore, the requirement of (a) above has been met. The requirement of (b) is also met because copper wires in the capsule to be withdrawn after approximately 10 years in operation will not saturate.

The USAR states that a separate neutron dosimeter will be used for fluence measurements during the first fuel cycle. This measurement will verify the predicted fluence at an early date in plant operation. However, there is adequate dosimetry data from similar vessels to evaluate the design basis flux without the first cycle dosimeter test results.

Safety Evaluation No.: 95-127 (cont'd.)

Safety Evaluation Summary:

The dosimeter does not interact with any equipment which could initiate or mitigate any accidents. Also, the existing vessel pressure-temperature limits are based on conservatively estimated fluence and will be accurately adjusted later in plant life. The conservatism of General Electric's estimated fluence has been verified by comparing measured fluence values with estimated fluence values at similar sister plants. The RPV surveillance capsules, which are currently in the vessel, are adequate to properly evaluate the brittlement fracture characteristics of the vessel long in advance of any chance of becoming a problem.

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

Safety Evaluation No.:	95-130
Implementation Document No.:	Nuclear Division Policy POL
USAR Affected Pages:	13.1-3, B.1-2; Figure 13.1-1
System:	N/A
Title of Change:	Reorganization; Change to Nuclear Division Policy to Reflect Establishment of the Corporate Officer Position "Executive Vice President-Generation Business Group/Chief Nuclear Officer"

Description of Change:

The Nuclear Division Policy "POL" has been revised to reflect the establishment of the corporate officer position Executive Vice President-Generation Business Group/Chief Nuclear Officer. This Executive Vice President reports directly to the Niagara Mohawk Power Corporation President. The Executive Vice President-Nuclear is subordinate to the Executive Vice President-Generation Business Group/Chief Nuclear Officer and continues to have overall responsibility for the administration and operation of the Nuclear SBU.

Safety Evaluation Summary:

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

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

Safety Evaluation No.: 95-131

Implementation Document No.: Simple Design Change SC2-0019-93

USAR Affected Pages: Figure 9.5-1g

System: Fire Protection Water (FPW)

Title of Change: Fire Protection Pressure Maintenance Supply

Description of Change:

This change utilizes the domestic water system as the primary source for makeup water to the fire protection pressure maintenance tank instead of the service water system. The service water system results in excessive silting in the pressure maintenance tank. In addition, the constant makeup process affects the Clamtrol treatment due to the dilution effect. The use of domestic water avoids these problems. The domestic water system was previously the secondary source for makeup water.

Safety Evaluation Summary:

The reliability and margin of safety equipment important to safety is not affected by this change. This change only affected the involved systems. This change will improve the availability of the fire protection water system by avoiding the present problems with pressure maintenance due to silting in the storage tank that provides water to the jockey pumps.

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

Safety Evaluation No.: 95-133

Implementation Document No.: Simple Design Change SC2-0082-94

USAR Affected Pages: Figure 5.4-16b

System: Reactor Water Cleanup (WCS)

Title of Change: WCS Pump Seal Flow Monitoring Improvement

Description of Change:

Modification 91-074 installed a seal purge supply from the control rod drive (CRD) to the WCS pump seals. The original change installed a seal valve and instrument station that could monitor flow to the individual pump seals. However, the original flow meters proved to be unreliable, continually sticking and not giving accurate readings. Existing pressure indicators were too small to provide accurate indication. Additionally, the temperature gauges, part of the original pump installation to give pump gland temperature readings, were small and difficult to read.

This simple design change removed the flow meters and installed a RO with a DP cell. The temperature gauges were replaced with a LED display. The new flow meter and pressure indicators required piping and tubing installation in the Secondary Containment el. 215, and the LED required instrument work in the Secondary Containment el. 289. Flow elements 2WCS-FI77A & B and 78A & B, pressure indicators 2WCS-PI64A & B and 65A & B, and temperature indicators 2WCS-TIS36A, B, C, & D were replaced. The flow and pressure indicators were mounted locally to the existing flow station.

Safety Evaluation Summary:

This change replaces flow, pressure, and temperature instrumentation enhancing the function of the WCS seal cooling water. In addition, it does not adversely impact the ability of the CRD to fulfill its design function.

The WCS system operates to remove minute particles from reactor coolant and thereby helps maintain reactor water chemistry levels within Technical Specification limits. This change will enhance WCS pump seal water flow indication. Therefore, this modification has no impact on Technical Specification 3/4.4.4.

Safety Evaluation No.: 95-133 (cont'd.)

Safety Evaluation Summary: (cont'd.)

The WCS ambient and differential temperature steam leak detection instrumentation is designed to detect a steam leak before break and minimize impact to offsite dose. The ambient temperature sensors detect increases in temperature due to leaks in rooms or areas where WCS piping is routed. The differential temperature sensors detect leaks by comparison of the temperature of the WCS pump and heat exchanger rooms to the temperature of the surrounding plant environment. An isolation signal is produced when the temperature differential exceeds the differential temperature setpoint. This change to the piping and instruments is designed to appropriate standards and will not create or increase the potential for any high temperature leaks. Therefore, no impact on the ambient and differential temperature Technical Specification limits will result.

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

Safety Evaluation No.: 95-134

Implementation Document No.: Mod. PN2Y94MX011

USAR Affected Pages: 9A.3-37, 9B.4-2, 9B.4-3, 9B.4-4; Tables 9.5-3 Sh 11, 9A.3-4 Sh 5 & 18, 9A.3-7 Sh 2 & 3, 9B.8-1 Sh 24, 38, 40, 9B.8-2 Sh 12; Figure 9A.3-5

System: Service Water (SWP), Control Building HVAC (HVC), Fire Protection

Title of Change: Abandonment/Removal of Thermo-Lag Fire Barriers

Description of Change:

This modification removed a fire barrier enclosure that was installed around the actuator for 2SWP*MOV50A and abandoned in place a fire barrier surrounding a short section of HVAC duct in the overhead of the Control Building corridor, Elevation 261'. Both these barriers were constructed of TSI Thermo-Lag 330-1 fire barrier material, which was identified by NRC Generic Letter 92-08 as potentially deficient if installed under the original specifications. In addition, this change removed an incorrect safe shutdown function designation of the 3-hour rated wall between the Control Building and the 115-kV switchyard, and added abandoned Thermo-Lag material to the Fire Hazards Analysis for Unit 2.

Safety Evaluation Summary:

Valve 2SWP*MOV50A is required to close in the event of a loss of offsite power (LOOP) to minimize the potential of hydrodynamic damage to the SWP system. The use of an assumption of a LOOP concurrent with a fire in the Unit 2 safe shutdown analysis resulted in the installation of a Thermo-Lag 330-1 enclosure for this valve's actuator. 10CFR50, Appendix R, Section III.G does not require the assumption of a LOOP in concurrence with a fire in any fire area. A LOOP is required for conformance with Section III.L of Appendix R. This does not affect service water pump room B where 2SWP*MOV50A is installed. Therefore, removal of the Thermo-Lag barrier from this valve will not result in noncompliance with Appendix R. In addition, previous evaluation of the valve has shown that even if this valve remains open in the event of a LOOP, the safety function of the SWP system can still be assured. The Thermo-Lag 330-1 barrier for the Control Building HVAC duct was installed to provide justification for not installing a fire damper at the air intake from the 115-kV switchyard side of the building. The missile-protected construction of the air intake provides sufficient assurance that a

Safety Evaluation No.: 95-134 (cont'd.)

Safety Evaluation Summary: (cont'd.)

fire originating in the switchyard would not propagate to the Control Building. The addition to the Fire Hazards Analysis of abandoned Thermo-Lag material does not compromise the analysis, and the removal of the safe shutdown designation of the Control Building outside wall does not result in any nonconformance to regulatory requirements.

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

Safety Evaluation No.: 95-135

Implementation Document No.: Simple Design Change SC2-0073-95

USAR Affected Pages: Figure 9.3-9a

System: Control Rod Drive (RDS), Reactor Building Equipment Drain (DER)

Title of Change: Upgrade Rupture Disc 2DER-PSE10A

Description of Change:

This simple design change installed a rupture disc, 2DER-PSE10A, rated at 30 psi burst pressure, in the Reactor Building drain (DER) system upstream of drain cooler 2DER-E2A. Its primary function was to burst with a high column of water in the control rod drive (RDS) hydraulic system scram discharge volume (SDV) drain, preventing high SDV level alarms and potential plant scrams. A secondary function was DER system pressure relief in the event the drain cooler was inadvertently isolated. Contrary to the above, the 30 psi rating was too high for two-phase flow conditions and bursts unnecessarily during post-scram reset evolutions. The subject rupture disc was replaced with a similar device with a 100 psi burst pressure; in addition, an isolation valve was installed upstream to facilitate replacement.

Safety Evaluation Summary:

All work associated with this change will be performed in the secondary containment elevations 175'-0" and 196'-0" in accordance with approved work control and radiation protection procedures. The constructibility aspects of this change have been reviewed, and appropriate work sequencing instructions included within the applicable design documentation and work orders. Construction aspects include isolating the reactor core isolation cooling (ICS) system and diverting RDS SDV drain flow. The RDS SRV drain flow diversion will bypass the existing cooler and drain tank (2DER-TK2A) and will be routed to tank 2DFR-TK2E. This will be accomplished by connecting a hose (minimum design requirements of 2" rated at 150 psi at 300°F) to the existing 2" pipe nipple at valve 2RDS-V2083. When the flow path to tank 2DFR-TK2E is established, valve 2RDS-V2083 will be opened, valve 2RDS-V2082 and/or 2RDS-V2084 will be closed, thus the alternate route established.

Safety Evaluation No.: 95-135 (cont'd.)

Safety Evaluation Summary: (cont'd.)

Temporary pipe rework required, and replacement back to the original design, will be controlled within the work order package and as described within the design change documentation. Temporary diversion of Reactor Building equipment drain effluent to the Reactor Building floor drain system has been approved and will be monitored by the Radwaste Department.

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

Safety Evaluation No.: 95-137

Implementation Document No.: DDC 2M10838, DDC 2E11017

USAR Affected Pages: Figures 1.2-20 Sh 2, 9.4-12b, 12.3-16, 12.3-49

System:

Title of Change: Abandonment of HVAC Equipment in East Switchgear Room

Description of Change:

This change abandoned HVAC equipment in the East Switchgear Room. The components consisted of two air handling units and a condensing unit with attached refrigerant tubing. This equipment was formerly used to resolve high room temperatures that were noted in the East Switchgear Room located on elev. 277'-6" of the Turbine Building. The air handling units are located within the East Switchgear Room, with the condensing unit located outside the Turbine Building. The refrigerant tubing is run through the Turbine Building to the outside of the building. The units were left in place since there is no technical or economic reason to remove the components.

Safety Evaluation Summary:

The abandonment of the components will not increase the probability of occurrence of an accident. The components will be in a de-energized state and will not have any attachment to permanent plant equipment. These components within the Turbine Building conform to plant standards for equipment installation. The condensing unit, located outside the Turbine Building, is currently considered sound and of no impact to other plant equipment. The elimination of the supplemental room cooling provided by the components will not impact the equipment located in the room.

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

Safety Evaluation No.: 95-139

Implementation Document No.: Procedure N2-OP-3

USAR Affected Pages: Figure 10.1-5d

System: Condensate (CNM)

Title of Change: Valve Position Indication

Description of Change:

There are two valves (2CNM-V366 and 2CNM-HV119) responsible for the continuous venting of the feedwater pump suction header. These two 3/4" globe valves were shown in the closed position in the USAR.

During plant startups, N2-OP-3 requires that 2CNM-V366 and 2CNM-HV119 be opened. Venting through these valves commences for about 30 seconds after water flow is heard and then both valves are required to be shut. 2CNM-HV119 is then opened, and 2CNM-V366 is throttled open approximately 1/2 turn for continuous venting of the feedwater pump suction header. By showing valves 2CNM-V366 and 2CNM-HV119 in the open position, the applicable design documents agree with the design intent and operating procedure. While at one point during the startup procedure these valves are closed, it is NMPC's standard to show valves in the USAR in their normal operating position.

Safety Evaluation Summary:

The result of leaving these valves in an open position will be to allow a minimal amount of flow from the feedwater system to the main condenser. This minimal amount of flow will neither reduce the feedwater systems' ability to provide makeup to the reactor pressure vessel (RPV) as required, nor reduce the main condensers' ability to provide a heat sink for the RPV as required.

Both valves are of a globe design and, therefore, will not be subject to the effects of steam-cutting when the pressurized water passes through them into a vacuum in the main condenser.

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

Safety Evaluation No.: 95-140

Implementation Document No.: Simple Design Change SC2-0056-95

USAR Affected Pages: Tables 3.9A-12 Sh 13 & 15, 5.2-1 Sh 10

System: Secondary Containment

Title of Change: Reactor Building Service Water Chemical Cleaning Penetrations

Description of Change:

This simple design change installed two Reactor Building penetrations through the outside wall of the north stair tower air lock to be used for chemical cleaning in the Reactor Building. This involved boring two holes in the 12-inch thick wall and installing penetration piping. Included in the penetration piping were two 6-inch in-line spring-loaded check valves located inside the air lock for automatic secondary containment isolation while cleaning was underway. Also, this change installed blind flanges both inside and outside the Reactor Building, the latter of which serves as the secondary containment barrier when the penetrations are not in use.

