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Enclosure A to
NMP2L 1838

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

ENCLOSURE A

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

UFSAR Appendix B Page/Section	Identification of Change	Reason for Change	Basis for Concluding that the Revised Program Continues to Satisfy 10CFR50 Appendix B and Commitments Previously Approved by the NRC
Page B.1-2, Section B.1.2.1.1 second and third paragraphs	Changed "Manager Human Resource Development" to "Director Human Resource Development". Deleted "and the General Supervisor Labor Relations".	Reorganization that established the position Director Human Resource Development. This position reports to the Chief Nuclear Officer and has responsibility for Employee and Labor Relations, Occupational Safety and Health, Quality First Program (Q1P) administrative issues, and the Fitness for Duty Program. The Director Q1P continues to report to the Chief Nuclear Officer on matters related to Q1P concerns. These changes improve NMPC's ability to maintain a safety conscious work place.	The assignment of these responsibilities to the Director Human Resource Development provides clear management control over related functional areas. The reporting of the functions to the Chief Nuclear Officer ensures effective lines of communication. The job functions and responsibilities assigned to the different groups remain the same. Therefore, the revised program continues to satisfy the criteria of 10CFR50 Appendix B and the QATR commitments previously accepted by the NRC.
Page B.1-4, Section B.1.2.1.1.4.b	Deleted previous Item b.	Reorganization. The functions were moved to the other QA supervisors.	Reorganization improves Quality Assurance effectiveness and value to the Nuclear Division. All responsibilities associated with the position of the Supervisor Quality Verification/Safety Assessment were assumed by the Supervisor Quality Assessment and /or General Supervisor Quality Services. The same qualified individuals continue to perform those functions. Also, the qualifications necessary to perform those functions remain the same.

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, Sections B.1.2.1.1.4.b and B.1.2.1.1.4.c	Renumbered Item c to b and Item d to c. Changed "Supervisor Quality Assurance Audits" to "Supervisor Quality Assessment." Added "and conducting performance-based surveillances" after "QA audits".	Reorganization. Combined surveillance and audit functions into the single functional area "Quality Assessment".	Reorganization improves Quality Assurance effectiveness and value to the Nuclear Division. Surveillance responsibilities associated with the position of the Supervisor Quality Verification/Safety Assessment were assumed by the Supervisor Quality Assessment. The same qualified individuals continue to perform those functions. Also, the qualifications necessary to perform those functions remain the same.
Page B.1-4, Section B.1.2.1.1.4.d	Renumbered Item e to d. Added "assessments determining applicability of industry and in-plant operating experience, assisting in root cause evaluations when requested, DER trend analysis," after "document control".	Combined all plant support and administrative functions under Quality Services.	Reorganization improves Quality Assurance effectiveness and value to the Nuclear Division. Plant support and administrative responsibilities associated with the position of the Supervisor Quality Verification/Safety Assessment were assumed by the General Supervisor Quality Services. The same qualified individuals continue to perform those functions. Also, the qualifications necessary to perform those functions remain the same.
Page B.2-4, Section B.2.2.11.1	Changed "Engineering" to "Implementing".	Clarification. The criteria used to identify structures, systems and components for which the QA Program applies was changed to a Nuclear Implementing Procedure from a Nuclear Engineering Procedure.	The procedure to determine the safety classification remained essentially the same and continues to meet NMPC and 10CFR50 Appendix B criteria.
Page B.2-4, Section B.2.2.11.2	Changed "Appendix B" to "Safety Classification".	The title of the process changed from Appendix B Determination to Safety Classification Determination.	The procedure to determine the safety classification remained essentially the same and continues to meet NMPC and 10CFR50 Appendix B criteria.
Page B.5-2, Section B.5.2.6.3	Deleted "emergency plan implementing procedures". Added "full" between "A" and "revision".	Clarification. Moved the emergency plan implementing procedures to the next paragraph. Periodic reviews require a full revision.	The periodic frequency was shortened; therefore, the level of commitment previously accepted by the NRC was not reduced. A full revision is more restrictive and is required by NMPC procedures to qualify as a periodic review.

UFSAR Appendix B Page/Section	Identification of Change	Reason for Change	Basis for Concluding that the Revised Program Continues to Satisfy 10CFR50 Appendix B and Commitments Previously Approved by the NRC
Page B.5-2, Sections B.5.2.6.4	Added "Emergency plan implementing procedures are reviewed at least annually and revised as appropriate. A full revision of a procedure, or detailed scrutiny of a procedure as part of a documented training program, drill, simulator exercise or other such activity, constitutes a procedure review".	Implementation of the requirements of NUREG-0654 Revision 01 and Regulatory Guide 1.101.	The periodic frequency was shortened; therefore, the level of commitment previously accepted by the NRC was not reduced.
Page B.15-1, Section B.15.1, second paragraph	Deleted entire paragraph.	Editorial. NMPC currently uses only one type of system (Deviation/Event Report) to identify, control and disposition nonconforming conditions in materials, parts, components or services.	Nuclear Implementing Procedures were generated several years ago. NIP-ECA-01 "Deviation/Event Report" (DER) was developed to incorporate the different departmental systems. The requirements of 10CFR50 Appendix B continue to be met.
Page B.15-1, Section B.15.2.2	Deleted "departmental".	Editorial. NMPC currently uses only one type of system (Deviation/Event Report) to identify, control and disposition nonconforming conditions in materials, parts, components or services.	Nuclear Implementing Procedures were generated several years ago. NIP-ECA-01 "Deviation/Event Report" (DER) was developed to incorporate the different departmental systems. The requirements of 10CFR50 Appendix B continue to be met.
Page B.15-2, Section B.15.2.12	Deleted "departmental".	Editorial. NMPC currently uses only one type of system (Deviation/Event Report) to identify, control and disposition nonconforming conditions in materials, parts, components or services.	Nuclear Implementing Procedures were generated several years ago. NIP-ECA-01 "Deviation/Event Report" (DER) was developed to incorporate the different departmental systems. The requirements of 10CFR50 Appendix B continue to be met.
Page B.15-2, Section B.15.2.13	Changed "senior nuclear division and corporate management" to "nuclear division management".	Reorganization. To line up with the current management organization described in Sections B.1 and B.2.	Reorganization approved by NRC via Unit 1 License Amendment 157 and Unit 2 License Amendment 71, dated February 20, 1996.
Page B.16-1, Section B.16.2.2	Deleted "departmental".	Editorial. NMPC currently uses only one type of system (Deviation/Event Report) to identify, control and disposition nonconforming conditions in materials, parts, components or services.	Nuclear Implementing Procedures were generated several years ago. NIP-ECA-01 "Deviation/Event Report" (DER) was developed to incorporate the different departmental systems. The requirements of 10CFR50 Appendix B continue to be met.

UFSAR Appendix B Page/Section	Identification of Change	Reason for Change	Basis for Concluding that the Revised Program Continues to Satisfy 10CFR50 Appendix B and Commitments Previously Approved by the NRC
Page B.17-1, Section B.17.2.2	Added "Quality Assurance" between "considered" and "records". Deleted "These records include: 1. Results of ...calibration procedures and reports". Added "Additionally, the Records Management Program includes those records identified in plant Technical Specifications."	Clarification. The addition of the words "Quality Assurance" provides a more precise and accurate description of what these documents are considered upon completion. The description of what types of documents become records upon completion is contained in the first sentence of Section B.17.2.2. The specific list of records was removed since it was not an all-inclusive list. The addition of the statement "Additionally, the Records Management ... in plant Technical Specifications" ensures that those records identified in Technical Specifications as requiring retention, but which do not meet the definition of a Quality Assurance record, will be captured under the Records Management Program.	Adding "Quality Assurance" between "considered" and "records" is consistent with the wording in 10CFR50 Appendix B Section XVII. The change is considered a clarification of an existing commitment and, therefore, does not contradict or alter any commitments previously approved by the NRC. The addition to the second statement is consistent with 10CFR50 Appendix B Section XVII and ANSI/ASME NQA-101983 (17, 17S-1). Inclusion of a partial list of documents considered to fall into this category allows the reader unnecessary room for misinterpretation. While a reader may interpret that a particular document need not be controlled by procedure because that document did not appear on the list of examples provided in the QATR, no such misinterpretation can be made if the partial list is eliminated. If the list is not all-inclusive and stand-alone it should not be included. The third statement ensures that those records identified in plant Technical Specifications as requiring retention, but which do not meet the definition of a Quality Assurance record, will be captured under the program.
Page B.17-1, Section B.17.2.3	Changed "permanent" to "lifetime".	Clarification. To be consistent with the terms used in NQA-1 to avoid any potential confusion.	The terms "lifetime" and "permanent," when applied to Quality Assurance records, are synonymous.
Page B.17-2, Section B.17.2.8	Changed "Except for records that are stored as originals, such as radiographs ... or features are used" to "Records are stored in appropriate fire rated facilities, or in remote dual facilities to prevent damage, deterioration, or loss due to natural or unnatural causes."	Clarification by eliminating redundant exception for records stored as originals. When only a single original can be retained, it will obviously not be stored in a remote, dual facility.	The intent of this section was not altered. This clarification eliminates a redundant exception for records stored as originals.

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-3, Sheet 4 of 8	Changed Exception wording in Item 3.r to "Installed plant instrumentation calibration status is tracked through the PMST database. Calibration status of portable measurement & test equipment (M&TE) may be labeled on the case or attached to the device. For instances where size or application precludes attaching the calibration labels on the device, the device shall be uniquely identified and traceable to its calibration record."	This was part of the corrective/preventive actions from a DER written during an ISEG assessment. The site was not implementing the exception as it was written.	The use of the PMST database for in-plant equipment allows for better tracking and scheduling of the calibration of this equipment. This database is addressed in the procedures and used in training. The portable M&TE still are required to maintain the same type of calibration labeling as the original exception. The requirements of ANSI/ANS-3-2 and 10CFR50 Appendix B continue to be met.
Table B-3, Sheet 5 of 8	Changed Exception in Item 4.c from "Personnel who perform audits for the SRAB are not required to be so qualified, since these audits are outside the scope of the audit program described in Section B.18 of this QATR" to "Personnel who perform SRAB audits that are outside the scope of 10CFR50 Appendix B are not required to be so qualified."	Clarification. Some of the SRAB required audits are in the scope of Section B.18 of the QATR.	Clarification. Some of the SRAB required audits are in the scope of Section B.18 of the QATR.

Enclosure B to
NMP2L 1838

NINE MILE POINT - UNIT 2

SAFETY EVALUATION SUMMARY REPORT

1998

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

Safety Evaluation No.: 90-129 Rev. 1 & 2

Implementation Document No.: Mod. PN2Y89MX151

USAR Affected Pages: 9.4-36; Figures 9.2-8a, 9.4-10e

System: Radwaste Building Ventilation (HVW)

Title of Change: Radwaste Control Room Humidification

Description of Change:

This modification installed a second electric steam humidifier to the Radwaste Control Room HVAC subsystem. The new humidifier may be operated in tandem with the existing unit or can serve as a backup during periods of maintenance or out-of-service conditions. The capacity of the new unit is comparable to the existing unit.

Revision 1 of this safety evaluation addressed the addition of a second humidifier identical to the existing unit. Revision 2 of this safety evaluation addresses the newer and improved model which has been installed. The manufacturer has made several improvements in the humidifier design since that time.

Safety Evaluation Summary:

The failure or malfunction of the new nonsafety-related humidifier does not adversely affect the integrity or function of any structure, system or component important to safety, but will enhance and facilitate the system performance.