Safety Evaluation Summary:

The permanent piping throughout the Reactor Building wall was designed to penetrate a QA Category I boundary to maintain secondary containment integrity. During normal plant operation, secondary containment integrity will be maintained, per Technical Specification 4.6.5.1.b.3, by a safety-related blind flange on the outboard side of the penetration. When the penetrations are in use, secondary containment integrity will be maintained by redundant (series) safety-related spring-loaded check valves.

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

Safety Evaluation No.:	95-142
Implementation Document No.:	Calculation EC-151
USAR Affected Pages:	N/A
System:	Normal 13.8-kV Switchgear System (NPS), Reserve Station Service Transformer (RTX)
Title of Change:	Alternate Transfer Scheme for Feeding Normal 13.8-kV Buses from Either Reserve Transformer

Description of Change:

This safety evaluation evaluated the blocking of the fast transfer scheme for one normal switchgear when both normal switchgear 2NPS-SWG001 and 2NPS-SWG003 are configured for transfer to the same reserve station service transformer. When this is done, one switchgear is allowed to attempt a fast transfer and one attempts a slow transfer only.. This change has no effect on the two sources of power to each emergency bus.

Safety Evaluation Summary:

The fast and slow transfers transfer only nonsafety-related loads from the station service transformer to the offsite power sources. Safety-related loads are always fed from dedicated offsite power sources and no transfer is needed. Neither a fast or slow transfer has any effect on the availability or reliability of the offsite power system or its ability to supply the Class 1E loads. The blocking of one fast transfer scheme will have no effect on the ability of the Class 1E loads to separate themselves from the offsite power system upon degraded or loss of voltage. The safety-related equipment remains energized from the offsite source continuously and does not require fast transfer.

This change does not alter the design of the electrical distribution system and the requirements of General Design Criterion 17 are still being met as designed. Two independent offsite power sources are available and are always connected to onsite emergency power.

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

Safety Evaluation No.:	95-143
Implementation Document No.:	DER 2-94-1596
USAR Affected Pages:	8.3-10
Systems:	Emergency 600-V Distribution System, EJS, EHS
Title of Change:	Momentary Parallel Operation of Emergency 600-Volt Unit Sub Transformers

Description of Change:

This safety evaluation evaluated the momentary paralleling of the emergency load center transformers. Allowing the load center transformers to be operated in parallel allows the 600-V loads to remain energized when the supply to the load center is swapped.

Safety Evaluation Summary:

While the transformers are operated in parallel, the equivalent impedance of the parallel combination is approximately half of the rated impedance. This reduction in circuit impedance increases the maximum available fault current to a value greater than that of the breakers downstream of the unit sub.

A probable risk analysis has been performed to determine the probability of a fault occurring while the transformers are in parallel. An estimate of 10 parallel operations a year at 8 seconds per transfer was used to obtain a probability of 3.3E-8 per year. Operating the transformers in parallel does not increase the probability of a fault occurring.

It has already been postulated that an electrical fault could cause the loss of an entire safety division. Due to maintained separation and redundancy of divisional power, the other division would be unaffected. Therefore, any adverse consequences due to the transformers being operated in parallel are encompassed by this analyzed event. The momentary paralleling of the emergency load center transformers does not degrade the emergency 600-V distribution system.

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

Safety Evaluation No.:	95-144
Implementation Document No.:	Design Change N2-95-031
USAR Affected Pages:	Figures 10.1-4a, 10.1-4b
System:	Electrohydraulic Control (EHC), Main Steam (MSS), (DSM)
Title of Change:	Add Redundant Level Switches for Turbine Trip Logic

Description of Change:

This design change added a redundant level switch in the turbine trip logic circuit to be in series with each of 2DSM-LS70A/B, the two existing high water level trip switches for the moisture separators/reheaters (MSRs). As part of the effort to reduce the possibility of an inadvertent turbine trip and subsequent reactor SCRAM, several modifications have been performed to add redundancy where the failure of a single component could initiate an inadvertent SCRAM. Two Magnetrol level switches of the same model as the previous switches were locally piped in parallel and wired in series with the existing components.

Safety Evaluation Summary:

This change has no effect on the probability of an accident occurring, because this change is limited to adding redundant level switches to the existing turbine trip logic, and a turbine trip is not an accident initiator. This also does not increase the probability of a transient since the turbine trip logic will function exactly the same as before, except that there will be two float switches in series for each MSR in this branch of the trip logic where previously there was one. If a switch failed and initiated a false trip signal, the redundant series switch would prevent the false turbine trip. If one of the switches failed to trip when it should have tripped, it is the same as the failure scenario that the existing switch would fail to trip when needed. There are numerous other inputs to the turbine master trip bus which would act to take the turbine off line. Also, there are level alarms in the drain tanks below the MSRs that should alarm before the subject level switches would be required to actuate. The mechanical piping tie-in will duplicate what is already there for the existing switch. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.:	95-145
Implementation Document No.:	Procedure N2-TDP-IIT-0104
USAR Affected Pages:	5.2-15
System:	Main Steam (MSS)
Title of Change:	Reduce the Required Population of the Main Steam Safety Relief Valves (SRV) Requiring As-Found Testing During Refueling Outages

Description of Change:

The SRV surveillance program has been revised to allow testing of only that quantity of main steam SRVs required by ANSI/ASME OM-1-1981, as adopted by the NMP2 IST Program Plan. Using current OM-1 sampling, all the valves shall be tested every 5-year period with a minimum of 20% of the valves tested within any 24 months.

Safety Evaluation Summary:

The previous 50% SRV test criteria was one element of a program implemented in an effort to reduce challenges and failures of boiling water reactor SRVs by an order of magnitude after the Three Mile Island accident. At that time, no Dikkers SRVs had seen operation, so an extremely conservative test program was adopted.

A historical evaluation of the performance of the Dikkers Model G471-6/125.04 SRV was performed. This performance history was compared to the performance of SRVs in service at the time that the 50% test criteria was implemented. The data shows that an order of magnitude performance improvement has been realized with the Dikkers SRV.

Testing of the Nine Mile Point Unit 2 SRVs will now be performed in accordance with the OM-1 code accepted relief valve testing sampling plan. The OM-1 code is adopted by the NMP2 IST Program Plan.

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

Safety Evaluation No.: 95-146

Implementation Document No.: Design Change N2-96-002

USAR Affected Pages: Figure 9.3-1d

System: Automatic Depressurization (ADS), Instrument Air (IAS)

Title of Change: ADS Nitrogen Tanks 2IAS*TK4 and 2IAS*TK5 Drain Redesign

Description of Change:

The ADS system provides for automatic depressurization of the reactor vessel due to small breaks or leaks in the reactor coolant system. ADS operates as a backup to the operation of the high-pressure core spray system (CSH) and is designated to provide a dedicated source of compressed nitrogen to actuate 7 of the 18 safety/relief valves (SRV). Supply tanks 2IAS*TK4 and 2IAS*TK5 each provide nitrogen to a group of ADS SRV air accumulators which in turn provide nitrogen to the pneumatic actuator of the associated SRV. This design change modified the drain from tank 2IAS*TK4, shortening the tail piece and fitting with a closure device (i.e., coupling with threaded plug). The new design was constructed using 600 psi minimum pressure class components from tank to cap.

Safety Evaluation Summary:

This design change will not impact the operation of the ADS system as required by Technical Specification 3.5.1. The change enhances the drain line from each nitrogen supply tank, preventing leakage during normal operation and ensuring compliance with Technical Specification leakage requirements by providing positive shutoff capability. A capped closure device helps ensure leak-tightness and is easily removed and maintained. The drain lines are used for tank blowdown and to satisfy system leak rate surveillance requirements in accordance with Technical Specification 4.5.1.e.2.e, and check valve forward flow exercise testing in accordance with Technical Specification 4.0.5. The reconfigured drain lines may still be used for blowdown by connecting a threaded fitting with hose and routing the hose to the drain hub. The new double isolation valve design with closure device will ensure system design pressure. Both drain lines in their entirety will be upgraded to a 600 psi pressure class. The piping changes shall be designed and installed in accordance with the original piping design and installation specifications and code requirements. The new configuration will be seismically analyzed and supported in accordance with safe shutdown earthquake design requirements.

Safety Evaluation No.: 95-146 (cont'd.)

Safety Evaluation Summary: (cont'd.)

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

Safety Evaluation No.:	96-002
Implementation Document No.:	Procedure NIP-FPP-01
USAR Affected Pages:	9A.2-4, 9A.3-2, 9A.3-3, 9A.3-4, 9A.3-30, 13.2-20
System:	N/A
Title of Change:	Fire Brigade Membership Requirements and Revision of NIP-FPP-01

Description of Change:

Procedure NIP-FPP-01 has been revised to allow the site fire brigade to be staffed by personnel trained and qualified per the fire brigade training program. The requirement that the brigade leader and the least two members be fire protection staff personnel is being deleted. This evaluation examined the requirements for fire brigade membership and the staff which may be qualified for membership in the fire brigade. Previously, the fire brigade leader and two of the fire brigade members were required to be part of the fire protection staff. This change allows plant staff members who are qualified in accordance with the fire brigade training program to serve as fire brigade members at the level to which they are assigned.

Safety Evaluation Summary:

Niagara Mohawk Power Corporation has traditionally staffed the fire brigade at Nine Mile Point with "professional" firefighters, based on the concept that personnel assigned to the fire brigade were dedicated to fire protection duties. In 1994, the composition of the fire brigade was modified to allow two of the fire brigade members to be non-fire protection staff personnel. Part of the philosophy for that modification was that each fire attack team could still have one full-time fire protection staff member ("professional" firefighter) assigned to lead the fire hose attack in fire suppression activities. As these teams consisting of fire protection and non-fire protection staff personnel have practiced as teams and matured as fire brigade members, it has become apparent that non-fire protection personnel can perform fire suppression activities effectively, given adequate training and practice sessions. Therefore, the fire brigade membership requirements have been revised to allow any individual receiving adequate training and practice to be assigned to the fire brigade. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.:	96-010
Implementation Document No.:	Procedure NEP-POL-01
USAR Affected Pages:	Figure 13.1-3
System:	N/A
Title of Change:	Restructuring of Unit 1 Engineering in Accordance with Revised Procedure NEP-POL-01

Description of Change:

Procedure NEP-POL-01, "Nuclear Engineering Department Organization", has been revised to reflect organizational changes in Unit 1 Engineering. The Unit 1 Plant Evaluation group, consisting of a supervisor and one engineer, was merged with the Unit 1 Project Management group. The Supervisor - Plant Evaluation position has been eliminated. The individuals in the Plant Evaluation group now report to the Unit 1 Supervisor - Project Management.

Safety Evaluation Summary:

The proposed procedure change revises lines of authority within the Unit 1 Engineering Branch. The proposed organizational structure satisfies the criteria of Standard Review Plan 13.1.1. The proposed changes do not impact accident or malfunction initiation or consequences.

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

Safety Evaluation No.:	96-026
Implementation Document No.:	Mod. PN2Y95MX013
USAR Affected Pages:	3.9A-33, 3.9A-34, 3.9B-30, 9A.3-5a, 9A.3-40, 12.4-3; Tables 1.8-1 Sh 46 & 47, 5.2-1 Sh 5 & 8, 9A.3-1 Sh 1-9, 9A.3-2 Sh 1-3 & 3a, 9A.3-3 Sh 1, 9A.3-4 Sh 10, 9A.3-6 Sh 7 & 11, 9A.3-7 Sh 2 & 3, 9A.3-8 Sh 4, 9A.3-12 Sh 1
System:	Various
Title of Change:	Allow the Installation of LISEGA Sealed Hydraulic Snubbers as "Drop-In" Replacements for Mechanical Snubbers

Description of Change:

The snubbers installed at NMP2 were Pacific Scientific mechanical snubbers. Due to the number of failures and subsequent number of additional snubbers to be tested as part of RF04 snubber functional testing, a change was initiated for the installation of LISEGA sealed hydraulic snubbers as replacements for mechanical snubbers. The primary causes of failure of mechanical snubbers at NMP2 have been vibration, dried grease and corrosion. The LISEGA snubbers are less susceptible to these causes due to their design and the materials they use. Therefore, their failure rate is projected to be significantly lower than that of the mechanical snubbers.

This modification revised the applicable design and installation specifications to generically allow the replacement of any mechanical snubber in the plant with an equivalent LISEGA hydraulic snubber in the future.

Safety Evaluation Summary:

The LISEGA sealed hydraulic snubbers meet all of NMP2 licensing and design requirements. They are designed, fabricated, analyzed, tested and certified to meet the requirements of the ASME III Code editions and addenda applicable to NMP2. The different weights and stiffnesses of the LISEGA hydraulic snubber and the Pacific Scientific mechanical snubbers produce similar and acceptable pipe stress analysis results. Clearance requirements in NMP2 specification P301F prevent the installation of the larger diameter hydraulic snubbers from resulting in unacceptable clearances. LISEGA hydraulic snubbers have been tested and qualified extensively for various environmental and dynamic conditions as

Safety Evaluation No.: 96-026 (cont'd.)

Safety Evaluation Summary: (cont'd.)

applicable. Operating experience at other nuclear power plants shows that the LISEGA sealed hydraulic snubbers are less susceptible to failure than the mechanical snubbers currently used at NMP2. The LISEGA sealed hydraulic snubbers installation manual, the implementation of the revised maintenance procedure, and the revised design specification satisfy the requirements of Technical Specification Surveillance Requirement 4.7.5.f.

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

Safety Evaluation No.:	96-027
Implementation Document No.:	DDC 2M10841, DDC 2E11032
USAR Affected Pages:	Tables 3.9A-1, 3.10A-1
System:	Reactor Core Isolation Cooling (ICS)
Title of Change:	Replace Actuator and Internals for 2ICS*PCV115

Description of Change:

This change replaced the electrohydraulic actuator and valve internals for 2ICS*PCV115 with an air-operated actuator and DRAG 100 control valve internals. This change also removed/disconnected some of the power and control cables for this valve. Also, it required the instrument air system to be connected to the actuator. The design function of the valve, supplying cooling water to the lube oil cooler for the ICS system, is adequately performed by an air-operated actuator and the DRAG 100 control valve.

Safety Evaluation Summary:

This change will not impact the safe operation of the plant by replacing the valve 2ICS*PCV115 actuator and internals. This valve is assumed to be open in plant accident analysis and this change does not alter that assumption or any operational modes for this valve.

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

Safety Evaluation No.: 96-028 Rev. 0 & 1

Implementation Document No.: DDC 2F01328

USAR Affected Pages: 10.4-19, 10.4-19a; Figures 10.4-7d, 10.4-8 Sh 8

System: Circulating Water (CWS)

Title of Change: Cooling Tower Bypass Gate Logic Changes

Description of Change:

The cooling tower bypass gates open automatically on low basin temperature or can be opened manually from the Control Room. The automatic control incorporates two-out-of-four logic using the four temperature elements located in the cooling tower basin. Failed detectors introduce the potential for spurious opening of the gates and plant shutdown.

This change modified the licensing basis description of the CWS to allow temporary or permanent use of logic other than the described two-out-of-four. Two-out-of-four remains the normal logic configuration since all of the temperature monitors have been repaired; however, other configurations are acceptable depending on the operability and reliability of the four temperature detectors. This includes the option of disabling power to the gates in order to maintain their position. Changes to the logic are controlled and approved in accordance with current design control procedures.

Safety Evaluation Summary:

Removal of faulty detectors from the automated logic increases system reliability by reducing the susceptibility to erroneous opening of the gates. In the event that basin water temperature does drop below 40°F, the remaining temperature element(s) will function to place the cooling tower in the bypass mode. Redundancy for opening the gates when temperature is actually low is sacrificed, but is of no consequence since the original design provides for a manual override. Disabling bypass gate automatic response to low basin temperature is not a concern since control room annunciation and computer points would remain functional. The proposed logic changes will have no impact on the discharge flume temperature indicator or the computer points for the basin temperature elements. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.: 96-029

Implementation Document No.: Simple Design Change SC2-0398-91

USAR Affected Pages: Figures 1.2-9 Sh 1, 12.3-9, 12.3-42

System: Standby Liquid Control System (SLS)

Title of Change: SLS Tank Platform Extension at Elevation 303'-6"

Description of Change:

This simple design change added a permanent work platform (top of grating, elevation 303'-6") attached to the north side of the existing standby liquid control tank (2SLS*TK1) platform located in the Secondary Containment of the Reactor Building, at floor elevation 289'-0", at azimuth 180°. This extension was added not only to resolve a problem concerning long-term use of temporary scaffolding, but also to improve general accessibility to the top of the storage tank. The extension consists of safety-related structural members, and is seismically designed due to the various safety-related components in the immediate vicinity of the original platform and the new extension.

Safety Evaluation Summary:

The SLS storage tank platform and the additional platform extension are designed and analyzed as safety-related structures. The combined platform structure is formally evaluated using normal and seismic loading conditions, and is determined to be adequate during a seismic event. During the construction activity for the installation of the structural steel, the protective measure of imposing the rigging scheme based on a 10:1 margin of safety precludes the potential of a load drop that would result in damage to safety-related equipment associated with the SLS system. As an added measure of protection, scaffolding shall be placed on the east side of the SLS tank platform, beginning near the north end of the hoistway and continuing around to the north side of the tank. Additional scaffolding shall extend north (as far as needed) as added protection for the safety-related components below. The platform changes do not modify any of the equipment malfunctions or procedural errors that are analyzed as accident initiators in the USAR.

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

Safety Evaluation No.: 96-031

Implementation Document No.: Simple Design Change SC2-0026-96

USAR Affected Pages: Table 3-3 Sh 2 & 3 and Figure 5-2 of Appendix C

System: MHR

Title of Change: Expanded Use of 25-Ton Auxiliary Hoist

Description of Change:

This simple design change allows the use of the 25-ton auxiliary hoist as an alternate to the 132-ton polar crane main hoist for the lifting of loads 12.5 tons or less, including the items listed below, to improve personnel safety and efficiency of the rigging activities associated with these lifts:

1. 10-ton spent fuel pool cooling (SFC) filter removal plugs
2. 4-ton SFC filter pump plug
3. 1.2-ton removal plates
4. 1.2-ton decontamination boom
5. 1.5-ton fuel inspection stand

Safety Evaluation Summary:

The implementation of this change will result in improved personnel safety and substantially reduce the rigging time required for these lifts which also improves ALARA considerations. The allowed use of this alternate hoist for the lifts specified above is in accordance with the heavy load commitments and single-failure-proof criteria as indicated in USAR Appendix 9C and referenced in NUREG-0612.

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

Safety Evaluation No.:	96-032
Implementation Document No.:	Design Change N2-95-026
USAR Affected Pages:	Tables 3.9A-12 Sh 13, 8.3-1 Sh 12 & 13, 8.3-2 Sh 11 & 12, 8.3-4 Sh 6 & 13, 8.3-5 Sh 1, 3, 4, 8.3-6 Sh 1-5
System:	Reactor Water Cleanup (WCS)
Title of Change:	Motor Replacement and Gear Set Change for 2WCS*MOV102 & 2WCS*MOV112

Description of Change:

NRC closure of the NMP2 Generic Letter 89-10 program changed the motor-operated valve (MOV) sizing calculation methodology. The new methodology increased design margin necessary to address degraded conditions, rate of loading, and diagnostic test equipment errors.

This design change added a new motor and gear set to 2WCS*MOV102 and 2WCS*MOV112 to permit sufficient thrust and torque to be developed to close the MOV under design basis events, to meet the required stroke time, and to provide two torque switch settings to allow setup and testing of the MOV. The overall gear ratio changed from 36.99 to 72.01 and the motor from 80 ft-lb, 1800 rpm, 5.2 hp to 60 ft-lb, 3600 rpm, 7.8 hp. The calculated stroke time decreased by 0.3 seconds.

Safety Evaluation Summary:

Motor-operated valves 2WCS*MOV102 and 2WCS*MOV112 are required to close within 14 seconds upon receipt of an automatic signal in response to a design basis event. Adding a new motor and gear set to these MOVs will provide additional thrust and torque capability to account for NRC concerns associated with Generic Letter 89-10 and still meet the 14-second stroke time requirement. The additional capability will account for design requirements, error analyses, diagnostic testing measurement errors, degraded operator condition and rate of loading associated with the design and operation of MOVs.

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

Safety Evaluation No.: 96-033

Implementation Document No.: Procedure N2-OP-61A

USAR Affected Pages: Figure 9.3-20a

System: Nitrogen (GSN)

Title of Change: Revise Figure Drawing 9.3-20a

Description of Change:

Valves 2GSN-V19A/B, 2GSN-V24A through D and 2GSN-V26A through D were previously shown in the open position in the USAR. The open position corresponds to inerting the primary and does not agree with the normal operational mode of the system.

The normal operational mode for valves 2GSN-V24A through D and 2GSN-V26A through D is to be in the closed position to isolate electric vaporizers 2GSN-EV2A through 2D. The normal operational mode for valves 2GSN-V19A/B is to be in the closed position to prevent liquid nitrogen from vaporizing in supply lines due to ambient air temperature. The vaporizing in nitrogen supply lines complicates nitrogen storage tank pressure control.

Safety Evaluation Summary:

The changes to show closure of valves simply reflect the proper valve lineup for the system to perform its original design intent. This system does not act as an accident precursor or initiator and, therefore, this change will not increase the probability of occurrence of an accident previously evaluated in the SAR. The original intent is to have the valves closed during normal operation and open only if high flow inert is in progress after an outage, which is the normal operational mode.

The drawing change related to nonsafety-related components of the GSN system showing the valve position from open to closed will make the applicable design documents agree with the original design intent and the current practice of operating procedure N2-OP-61A.

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

Safety Evaluation No.: 96-036

Implementation Document No.: Calcs. PX-70155, PX-70156

USAR Affected Pages: 3.7A-26, 3A-1, 3A-2, 3A.15-1, 3A.15-2, 3A.19-1; Figures 3A.19-1 (Deleted), 3A.19-2 (Deleted)

System: N/A

Title of Change: Change USAR Section 3.7 and Appendix 3A Computer Program Information and Allow use of a Personal Computer Version of NUPIPE

Description of Change:

This safety evaluation addresses two issues: 1) use of a personal computer (PC) version of NUPIPE for piping analysis, and 2) updates to USAR Section 3.7 and Appendix 3A, "Computer Programs for Dynamic and Static Analysis of Category I Structures, Equipment and Components" (SSCs).

Seismic Category I SSCs are qualified for static and dynamics design conditions by testing or analysis, or by a combination of both. A PC-based version of the NUPIPE computer program is now used as an alternate to the mainframe versions previously valid for piping analysis at Unit 2.

Revisions to USAR Section 3.7 and Appendix 3A updated valid versions of the piping analysis computer program NUPIPE and deleted references to the computer program PSPECTRA.

Safety Evaluation Summary:

Changes in analytical results from computer programs used to evaluate piping and structural components could change locations of assumed pipe breaks and related accidents described in the USAR. The use of NUPIPE-SW, Versions 04, 05, and 06, and NUPIPE-SWPC, Version 00, will not significantly change these analytical results. Removing unnecessary references to NUPIPE-SW version levels and the computer program PSPECTRA from the USAR is an administrative change and will not have any effect on accident initiators described in the USAR. Since the results from new or PC versions of NUPIPE are insignificantly different from the existing

Safety Evaluation No.: 96-036 (cont'd.)

Safety Evaluation Summary: (cont'd.)

evaluations, and reference changes will not modify accident initiators, these proposed changes will not increase the probability 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.: 96-037

Implementation Document No.: Design Change N2-95-020

USAR Affected Pages: Table 6.2-56 Sh 1

System: Main Steam (MSS)

Title of Change: Gear Set Change for 2MSS*MOV208

Description of Change:

Valve 2MSS*MOV208 is a 2-inch, normally closed outboard containment isolation valve on the MSS drain line to the condenser. Its safety function is to close upon an automatic isolation signal to provide containment isolation during a design basis event. The valve is normally closed and is only opened during plant startup and cooldown.

As a result of changes to the NMP2 motor-operated valve (MOV) sizing calculation methodology required for closure of Nuclear Regulatory Commission Generic Letter 89-10, the sizing calculation for 2MSS*MOV208 was revised. Due to increased design margin necessary to address uncertainty due to degradation, rate of loading, and measuring and test equipment errors, the design window shifted, resulting in the as-left torque switch setting of 2MSS*MOV208 being outside the new design window.

In order to comply with the revised sizing calculation, a gear set change and spring pack change were required for 2MSS*MOV208 to produce sufficient thrust to function under design basis conditions (considering sufficient design margin). This design change installed a new gear set and spring pack to the valve operator to permit sufficient thrust and torque to be developed to close the MOV under design basis conditions in order to meet the required stroke time and to provide margin for setup and testing of the MOV. The overall gear ratio was changed from 47.85 to 75 and spring pack 0101-092 replaced spring pack 0101-091. The calculated valve opening and closing stroke time increased from approximately 9 seconds to approximately 12 seconds.

Safety Evaluation Summary:

The increased output thrust/torque of the modified actuator is sufficient to close the MOV under design basis conditions in order to meet the required stroke time and to provide margin for setup and testing of the MOV, without adversely affecting the safety function of the valve or the MSS system. This valve is

Safety Evaluation No.: 96-037 (cont'd.)

Safety Evaluation Summary: (cont'd.)

assumed to be closed in plant accident analyses and this change does not alter that assumption or any operational modes for this valve.