This modification adds a small quantity of combustible material to some fire zones. However, the fire loading increase is not significant enough to affect the Fire Protection Program.

Installation for this modification can proceed during normal plant operation with only a minor interruption to Radwaste Control Room HVAC operation.

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

Safety Evaluation No.: 92-048 Rev. 1

Implementation Document No.: Temporary Mod. 91-107

USAR Affected Pages: N/A

System: Service Water (SWP)

Title of Change: Installation and Use of Service Water Biocide Injection and Monitoring Equipment

Description of Change:

This temporary modification installed a biocide injection system to control microbiologically-induced corrosion (MIC) and reduce the fouling of heat transfer surfaces.

Safety Evaluation Summary:

Use of this equipment will have no adverse effect on any design characteristics of the SWP system. Water analysis and the use of this temporary equipment will be controlled by the Chemistry Department. Environmental parameters of the SWP system will be maintained within those limits specified by the New York State Department of Environmental Conservation. Operation of the SWP system will continue to be within the original design and l
icensing bases.

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

Safety Evaluation No.: 92-061 Rev. 0 & 1

Implementation Document No.: Mod. PN2Y91MX052

USAR Affected Pages: 3.8-45, 11.5-6, 11.5-9; Tables 9.5-1 Sh 7, 9.5-2 Sh 6, 9A.3-9 Sh 9, 3-2 Sh 1 (Appendix 9C), 11.5-1 Sh 3, 12.3-1 Sh 3; Figures 1.2-13 Sh 1 & 3, 1.2-14, 9.4-11 Sh 16, 9.5-21, 9A.3-4, 11.4-1a, 11.4-1b, 12.3-24, 12.3-26, 12.3-27, 12.3-57, 12.3-59, 12.3-60

System: Solid Radwaste (WSS)

Title of Change: Upgrade Radwaste 245' Elevation Storage

Description of Change:

This modification installed lighting (general and emergency), radiation monitoring, communications system, and fire detectors in the Radwaste El. 245' storage area, and a four-ton bridge crane on Radwaste El. 261'. These changes ensure personnel safety and safe handling of the low-level radioactive waste (LLRW) generated at NMP2 and NMP1 and being transferred between units. The waste is being stored on an interim basis in the Radwaste El. 245' storage area.

Safety Evaluation Summary:

The addition of lighting, communications, fire detection, radiation monitoring, and bridge crane for the Radwaste El. 245' storage area is a system enhancement which will facilitate handling and storage of LLRW. These changes have no impact on safety-related systems or equipment and, therefore, will have no impact on the safe operation or shutdown of the plant.

The proposed transfer of LLRW between units and storage of LLRW in the Radwaste El. 245' storage area are well within the design basis of the site and Radwaste facility. The type and nature of the material to be stored is such that any postulated accident or occurrence is already bounded by the existing Radwaste facility analysis.

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

Safety Evaluation No.: 92-073 Rev. 1

Implementation Document No.: Simple Design Change SC2-0310-92

USAR Affected Pages: Figure 10.1-6c

System: Reactor Feed Pump Seal Water

Title of Change: Reactor Feed Pump Seal Water Instrumentation

Description of Change:

This simple design change installed monitoring instrumentation for the reactor feed pump seal water flow and temperature. Revision 1 eliminated the seal cavity temperature indication since seals of the original design did not have threaded ports to accommodate the seal cavity temperature probes. The instrumentation allows seal degradation to be monitored so that seal replacement might be a planned maintenance evolution instead of a catastrophic failure causing feedwater system transients.

Safety Evaluation Summary:

This change will have no impact on the safe operation of the plant as the instrumentation will allow seal water parameters to be monitored with no adverse effect on the operation of the feedwater system.

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

Safety Evaluation No.: 92-077

Implementation Document No.: Procedure N2-CTP-CWS-807

USAR Affected Pages: 10.4-17

System: Circulating Water (CWS), Chemical Feed-Hypochlorite (WTH)

Title of Change: Justification of Circulating Water Sodium Hypochlorite Addition

Description of Change:

This safety evaluation evaluated a procedurally controlled temporary alteration used in lieu of the permanent plant WTH system to accomplish hypochlorite injection into the CWS. Effects on station equipment, personnel safety and environmental limits were evaluated and no adverse consequences have been experienced due to this change.

Safety Evaluation Summary:

This temporary alteration will have no impact on the safe operation or capability to keep the plant in a safe shutdown condition.

CWS capabilities are not degraded at any time by this temporary alteration because microbiological control shall continue to be maintained by the addition of sodium hypochlorite. Precautions are being taken to prevent chlorine gas generation and sodium hypochlorite spill.

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

Safety Evaluation No.: 93-096 Rev. 1

Implementation Document No.: Temporary Mod. 96-026, Procedure GAP-DES-03

USAR Affected Pages: N/A

System: N/A

Title of Change: Installation and Control of Temporary Communications Equipment per GAP-DES-03

Description of Change:

Procedure GAP-DES-03 was revised to include an exclusion for temporary communications installed in accordance with the Technical Support Administrative Procedure. A new procedure was developed to allow the installation and control of temporary communications equipment (GAI-TRONICS) in facilities at the Nine Mile Point site (e.g., temporary trailers installed in support of refueling outage activities).

Safety Evaluation Summary:

The permanent plant emergency notification public address system will remain the primary means of notification for all site personnel. Testing shall be conducted, upon installation and removal, to ensure the operation of the existing communication equipment is not adversely impacted. The change to the communications system will interface with the existing system's signal in the normal design manner. The equipment design isolates the non-UPS power source from the permanent system signals.

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

Safety Evaluation No.: 94-072

Implementation Document No.: Simple Design Change SC2-0013-94

USAR Affected Pages: Figure 9.4-9 Sh 8, 20, 26

System: Reactor Building Ventilation (HVR)

Title of Change: Alarm Logic Change for Unit Coolers 2HVR*UC408A/B, *409A/B

Description of Change:

This simple design change revised alarm logic for unit coolers 2HVR*UC408A/B and *409A/B so that, regardless of which unit is selected as the standby unit in each pair, the alarm for auto start of the standby unit actuates properly. Previously, the alarm was tied only to the B unit of each pair, because originally the A unit was the primary and the B unit was always the standby cooler. It was determined that rotating the equipment and alternating the A and B units as standby would prevent a nuisance alarm when the B unit actuated first. When the B unit was set to start first, this caused a nuisance alarm. This change connected an auxiliary contact from each motor starter in each pair in series to actuate the alarms so that when the second cooler in each pair starts, regardless of whether it is the A or B unit, the alarm will actuate. This change involved running one conduit and wires between each pair of local starters mounted on secondary containment el. 240'. Also, internal wiring in 2CEC*PNL870 and *PNL871 was modified to remove the control switch interlocks that were in the alarm circuits. Most of the existing wiring and alarm circuitry remained in place, and this represents a minor change to the two existing alarm circuits.

Safety Evaluation Summary:

This change is limited to minor rewiring of the alarm circuits which warn the operator that the standby unit cooler has auto started. The existing circuit alarms only upon start of the B cooler in each pair. The system logic is being changed to maintain the originally desired alarm logic, but with the new provision that either the A or B unit cooler of each pair, whichever is assigned to be the standby unit and is the second to start, will activate the existing alarm. The new circuit changes will be installed to maintain the existing standards for safety-related circuits and this change will not increase the probability of any type of failure.

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

Safety Evaluation No.: 94-074 Rev. 0 & 1

Implementation Document No.: Mod. PN2Y94MX015

USAR Affected Pages: 11.2-8 thru 11.2-11, 11.2-14, 11.4-3, 11.4-4; Tables 11.2-1 Sh 4, 11.2-2, 11.4-1, 11.4-2; Figures 1.2-13 Sh 2 & 3, 1.2-14, 9.3-1b, 9.3-1k, 9.3-11e, 9.4-10d, 11.2-1e, 12.3-25, 12.3-26, 12.3-59

System: N/A

Title of Change: Installation of the Thermex System

Description of Change:

This modification replaced the advance liquid processing system with the Thermex system as the primary liquid radwaste treatment system. Thermex produces better quality recycled water and significantly less solid waste. The Thermex system uses principles of reverse osmosis, electrode ionization, ultraviolet photo degradation and demineralization to purify plant waste water. This change also removed the flatbed filter and its associated piping and instrumentation.

Safety Evaluation Summary:

The design, testing, ALARA considerations, floor loading, fire protection and plant service requirements of the Thermex System were reviewed and analyzed for compatibility with NMP2's systems. In addition, system conformance including Thermex performance and waste processing was reviewed. Based on the reviews performed, this change will enhance NMP2's liquid waste processing capabilities and create less radioactive waste.

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

Safety Evaluation No.: 95-047

Implementation Document No.: IST Program Plan - NMP2-IST-001

USAR Affected Pages: Table 3.9A-12 Sh 1 thru 5, 8, 9, 11 thru 14

System: CCP, CPS, CSL, CSH, EGF, GTS, HCS, HVK, IAS, ICS, ISC, RHS, SFC, SLS, SWP

Title of Change: USAR Table 3.9A-12 Update

Description of Change:

This change involved a general update of the valve listings in USAR Table 3.9A-12. The changes made involved: (1) changing from "safety related with active function" to "safety related with passive function"; (2) specifying a different safety-related active function for selected valves than the presently identified safety-related active function; (3) clarification of selected valves which have an additional active safety-related function in addition to the existing safety-related active function presented; (4) additional valves that were not originally included in this table, but were described elsewhere, that have active safety-related functions; (5) reclassification of thermal relief valves which do not have an active function but rather passive based on the thermal relief function; and (6) correction of various typographical errors in table descriptions, mark number, and valve type.

Safety Evaluation Summary:

These changes update USAR Table 3.9A-12 to provide consistency with discussion and description elsewhere in the USAR. These changes are the result of a design review of existing plant design documentation and do not make any changes that impact the plant configuration.

Testing of the safety-related valves to their safety functions is in compliance with the ASME Code as specified in the NMP2 IST Program Plan.

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

Safety Evaluation No.: 95-070

Implementation Document No.: Simple Design Change SC2-0025-95

USAR Affected Pages: 9.3-19, 9.5-67; Tables 9.2-9, 9.3-1 Sh 4; Figures 9.2-19f, 9.3-5a, 9.3-5b, 9.3-7 Sh 6, 8, 10.1-6e

System: Instrument Air Supply (IAS), Zinc Injection Passivation (ZIP), Turbine Building Closed Loop Cooling Water (CCS), Turbine Sampling (SST)

Title of Change: Removal of Feedwater Corrosion Monitor and Associated Equipment

Description of Change:

This simple design change removed feedwater corrosion product monitor (FCPM) 2SST-IPNL172, zinc injection panel 2ZIP-PNL101, and associated components. The FCPM and associated components have been very maintenance and labor intensive, have not been used for some time, and have been determined not to be required for plant support. The FCPM provided a means for continuously monitoring the level of soluble and insoluble materials in reactor feedwater; provided an instantaneous readout of corrosion product levels; and provided a means for collecting samples of suspended and dissolved solids by filtration and/or ion exchange.

Safety Evaluation Summary:

The FCPM is a component of the SST system. The SST system is nonsafety related, is not required to function during or following an accident, and is not required to safely shut down the reactor.