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

Safety Evaluation No.:	96-038
Implementation Document No.:	Calc. EC-137 Rev. 1
USAR Affected Pages:	8.2-9, 8.2-11, 8.3-3, 8.3-4, 8.3-4a
System:	Normal 13.8-kV Switchgear System (NPS), Reserve Station Service Transformer (RTX)
Title of Change:	Fast Transfer Scheme for Feeding Normal 13.8-kV Buses from Either Reserve Transformer

Description of Change:

This change configured normal switchgear 2NPS-SWG001 and 2NPS-SWG003 for automatic fast transfer to one reserve station service transformer. Both switchgears are now able to attempt a fast transfer upon loss of normal station service transformer following a turbine/generator trip. This change has no effect on the two sources of power to each emergency bus 2ENS*SWG101 and 2ENS*SWG103.

Safety Evaluation Summary:

At Unit 2, the fast transfer scheme within 6 cycles will transfer only nonsafety-related loads from the station service transformer to the offsite power sources. Safety-related loads are always fed from dedicated offsite power sources and no transfer is needed. A fast transfer has no effect on the availability or reliability of the offsite power system or its ability to supply the class 1E loads. Allowing fast transfer of nonsafety-related loads to a single reserve station service transformer will have no effect on the ability of the Class 1E loads to separate themselves from the offsite power system upon degraded or loss of voltage. The nonsafety-related motor may be stressed by higher volts/HZ. The largest nonsafety-related feedwater pump motor will be inspected at the next earliest outage.

This change does not alter the design of the electrical distribution system and the requirements of GDC 17 are still being met as designed. Two independent offsite power sources are available and always connected to onsite emergency power.

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

Safety Evaluation No.: 96-039

Implementation Document No.: Design Changes N2-95-014, N2-95-017, N2-95-018

USAR Affected Pages: 6.2-106; Figures 5.4-9a, 6.3-6a

System: High Pressure Core Spray (CSH), Reactor Core Isolation Cooling (ICS)

Title of Change: Pressure Locking Bonnet Vent

Description of Change:

Pressure locking is a term that describes the occurrence of high pressure in the bonnet of a closed gate valve relative to upstream and downstream system pressures. This high bonnet pressure wedges the valve discs more tightly on their seats so that more thrust is required for the valve to open. The causes of pressure locking are generally either thermal expansion of an incompressible fluid (e.g., water) in the bonnet or rapid depressurization of the system which traps initial system pressure in the bonnet. These design changes utilized existing bonnet connections on valves 2ICS*MOV136 and 2CSH*MOV118 to install bonnet vent lines, which connected the bonnet with the system piping to continuously relieve the high bonnet pressure. Valve 2CSH*MOV105 did not have a bonnet connection which could be used to relieve the high pressure in the bonnet; therefore, a hole was drilled into the downstream side of the flex disc for bonnet pressure relief.

Safety Evaluation Summary:

Valve bonnet vent pathways will be installed between the motor-operated valve (MOV) bonnet and the system piping in order to relieve high pressure in the bonnet and allow the MOV to open on demand. The bonnet vent pathways will be designed and installed in accordance with the required specifications, procedures, and ASME III Code to ensure system piping integrity. The bonnet vent pathways will connect the MOV bonnet with the system piping such that the bonnet vent pathways will not bypass the disc seat used for MOV isolation. This will ensure that applicable requirements will be met to maintain primary containment integrity.

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

Safety Evaluation No.: 96-040

Implementation Document No.: Design Change N2-95-016

USAR Affected Pages: 6.3-11

System: High-Pressure Core Spray (CSH)

Title of Change: Gear Set Change for 2CSH*MOV107

Description of Change:

Motor-operated gate valve 2CSH*MOV107 is a 12-inch, normally closed outboard containment isolation valve located as close as practical to the high-pressure core spray discharge line penetration into the containment. Its safety function is to open following receipt of an automatic initiation signal (High Drywell Pressure and/or Reactor Water Level 2) to provide makeup water to the reactor vessel and to close automatically when reactor water level is restored (Reactor Water Level 8). Remote manual operation of the valve is possible at all times.

As a result of changes to the NMP2 motor-operated valve (MOV) sizing calculation methodology required for closure of NRC Generic Letter 89-10, the sizing calculation for 2CSH*MOV107 was revised. Due to increased design margin necessary to address uncertainty due to degradation, rate of loading, and measuring and test equipment errors, the design window shifted, resulting in the as-left torque switch setting of 2CSH*MOV107 being outside the new design window.

This design change installed a new gear set to the valve operator to permit sufficient thrust and torque to be developed to open and close the MOV under design basis conditions in order to meet the required stroke time and to provide margin for setup and testing of the MOV. The overall gear ratio changed from 50.02 to 61.5.

Safety Evaluation Summary:

This design change will not adversely impact the performance of valve 2CSH*MOV107 with respect to performing its safety function as a containment isolation valve, or its ability to open and close following receipt of an initiation signal. This change will increase the actuator's thrust/torque output capacity and

Safety Evaluation No.: 96-040 (cont'd.)

Safety Evaluation Summary: (cont'd.)

increase the valve calculated stroke time from approximately 10.1 seconds to approximately 12.4 seconds.

This design change will have no impact on the valve ASME III Class 1 pressure boundary. All installation work will be performed on the actuator itself outside of the ASME pressure boundary.

In accordance with the NMP2 Appendix J Program Plan, an engineering evaluation has determined that no as-found or as-left local leak rate testing is required due to this design change. The increased thrust/torque generated by the valve will tend to seat the valve more tightly, thereby not adversely affecting previous leak rate test results. The increased torque/thrust has been limited by design and torque switch setting to prevent overstressing of the valve and/or actuator.

The increased margin provided as a result of performing this design change allows for margin in testing and setup of the valve while ensuring that it will meet its safety function requirements for all design conditions.

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

Safety Evaluation No.: 96-041

Implementation Document No.: Dwg. EE-4CB Rev. 6

USAR Affected Pages: 7.7-32

Systems: Performance Monitoring (PMS), CEC

Title of Change: Audible Alarm, Panel 2CEC-PNL800A

Description of Change:

This safety evaluation evaluated the current as-built (disconnected) configuration of the PMS audible alarm, located on panel 2CEC-PNL800A, to permit the audible alarm to remain disconnected.

The purpose of the PMS alarm features is to provide Operators with information on plant status. Printers, video displays, and audible alarms are devices used to provide Operator notification of an abnormal condition. The as-built condition of the audible alarm has the wires to the alarm disconnected, thereby disabling the audible tone generator.

Electrical configuration documents were revised to represent the current as-built configuration.

Safety Evaluation Summary:

The PMS audible alarm located on panel 2CEC-PNL800A does not provide control functions and is not connected to any system that is used to mitigate an accident. Audible alarms from the PMS computer that are required for plant operation or accident mitigation are provided by the plant annunciator system. The audible alarm has been disabled for several years and is not used to alert Operators to actions required to mitigate accidents.

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

Safety Evaluation No.:	96-045
Implementation Document No.:	Simple Design Change SC2-0430-91
USAR Affected Pages:	Figure 7.7-2 Sh 1, 26, 28
System:	Control Rod Drive (RDS)
Title of Change:	Replace CRD Temperature Recorder 2RDS-TRS165

Description of Change:

This simple design change replaced the local panel-mounted control rod drive (CRD) temperature recorder (2RDS-TRS165) and panel internally-mounted input scanners with state-of-the-art recorder and scanners. The original components located in 2CES-RAK007 required excessive maintenance due to obsolete technology and unavailability of replacement parts. Implementation of this change required de-terminating the existing nine scanners and re-terminating on the three replacement scanners. Additionally, the obsolete recorder in the same panel was replaced with a new recorder, but the alarm function output is still wired to the Control Room annunciator and computer point to indicate CRD high temperature.

Safety Evaluation Summary:

The replacement components still provide the function of alarming on high temperature in the main Control Room. The new state-of-the-art recorder has the ability to monitor and alarm the entire range of anticipated CRD temperatures, as did the obsolete recorder which it replaced. There is no change in function or performance characteristics required of the replacement recorder, and implementing this change satisfies all temperature monitoring and alarming requirements. Implementation of this change has not created any difference in the function of providing the CRD high temperature alarm, whether due to CRD inadequate cooling water flow (which is the primary function) or assisting in detecting CRD mechanism failures. All applicable requirements are maintained for this change.

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

Safety Evaluation No.: 96-047

Implementation Document No.: N/A

USAR Affected Pages: 1.2-30; Table 9.3-1 Sh 1, 2, 3, 9

System: N/A

Title of Change: Elimination of Turbidity Analysis

Description of Change:

Turbidity measurements provide a qualitative assessment of suspended solids for a given water sample and do not quantify the amount of suspended solids that can be obtained from suspended solids analysis. The performance of turbidity analyses has been discontinued and suspended solids analyses have been designated as backup laboratory analyses for out-of-service turbidimeters.

Safety Evaluation Summary:

Suspended solids analyses provide a quantitative method for evaluating and measuring suspended solids as compared to turbidity, which provides a qualitative assessment of suspended solids. This change results in the elimination of turbidity as one of several parameters used to assess water quality and designates suspended solids analysis as an alternative to turbidity analysis. As a result, this change does not introduce new, or changes to, initiators or precursors to accidents 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.: 96-048

Implementation Document No.: Procedures N2-ISP-RTT-AT205, AT206; N2-ISP-ISC-R201, R202, R204; N2-ISP-MSS-R202, R204

USAR Affected Pages: Tables 7.2-3, 7.3-18, 7.3-19 Sh 1 & 2

System: Reactor Protection (RPS), Emergency Core Cooling (ECCS), Isolation Actuation (IAS)

Title of Change: Elimination of Selected Response Time Testing (RTT)

Description of Change:

This change eliminated RTT for sensors in RPS circuits, main steam isolation valve (MSIV) actuation logic circuits, and ECCS actuation instrumentation (sensors, trip unit, logic circuits) from the USAR. This change was in accordance with the Nuclear Regulatory Commission (NRC) safety evaluation for BWROG Topical Report NEDO-32291.

Safety Evaluation Summary:

The BWROG analysis provided the basis for the proposed elimination of RTT for sensors in the RPS and MSIV actuation logic circuits. ECCS actuation instrumentation RTT is also eliminated. Analysis demonstrated that other periodic tests required by Technical Specifications, such as channel calibrations, channel functional tests, and logic system functional tests (LSFT), in conjunction with actions taken in response to NRC Bulletin 90-01, "Loss of Fill-Oil in Transmitters Manufactured by Rosemount," and Supplement 1, provide adequate assurance that response times are within acceptable limits. The elimination of RTT would not involve a test or experiment not previously described in the USAR and has no effect on nuclear safety. The elimination of RTT provides an improvement to plant safety and operation by: a) reducing the time that safety systems are unavailable, b) reducing safety system actuations, c) reducing shutdown risk, d) limiting radiation exposure to plant personnel, and e) eliminating the diversion of key personnel to conduct unnecessary testing. No physical changes to the station are required.

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

Safety Evaluation No.: 96-050

Implementation Document No.: Procedure N2-ESP-RPS-T742

USAR Affected Pages: N/A

System: Reactor Protection (RPS), Reactor Building Closed Loop Cooling Water (CCP), Containment Monitoring (CMS), Reactor Water Cleanup (WCS), Instrument Air (IAS)

Title of Change: Procedure for the Repair of EPA 2VBS*ACB2A

Description of Change:

This change temporarily installed jumpers around electrical protection assembly (EPA) 2VBS*ACB2A to allow replacement of the EPA without disrupting RPS power normally supplied through the EPA in accordance with the above-referenced procedure. In addition, during the installation of the jumpers, selected Division I containment isolation valves were kept open by administrative controls.

Safety Evaluation Summary:

Power for the RPS trip channels is supplied by uninterruptible power supply (UPS) 2VBB-UPS3A through EPAs 2VBS*ACB1A and 2VBS*ACB2A connected in series. The purpose of having two EPAs in series is to provide Class 1E isolation capability for the RPS power supplies due to nonsafety-related UPS.

EPA 2VBS*ACB2A tripped and failed to reset on the first try. The second reset attempt was successful. It is suspected that the circuit breaker associated with EPA 2VBS*ACB2A caused the trip. In order to replace the EPA circuit breaker while the plant is on line without interrupting RPS power, jumpers will be temporarily installed across the EPA. As a precautionary step while the jumpers are being installed, the selected containment isolation valves will be administratively controlled to ensure they stay open should the EPA inadvertently trip during jumper installation. While the administrative controls are in place, the valves are inoperable and the associated Limiting Conditions of Operation (LCO) will be entered. However, the redundant isolation valves will be available to provide the isolation function should it be needed. For additional assurance, compensatory action of stationing Operators at breakers and overriding switches will be taken for those valves that are defeated in the open position. These compensatory measures provide additional assurance that primary containment isolation will be obtained under accident conditions.

Safety Evaluation No.: 96-050 (cont'd.)

Safety Evaluation Summary: (cont'd.)