All work associated with this change will be performed in the Turbine Building and Control Building in accordance with approved site procedures and specifications. The removal of the FCPM, one zinc injection panel, and associated equipment located in and around reactor feedwater pump 2FWS-P1C will facilitate future maintenance activities on the feed pumps.

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:

The CCS system is an intermediate cooling distribution loop that transfers heat from designated equipment to the service water system. The CCS system is designed to remove heat with sufficient flow from designated heat exchangers in the Turbine and Radwaste Building. This simple design change replaced the existing packing from within the stuffing boxes of pumps 2CCS-P1A, P1B, and P1C. No cooling water is required for this packing, thus the stuffing box cooling water lines were removed and ports capped.

Safety Evaluation Summary:

The new self lubricating and heat conductive packing is recommended by NMPC Maintenance based on successful performance at other plants in similar applications. Any potential malfunction associated with this change would be bounded by those previously reviewed by the NRC.

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

Safety Evaluation No.: 95-129

Implementation Document No.: Simple Design Change SC2-0065-95

USAR Affected Pages: Table 6.2-62; Figures 1.2-8 Sh 1, 9A.3-5

System: N/A

Title of Change: Explosion-Proof Door

Description of Change:

This simple design change eliminated access through doors NA262-1 and TS261-4 as part of the nuclear security response plans for NMP2 in accordance with 10CFR73. This change eliminated these doors for use as building access points and emergency egress points. These doors are positively assured closed, which ensures their performance associated with maintaining negative building pressure for radiological control purposes.

Safety Evaluation Summary:

The concrete blocks used to eliminate access through the doors will not compromise adjacent Category 1 structures or become a missile problem. The doors' function associated with confinement of radiological material will be maintained. This change will delete use of these doors as normal building access and emergency egress. Adequate alternative building access is maintained as well as emergency egress and areas of refuge. These doors can be restored to functional by removing the blocks. Therefore, use of door NA262-1 post-LOCA for access to the North Auxiliary Bay is not adversely impacted. This change will not compromise safe operation and will enhance nuclear security provisions for NMP2.

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

Safety Evaluation No.: 95-138

Implementation Document No.: DDCs 2M10846, 2M10848

USAR Affected Pages: Figure 11.3-1b

System: Offgas (OFG)

Title of Change: Freezeout Dryer Flushing Connections

Description of Change:

This change installed tees at the air exhaust and the water drain in the freezer section of freezeout dryers 2OFG-DRY1A, C, just downstream of 2OFG-TE61A, C approximately midpoint on the dryers. This allows for inspection and flushing of the freezeout dryers as required to prevent clogging in the freezer section.

Safety Evaluation Summary:

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

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

Safety Evaluation No.:	96-006
Implementation Document No.:	Procedure NSAS-POL-01
USAR Affected Pages:	13.1-6, 13.2-16, 13.2-18; Figure 13.1-5
System:	N/A
Title of Change:	Reorganization; Change to NSAS-POL-01 to Reflect Consolidation of the Maintenance and Technical Training Groups

Description of Change:

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

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

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

Safety Evaluation Summary:

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

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

Safety Evaluation No.: 96-019

Implementation Document No.: Site Emergency Plan

USAR Affected Pages: N/A

System: N/A

Title of Change: Emergency Operations Facility (EOF) Move from the Nuclear Learning Center at 9 Mile Point to the Existing EOF on Route 176 in Fulton, NY

Description of Change:

The EOF is a near-site support facility for the management of overall licensee emergency response, coordination of radiological and environmental assessments, and determination of recommended public protective actions. The EOF is equipped with administrative, communication, and computer equipment.

This safety evaluation evaluated the relocation of the Nine Mile Point EOF located in the Nuclear Learning Center (NLC) to the new EOF located on Route 176 in Fulton, N.Y.

Safety Evaluation Summary:

Chapter 15 of the Unit 2 USAR has been reviewed and this change will have no effect on the probability of occurrence of the accidents analyzed therein. The EOF located at the NLC does not provide plant control functions and is not connected to any system that is used to mitigate an accident. The EOF operates in accordance with design configuration and site procedures to comply with NUREG-0696, the Site Emergency Plan (SEP), Unit 1 UFSAR, and Unit 2 USAR. Changes to the SEP, as a result of moving the EOF to the new location, will not affect any plant system used to mitigate an accident or any system associated with accidents previously analyzed. Therefore, the probability of occurrence of the accidents previously evaluated in the USAR is not affected.

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

Safety Evaluation No.:	96-027 Rev. 2
Implementation Document No.:	Design Change N2-97-001
USAR Affected Pages:	5.4-24, 15.0-14; Tables 3.9A-12 Sh 5, 12, 9B.8-1 Sh 3; Figure 5.4-9c
System:	Reactor Core Isolation Cooling (ICS)
Title of Change:	Replace Actuator and Internals for 2ICS*PCV115

Description of Change:

This design 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 or disconnected some of the power and control cables for this valve. The instrument air system was connected to the actuator, and restricting orifice 2ICS*RO207 was rebored.

Safety Evaluation Summary:

The design function of the valve, supplying cooling water to the lube oil cooler for the ICS system, can be adequately performed by an air-operated actuator, the DRAG 100 control valve, with the rebore as the restricting orifice. The replacing of valve 2ICS*PCV115 actuator and internals and reboring of restricting orifice 2ICS*RO207 will not impact the safe operation of the plant. This valve is assumed to be open in the plant accident analysis and, thus, this change does not alter that assumption or any operational modes for this valve. Any malfunctions associated with this change would be bounded by those previously reviewed by the NRC.

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

Safety Evaluation No.: 96-030

Implementation Document No.: Mod. PN2Y94MX001

USAR Affected Pages: Tables 9.4-4 Sh 4, 5, 9.5-2 Sh 6; Figures 1.2-19, 9.2-1q, 9.2-8b, 9.2-17b, 9.3-1h, 9.3-3a, 9.3-11d, 9.3-11e, 9.4-10b, 9.4-10e, 9.5-21, 9.5-22, 11.5-7, 12.3-14, 12.3-47

System: N/A

Title of Change: Chemistry Lab Expansion

Description of Change:

This modification expanded the chemistry lab facility into the large tool and equipment decontamination area, Radwaste El. 261'-0", across from the existing chemistry lab. The expansion made available approximately 2,000 sq. ft. of floor space overall for the chemistry facilities by: (1) removing maintenance equipment, and (2) routing/rerouting demineralizer water, service water, breathing air, service air, piping and drainage. This also required HVAC tie-ins to existing lines, electrical power feeds, removal of curb, minor changes to existing lab furniture, and relocation and installation of new chemistry lab furniture (consisting of hoods, cabinets, sinks, etc.)

Safety Evaluation Summary:

Chemistry lab expansion in the dirty shop area is required due to the inadequacy of space in the existing lab. Since these are passive area changes, they will not affect any equipment operation important to safety. This change does not have any interaction effects.

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

Safety Evaluation No.:	96-042 Rev. 0 & 1
Implementation Document No.:	Mod. PN2Y94MX003
USAR Affected Pages:	9.2-25, 9.4-60, 12.5-2; Tables 9.4-10 Sh 1, 9.5-1 Sh 4, 9.5-2 Sh 3; Figures 1.2-8 Sh 1, 1.2-19 Sh 2, 9.2-8b, 9.2-9b, 9.3-10h, 9.4-2d, 9.4-2e, 9.4-12b, 9.4-19 Sh 1, 9.5-8 Sh 2, 9.5-30, 9A.3-5, 12.3-1, 12.3-8, 12.3-14, 12.3-34, 12.3-41, 12.3-47, 12.3-69 Sh 2
System:	DWS, PBS, DFT, HVL, HVT
Title of Change:	Auxiliary Service Building Renovation, Phase II

Description of Change:

This modification renovated the existing Auxiliary Service Building (ASB) el. 261'-0" men's locker room and restroom, women's restroom, and decontamination shower area to provide a controlled permanent access/egress to both the Turbine and Reactor Building from the Access Control Building and associated linkway. A portion of Turbine Building el. 250'-0" has also been renovated to allow access for the equipment lift.

The facilities previously located in ASB el. 261'-0" have been dismantled and rearranged to provide a radiation protection (RP) calibration room, RP storage room, male and female decontamination shower areas, and an additional personnel access from the linkway to the ASB inside the radiologically-controlled area (RCA). The entire facility has been classified as part of the RCA.

Safety Evaluation Summary:

This modification provides controlled access and egress to the Turbine and Reactor Buildings from the Access Control Building and linkway. All structures and systems involved are nonsafety related and nonseismic, except for some pipe supports for drain piping located in ASB el. 237'-0" which are seismically supported to protect safety-related equipment located below the piping.

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

Safety Evaluation No.: 96-044

Implementation Document No.: Temporary Mod. 96-014

USAR Affected Pages: N/A

System: Reactor Protection (RPS), Nuclear Steam Supply Shutoff (NS^4), Main Steam (MSS)

Title of Change: Defeat of Main Steam Line Rad Monitoring Trip Signal Channel A1

Description of Change:

This change installed a jumper in panel 2CEC*PNL606 Bay A in order to defeat a trip signal (Channel A1) which would normally be provided whenever detector 2MSS*RE46A is inoperable.

Safety Evaluation Summary:

This proposed change will defeat the RPS Channel A1 trip signal during maintenance and would allow time for maintenance of the faulty detector without initiating a 1/2 RPS trip and NS^4 isolation signal. Compliance with Technical Specifications 3.3.1.a and 3.3.2.b.1.b will be maintained. The station would immediately enter the required 12-hour LCO.

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 maintenance is being performed. 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 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.: 96-046

Implementation Document No.: DDC 2F01333

USAR Affected Pages: Figures 10.1-7L, 10.1-7m, 10.1-7n

System: Feedwater Heater Relief Vents and Drains (SVH), Condensate (CNM), Low-Pressure Feedwater Heater Drains (HDL)

Title of Change: Replacement of Flanges with Stainless Steel Couplings on 4th Point Heaters 2CNM-E4A, 2CNM-E4B, and 2CNM-E4C Vent and Drain Lines

Description of Change:

This portion of the SVH system supports the CNM system by venting gases from the 4th point feedwater heaters 2CNM-E4A, 2CNM-E4B, and 2CNM-E4C and returning the gases to the condenser. The drain lines drain water from these 4th point heaters and return it to the condenser.

The original vent lines 2-SVH-150-406-4, 2-SVH-150-436-4, and 2-SVH-150-466-4, and original drain lines 2-SVH-002-407-4, 2-SVH-002-437-4 and 2-SVH-002-467-4 were designed utilizing flanges. The flanged connection in drain line 2-SVH-002-4674 leaked and a temporary repair was installed. The flanged design for the vent and drain lines was proven inadequate due to continual leakage and was maintenance labor intensive.

The new design for each vent and drain line removed the flanged connections and installed stainless steel, socket-welded couplings in their place.

Safety Evaluation Summary:

The flanges are unnecessary since the large bore lines connected to the feedwater heaters would have to be cut and rewelded if the feedwater heaters were to be replaced. The savings of having these lines flanged is minimal. The new design will ensure system leak-tightness and alleviate increased maintenance and ALARA issues. The new design will be constructed utilizing materials compatible with the original material and pressure class of the system.

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

Safety Evaluation No.: 96-062

Implementation Document No.: Mod. PN2Y89MX086

USAR Affected Pages: Figures 1.2-10 Sh 1, 9.5-18, 9A.3-8, 12.3-11, 12.3-44

System: Reactor Water Cleanup (WCS)

Title of Change: Install a 3'-0" x 6'-0" Door in the South Wall of the Precoat Room, Located in the Reactor Building Elevation 328'-0" (Phase 1)

Description of Change:

This modification added a permanent doorway (Door No. R328-10) into the WCS precoat room, located in the secondary containment area of Reactor Building el. 328'-10". Door R328-10 is located near the southwest corner of the WCS precoat room, allowing plant personnel to readily transport boron containers in and out of the room. The addition of Door R328-10 required the redesign of pipe support BZ-426FT, and removal of a portion of an existing Type XVI embedded plate to allow clearance for cutting the concrete opening.

Safety Evaluation Summary:

The rough opening of approximately 3'-4" x 6'-2" for the installation of Door No. R328-10, located near the southwest corner of the south wall of the WCS precoat room, will improve accessibility for plant personnel to transport boron containers in and out of the room. Appropriate engineering design measures for seismic and other operating conditions were followed to maintain the structural integrity and intended function of the concrete wall about the WCS precoat room. Calculation ES-106 was reviewed to determine applicable environment criteria for the new door. This review indicated that a ΔP of approximately 10 psi is expected in the cubicle due to high-energy line break in an adjacent cubicle and a concurrent thimble seal failure between the cubicles. The new door will have a louver installed to allow dissipation of this pressure to preclude any damage to safety-related equipment in the area.

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

Safety Evaluation No.: 96-063

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

USAR Affected Pages: N/A

System: Service Water (SWP), Reactor Building Ventilation (HVR)

Title of Change: Freeze Seal for Maintenance of 2SWP*V152A and 2SWP*V153A

Description of Change:

This change installed temporary freeze seals in the supply and return service water piping to unit coolers 2HVR*UC401A and 2HVR*UC401D. The freeze seals allow internal repair of the service water isolation valves for 2HVR*UC401A. Repair of these isolation valves was necessary to support unit cooler testing in response to NRC Generic Letter 89-13. The freeze seal installation invokes the appropriate plant Technical Specification Limiting Conditions for Operation (LCO).

Safety Evaluation Summary:

Freeze sealing is a common industry application when it is necessary to repair or modify plant piping systems and components where no ready means of isolation is available, and it is uneconomical or otherwise not desirable to take the entire system out of operation. These freeze seals are in 1 1/2-inch and 2 1/2-inch diameter NPS carbon steel service water piping. Adequate contingency measures will be in place such that any freeze seal failure would not affect the operability of the SWP system. The freeze seal procedures employed are in conformance with guidance provided in applicable regulatory and industry documents relating to pipe freeze seals.

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

Safety Evaluation No.: 96-064

Implementation Document No.: Mod. PN2Y95MX011

USAR Affected Pages: Figure 9.4-12b

System: Turbine Building Ventilation (HVT)

Title of Change: Flow Adjustments on Turbine Building Elevation 277' and the Main Steam Lead Enclosure

Description of Change:

This modification installed one additional register and volume damper in the lead enclosure area. This circulates the air around the main steam lead enclosure area that had shown higher temperature readings, and decreases the circulation in other parts of the area. This change provides a more even distribution of the supply air throughout the main steam lead enclosure.

In addition, a register was installed in the exhaust of the turbine stop valve area. This register exhausts air in the hotter area of Turbine Building El. 277' to help in cooling this area.

Safety Evaluation Summary:

The proposed change will improve ambient conditions in the area by providing a better distribution of cooling air. These changes do not adversely impact the function of any safety-related structural system or component. The capability to safely shut down the reactor and maintain it in that condition is also not adversely affected.

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

Safety Evaluation No.: 96-066

Implementation Document No.: Simple Design Change SC2-0122-93

USAR Affected Pages: 6.2-55, 6.2-56, 6.2-57, 6.2-58, 6.2-64, 6.2-65, 6.2-66, 6.2-68, 6.2-70

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

Title of Change: D/P Setpoint Changes for GTS and HVR Systems

Description of Change:

The negative pressure differential in the secondary containment is required to be no less than 0.25 in. WG when compared with the outside ambient atmosphere. The setpoint is conservative provided the outside air temperature (ambient air) is below the secondary containment temperature. Normally, for most of the time during the year, the secondary containment air is warmer than the outside air. However, the potential of secondary containment temperature reversal (resulting in secondary containment being cooler than the outside ambient air) exists. The control and alarm setpoints of the GTS and HVR systems do not take into consideration the effect of such temperature reversals. This situation has the potential for decreasing the negative differential pressure to less than 0.25 in. WG. Such a situation is possible during the summer months because the latest analysis for one-hour drawdown time has resulted in suspension of the secondary containment heating. This safety evaluation addressed the discrepancy noted and revised the GTS and HVR setpoints and other affected analyses.

Safety Evaluation Summary:

The setpoint calculations for the GTS system have been revised to address the effect of temperature reversal. Based on the calculation, a worst case (30°F temperature reversal) analytical limit of -0.47 (0.444 + 5% margin) in. WG has been identified. The actual setpoint for the GTS system shall be established such that the Technical Specification 6.2.3 requirement of maintaining a minimum negative differential pressure of 0.25 in. WG anywhere in the secondary containment is maintained during normal plant operating conditions. The weak link in the secondary containment structure is the roof decking. The decking is designed to withstand a negative internal pressure of 0.45 psi (-12.45 in. WG). The revised setpoints will assure that there is no detrimental effect on penetration seals, and will also address the operational concerns with respect to the GTS and HVR systems.

Safety Evaluation No.: **96-066 (cont'd.)**

Safety Evaluation Summary: (cont'd.)

The drawdown requirement under worst case 30°F temperature reversal is less limiting than when outside air is colder than the secondary containment air. Therefore, the GTS function of establishing and maintaining 0.25 in. WG is unaffected by the GTS controller setpoint change to accommodate the temperature reversal phenomenon. Additionally, the secondary containment/ service water differential temperature requirement is unaffected because the in-leakage with 30°F temperature reversal is bounded by cases analyzed previously.

During normal plant operation, the secondary containment negative differential pressure of -0.25 in. WG is maintained by the HVR system. Therefore, the same considerations apply to the HVR system differential pressure setpoints as for the GTS system. The pressure control and alarm setpoints for the HVR system shall also be revised in order to eliminate conflict with the GTS requirements. Review of the GTS and HVR systems indicates that they are capable of accommodating the increased setpoints.

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

Safety Evaluation No.: 96-073

Implementation Document No.: GE Report No. CFAD-93056, NETCO Report No. NET-113-01 Rev. 1.0

USAR Affected Pages: 9.1-6, 9.1-10, 9.1-11, 9.1-55

System: Spent Fuel Storage Racks

Title of Change: Update of USAR Section 9.1 Per New Design Basis

Description of Change:

This evaluation updated the design basis for the spent fuel storage racks in USAR Section 9.1. Previously, USAR Section 9.1 was based on 3.6 weight percent U-235 without Gd_2O_3 rods, which does not reflect the existing core bundle design. This change ensures that the criticality criteria for spent fuel racks are maintained as long as enrichment, in-core K_∞, and Gd_2O_3 concentrations are satisfied. This evaluation updates the current USAR Section 9.1 to ensure that the criticality calculation for the fuel designs with a maximum average U-235 enrichment up to 4.65 weight percent meets the spent fuel criticality criterion.

Safety Evaluation Summary:

The acceptance criterion for spent fuel criticality will not change. The calculations performed to support the update are consistent with regulatory criteria and are not being changed in a manner not previously assessed. The USAR has been updated for the spent fuel storage racks for U-235 enrichment up to 4.65 weight percent. No fuel handling equipment or procedures are being changed. Therefore, the potential for an inadvertent criticality will not increase.

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

Safety Evaluation No.: 96-082 Rev. 0 & 1

Implementation Document No.: Design Change N2-96-030

USAR Affected Pages: Tables 3.9A-12 Sh 11, 9.2-1A Sh 2, 3, 9.2-1B Sh 2, 3, 9.4-2 Sh 6; Figure 9.2-1f

System: Service Water (SWP)

Title of Change: Removal of Service Water Check Valves 2SWP*V201A and *V201B

Description of Change:

Check valves 2SWP*V201A and *V201B are installed in the divisional service water supply lines to HPCS Switchgear Room unit cooler 2HVC*UC102. These valves were subject to low flow conditions during normal operation when the service water flow is split between two divisions. The low flow condition was exacerbated by increased flow resistance due to fouling. The original piston-style check valves repeatedly failed during quarterly forward flow tests because they were installed in a non-preferred orientation and the available flow rate was less than the vendor-specified critical velocity. Silt and corrosion product deposition within the valves also restricted movement of the disc, producing a throttling effect. The valves were replaced during refueling outage RFO4 with nozzle-style check valves with a suitable design for achieving and maintaining full disc open during normal operating flow conditions. Quarterly tests of the new valves revealed a vulnerability to reverse flow failures due to debris and corrosion product deposition. A vendor-specified design enhancement successfully increased internal clearances, but did not eliminate reverse flow failure due to foreign particles on the seating surfaces. This design change eliminated the problem by re-evaluating unit cooler 2HVC*UC102 for a reduction in service water flow rate and replacing check valves 2SWP*V201A and *V201B with flanged spool pieces.

Safety Evaluation Summary:

The proposed changes satisfy the design, material, construction standards, and practices applicable to the SWP system. As such, the likelihood of an accident is not affected by the proposed change. The hydraulic impact of removing check valves 2SWP*V201A and *V201B and the reduced heat removal capability of unit cooler 2HVC*UC102 have been reconciled in NMP2 design calculations. The proposed change could not initiate, nor act as a precursor to, an accident 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.: 96-091

Implementation Document No.: Procedure N2-ISP-MSS-R109

USAR Affected Pages: N/A

System: Reactor Protection (RPS), Nuclear Steam Supply (NSSS), Main Steam (MSS)

Title of Change: Defeat of Main Steam Line Rad Monitoring Trip Signal During Maintenance

Description of Change:

This change will defeat the affected RPS channel trip signal during maintenance of main steam line radiation monitors by installing a jumper in panel 2CEC*PNL606 or 2CEC*PNL633, depending on the affected monitor, in order to defeat a trip signal which would normally be provided whenever a main steam line radiation monitor detector is inoperable.

Safety Evaluation Summary:

Compliance with Technical Specifications 3/4.3.1.a and 3/4.3.2.b.1.b will be maintained. The station will immediately enter the required 12-hour LCO. If the affected detector can not be returned to operable status within the 12-hour LCO time limit, LCO action will be entered as required per the Technical Specifications.

This procedural change reduces the plant's vulnerability to a full scram and main steam isolation valve isolation by eliminating the half scram and NS[4] signal which would otherwise be present during the time period that maintenance is being performed. In the event of gross fuel failure, 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 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.: 96-094

Implementation Document No.: N/A

USAR Affected Pages: 12.5-12

System: N/A

Title of Change: Airborne Radioactivity Survey Program Refinement

Description of Change:

This change clarifies and enhances portable air sampling program requirements by:

1. Eliminating the specific value of 3 cfm from Unit 2 USAR Section 12.5.3.3.4, which contradicts Table 12.5-4. Table 12.5-4 provides a range of 1-6 cfm at an accuracy of $\pm 10\%$. This change provides consistent guidance concerning acceptable flow rates when performing high volume air samples.

2. Requiring the use of iodine cartridges for high volume air samples as deemed appropriate after an evaluation of plant conditions by Radiation Protection personnel.