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

Safety Evaluation No.:	96-051
Implementation Document No.:	Design Changes N2-95-021, N2-95-022
USAR Affected Pages:	Table 3.9A-12 Sh 9; Figures 5.4-13a, 5.4-13b
System:	Residual Heat Removal (RHS)
Title of Change:	Pressure Locking Bonnet Vent for 2RHS*MOV15A/B

Description of Change:

Pressure locking is a term that describes the occurrence of high pressure in the bonnet of a closed gate valve relative to upstream and downstream system pressures. This high bonnet pressure wedges the valve discs more tightly on their seats so that more thrust is required for the valve to open. The causes of pressure locking are generally either thermal expansion of an incompressible fluid (e.g., water) in the bonnet or rapid depressurization of the system which traps initial system pressure in the bonnet. These design changes utilized existing bonnet connections on valves 2RHS*MOV15A/B to install bonnet vent lines, which connected the bonnet with the system piping to relieve the high bonnet pressure. These motor-operated valves (MOV) are primary containment isolation valves and are leak tested to verify isolation integrity. The leak test methods were not affected.

Safety Evaluation Summary:

Valve bonnet vent pathways will be installed between the MOV bonnet and the system piping in order to relieve overpressurization of the bonnet and allow the MOV to open on demand. The bonnet vent lines will be designed and installed in accordance with the required specifications and ASME III Code to ensure system piping integrity. The bonnet vent lines will connect the MOV bonnet with the system piping. A relief valve will be installed between the MOV and the system piping and become part of the MOV isolation boundary. The relief valve will open to relieve high bonnet pressure and seat for isolation. The MOV will continue to be leak rate tested with the total leakage to include any relief valve leakage. This will ensure that 10CFR50 Appendix J requirements will be met to maintain primary containment integrity. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.:	96-052
Implementation Document No.:	Mod. PN2Y95MX011
USAR Affected Pages:	9.4-40; Figures 9.4-12a, 9.4-13 Sh 2, 11.5-7
System:	Turbine Building Ventilation (HVT)
Title of Change:	Three Fan Operation of the Turbine Building Exhaust

Description of Change:

Normal operation of the HVT system has two 50% capacity exhaust fans running with a third fan in standby. Electrical interlocks prevent simultaneous operation of all three fans. In order to provide additional outside supply air for cooling in summer and help maintain subatmospheric conditions inside the Turbine Building, this modification allows manual simultaneous operation of all three fans in an effort to provide enhanced pressure and temperature control.

This change was previously implemented as Temporary Modification No. 94-025. Safety Evaluation No. 95-048 Rev. 0 addressed the implementation of the temporary modification and concluded that an unreviewed safety question did not exist. This safety evaluation addresses the conversion of the temporary modification to a permanent modification.

Safety Evaluation Summary:

Parallel operation of the three exhaust fans will improve performance of the HVT system in maintaining the pressure and temperature controls inside the building. The three fans operation will increase the exhaust by approximately 17,500 cfm, thus allowing a corresponding increase in the outside supply air. In warmer months, the outside cooler air helps in keeping the temperatures inside the building lower.

The steam tunnel temperature and differential temperature will be monitored during initial three fan operation to verify that the potential change in exhaust flow from the main steam tunnel remains within current design. In respect to operability considerations of the leak detection system temperature elements, the steam tunnel has been evaluated in Calculation No. HVT-33 for a total loss of heating, ventilating and air conditioning. Running the third fan does not significantly impact the air balance for the steam tunnel which was verified during final testing

Safety Evaluation No.: 96-052 (cont'd.)

Safety Evaluation Summary: (cont'd.)

of the temporary modification. Therefore, the proposed change will not impact the function of the leak detection instruments.

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

Safety Evaluation No.:	96-053 Rev. 0 & 1
Implementation Document No.:	Design Change N2-96-003
USAR Affected Pages:	11.3-4, 11.3-5, 11.3-6, 11.3-7, 15.7-2, 15.7-3, 15.7-4; Tables 1.9-1 Sh 47, 11.3-2 Sh 1, 15.7-1, 15.7-3, 15.7-4, 15.7-5; Figures 9.2-19c, 11.3-1a, 11.3-1b
System:	Offgas (OFG)
Title of Change:	Offgas Freezeout Dryer Modification

Description of Change:

The OFG system refrigeration units 2OFG-REF1A, B, and C and associated freezeout coil in dryers 2OFG-DRY1A, B and C were retired in place. In addition, the low flow air dilution makeup and low-low flow isolation automatic controls were eliminated. Administrative procedures were enhanced to maintain a minimum flow of greater than 6 scfm per train through the system to prevent combustible hydrogen concentrations, and require adsorption coefficients to be determined and evaluated prior to replacing the charcoal beds.

To support the change and eliminate the freezeout and defrosting portion of the system, the dryer automatic high differential pressure swapover was removed, the high differential pressure setpoint was lowered, and the high dryer outlet temperature increased. In addition, the moisture transmitter high moisture setpoint was changed to reflect the new dew point.

Even though the actual adsorption coefficients are less conservative than the adsorption coefficients used in determining the shielding design for the offgas charcoal adsorber cubicle, Calculation Disposition PR-C-26-A-00B proves that the current design is adequate to ensure that the radiation zone limits for the adjoining areas are not exceeded.

Safety Evaluation Summary:

This change does not affect the design intent of the OFG system. The dew point temperature of the inlet air to the charcoal beds will be raised to a maximum of 50°F; however, analysis determined that the OFG system provides adequate protection with a 50°F (max.) dew point.

Safety Evaluation No.: 96-053 Rev. 0 & 1 (cont'd.)

Safety Evaluation Summary: (cont'd.)

The low flow makeup air and the low-low flow isolation were originally provided to meet the specification requirement to maintain below 0.5-percent hydrogen concentrations with 0.5 scfm of condenser in-leakage. The requirement per the Technical Specifications is that hydrogen concentrations be maintained below the combustible limit of 4 percent. Administrative controls were established to maintain minimum flow by inducing air in-leakage, if needed, to preclude development of hydrogen pockets and maintain hydrogen concentrations below combustible limits.

Automatic isolation on low-low flow is not required because: a) low flow annunciation is provided within sufficient time to initiate manual action, b) high hydrogen annunciation is provided prior to isolation, c) automatic isolation of either offgas train will still occur on high hydrogen, and d) the USAR Chapter 15 Accident Analysis takes credit only for manual action in isolating the system on low flow. A review of the Safety Evaluation Report verified that the automatic feature was not credited in the Nuclear Regulatory Commission's acceptance of the system design. It should be noted that Standard Review Plan 11.3 does not require automatic dilution air for the NMP2 OFG system because the system is designed to withstand a hydrogen detonation.

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

Safety Evaluation No.: 96-055

Implementation Document No.: Calc. AX-076A-02C

USAR Affected Pages: Table 6A.9-4

System: Reactor Core Isolation Cooling (ICS)

Title of Change: Revise Stress Analysis Results at Sockolet (AX-076A, Node 31) DER #2-94-2422

Description of Change:

A review of Calculation AX-076A revealed that stress indices for a 22" X 12" vesselet (WFI) from an industry published document were inappropriately used for the qualification of a sockolet (Bonney Forge) for vent valves 2ICS*V178 and *V179.

The discrepancy was corrected in Calculation AX-076A-02C by recalculating stress intensities and cumulative usage factor (CUF) at the sockolet using stress indices based on ASME III Table NB-3683.2-1.

Safety Evaluation Summary:

Piping was stress reanalyzed using sockolet stress indices based on ASME Code and vendor-supplied data. The reanalysis also included updates for as-built data and power uprate, and correction of some discrepancies in thermal transient analysis. The reevaluation indicated that stresses and CUF for the sockolet are within acceptable limits. The proposed change does not affect nuclear safety.

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

Safety Evaluation No.:	96-056
Implementation Document No.:	Design Change N2-96-004
USAR Affected Pages:	12.3-25; Figures 1.2-10 Sh 2, 9.1-25, 5-2 (Appendix 9C), 12.3-12, 12.3-45
System:	Fuel Nuclear Refueling (FNR)
Title of Change:	Relocation of New Fuel Inspection Stand

Description of Change:

This change relocated the new fuel inspection stand to a more optimum service area for the polar crane just south of the new fuel vault at elev. 353'-10" refuel floor in the Reactor Building. Relocation of the new fuel inspection stand improves inspection activities associated with new fuel receipt and facilitates the installation of the piping system for the alternate decay heat removal system. The relocation required remounting the inspection stand and modifying the existing handrails around the inspection platform.

Safety Evaluation Summary:

The relocation of the new fuel inspection stand from the north side to the south side of the new fuel storage vault will improve inspection activities associated with new fuel receipt and facilitate the installation of the piping system for the alternate decay heat removal system. A Category II/I evaluation has determined that the failure of the mounting of the stand's support during a seismic event would not result in damage to safety-related system components or structures on the refuel floor, nor would the falling of a maximum of two new fuel bundles from the stand result in a nuclear safety concern. The existing area radiation monitor system meets the requirements of 10CFR70.24, Regulatory Guide 8.12, and ANSI N16.2, as indicated in the Standard Review Plan, Section 12.3.

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

Safety Evaluation No.:	96-058
Implementation Document No.:	N/A
USAR Affected Pages:	N/A
System:	Feedwater (FWS), Feedwater Control (FWC)
Title of Change:	Replacement of Reactor Water Control Circuit Logic Card While Plant is Operating

Description of Change:

The control circuit for a feedwater flow control valve would not stay in the automatic mode of operation. The control circuit logic card was replaced while the plant was operating. The valve for the affected control circuit was locked in place and the control for the other operating feedwater flow control valve was placed in local-manual to prevent inadvertent valve movement while the logic card was replaced. Operations provided appropriate controls/procedures to maintain manual control of the feedwater flow control valve(s) to maintain reactor water level. No adjustments to valves were required during the short time when the logic card was replaced.

Safety Evaluation Summary:

During performance of maintenance on the feedwater control system, there is a potential to initiate a malfunction in the feedwater system. A malfunction of the feedwater control system could cause a loss of feedwater flow event or a maximum demand feedwater flow event, both of which are analyzed in the USAR. However, for this activity, one feedwater flow control valve will be locked in place and the other valve will be placed in local-manual control. This prevents the valves from inadvertently moving during the maintenance activity, which in turn prevents initiation of either of the two feedwater events. Therefore, the proposed activity 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.: 96-060

Implementation Document No.: Simple Design Change SC2-0005-95

USAR Affected Pages: 7.7-26

System: Fuel Nuclear Refueling (FNR)

Title of Change: Replace Dillon Force Switches

Description of Change:

This simple design change replaced force switches located on top of load cells at the refuel platform frame-mounted auxiliary hoist and monorail auxiliary hoist with electronic setpoint modules. The force switches provided electrical weighing interlocks for hoist loaded and hoist overload. However, the force switches were prone to drift and were difficult to calibrate. The new electronic setpoint modules provide the same interlocks which were previously performed by the force switches, while improving operation and maintainability of the load weighing systems for auxiliary hoists. Required circuit redundancy is maintained with the setpoint modules.

Safety Evaluation Summary:

This change maintains all applicable USAR requirements with the exception of the following statement in section 7.7.1.4.2: "Associated interlock is performed by force switches located on top of the load cells." Contacts from the new setpoint modules will replace the force switches and will maintain the function currently being performed by the force switches. The setpoint module contacts will be wired in the load weighing circuits in the same location as the existing force switches; the same load setpoints will be used; and redundancy of the circuits will be maintained by the replacement of each force switch with a corresponding setpoint module. The two setpoint modules for each interlock circuit will each have a contact wired in series; each contact will open when the load setpoint is reached to stop hoist lifting. The series wiring configuration provides redundancy such that, if one of the setpoint modules fails (contact remaining closed), then the other setpoint module operates properly (contact opening) to ensure that the circuit opens to stop hoist lifting.

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

Safety Evaluation No.:	96-061 Rev. 0 through 7
Implementation Document No.:	Design Change N2-96-004
USAR Affected Pages:	3.3-4, 3.5-18, 3A.35-1, 3A.35-2, 9.1-14, 9.1-18, 9.1-18b, 9.1-48, 9.1-49, 9.1-50, 9.1-51, 9.1-52, 11.5-9, 11.5-10; Tables 1.9-1 Sh 7 & 53, 3.2-1 Sh 18a, 3.9A-12 Sh 13 & 15, 3C.4-1, 9.1-7, 11.5-2 Sh 4; Figures 1.2-1, 1.2-2, 1.2-10 Sh 1, 9.1-5a, 9.1-28a, 12.3-44
System:	Alternate Decay Heat Removal (ADH), Spent Fuel Pool Cooling and Cleanup (SFC), Domestic Water (DWS), Residual Heat Removal (RHR)
Title of Change:	Alternate Decay Heat Removal

Description of Change:

This design change installed a new system capable of removing decay heat from the reactor core and the spent fuel pool during Operational Condition 5, "Refueling". The system is designed in conjunction with natural circulation as an alternate method of decay heat removal in accordance with Technical Specification 3.9.11.1. The system has been sized to support spring and fall outages as early as 96 hours after shutdown (heat load of 54.9×10^6 Btu/hr), but can be used for outages between June and September. The system is designed to be used with the head removed from the vessel, the reactor cavity flooded to an elevation greater than 22'-3" above the vessel flange, and the spent fuel pool interconnected (open) to the reactor cavity pool. Under these conditions, the ADH system is capable of maintaining pool surface temperatures and core exit temperatures at the limits needed to support refueling operations. This in turn allows both trains of RHR to be taken out of service for maintenance as allowed by Technical Specification 3.9.11.1. Operation of the system is not limited to Mode 5 only; it may be used during other modes as necessary.

Safety Evaluation Summary:

The ADH system does not perform any safety-related functions; it's primary function is to perform RHR shutdown cooling functions during refueling operations with the head removed from the vessel, the reactor cavity flooded to an elevation of greater than 22'-3" above the vessel flange, and the spent fuel pool interconnected to the reactor cavity pool. However, the use of ADH is not

Safety Evaluation No.: 96-061 Rev. 0 through 7 (cont'd.)

Safety Evaluation Summary: (cont'd.)

restricted to Mode 5 only. The SFC system can backup the ADH system in the event ADH is lost. Operation of ADH during refueling and during reactor operation has been evaluated to insure SFC can be maintained within the specifications of USAR section 9.1.3.3.

The impact of internally-generated missiles and missiles generated by natural phenomena is evaluated. Internally-generated missiles from the ADH primary loop pumps is not credible based on USAR section 3.5.1.1.5, which considers catastrophic failure of rotating equipment that leads to the generation of missiles to be not credible. Postulated missiles generated by natural phenomena are evaluated based on an analysis of the probability of a missile strike on the Reactor Building penetrations for the ADH components. The analysis calculated the probability of a missile strike to these openings to be less than the accepted Regulatory Guide 1.117 value of 1.0×10^{-7} per year and, therefore, the new penetrations are not protected against missile strikes from external missiles. The probability analysis described above is based on the methodology developed by L.A. Twisdale as documented in EPRI Report No. NP-2005.

The nonsafety portions of the ADH system housed inside the Reactor Building are designed and supported, where required, for seismic forces to ensure that failure of these components does not affect the operation of any Category I equipment or cause damage to Category I structures.

The ADH components in the yard will not impact atmosphere dispersion factors, nor will they impact external flood protection features.

The ADH piping is moderate-energy piping. The flood height in the Reactor Building resulting from a postulated crack in the moderate-energy ADH piping is below the level which results from the limiting case 18" RHS line. Therefore, existing flood protection features are unaffected by the ADH piping.

Based on the evaluation performed, it is concluded that installation and use of ADH system does not involve an unreviewed safety question.

Safety Evaluation No.:	96-068
Implementation Document No.:	N/A
USAR Affected Pages:	A.0-1, A.0-3, A.4.3-1, A.4.4-1, A.4.4-3, A.5.2-4, A.6-2, A.15.0-2, A.15.0-7, A.15.1-4, A.15.1-5, A.15.1-6, A.15.1-8, A.15.2-2, A.15.2-4, A.15.2-5, A.15.2-6, A.15.2-9, A.15.4-1, A.15.4-7, A.15B-1, A.15D-1; Tables A.5.2-1, A.6-2, A.15.0-4 Sh 1 & 2
System:	Various
Title of Change:	Operation of NMP2 Reload 5/Cycle 6

Description of Change:

This change consisted of the addition of new fuel bundles and the establishment of a new core loading pattern for Reload 5/Cycle 6 operation of NMP2. Two hundred forty (240) new fuel bundles of the GE11 design were loaded. All 32 of the remaining GE6B-P8CIB219-4GZ-100M-150-T bundles and all 40 of the remaining GE9B-P8CWB299-7GZ-100M-150-T bundles from Reload 4/Cycle 5 were discharged to the spent fuel pool. In addition, 168 of 248 GE9B-P8CWB320-9GZ1-100M-150-T bundles were discharged to the spent fuel pool. Various evaluations and analyses were performed to establish appropriate operating limits for the reload core. These cycle-specific limits were documented in the Core Operating Limits Report.

Safety Evaluation Summary:

The reload analyses and evaluations are performed based on the General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A-12 and NEDE-24011-P-A-12-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 5, the evaluations included transients and accidents likely to limit operation because of minimum critical power ratio considerations; overpressurization events; loss-of-coolant accident; and stability analysis. Appropriate consideration of equipment out of service was included. Limits on plant operation were established to assure that applicable fuel and reactor coolant system safety limits are not exceeded.

Safety Evaluation No.: 96-068 (cont'd.)

Safety Evaluation Summary: (cont'd.)

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

Safety Evaluation No.: 96-069

Implementation Document No.: DDC 2F01363

USAR Affected Pages: Figure 9.3-1d

System: Automatic Depressurization (ADS), Instrument Air (IAS)

Title of Change: ADS Valves 2IAS*V407 and 2IAS*V508 Replacement

Description of Change:

Valves 2IAS*V407 and 2IAS*V508 comprise a double-valved test connection, installed in the secondary containment at approximately elevation 294'-3". The test connection is used during surveillance testing and is installed between containment isolation valves 2IAS*V448 and 2IAS*SOV164. The original design utilized two 3/4", 600# stainless steel Velan packless diaphragm valves. The associated piping and piping components were stainless steel, designed and fabricated in accordance with ASME Section III, Subsection NC with a threaded plug closure device. The original valves leaked and identical replacements were not readily available to support the maintenance replacement activity scheduled for RFO-5. To facilitate maintenance, approved replacements were required. The new design was constructed using two stainless steel 3/4", 2680# Velan globe valves. The associated piping and components were designed and fabricated in accordance with the original requirements. A threaded cap closure device is utilized in lieu of the plug.

Safety Evaluation Summary:

The 3/4" Velan globe valve model W04-9076Z-13AA has been evaluated for potential leakage criteria. It is concluded that a catastrophic failure will not occur. Minor nitrogen leakage (i.e., stem, seat) may occur, but will be detected by surveillance and necessary maintenance performed accordingly.

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

Safety Evaluation No.:	96-070
Implementation Document No.:	Mod. PN2Y94MX007
USAR Affected Pages:	9.2-5, 9A.3-39, 9A.3-40; Table 3.2-1 Sh 11; Figures 9.2-1h, 9.2-1L, 9.2-6a, 9.3-12e, 10.4-7e, 10.4-7h
System:	Service Water Chemical Treatment (SCT), Service Water (SWP)
Title of Change:	Service Water Chemical Treatment

Description of Change:

This modification provided a SCT system for the storage and injection of a biocide (sodium hypochlorite and sodium bromide) and detoxicant (sodium bisulfite) into the SWP system to control microbiologically-influenced corrosion. This system consists of three pump skids and a local control panel located in the acid storage area (northwest corner) of the Screenwell Building. The pumps draw suction on three chemical storage tanks located in the same area and discharge the appropriate chemical into the SWP pump intake bays for treatment, and into the SWP discharge lines for detoxification prior to release to the lake. In addition, this modification makes permanent a service water intake/discharge corrosion monitoring station installed under Temporary Modification 91-107. This station allows for sample coupon analysis of the intake and discharge streams of the SWP system to monitor biocide treatment. Finally, a SWP sample line now extends from valve 2SWP-V746 (elevation 243') to a sample valve located on a nearby staircase platform (elevation 243') for simplified SWP discharge bay sampling.

Safety Evaluation Summary:

The SCT system does not perform any safety-related functions and is only intended to destroy microbiological organisms in the SWP system to minimize fouling of safety-related and nonsafety-related components. The treatment system interfaces solely with service water and will not change or impact the operation or performance of the SWP system. The SWP system acts to mitigate an accident and is not considered an accident initiator or precursor as evaluated in the USAR. Adequate provisions are in place in the plant to preclude any potential flooding that could occur due to a malfunction with the SCT system. Based on the evaluation performed, it is concluded that these changes do not involve an unreviewed safety question.

Safety Evaluation No.: 96-071

Implementation Document No.: Procedures OT-EDS-005, OT-EDS-004

USAR Affected Pages: 4.6-26

System: Control Rod Drive (RDS)

Title of Change: Updating CRD Friction Testing Requirements

Description of Change:

The USAR previously required that friction testing be performed during refueling outages on control rods that have channel exposure exceeding 30,000 MWD/T in their control cells at the beginning of the cycle. No channel at NMP2 has seen exposure greater than 30,000 MWD/T at the beginning of cycle; however, next cycle channels would exceed this exposure. DER 2-96-0878 was written because there existed no procedural requirement to perform friction testing on channels exceeding 30,000 MWD/T. During an investigation, it was determined that this requirement was no longer applicable. In 1984 when this USAR requirement was developed, it was general practice to reuse channels. This requirement was needed since channels could exceed their design lifetime if they were reused. Since NMP2's general practice is not to reuse channels, the exposure requirement has been replaced with the requirement that a channel remains with its original fuel bundle. The assembly lifetimes (which include fuel channels and fuel bundles) are limited by fuel limits; therefore, it is not possible to exceed a channel design limit without exceeding the fuel limits. The fuel limits are that the peak pellet exposures do not exceed 70,000 MWD/MT for GE11 assemblies and 60,000 MWD/MT for GE9 assemblies. The initial GE6 core fuel assemblies had a peak pellet exposure limit of 46,000 MWD/MT.

Safety Evaluation Summary:

The revisions being made to the USAR ensure practices are in place to mitigate channel bowing. These controls will implement the recommendations of General Electric Service Information Letter SIL-320 and assure proper practices are implemented for future core designs. An exposure limit on the channel is being replaced with a requirement that the channels remain with their original fuel assembly. Channels are designed for the lifetime of the fuel assembly, and as long as the channel is not reused, the channel design lifetime will not be exceeded. Based on the evaluation performed, it is concluded that these changes do not involve an unreviewed safety question.

Safety Evaluation No.: 96-072

Implementation Document No.: Simple Design Change SC2-0022-93

USAR Affected Pages: Figures 5.4-9c, 5.4-13d, 9.3-9d

System: Reactor Core Isolation Cooling (ICS), Residual Heat Removal (RHS), Reactor Building Equipment Drain (DER), Reactor Building Floor Drain (DFR)

Title of Change: Reactor Building Equipment Drain Modification

Description of Change:

This modification installed a new drain cooler for the ICS and RHS drain lines. The modification was performed due to undesirable interactions/problems in the past that resurfaced with respect to the ICS, RHS, SDV, and the gravity drains. The modification was implemented such that flexibility to switch the ICS/RHS drains to the existing drain cooler, 2DER-E2A, is still possible if desired at any time. The new cooler obtains cooling water from the reactor building closed loop cooling system (CCP).

Installation of the new drain cooler was divided into two phases. Phase 1 was comprised of modifications to be made during RFO-5 so that the balance of the work, Phase 2 (actual installation of new drain cooler, including final tie-in to ICS, RHS, DER, and CCP), could be implemented on line. This safety evaluation addresses the RFO-5 scope only of the modification. The Phase 2 portion of the modification will be addressed by means of a revision to this safety evaluation, which will discuss further plant impacts. The RFO-5 scope included: a) installation of ICS/RHS drain line valve tie-ins for Phase 2 installation of the new drain cooler; b) installation of isolation valves on existing ICS and RHS lines to allow flexibility for switching drain flow from the proposed drain cooler to existing drain cooler 2DER-E2A, if desired; c) installation of the relief devices (Rupture Discs 2ICS-PSE1, 2RHS-PSE1A, and 2RHS-PSE2A) for overpressure protection of Class 150 piping upstream of isolation valves which were installed.

Safety Evaluation Summary:

All isolation valves to be installed per this change shall meet the requirements of ANSI B31.1 and be designed for a pressure rating of Class 150 (minimum).

Safety Evaluation No.: 96-072 (cont'd.)

Safety Evaluation Summary: (cont'd.)

The ICS drain will continue to flow directly to the 4" DER header and through existing cooler 2DER-E2A (until Phase 2 installation of new cooler) to tank 2DER-TK2A, as previously designed. Isolation valve 2ICS-V271, installed for tie-in to the proposed (new) drain cooler, will ensure that appropriate pressure boundary for line 2ICS-001-128-4 is maintained.

Steam-condensing effluents will continue to drain to the 4" DER header, the existing cooler, and into the Reactor Building equipment drain tank (until Phase 2 installation of new cooler). Isolation valves 2RHS-V437 and 2RHS-V439, installed for tie-in to the proposed (new) drain cooler, will ensure that appropriate pressure boundary for lines 2RHS-001-233-4 and 2RHS-001-373-4 is maintained.

The existing rupture disc, 2DER-PSE10A, on the drain header is routed to equipment drain funnel 2DER-ED1506, which discharges into drain sump 2DFR-TK2E. The rupture discs installed per this modification, 2ICS-PSE1 and 2RHS-PSE1A (2A), will also relieve to tank 2DFR-TK2E.