Safety Evaluation Summary:

Deletion of the specific value of 3 cfm from USAR Section 12.5.3.3.4 will eliminate contradictory guidance given in the USAR. Revising iodine cartridge sampling requirements to "as required based on plant conditions" will provide flexibility in line with current acceptable industry guidance for monitoring radioactive airborne contamination.

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

Safety Evaluation No.: 96-126

Implementation Document No.: DDC 2E11219

USAR Affected Pages: Figure 11.5-1

System: Containment Monitoring (CMS)

Title of Change: Disable 2CMS*CAB10A Automatic Flow Control

Description of Change:

This change deleted the automatic sample flow control feature of off-line gaseous/particulate radiation monitor 2CMS*CAB10A.

Safety Evaluation Summary:

Deleting the automatic sample flow control does not affect the monitor's ability to detect small unidentified leaks within the primary containment. Flow will nominally be set between 1.0-3.0 scfm (2.25 scfm). Flow indication and alarm functions will remain operable. If flow exceeds the sample flow alarm setpoints, alarm annunciation will occur in the Main Control Room and the Radiation Protection (Health Physics) Room to alert the operators and radiation protection technicians to adjust flow to within the normal operating band. Experience has shown that 2CMS*CAB10A sample flow has only minor flow variations because the process flow (drywell atmosphere) is essentially constant. Therefore, sample flow does not require frequent adjustment.

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

Safety Evaluation No.:	96-127
Implementation Document No.:	DDC 2M11110
USAR Affected Pages:	Figure 10.1-5a
System:	Condensate (CNM), Condenser Air Removal (ARC)
Title of Change:	Intercondenser 2ARC-E3A/B Valve Line-up Change

Description of Change:

One steam jet air ejector (SJAE) intercondenser train is required to operate during normal operation while the other intercondenser train is in standby. Accordingly, operating procedures N2-OP-3 and N2-OP-9 require that condensate inlet isolation valves 2CNM-MOV64A or B be closed to the out-of-service SJAE intercondenser. In order to minimize the potential of pressure pulsations experienced by the out-of-service SJAE intercondenser (due to feedwater pump trip, etc.), causing increased leakage to the offgas system due to a potential tube leak, this change isolated the standby intercondenser by means of condensate outlet isolation valves 2CNM-MOV65A or B in lieu of inlet isolation valves 2CNM-MOV64A or B.

Safety Evaluation Summary:

This change will not affect performance of either the CNM or the ARC systems as required per original design. The change being made will not add or delete a safety-related function. No system interaction impact is introduced, and malfunction or failure of the systems or changes involved will not compromise any safety-related system or component, or prevent a safe reactor shutdown.

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

Safety Evaluation No.: 96-128

Implementation Document No.: Calculation H06E-201

USAR Affected Pages: 9.2-28, 9.2-31, 9.2-40

System: Circulating Water (CWS), Service Water (SWP)

Title of Change: Service Water Tempering

Description of Change:

The USAR has been revised to state that service water intake may be tempered, and tempering control may be in automatic or manual as required to comply with the Environmental Protection Plan. Because of compliance issues with the SPDES permit, the minimum service water inlet temperature of 38°F, as described in the USAR, is not always maintained during winter; therefore, tempering flow does not provide freeze protection nor frazil ice protection to the service water intake structure during winter operations.

Safety Evaluation Summary:

Tempering is not a design requirement to satisfy the requirements for ultimate heat sink. Compliance with the SPDES permit is a license requirement for plant operation. It is not always possible to remain in compliance with the minimum 38°F service water intake temperature, as stated in the USAR. Maintaining the service water intake water temperature greater than or equal to 32°F is acceptable since this is within the SWP system design bases.

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

Safety Evaluation No.:	96-129
Implementation Document No.:	DDC 2M11112
USAR Affected Pages:	Figure 9.3-20b
System:	Nitrogen (GSN)
Title of Change:	Close Manual Supply Valves 2GSN-V119 and V116

Description of Change:

Pressure control valve 2GSN-PCV144 supplies 100 psig nitrogen to the non-ADS accumulators, the inboard MSIV accumulators, sampling, and primary containment instrumentation and controls through receiver tank 2GSN-TK2. To correct for 2GSN-PCV24A/B leakage, procedure N2-OP-61A was revised to maintain 2GSN-V116 and V119 normally closed. However, valve 2GSN-V116 was shown normally closed, and 2GSN-V119 was shown normally open on the USAR figure. This configuration showed that one pressure-reducing station was in standby while the other was operating.

This change revised the USAR to show both 2GSN-V116 and V119 normally closed and allow nitrogen supply to 2GSN-TK2 to be placed into service manually.

Safety Evaluation Summary:

Isolating the high-pressure nitrogen supply by manual block valves 2GSN-V116 and V119 does not adversely affect operation of the nitrogen system. To place the backup high-pressure nitrogen supply into service, either of the block valves would have to be manually opened. This manual procedure will be completed within one hour.

Nitrogen gas will continue to be supplied to safety-related instrumentation and controls required for safe shutdown. This nitrogen gas supply is normally isolated and includes supplying nitrogen for post-accident sampling purge and testing of drywell vacuum breakers and testable check valves.

Any potential loss or decrease in nitrogen receiver tank pressure for a short duration does not impact the safety function of the non-ADS SRVs.

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

Safety Evaluation No.: 96-130

Implementation Document No.: DDC 2M11111, Calculation WM-B-027 Rev. 2, Disposition WM-B-027-02C

USAR Affected Pages: 2.2-7, 2.2-8, 2.2-12; Tables 2.2-7, 2.2-8

System: Service Water Chemical Treatment (SCT)

Title of Change: Outside Venting of Sodium Bisulfite Tank 2SCT-TK1

Description of Change:

To ensure personnel safety from sulfur dioxide emission and to meet pending New York State Department of Environmental Conservation regulations (Part 599.18.a.5) which take effect on December 22, 1999, the sodium bisulfite tank for the SCT system (2SCT-TK1) has been vented outside the acid storage section of the Screenwell Building. This required modification of the existing vent by removing the unnecessary desiccant chamber (2SCT-DRY1) and the inlet vent check valve (2SCT-V76). A vent was installed in its place which runs up and out the west wall of the building. In order to meet an additional pending environmental regulation (Part 599.9.a.2.iii) which requires that overfills from vents occur within the secondary containment (existing berm), the existing overflow check valve (2SCT-V75) was replaced with a rupture disc. The new vent line includes a float-actuated vent valve so that, during an unexpected overfill event, the vent valve will isolate on rising liquid level. Thus, if overfilling continued, the tank would pressurize to the setpoint of the rupture disc and the liquid would relieve through the existing overflow line to the sump in the berm.

Safety Evaluation Summary:

The use of the sodium bisulfite tank, previously evaluated under Safety Evaluation 96-071, will not functionally change. The method of venting the tank will be modified to allow fumes from the stored chemical to discharge harmlessly to the atmosphere. The tank supports the operation of the SCT system, which chemically treats service water but is not required for the continued operation of the service water system. Gases produced by normal venting or unexpected spills/overflows will not impact Control Room habitability for Units 1 and 2, or the function of Control Room ventilation system equipment. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.: 96-131

Implementation Document No.: Procedure GAP-RPP-02

USAR Affected Pages: 12.5-4, 12.5-11 thru 12.5-16

System: N/A

Title of Change: USAR RWP Amendments

Description of Change:

This change revised the USAR to agree with current Radiation Protection Program requirements associated with the administration of the radiation work permit (RWP) program. The change eliminated outdated terminology and practice and replaced it with conservative current terminology and practice.

Safety Evaluation Summary:

The proposed changes to the USAR will meet the intent of Regulatory Guide 8.8, the Standard Review Plan, INPO 91-014, and Technical Specification 6.12.

RWPs will continue to contain data on radiation levels in the area, allowable working time, protective clothing and respiratory equipment, special tools, portable shielding, and special personnel monitoring devices.

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

Safety Evaluation No.: 96-132

Implementation Document No.: Procedure EPIP-EPP-08

USAR Affected Pages: N/A

System: Meteorological Monitoring (MMS)

Title of Change: Meteorological Instrumentation

Description of Change:

This change identifies the existing meteorological instrumentation located on the primary tower as the primary backup meteorological instrumentation. This change also corrected reference elevations for the existing instrumentation on the primary, J. A. FitzPatrick plant (JAF) backup, and inland towers. This safety evaluation justifies the use of the existing instrumentation as backup to the Technical Specification instrumentation until a Technical Specification Amendment can be approved. Since the Technical Specifications require only a report to be submitted to the NRC when the meteorological instrumentation is inoperable, the Technical Specifications do not prevent the identification of backup instrumentation. The NMP2 USAR identifies the JAF Meteorological Tower as the backup tower. This change added the existing instrumentation on the primary meteorological tower as the primary backup and the JAF Meteorological Tower as the secondary backup.

Safety Evaluation Summary:

The existing meteorological instrumentation and the meteorological measurements program meet the intent and recommendations of Regulatory Guide 1.23 and NUREG-0654. This change will not increase the probability of occurrence or the consequences of an accident or malfunction 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.: 96-134

Implementation Document No.: Plant Change Request PC2-0241-96

USAR Affected Pages: 11.3-5, 11.3-6; Figure 11.3-1b

System: Offgas (OFG)

Title of Change: Offgas System Dryer Dew Point and Temperature Alarm Changes

Description of Change:

The offgas dryers (2OFG-DRY1A, B, and C) have a moisture element (2OFG-ME116) which measures the outlet dew point from the dryers. It was determined that this element was susceptible to becoming wetted, which artificially increased its dew point reading and caused nuisance alarms. When this occurred, the actual dew point was determined by reading temperature indicators which read the temperature in the dryers and equated directly to dew point temperature (2OFG-TI61A, B, and C).

This change eliminated the alarm function of the moisture element to remove the nuisance alarm and replaced it with a temperature alarm from 2OFG-TE61A, B, and C which, at 100% relative humidity, equates directly to dew point temperature.

Safety Evaluation Summary:

This change will maintain the intent of the moisture element alarm function, which serves to guard the charcoal adsorbers from moisture contamination resulting from dryer or condenser damage. Specifically, a new setpoint of 50°F dry-bulb temperature equals the current moisture element setpoint of 50°F dew point. This will bound the maximum 50°F dew point assumed in calculating the annual releases of radioactive gaseous effluents. In addition, there will be no effect on the calculated radiological consequences of the postulated radioactive gaseous waste system leak or failure. Finally, shielding requirements for the charcoal adsorber cubicles, as documented in Calculation PR-C-26-A, will remain unaffected, thereby ensuring that the radiation zone limits for the adjoining areas are not exceeded.

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

Safety Evaluation No.: 96-135

Implementation Document No.: Plant Change Request PC2-0028-96

USAR Affected Pages: Figure 9.3-20a

System: Nitrogen (GSN)

Title of Change: Disabling Nitrogen Low Temperature Alarm Input

Description of Change:

Low temperature alarm in the outdoor high flow section of the nitrogen inerting system causes a nuisance alarm due to cold weather conditions. This simple design change disabled the alarm while still relying on the dedicated low temperature alarm sensing the nitrogen in the downstream common header used for inerting.