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

Safety Evaluation No.: 96-074

Implementation Document No.: Design Change N2-96-033

USAR Affected Pages: 3.1-59, 9.1-15, 9.1-16, 9.1-17, 9.1-18, 9.1-18a, 9.1-18b, 9.1-41; Table 1.9-1 Sh 40

System: Alternate Decay Heat Removal (ADH), Spent Fuel Pool Cooling and Cleanup (SFC), Residual Heat Removal (RHR)

Title of Change: SFC UFSAR Update

Description of Change:

This design change revised the SFC system design basis as described in USAR section 9.1.3. These changes:

- Allow fuel transfer from the reactor core to the spent fuel pool at a rate of 10 fuel bundles per hour as early as 96 hours after reactor shutdown versus 12 days after reactor shutdown.
- Allow a full core offload to the spent fuel pool during normal refueling outages.
- Allow planned maintenance of one SFC train during normal plant operation.
- Address SFC operation with divisional bus and service water system outages.

Safety Evaluation Summary:

The change to allow fuel transfer to the spent fuel pool as early as 96 hours after shutdown at a rate of 10 bundles/hour has been evaluated. For the normal refueling case of 324 bundles with the spent fuel pool isolated from the reactor cavity, one loop of SFC will maintain the pool at or below 125°F. For the emergency offload case, two SFC loops will maintain the pool at or below 150°F.

The change to allow a normal full core offload during refueling operations has been evaluated for when the spent fuel pool is open to the reactor cavity and when the pool is isolated. For a normal full core offload, the pool will be maintained at or below 125°F.

Safety Evaluation No.: 96-074 (cont'd.)

Safety Evaluation Summary: (cont'd.)

The change which describes decay heat removal functions during refueling when the spent fuel pool is open to the reactor cavity has been evaluated. It has been concluded that use of the defense-in-depth approach for spent fuel pool cooling and core decay heat removal is consistent with current requirements.

The change which allows for planned SFC maintenance outages has been evaluated. If the planned maintenance is expected to exceed the time required for the pool to heat up to its maximum design temperature should the available cooling system be lost, compensatory methods shall be made available to provide redundancy for pool cooling.

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

Safety Evaluation No.: 96-075

Implementation Document No.: NMP2 Emergency Operating Procedures, Rev. 7

USAR Affected Pages: N/A

System: Various

Title of Change: Revision 6 of the NMP2 Emergency Operating Procedures (EOP)

Description of Change:

Revision 7 of the EOPs changed some operating parameters as a result of the use of more GE-11 fuel and less GE-9 fuel. It also incorporated other minor changes to facilitate use of the EOPs. The parameters which were revised are as follows:

- Heat capacity level limit.
- Minimum core flooding interval.
- Direct the operators to exit the main steam isolation valve (MSIV) leakage control EOP if the MSIVs are required to be open.
- Update of the minimum indicated level for reactor pressure vessel water level instrument accuracy.
- Revise the entry condition to the radioactivity release control EOP to read: Offsite radioactivity release rate above the Emergency Plan Alert level.

Safety Evaluation Summary:

The operator actions prescribed in Revision 7 to the EOPs are in accordance with the Boiling Water Reactor Owners' Group Emergency Procedure Guidelines (BWROG EPGs). When applied to licensing basis accidents and transients, the EOPs will not increase the probability or the consequences of these events as depicted in the USAR.

None of these changes has altered the philosophy, logic, or validity of the NMP2 EOPs, nor did they affect the capability of the operators to recover from an accident.

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

Safety Evaluation No.:	96-076
Implementation Document No.:	Procedures FHP13.3, N2-REP-34
USAR Affected Pages:	N/A
System:	N/A
Title of Change:	The Use of COSMOS for Calculating Shutdown Margin During Shuffle for Reload 5

Description of Change:

This safety evaluation provided justification for using computer code COSMOS for calculating shutdown margin during the Reload 5 shuffle, should the planned move sheets require changes.

Safety Evaluation Summary:

The current planned moved sheets have been analyzed with a qualified computer model (ARROTTA) under a safety-related program which assures adequate shutdown margin is maintained during the shuffle. However, it may be necessary to make changes to the move sheets. This safety evaluation provides the justification to change the move sheets with the use of the computer code, COSMOS.

COSMOS was shown to provide conservative results when it compared against a qualified code. Design margins are being maintained during the shuffle; therefore, there is no reduction of margin.

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

Safety Evaluation No.:	96-078
Implementation Document No.:	NFPA Standard 12A
USAR Affected Pages:	9A.3-57
System:	N/A
Title of Change:	Deletion of Halon Air Flow Test from USAR

Description of Change:

This change deleted the requirement to perform an air flow test to demonstrate operability of the Halon fire suppression systems. Such testing is not required by the national consensus standard and adequate precautions are implemented when systems are most subject to foreign material intrusion. Periodic visual examinations of piping systems and nozzles are performed, which assure piping system continuity.

Safety Evaluation Summary:

This evaluation examines the requirement to perform an air flow test through headers and nozzles of Halon systems to assure no blockage. This is currently required by the USAR to demonstrate system operability. However, there is no technical/regulatory basis for such a test. The applicable national consensus standard does not require such testing to be performed. The proposed change will result in deletion of the requirement to perform air flow testing to demonstrate operability of Halon fire suppression systems.

System operability will be verified in accordance with National Fire Protection Association (NFPA) Standard 12A, Halon 1301 Fire Extinguishing Systems. Appropriate precautions are taken to preclude foreign material intrusion, including protective covers over piping which is disassembled. The Halon system operability does not affect the ability to safely shut down in the event of a fire.

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

Safety Evaluation No.: 96-079

Implementation Document No.: Procedure N2-OP-3

USAR Affected Pages: N/A

System: Redundant Reactivity Control (RRS), Feedwater (FWR)

Title of Change: Isolation of Feed Pump Min Flow Valve

Description of Change:

This temporary change allowed operation of feedwater pump 2FWS-P1C with its minimum flow valve, 2FWR-FV2C, isolated. This minimized high-pressure, high-temperature emission due to a leak that had developed in the minimum flow valve. The minimum flow valve was not fully isolated until the leak was repaired.

Safety Evaluation Summary:

Based on the reviews performed, it has been determined that isolation of minimum flow valve 2FWR-FV2C will not adversely affect the safety functions of the systems and equipment involved. To prevent a potential loss of condenser vacuum resulting from this activity, the blocking valve will not be isolated until the minimum flow valve is repaired. The systems will be able to perform all of their safety functions with this temporary change in place. In addition, this change is not an initiator or precursor to the accidents analyzed.

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

Safety Evaluation No.:	96-080
Implementation Document No.:	Procedure N2-PM-R@029
USAR Affected Pages:	Tables 8.3-2 Sh 30, 8.3-6 Sh 6; Figures 8.3-1, 8.3-3 Sh 2, 8.3-11
System:	NJS, LAS
Title of Change:	Power Supply to Panel 2LAS-PNL400 During the Outages

Description of Change:

The 600-V ac, 3-phase nonsafety-related normal distribution panel 2LAS-PNL400 is supplied power from normal 600-V ac load center 2NJS-US4. This panel provides power to the lighting of the Control Building, Control Room, Normal Switchgear Building, Diesel Generator Building, and Transformers area. It also provides power to the Channel B scram pilot solenoids of the reactor protection system.

During the outages when load center 2NJS-US4 may be de-energized, the above loads may be lost, interfering with outage activities. To reduce the impact of the 2NJS-US4 shutdown and to facilitate the outage activities, temporary 600-V ac, 3-phase power was provided to panel 2LAS-PNL400. The source of the temporary power was normal 600-V ac, 3-phase load center 2NJS-US6, which remained energized during 2NJS-US4 shutdown.

This change lasted for the outage duration. Upon re-energization of load center 2NJS-US4, the permanent power supply to panel 2LAS-PNL400 from load center 2NJS-US4 was restored.

The temporary power supply to panel 2LAS-PNL400 was performed in accordance with Section 6.13 of Operations Preventive Maintenance Procedure N2-PM-R@029, Outage of Non-divisional Electrical Bus 2NJS-US4 and Associated Motor Control Centers and Distribution Panels.

Safety Evaluation Summary:

The analysis performed determined that the proposed change has no impact on function and performance of the normal distribution panel 2LAS-PNL400, or 600-V ac nonsafety-related load center 2NJS-US6. The loads connected to the panel and to the load center will perform as designed.

Safety Evaluation No.: 96-080 (cont'd.)

Safety Evaluation Summary: (cont'd.)

Load center 2NJS-US6 is fed from nonsafety-related 13.8-kV stub bus 2NNS-SWG015, which can be connected to Div. II Emergency Diesel Generator (EDG) 2EGS*EG3. This connection is performed manually and is permitted only in case of loss of offsite power (LOOP) without an accident. If a LOCA occurs when the stub bus is connected to the diesel generator, it is automatically disconnected. The additional load increases the noncoincident loading of EDG 2EGS*EG3 by 150 kVA (125 kW). These noncoincident loads will be added to the EDG at the operator's discretion under existing administrative controls to ensure the loading on the EDG is within its rating. The connection of the stub bus to the emergency switchgear in case of LOOP without loss-of-coolant accident (LOCA) is already 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.: 96-081

Implementation Document No.: Procedure N2-ODP-OPS-0113

USAR Affected Pages: N/A

System: Standby Gas Treatment (GTS), Reactor Building Ventilation (HVR)

Title of Change: Control of Secondary Containment Leakage Paths During Outages

Description of Change:

Secondary containment integrity requirements, as defined in the Technical Specifications, are applicable during plant operational conditions 1, 2, 3, and *. During outages, various piping systems and components, including those in the secondary containment, are dismantled for modification, repair, and/or maintenance. This activity in the secondary containment has the potential for creating breaches between the secondary containment environment and the outside air, adversely affecting the secondary containment integrity under operational condition *. If not properly controlled, the area of these breaches, coupled with the area of the leakage paths that are inherent to the secondary containment, could exceed the "equivalent area" used in the drawdown analysis that provides the basis for the one-hour drawdown time.

This safety evaluation addressed implementing Procedure N2-ODP-OPS-0133 during the outages such that Technical Specifications secondary containment in-leakage is not violated and the GTS system's ability to drawdown the secondary containment and maintain it at the required negative pressure is not compromised.

Safety Evaluation Summary:

Calculation No. ES-280 has been prepared to define breaches that are permissible during outages so that total combined in-leakage through the leakage paths inherent to secondary containment and through these breaches does not exceed the Technical Specifications value.

Typically, the actual in-leakage (as measured during the drawdown test, Procedure N2-OSP-GTS-R001, and converted to the Technical Specifications basis) is lower than the Technical Specifications value. For permitting breaches that are generated by dismantling of various systems for modification, repair, and/or maintenance during outages, advantage is taken of the difference between the actual in-leakage and the Technical Specifications permitted in-leakage.

Safety Evaluation No.: 96-081 (cont'd.)

Safety Evaluation Summary: (cont'd.)

Limiting breaches in accordance with Calculation ES-280 will ensure that the total in-leakage at any time during the outages is limited to the Technical Specifications value. This will ensure that the GTS system capacity will not be exceeded with excessive in-leakage and the GTS system can drawdown the secondary containment to, and maintain it at, a negative pressure of at least -0.25 in. WG under all design bases, including postulated accident conditions during operational condition *. No testing is needed to ensure that the in-leakage through these breaches will not result in the total in-leakage exceeding the Technical Specifications value, because the "k" values used in Calculation ES-280 for these are conservative. Therefore, permitting breaches in accordance with Calculation ES-280 during outages will not result in violation of Technical Specifications in-leakage. The additional in-leakage will not prevent the GTS system from performing its intended safety function under all design basis conditions, including postulated accidents.

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

Safety Evaluation No.: 96-084 Rev. 0 & 1

Implementation Document No.: Procedure N2-TTP-Chemcln-@001

USAR Affected Pages: N/A

System: Service Water (SWP)

Title of Change: Service Water Chemical Cleaning

Description of Change:

An assessment of the SWP system was performed in response to industry concerns, and Unit 2 performance problems were noted during SWP check valve and unit cooler tests. A study was performed which noted several areas for improvement of the system performance. This safety evaluation addressed one aspect of the proposed system improvements. Flow degradation had been recorded since initial startup. Visual inspection of SWP piping and equipment internals verified the presence of general corrosion and microbiological fouling. Independent evaluations of SWP pipe samples confirmed the presence of microbiologically influenced corrosion (MIC). A chemical cleaning operation was initiated to remove resultant corrosion products and enhance the effectiveness of future chemical treatment. The cleaning operation was designed to address SWP piping segments and unit cooler coils in the Reactor and Control Buildings that were considered most susceptible to MIC.

The selected cleaning method was the tannin/citric acid process described in EPRI Document No. RP3232-1, "Recommended Cleaning Practices for Service Water Systems." This process was selected based on its comparatively low corrosion rate, compatibility with nonmetallic materials, and its overall effectiveness in removing the fouling product. A series of prototypical tests were performed on Unit 2 service water pipe samples to qualify the process. The prototypical tests were designed to simulate critical system characteristics, including surface area to volume ratios, carbon steel to copper ratios, wetted materials, temperature and pressure limitations and flow rate distributions. The results of the prototypical cleaning demonstrated effective removal of all fouling product from the pipe and left the sample with a passivated bare metal surface. Data obtained during the prototypical tests was used in evaluating metal wall loss, compatibility with nonmetallic wetted materials, expected cleaning times, chemical requirements and waste volume. Parameters such as cleaning temperature, pressure, and flow rate were established based on SWP system design constraints and hydraulic considerations.