Safety Evaluation Summary:

The outdoor nitrogen system low temperature alarm from the high flow inerting section is not needed since there is a separate and dedicated low temperature alarm in the common nitrogen inerting header. There is no change to the valve isolation logic associated with low temperature downstream closure which protects downstream components from liquid/low temperature gaseous nitrogen. There is no change to the instrument nitrogen portion of the system since there is no change to the nitrogen system economizer section from which instrument nitrogen is supplied. The disabled alarm does not perform any safety-related function nor is it required for 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.: 96-137

Implementation Document No.: DDC 2E11278

USAR Affected Pages: 5.4-35, 7.4-6, 9B.8-2; Tables 3.9A-12 Sh 8, 14, 9B.8-3 Sh 9, Figures 5.4-13a, 5.4-13f

System: Residual Heat Removal (RHS), Remote Shutdown (RSS)

Title of Change: De-energize and Disconnect Power Source to MOVs 2RHS*MOV112, 142, and 149

Description of Change:

This change de-energized motor-operated valves (MOV) 2RHS*MOV112, 142, and 149 in the closed position and disconnected them from their power sources. In addition, the nuisance alarms caused by the disconnection of power sources were eliminated by lifting leads at the respective motor control centers of these MOVs or by addition of a control switch in the alarm circuit such that the Control Room will not have a continuous nuisance alarm. The component level inoperable indication lights are still available in the Main Control Room.

Safety Evaluation Summary:

This change will improve the availability of the shutdown cooling mode of the RHR system from the remote shutdown panel (RSP) rooms by precluding mechanical damage to MOVs involved during a Main Control Room fire and subsequent Main Control Room evacuation. Review of this change revealed no adverse impact to safe shutdown of the plant from the RSP rooms if the Main Control Room is evacuated due to a fire.

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

Safety Evaluation No.:	97-003
Implementation Document No.:	Procedure NIP-FFD-02
USAR Affected Pages:	Table 1.8-2 Sh 3
System:	N/A
Title of Change:	Change to NIP-FFD-02 Which Extends Respirator Physicals to Once per 2 Years for Select Groups of Personnel

Description of Change:

In February 1995, 10CFR Part 20 was changed to state that respirator qualifications will include a "physician's determination prior to initial fitting of respirators and...periodically at a frequency determined by a physician that the individual is medically fit to use the respiratory protection equipment". Previously, the guidance of Regulatory Guide 8.15, paragraph C.4.h, required "...the medical status of each respirator user is to be reviewed at least annually". Because the requirement for respiratory physicals was relaxed by the NRC, the requirements have been relaxed as stated in the USAR.

Safety Evaluation Summary:

The changes to NIP-FFD-02 are based on the current regulations of 10CFR Part 20 and as prescribed by the company physician. The changes meet or exceed all current requirements for respirator qualification physical periodicity.

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

Safety Evaluation No.:	97-026 Rev. 0 & 1
Implementation Document No.:	Mod. PN2Y95MX009
USAR Affected Pages:	9.5-50, 9.5-51, 9.5-55, 9.5-60; Figure 9.5-40a
System:	Diesel Generator Air Start
Title of Change:	Replace Division I and II Diesel Generator Air Dryers

Description of Change:

Air in the diesel generator air start receiver tanks was constantly being bled down by pneumatic instrumentation in the air start system. The compressors turned on when the pressure in the air receiver tanks reached the low setpoint and would turn off when the high setpoint was reached. Thus, flow to the air receiver tanks through the dryers was intermittent.

The intermittent compressor cycle caused the shuttle valves in the desiccant-type air dryers to misposition and ultimately fail. The desiccant-type air dryers were not designed for this intermittent application.

This modification replaced the desiccant-type air dryers with nonsafety-related refrigerant air dryers.

Safety Evaluation Summary:

The refrigerant air dryers will provide clean, dry air at design pressure and temperature. The dew point of the outlet air will be in accordance with the requirements of the Standard Review Plan (NUREG-0800). The new air dryers will remain capable of being isolated from the safety-related air receiver tanks such that failure of the dryers will not affect the air supply required to start the diesel generator. The reliability of the emergency diesel generator air start system will continue to be maintained to the standards required.

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

Safety Evaluation No.: 97-027

Implementation Document No.: Procedure N2-OP-78

USAR Affected Pages: 5.4-32, 7.4-5, 7.4-6, 9B.8-3; Tables 3.9A-12 Sh 8, 14; Figure 9B.4-1

System: Residual Heat Removal (RHS)

Title of Change: "Pseudo LPCI" Injection

Description of Change:

In response to Information Notice 92-18, "Potential for Loss of Remote Shutdown Capability During a Control Room Fire," and with insights gained from the implementation of Generic Letter 89-10, "Safety-Related MOV Testing," it was determined that an electrical short could cause damage to one or more reactor core isolation cooling (RCIC) system MOVs such that the RCIC system would not be available to the operator at the remote shutdown panel (RSP) following a Main Control Room evacuation, as described in 10CFR50 Appendix R. Since RCIC is the only proceduralized method of inventory control available to the operator at the RSP, an alternate method of inventory control was required to maintain the reactor coolant above the top of active fuel.

This safety evaluation evaluated an alternate low-pressure injection methodology available to the operator utilizing existing controls at the RSP. Because this alternate low-pressure injection is essentially equivalent to the low-pressure coolant injection (LPCI) mode of RHR, this methodology is termed "pseudo LPCI". The "pseudo LPCI" mode of RHR takes suction from the suppression pool and delivers water to the reactor coolant system through remote manual shutdown cooling valve 2RHS*MOV40A(B), which is capable of being remotely operated from the Remote Shutdown Room. This mode differs from the normal shutdown cooling mode of RHR in that relatively cold water (70°F) from the suppression pool is delivered to the reactor vessel at near rated temperature.

Safety Evaluation Summary:

Recirculation piping and RHR shutdown cooling piping to the recirculation loops "A" and "B" have been evaluated for injection of cold water at 70°F to the recirculation piping at 535°F. The postulated event would have no adverse effect on the integrity of the RHR and recirculation piping inside the primary containment. The containment and the reactor pressure vessel (RPV) and internals were evaluated to be acceptable during this postulated event. For this "pseudo LPCI" mode of RHR, it was postulated that the high-pressure injection systems of high

Safety Evaluation No.: 97-027 (cont'd.)

Safety Evaluation Summary: (cont'd.)

pressure core spray (HPCS) and RCIC would not be available. Controls for HPCS would not be available from the RSP, and RCIC was postulated to be inoperable due to hot shorts resulting from a fire in the Main Control Room. Thus, to depressurize the RPV to allow the RHR pump injection, four of the safety relief valves (SRV) will be used since only four SRVs are available from the RSP. With the RPV being depressurized by the four SRVs to allow the "pseudo LPCI" mode, the temperature decreases at a greater rate than specified in Technical Specification 3.4.6.1 of 100°F/hour. However, the rapid decrease in reactor coolant temperature is bounded by the analysis for all seven ADS SRVs and documented in Section 3.9B of the USAR. This analysis allows a one-time only automatic blowdown which bounds depressurization of the RPV with only four SRVs. By cooling the RPV at a rate greater than 100°F/hour, the Technical Specification 3.4.6.1 action statement will be entered and the required actions performed.

Once the decision to evacuate the Main Control Room has been made and the reactor scrammed, the operators have nine minutes to initiate depressurization of the RPV with the four SRVs available at the RSP in order to use the "pseudo LPCI" injection. The specified time of nine minutes to initiate depressurization with the four SRVs allows sufficient time for the "pseudo LPCI" mode to supply water to the reactor coolant system before the core is uncovered. The specified time of nine minutes has been evaluated and provided by analysis.

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

Safety Evaluation No.: 97-028

Implementation Document No.: DDC 2E11295

USAR Affected Pages: 7.7-26

System: Fuel Nuclear Refueling (FNR)

Title of Change: Replace Dillon Force Switches on Refuel Bridge Main Hoist

Description of Change:

This change replaced existing force switches located on top of the load cell at the refuel platform main hoist with programmable logic controller (PLC) logic. The force switches provided electrical weighing interlocks for "Hoist Loaded", "Redundant Hoist Loaded", and "Hoist Jam". However, the force switches were prone to drift and were difficult to calibrate. The new PLC logic provides the same interlocks which were previously performed by the force switches, while improving operation and maintainability of the load weighing system for the main hoist. Required circuit redundancy is maintained with the PLC logic.

Safety Evaluation Summary:

The PLC program is utilized in the same manner as the previously installed force switches. PLC logic for each setpoint is used in place of the previously installed force switches for each setpoint. Load weighing system circuit redundancy has been maintained by configuring the PLC logic for each setpoint to operate electrically the same as its corresponding force switch. The analog-to-digital upgrade did not introduce any new performance characteristics or new design features that would give an increased probability of a system malfunction resulting in an accident. The circuits continue to provide the same logic for each setpoint. Thus, a single switch or contact failure will not prevent the circuits from opening as intended when the load setpoint is reached. With this change, the main hoist and fuel grapple continue to perform their functions.

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

Safety Evaluation No.: 97-031

Implementation Document No.: DER 2-95-1143

USAR Affected Pages: Figure 4.6-5C

System: Control Rod Drive (RDS)

Title of Change: 2RDS*AOV123 and 2RDS*AOV130 USAR Figure Discrepancy

Description of Change:

This change revised the piping and instrumentation diagram and corresponding USAR figure to illustrate that valves 2RDS*AOV123 and 2RDS*AOV130 are air-operated globe valves which support the RDS system. These valves were previously indicated as diaphragm valves. In addition, the In-service Testing Program Plan was revised to reflect the same change.

Safety Evaluation Summary:

No field changes are being performed to these valves, and the design and operation will remain as is. Overall system performance is not affected, nor does this change affect the normal operating position of the valves. These valves will remain normally open and continue to keep the scram discharge volume empty. The control circuitry is not being revised by this change. These valves will continue to close after a scram which will support retention of reactor water.

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

Safety Evaluation No.: 97-032

Implementation Document No.: DDC 2S11004

USAR Affected Pages: 9C.8-4; Tables 3-1, 3-4, 4-1 Sh 2 (Appendix 9C); Figure 5-1 (Appendix 9C)

System: Reactor Building Cranes and Elevators (MHR)

Title of Change: Remove MSIV Cranes 67A - 67D References from USAR and Drawings

Description of Change:

This change removed USAR Appendix 9C references to cranes 2MHR-CRN-67A through 67D which previously reflected the actual lifting arrangements for MSIV equipment removals. In lieu of using these four cranes, the actual lifting arrangements typically consist of chain hoists and slings used in accordance with miscellaneous heavy load procedure N2-MMP-GEN-923.

Safety Evaluation Summary:

The loads lifted during the MSIV work are classified as Hazard Category "f" per Table 4-1 of USAR Appendix 9C and, therefore, would not impact equipment important to safety in the event a load drop was to occur. Use of procedure N2-MMP-GEN-923 satisfies the heavy load commitments of NUREG-0612 as discussed in USAR Appendix 9C.

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

Safety Evaluation No.: 97-033

Implementation Document No.: Specifications GEK-83337B, GEK-90425A

USAR Affected Pages: 5.4-40

System: Residual Heat Removal (RHS)

Title of Change: RHR Heat Exchanger Operation

Description of Change:

The USAR has been revised to reflect the current practice of placing the RHR heat exchanger in service during shutdown cooling. The recommended industry practice is to establish the cold medium (service water) first, followed by the gradual introduction of the hot medium (reactor coolant). This practice ensures equipment reliability, and is consistent with the vendor design documents and industry practice.

Safety Evaluation Summary:

This change revises the method of equipment startup to ensure equipment reliability as outlined in vendor documentation. It does not change the functional requirements of the RHS system or introduce any new failure mode that has not been 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.:	97-034
Implementation Document No.:	Procedure N2-OP-11
USAR Affected Pages:	N/A
System:	Service Water (SWP)
Title of Change:	Closure of 2SWP*MOV599 and Opening of SWP-V8

Description of Change:

This safety evaluation reviewed the procedural changes necessary to close 2SWP*MOV599 and open 2SWP-V8 while at power. This procedural change was necessary to allow flow to be diverted from the nonsafety Turbine Building return header to augment the flow to the nonsafety Reactor Building return header. Additional flow was required to ensure that the circulating water system (CWS) basin level would not be impacted during maintenance on the Division II CWS makeup control valves.

Safety Evaluation Summary:

A change to procedure N2-OP-11 will allow closure of 2SWP*MOV599 and opening of 2SWP-V8 while at power. This change will support maintenance activities on Division II CWS makeup valves 2SWP*FV47B and 2SWP*FV54B. Original plant operation is designed with service water Turbine Building return isolation valve 2SWP*MOV599 in the open position during power operations and 2SWP-V8 closed. Closure of 2SWP*MOV599 will divert flow from the Turbine Building nonsafety return header via valve 2SWP-V8 to the Reactor Building nonsafety return header. The lineup change, as supported by Calculation A10.1-N-400, will ensure sufficient flow to the Division I header to maintain CWS basin level during maintenance of 2SWP*FV47B and 2SWP*FV54B.

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

Safety Evaluation No.:	97-037
Implementation Document No.:	DDC 2E11352
USAR Affected Pages:	Figure 5-3 (Appendix 9C)
System:	Traveling Screens Wash (SWT)
Title of Change:	Revise USAR Figure for Traveling Water Screens Control Panel

Description of Change:

USAR Figure 5-3 incorrectly identified SWT local control panel 2SWT-PNL108 as safety related, therefore restricting crane travel over the panel. The USAR figure has been revised to properly identify the panel.

Safety Evaluation Summary:

The removal of load travel restrictions over the traveling screens control panel will not alter the ability of the panel or its associated components to perform their intended functions. With this change, the panel will be correctly identified as nonsafety related and the appropriate heavy loads criteria will be maintained. The proposed change maintains all other applicable USAR criteria associated with the service water system, the ultimate heat sink, and traveling water screens. Neither the panel nor any of its associated equipment is required for safe shutdown or decay heat removal.

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

Safety Evaluation No.: 97-038

Implementation Document No.: DDC 2F01589

USAR Affected Pages: 9.2-23; Figure 9.2-5e

System: Water Treatment (WTS)

Title of Change: (Hold Out #2-94-HO532) Closure of Manual Valve 2WTS-V540

Description of Change:

Valve 2WTS-V540 was normally open between makeup waste neutralizing tank 2WTS-TK1 and the service water discharge bay. This piping and associated equipment provides the means of discharging makeup waste directly into Lake Ontario via the service water system. This change makes valve 2WTS-V540 closed as its normal operating configuration. Valve 2WTS-AOV247 closes automatically when pH levels are unacceptable.

Safety Evaluation Summary:

WTS is currently not in use. An administratively controlled trailer is available to provide demineralized water. However, with sufficient startup preparation, WTS is capable of fulfilling the normal operating requirements of the facility for acceptable makeup water with the necessary component redundancy if desired by the plant. This change is not affecting the ability of the WTS system to supply demineralized water nor the system's originally designed component redundancy.

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

Safety Evaluation No.: 97-040

Implementation Document No.: DDC 2E11325

USAR Affected Pages: Figure 7.7-2 Sh 8, 13, 14

System: Reactor Manual Control (RMCS)

Title of Change: RMCS Refueling Interlocks - Grapple Position

Description of Change:

This change deleted a notation on USAR Figure 7.7-2 that the "grapple down position" is a refueling interlock to the RMCS system. This change agrees with the logic described in the text sections of the USAR as well as the USAR logic diagrams and the physical as-built configuration.

Safety Evaluation Summary:

The refueling interlocks of the RMCS system are provided to restrict movement of refueling equipment or control rods during the refueling mode of operation under conditions that might place refueling floor operating personnel in potentially unsafe situations. These interlocks reinforce the refueling procedures and reduce the probability of inadvertent criticality, damage to reactor internals or fuel assemblies, and exposure of personnel to excessive radioactivity.

No Regulatory Guides nor industry standards were identified which form a basis to require "grapple down position" as an interlock to the RMCS. The operating procedures along with grapple vertical position indication, and the protective interlocks described in the text section of the USAR, provide adequate protection to reduce the probability of inadvertent criticality, damage to reactor internals or fuel assemblies, and exposure of personnel to excessive radioactivity. Therefore, grapple down position is not required as an interlock to the RMCS.

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

Safety Evaluation No.: 97-041

Implementation Document No.: DDC 2M11149

USAR Affected Pages: Figures 9.4-3b, 9.4-3c, 9.4-3d

System: Ventilation Chilled Water (HVN)

Title of Change: Closure of Valves 2HVN-V611A, B, C

Description of Change:

This safety evaluation evaluated a change to the normal operating position of 2HVN-V611A, B, C and 612A, B, C from open to closed. The HVN system is located in the Chiller Building adjacent to the Turbine Building and provides cooling for the Turbine Building, the Normal Switchgear Building, and the Radwaste Building. Each chiller utilizes two valves for maintenance and testing purposes (2HVN-V611A, B, C and 612A, B, C for chillers 2HVN-CHL1A, B, C, respectively). These valves are part of the vendor-supplied chiller assembly.

Safety Evaluation Summary:

The changes to the normal operating position of the subject valves will not affect their compliance with ANSI B31.1. The chillers' compliance with IEEE and NEMA MG1 codes and standards is not affected.

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

Safety Evaluation No.: 97-042

Implementation Document No.: Plant Change Request PC2-0128-96

USAR Affected Pages: Figures 9.5-15, 9.5-32

System: Communications (COP)

Title of Change: Provide Additional Communications at Access Points in Radiological Area and to Unit 2 Fire Chief's Office

Description of Change:

This change provided a permanent NMP2 Gai-tronics feed to the Fire Chief's office, which is located in the NMP1 Administrative Building. This Gai-tronics unit is required to monitor and communicate with Fire Response Teams at NMP2. Also, this change provides permanent NMP2 Gai-tronics at the access point in the radiological area to support Radiation Protection during drywell access for refueling and/or forced outages. This access point has been in essentially the same location each outage and is required to ensure adequate communication capabilities for working safely and productively.

Safety Evaluation Summary:

This change to the plant communications system will add/improve communication capabilities to meet the requirements in USAR Section 9.5.2. The change will add communication equipment in areas that have been identified as in need of these capabilities.

Components are identified as nonsafety related and do not impact safe operation or shutdown of the plant. The system function will not change.

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

Safety Evaluation No.: 97-043

Implementation Document No.: DDC 2E11328

USAR Affected Pages: 9.4-10; Figures 9.4-1c, 9.4-4 Sh 4

System: Control Building Ventilation (HVC)

Title of Change: De-energization of Duct Heaters 2HVC-CH8 and 2HVC-CH9 When Not Needed (Hold Out # 2-96-H0082)

Description of Change:

Duct heaters 2HVC-CH8 and 2HVC-CH9 supply area heating to the Control Room Supervisor's Office and the Training Room. Heater 2HVC-CH2 supplies area heating to the Main Control Room. Normally, the doors between these three rooms are open and heater 2HVC-CH2 furnishes the heating required for the Main Control Room and the Control Room Supervisor's Office and the Training Room. This safety evaluation evaluated de-energizing duct heaters 2HVC-CH8 and 2HVC-CH9 when they are not needed for additional area heating.

Safety Evaluation Summary:

Based on the reviews performed, it has been determined that duct heater 2HVC-CH2 is normally able to provide required heating for all three rooms and, therefore, maintain habitability of the Main Control Room. De-energizing duct heaters 2HVC-CH8 and 2HVC-CH9 will reduce the maintenance of the heaters and disruptions to normal plant operations made while maintenance is being performed. This change has no impact on radiological barriers, systems or 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.: 97-044

Implementation Document No.: Calculation FOF-004

USAR Affected Pages: 9A.3-18

System: Fire Protection Fuel Oil Supply (FOF)

Title of Change: Diesel Fire Pump Fuel Oil Tank Capacity and Sump

Description of Change:

USAR Section 9A.3.1.2.5.6 stated that the 660-gallon fuel oil storage tank 2FOF-TK1 diesel fire pump is located in the diesel fire pump room within a diked area. However, there is no dike installed. Rather, the area is provided with a sump below the fuel oil storage tank. In addition, the design capacity of the fuel oil storage tank 2FOF-TK1 is 650 gallons.

Safety Evaluation Summary:

The installation of a sump underneath the diesel fuel oil storage tank rather than a diked area is an acceptable method to contain any spilled fuel oil from the storage tank in order to prevent the spread of flammable combustible liquid. The size of the sump is adequate to contain the 650 gallons of fuel oil in the storage tank. The containment of fuel oil in the sump will mitigate the fire hazard due to spillage.

Revising the tank capacity from 660 to 650 gallons has no impact, since the useable amount of fuel oil in the storage tank of 585 gallons remains the same.

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

Safety Evaluation No.: 97-046

Implementation Document No.: Mod. PN2Y95MX011

USAR Affected Pages: 9.4-57, 9.4-58, 9.4-59; Table 9.4-1 Sh 2; Figures 9.4-2C, 9.4-18

System: Auxiliary Boiler Room Ventilation (HVI)

Title of Change: Manual Operation of the Auxiliary Boiler Building Ventilation System

Description of Change:

In order to prevent occurrence of a radioactive release event, Holdout tags were placed on two of the HVI system modulating dampers (2HVI-AOD106 and 2HVI-AOD107) and the supply fans (2HVI-HVU1A/1B). The fans were placed in "pull to lock" position, damper AOD106 was placed in the closed position, and AOD107 was placed in the open position permanently. With these Holdout tags in place, the system could not be operated in the automodulating mode and air was being drawn from the room by fans 2HVI-FN2A/2B and discharged to the Reactor Building vent. This change allowed operation of the HVI system in the current configuration and use of supplemental portable heaters (if needed) for an extended period. The Holdout tags were removed after the design change was implemented.

Safety Evaluation Summary:

Damper DMPV1 was permanently closed to provide additional assurance against unmonitored release of radioactivity to the atmosphere. Dampers AOD8A and AOD8B are normally closed if the corresponding air handling unit is not in operation. These dampers can, however, be manually operated using their local handwheels, along with the associated dampers for the HVUs, as necessary, to allow some outside air into the room. In addition, the supply fans may be operated manually provided dampers DMPV1 and/or AOD106 are closed and the pressure inside the room is subatmospheric. Dampers AOD8A, B will open automatically when the corresponding HVU is started. The configuration and subatmospheric pressure ensure that the exhaust air does not escape to the atmosphere via the supply fan dampers where the effluents are not monitored. Therefore, the requirements of GDC 60 are satisfied.

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

Safety Evaluation No.: 97-047

Implementation Document No.: Procedure N2-OP-19

USAR Affected Pages: Figure 9.3-1c

System: Instrument Air (IAS)

Title of Change: Changing the Normal Position of the Out-of-Service Instrument Air Afterfilter Outlet Isolation Valve to the Open Position

Description of Change:

The normal position of the out-of-service afterfilter outlet isolation valve has been changed to the open position. The IAS system is equipped with two afterfilters located downstream of the IAS dryers. The afterfilters (2IAS-FLT3A and 2IAS-FLT3B) are configured parallel to each other and have an inlet and outlet isolation valve.