Safety Evaluation No.: 96-084 Rev. 0 & 1 (cont'd.)

Safety Evaluation Summary:

The chemical cleaning process is a maintenance activity which will improve the hydraulic performance of the system. There are no permanent plant changes associated with the proposed operation. The location of the cleaning process equipment will have no impact on any permanent plant equipment, including the 115-kV switchyard. The temporary alterations will be made during Operational Conditions 4, 5 and when handling irradiated fuel in the secondary containment, with the affected headers out of service. There will always be a division of service water operable for the plant requirements during this process. The system will have sufficient isolation from other plant components to preclude any possible cross-contamination that could impact neighboring systems or components. All chemicals will be handled in accordance with NMPC procedures. The system will be restored to a near-original surface cleanliness, allowing for the system to achieve design flow rates upon completion of the cleaning process. Analysis of wetted materials within the cleaning loops determined the cleaning process does not affect the design life or operability of any material component or part. The SWP system is not considered an accident initiator or precursor as 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.:	96-086 Rev. 0 & 1
Implementation Document No.:	Temporary Mod. 96-029
USAR Affected Pages:	N/A
System:	Reactor Protection System (RPS), Nuclear Steam Supply System (NSSS), Main Steam (MSS)
Title of Change:	Defeat of Main Steam Line Rad Monitoring Trip Signal Channel B1

Description of Change:

Main steam line radiation monitor 2MSS*RE46B was experiencing periodic spiking in its signal which was not reflected in the other three associated main steam line radiation detectors. The spiking was developing progressively in the number of occurrences and in severity to a point of initiating RPS half scrams and nuclear steam supply shutoff system (NS4) isolation signals. It was suspected that the detector was defective and needed replacement. However, in order for replacement, the detector had to be de-energized which, similar to the main steam line high radiation signal, brings in a half scram signal on the RPS, an isolation signal to the 1 and 2 valve groups, trip signal to the condensate air removal pumps, annunciation and performance monitoring system computer points. This temporary modification installed a jumper in panel 2CEC*PNL633 Bay B in order to defeat a trip signal (Channel B1) which would normally be provided whenever detector 2MSS*RE46B is inoperable.

Safety Evaluation Summary:

This temporary modification will defeat the RPS Channel B1 trip signal during maintenance and will allow for ample time for troubleshooting and replacement of the faulty detector without initiating a 1/2 RPS trip and NS4 isolation signal. Compliance with Technical Specifications 3/4.3.1.a and 3/4.3.2.b.1.b will be maintained. The station would immediately enter the required 12-hour Limiting Condition of Operation (LCO). If the affected detector could not be returned to operable status within the 12-hour LCO time limit, LCO action would be entered as required per Technical Specifications 3/4.3.1.a and 3/4.3.2.b.1.b. This change reduces the plant's vulnerability to a full scram by eliminating the half scram signal which would otherwise be present during the time period that the detector is being replaced. In the event of fuel damage, the remaining main steam line radiation monitors will function to detect the release of fission products and initiate the appropriate mitigating actions to limit the release and to shut down the plant. This

Safety Evaluation No.: 96-086 Rev. 0 & 1 (cont'd.)

Safety Evaluation Summary: (cont'd.)

change does not impact the remaining detectors from performing their safety functions as originally designed.

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

Safety Evaluation No.: 96-087

Implementation Document No.: PCE to Procedure N2-OP-11

USAR Affected Pages: N/A

System: Service Water (SWP)

Title of Change: Defeat SWP Pumps Not Running Isolation To SWP System Isolation Valves 2SWP*MOV3A, 19A, 93A and 599

Description of Change:

Isolation valves 2SWP*MOV3A, 19A, 93A, and 599 automatically close upon restoration of either a full or partial loss of offsite power (LOOP). The isolation signal is comprised of two inputs: a) LOOP in associated division, and b) loss of service water flow in redundant division. Both signals were designed to simulate a total LOOP or partial LOOP to ensure that single failure criteria was met.

This procedure change temporarily defeated the signal indicating pumps not running but did not prevent these valves from closing on LOOP.

Safety Evaluation Summary:

The PCE to Procedure N2-OP-11 provides instruction for implementation of this change and restoration of logic to its original design upon completion of maintenance.

The defeating of the service water pumps not running signal to isolation valves 2SWP*MOV3A, 19A, 93A and 599 will not adversely affect the safety function of the SWP system. The single failure criterion will be satisfied by entering Technical Specification LCO 3.7.1.2.c. Single failure criterion is normally preserved by specifying limiting conditions for operation. The analysis performed determines that the proposed change does not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety.

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

Safety Evaluation No.: 96-088

Implementation Document No.: EDC 2M10386C

USAR Affected Pages: 12.3-3, 12.3-20

System: Various

Title of Change: Valve Packing

Description of Change:

This change revised the USAR to represent actual valve packing practices at Unit 2. EDC 2M10386 was issued to incorporate EPRI NP-5697 recommendations which have been proven to be the most effective in reducing valve packing leakage. This packing program eliminated the double set of packing and lantern ring. However, the packing function to minimize valve stem leakage was not changed.

Safety Evaluation Summary:

The valve packing change will exceed the original valve packing requirements, which will further compliment Unit 2's ALARA program. This is in compliance with Regulatory Guide 8.8 requirements.

Appendix B Determination 87-002 classifies valve packing as nonsafety related. Valve packing is not considered an integral part of the pressure boundary of valves as defined by ASME Boiler and Pressure Code. As such, these items cannot significantly degrade the pressure retention function of the valve in which they are installed. A change in valve packing will not change the design basis and safety function of the valve that is to be repacked. The EPRI packing program affords the same protection as the double set packing and the lantern ring arrangement, since the valve leak-offs stems were never functional. These changes represent an enhancement over the original design. The EPRI study concluded that deep stuffing boxes filled with packing actually degrade packing performance, and reducing the number of packing rings improves sealing performance.

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

Safety Evaluation No.:	96-089
Implementation Document No.:	Design Change N2-96-036
USAR Affected Pages:	4.4-8, 4.4-9
System:	Loose Parts Monitor (LPM)
Title of Change:	Disable Loose Parts Monitor Nuisance Alarms

Description of Change:

This design change disabled certain LPM system channel loose parts alarms associated with the reactor recirculation loops at the vibration and loose parts monitoring system (V&LPM) panel. The affected channels were Channel 3-recirculation loop A suction, Channel 4-recirculation loop B suction, Channel 5-recirculation loop A discharge, and Channel 6-recirculation loop B discharge. This change prevents nuisance alarms during periods of throttled recirculation flow (i.e., feedwater pump swap-over). The reduced recirculation flow was resulting in increased hydraulic noise in the recirculation loops causing continuous alarms to be received by the loose parts event analysis computer (LPEAC). The alarms were received in rapid succession causing diagnostic overload of the LPEAC before completion of first-received alarm validation (to determine if the alarm was false or valid). This conclusion forced the LPEAC computer to pass on all alarms to the Control Room creating an operator nuisance.

Safety Evaluation Summary:

The recirculation loops are not defined in Regulatory Guide (RG) 1.133 as natural collection regions and, therefore, the associated loose parts alarms are not required for RG 1.133 compliance. Thus, by eliminating the loose parts alarms which automatically trigger the LPEAC validation process associated with the recirculation loops (but still retaining diagnostic capabilities), the condition for LPEAC overloads is removed. The V&LPM system will then function properly to detect and alarm those required channels associated with the reactor vessel internals to comply with RG 1.133.

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

Safety Evaluation No.:	96-090
Implementation Document No.:	Procedure N2-WHP-6
USAR Affected Pages:	11.4-5, 11.4-6
System:	Solid Radioactive Waste
Title of Change:	Shipment of Dry Compressible Waste to Off-site Vendors for Processing

Description of Change:

All dry active waste (DAW) was previously required to be compacted prior to shipment offsite. This change allows DAW to be shipped offsite without prior compaction. After packaging, the waste is transported to an approved burial site or an offsite vendor for processing (i.e., incineration, super compaction, sorting) if processed waste will be shipped to a licensed disposal facility, or returned to the station for interim storage.

Safety Evaluation Summary:

Collection of trash in an uncompacted form is within the design of the Radwaste Building collection area. The volume of collected trash will not exceed design before being packaged for shipment. Requirements for packaging and shipment of uncompacted trash are the same as those for compacted trash. All waste is packaged and transported in accordance with Nuclear Regulatory Commission and Department of Transportation requirements.

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

Safety Evaluation No.: 96-092

Implementation Document No.: N2-STP-046

USAR Affected Pages: N/A

System: Condensate (CNM), Feedwater (FWS)

Title of Change: Single Feedwater Pump Flow Capability Testing

Description of Change:

Testing of the single failure pump flow capability demonstrated the maximum flow that can be produced by a single feedwater pump at reactor power levels normally achieved with two pumps in service. The data obtained from this testing is used to determine whether the condensate/feedwater and reactor recirculation system will respond such that a reactor trip will not occur in the event of a feedwater pump trip when the plant is operated normally with two pumps in service at high rod lines. Based on the results, the data may also be used to determine the most appropriate future plant modifications required to assure the desired system operation is achieved.

Testing was completed with the associated feedwater pump low and low-low suction pressure trip functions defeated to provide added assurance that all feedwater will not be lost due to setpoint drift on the suction pressure switches.

Safety Evaluation Summary:

The plant will be operated in accordance with approved operating procedures and the administrative controls imposed in N2-STP-046 will provide compensatory actions for defeating the low and low-low suction pressure trips. The systems and equipment will still be operated within their design. The activity does not impact any precursors or initiators of an accident previously evaluated in the USAR. The anticipated operational occurrences due to loss of all feedwater flow, feedwater controller failure to maximum demand, and loss of feedwater heaters, have already been 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.:	96-095
Implementation Document No.:	Design Changes N2-96-043, N2-96-045
USAR Affected Pages:	Figures 5.4-9c, 6.3-6a
System:	High Pressure Core Spray (CSH), Reactor Core Isolation Cooling (ICS)
Title of Change:	Pressure Locking Bonnet Vent for 2ICS*MOV126 and 2CSH*MOV107

Description of Change:

Pressure locking is a term that describes the occurrence of high pressure in the bonnet of a closed gate valve relative to upstream and downstream system pressures. This high bonnet pressure wedges the valve discs more tightly on their seats so that more thrust is required for the valve to open. The causes of pressure locking are generally either thermal expansion of an incompressible fluid (e.g., water) in the bonnet or rapid depressurization of the system which traps initial system pressure in the bonnet.

These design changes utilized existing bonnet connections on valves 2ICS*MOV126 and 2CSH*MOV107 to install bonnet vent lines, which connected the bonnet with the system piping to continuously relieve the high bonnet pressure.

Safety Evaluation Summary:

Valve bonnet vent pathways will be installed between the motor-operated valve (MOV) bonnet and the system piping in order to relieve high pressure in the bonnet and allow the MOVs to open on demand. The bonnet vent pathways will be designed and installed in accordance with the required specifications, procedures, and ASME III Code to ensure system piping integrity.

The bonnet vent pathways will connect the MOV bonnet with the system piping such that the bonnet vent pathways will not bypass the disc seat used for MOV isolation. This will ensure that applicable requirements will be met to maintain primary containment integrity.

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

Safety Evaluation No.: 96-097

Implementation Document No.: Design Change N2-96-044

USAR Affected Pages: Figures 5.4-13a, 5.4-13b

System: Residual Heat Removal System (RHS)

Title of Change: Pressure Locking Bonnet Vent for 2RHS*MOV25A/B

Description of Change:

Pressure locking is a term that describes the occurrence of high pressure in the bonnet of a closed gate valve relative to upstream and downstream system pressures. This high bonnet pressure wedges the valve discs more tightly on their seats so that more thrust is required for the valve to open. The causes of pressure locking are generally either thermal expansion of an incompressible fluid (e.g., water) in the bonnet or rapid depressurization of the system which traps initial system pressure in the bonnet.

This safety evaluation evaluated design changes to preclude the potential pressure locking problem with valves 2RHS*MOV25A/B. These valves had existing bonnet connections which were utilized to install bonnet vent lines which connected the bonnet with the system piping to continuously relieve the high bonnet pressure.

Safety Evaluation Summary:

Valve bonnet vent pathways will be installed between the motor-operated valve (MOV) bonnet and the system piping in order to relieve high pressure in the bonnet and allow the MOVs to open on demand. The bonnet vent pathways will be designed and installed in accordance with the required specifications, procedures, and ASME III Code to ensure system piping integrity.

The bonnet vent pathways will connect the MOV bonnet with the system piping such that the bonnet vent pathways will not bypass the disc seat used for MOV isolation. This will ensure that applicable requirements will be met to maintain primary containment integrity.

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