Safety Evaluation Summary:

Changing the normal position of the out-of-service IAS afterfilter outlet valve to the open position does not affect the design requirements of ANSI B31.1. System design pressure and the design temperature will not be affected. With the out-of-service IAS afterfilter outlet valve in the open position, the ability of instrument air being supplied to system end users will not be impacted.

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

Safety Evaluation No.: 97-048

Implementation Document No.: DDC 2F01564

USAR Affected Pages: Figure 5.4-9d

System: Reactor Core Isolation Cooling (ICS)

Title of Change: 2ICS*V234 Spare Instrument Connection, As-Installed Condition

Description of Change:

Valve 2ICS*V234, located in the RCIC room, was shown on the USAR figure as a 3/4" drain connection (root nipple, drain valve, outer nipple with cap) located between RCIC pressure pump 2ICS*P2 and the pump discharge check and isolation valve. Contrary to this, the original installed configuration is a 3/4" root nipple, valve, 3/4" pipe to ½" tube adapter, ½" flex hose to ½" tubing which penetrates outside the RCIC room (penetration W-3621-C) and is then capped. The original design function was to be an instrument connection for pressure transmitter 2ICS*PT175 and pressure indicator 2ICS*PI175. These instruments were never installed during original construction and the remaining installation became a spare instrument connection. The connection no longer served a useful function, nor was it configured or located at a low point which could be used as a drain. This change revised the USAR figure to reflect the as-installed condition.

Safety Evaluation Summary:

This change updates the USAR figure to reflect the existing field installation. The subject spare instrument connection, as described in approved design documentation, was designed and installed in accordance with all applicable design specifications and codes. The connection serves no function, thus has no adverse impact on the design function of the RCIC system.

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

Safety Evaluation No.: 97-049 Rev. 0 & 1

Implementation Document No.: Procedure NIP-CHE-01

USAR Affected Pages: 9.2-4, 9.2-4a

System: Service Water (SWP)

Title of Change: Molluscicide Treatment for the Unit 2 Service Water System

Description of Change:

This change eliminated the specific manufacturer and brand name product used at Unit 2 for treating the service water system for zebra mussels/asiatic clams, as indicated in the USAR. Additionally, this change eliminated the exact locations where the molluscicide and detoxification agent, if applicable, are introduced into the SWP system.

Revision 1 of this safety evaluation incorporated the use of a new product for treatment of mollusks at Nine Mile Point. This product is called EVAC and is marketed by Calgon Corporation. The major feature of the product is that it requires no addition of a detoxification agent. With no detoxification agent added, the product is substantially more environmentally friendly, requires less manpower to implement, and will significantly reduce the cost of a treatment at NMP2. The new chemical will perform the same function as the chemicals currently used for mollusk control as well as having no adverse impact on the SWP system or the components it services.

Safety Evaluation Summary:

New chemicals to treat zebra mussels/asiatic clams will be as effective as the original chemicals without any adverse effects on the operability of the SWP system. All chemicals used in the zebra mussel program must be approved by the New York State Department of Environmental Conservation. Prior to use of any newly approved chemicals, the SPDES permit must be amended to include these chemicals.

The feature of no detoxification agent in this chemical makes it extremely attractive to use from an implementation, cost and risk standpoint. The intent for the use of this chemical is the same as previous chemicals used to control

Safety Evaluation No.: 97-049 Rev. 0 & 1 (cont'd.)

Safety Evaluation Summary: (cont'd.)

mollusks, to kill mollusks (veligers and adults) in the SWP system to minimize fouling of safety-related and nonsafety-related system components. The treatment program interacts with the SWP system and the circulating water system but has no impact on their operability or operation.

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

Safety Evaluation No.:	97-051 Rev. 0, 1 & 2
Implementation Document No.:	Design Change N2-97-016
USAR Affected Pages:	8.3-69; Tables 3.9A-12 Sh 5, 8.3-4 Sh 2, 8.3-8
System:	Reactor Core Isolation Cooling (ICS)
Title of Change:	2ICS*MOV126 Actuator and Stem Replacement

Description of Change:

This design change replaced the actuator SMB-0-40 with SMB-1-60; increased the gear set ratio from 39.11 to 40.15; and replaced the stem to maintain a 15-second maximum stroke time.

The larger actuator and motor, along with the higher gear ratio, increases the actuator thrust/torque capability under the reduced voltage condition so that the valve can perform its safety function under the worst case design basis conditions. The new stem with higher lead and pitch increases the stem speed and as such reduces the stroke time from 14.51 to 11.17 seconds, calculated under normal voltage condition. The reduction in stroke time was required to assure that the valve stroke time remains below 15 seconds maximum allowable, even under the reduced voltage conditions. Under reduced voltage condition, the calculated stroke time is 14.44 seconds, which complies with the original basis. The selection of the new actuator, motor, gear ratio, and stem was based on the approved calculation.

Safety Evaluation Summary:

The new, larger actuator and stem will be designed and installed to the original criteria. The change will allow implementing improvements required to satisfy Generic Letter 89-10 criteria. The performance of the valve will remain within the existing allowable parameters.

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

Safety Evaluation No.: 97-053

Implementation Document No.: Calculations AX-075B, A10.1-D-10, HVR-032

USAR Affected Pages: 9.3-35; Table 9.3-2 Sh 1

System: Standby Liquid Control (SLS)

Title of Change: SLS Pump NPSH

Description of Change:

This change allowed a maximum temperature of 110°F vs. 100°F for testing the SLS system. Additionally, a change was made to indicate that the SLS equipment is located in a general area where the maximum temperature is 104°F vs. 100°F.

Safety Evaluation Summary:

During testing of the SLS system, the proposed change allows the fluid temperature to be as high as 110°F. The main concern in allowing this temperature increase is the net positive suction head available ($NPSH_A$) for the SLS pumps. Calculation A10.1-D-10 Rev. 2 shows that there is sufficient $NPSH_A$ when the test medium is 110°F or less. This calculation was done with sodium pentaborate as the pumped fluid. The test medium will be demineralized water. Sufficient NPSH is available for the SLS pumps with water at 110°F.

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

Safety Evaluation No.: 97-054

Implementation Document No.: Design Change N2-97-046

USAR Affected Pages: 9.4-65; Figures 9.2-1j; 9.2-2 Sh 9, 9.4-1a

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

Title of Change: Removal of Trip Function of Chiller Low Service Water Flow - 2SWP*FSL29A and *FSL29B

Description of Change:

Condensing water for the HVK system chiller units 2HVK*CHL1A and *CHL1B is provided by the SWP system. The previous design incorporated a low condensing water (i.e., service water) flow trip, as indicated by flow switches 2SWP*FSL29A and *FSL29B. The purpose of this trip was to prevent operation of the chiller units with low condensing water flow that could lead to a high refrigerant pressure cutout. The low condensing water flow trip also prevented cycling of the compressor following a trip and prevented operation of the unit with no cooling water flow. Contrary to the design intent, the low service water flow trip reduced the reliability of the HVK chiller units by potentially initiating unnecessary trips.

This design change eliminated the low service water flow trip while maintaining the associated alarms.

Safety Evaluation Summary:

Discussions with the manufacturer, York Division Borg-Warner Corporation, confirmed that the function of the switch is to protect the compressor from damage when the equipment is operated outside its design envelope. Eliminating the low flow trip has no impact on the operation of the primary refrigerant-based protection. A review of the previous trips confirmed that the initiating flow parameters were sufficient to support operation of the chillers, without producing high refrigerant conditions. A review of historical plant data confirmed that service water flow conditions exceed minimum specified levels, even during past inadvertent trips. Added assurance is provided by maintaining the low flow alarm.

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

Safety Evaluation No.: 97-055

Implementation Document No.: EDC 2F00815, EDC 2F00632

USAR Affected Pages: 4.6-6, 4.6-7

System: Control Rod Drive (RDS)

Title of Change: USAR Update for Change in Seal and Bushing Material on CRD Mechanism

Description of Change:

The control rod drive mechanism (CRDM) is used to rapidly insert the control rods in response to a manual signal or an automatic signal from the reactor protection system (RPS). The CRDM is also used to change the position of the control rods within the core in response to the reactor manual control system for the control of reactivity. The CRDMs are provided by General Electric (GE), the original equipment manufacturer.

The material used in the CRDM seals and bushings has been changed as a result of the experience gained in operating and maintaining the CRDMs. This change in material is supported technically by GE's extensive testing of the material before recommending its use in the CRDMs.

Safety Evaluation Summary:

The change in seal and bushing material was made by GE to take advantage of their experience in working with utilities to improve plant operation and maintenance. The change does not adversely affect the ability of the CRDMs to scram the reactor in response to signals from the RPS, nor does it adversely affect the ability of the CRDMs to control reactivity. The results of the change provide increased CRDM reliability and reduction in maintenance, such that there is continued assurance that the CRDMs will continue to be able to perform their design functions.

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

Safety Evaluation No.: 97-056

Implementation Document No.: DDC 2F01631

USAR Affected Pages: Figure 10.1-9g

System: Condensate Demineralizer (CND)

Title of Change: Hold Out for the Condensate Demineralizer System (2CND-TK8)

Description of Change:

Valves 2CND-V95 and 2CND-V158 previously were normally open valves in the CND system for the caustic dilution water tank (2CND-TK8). These valves provide a flow path for condensate during resin regeneration of the CND system. Resin replacement was performed in lieu of resin regeneration for the condensate demineralizers. Therefore, it was desirable to isolate the caustic dilution water tank.

The resin regeneration portion of the CND system is not currently being used. In order to provide isolation for equipment still installed in this section of the CND system, the normal operating position has been changed to closed. This safety evaluation evaluated the change in the normal operating position of the above-listed valves from open to closed and allowed Hold Out #2-95-H1025 for these valves to remain in place until final design paperwork was issued and operations accepted.

Safety Evaluation Summary:

Changing the position of 2CND-V95 and 2CND-V158 from open to closed will not affect the system's ability to regenerate resins. Procedural controls will be put in place to manually open these valves if this equipment is needed for operation to support resin regeneration. If the system is not required for processing, the Hold Out tags can remain in place until design paper is issued and operations accepted. This change will not affect the CND system's ability to maintain high-quality condensate and a reliable supply of feedwater to the reactor as per original design requirements.

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

Safety Evaluation No.: 97-057

Implementation Document No.: DDC 2F01641

USAR Affected Pages: Figure 9.3-1b

System: Instrument Air (IAS), Chemical Feed-Acid (WTA), Chemical Feed-Hypochlorite (WTH)

Title of Change: Holdout for the Instrument Air System (2IAS-V329)

Description of Change:

This safety evaluation changed the normal operating position of valve 2IAS-V329 from open to closed, allowing the Holdout for this valve to remain in place until final design paperwork was issued and operations accepted. Valve 2IAS-V329 was a normally open valve in the IAS system for the isolation of various solenoid and air-operated valves. This valve provides a flow path of instrument air for manipulation and operation of 2WTA-AOV13A, B, C, D, E, F and 2WTH-AOV28A, B, C, D, E, F. Air leakage by the seat of various abandoned WTA solenoid valves was identified. Therefore, isolation of the instrument air to these valves was required. The affected solenoid and air-operated valves have been left in a fail-safe position, which is their normal operating position.

Safety Evaluation Summary:

The affected section of WTA has been abandoned and is no longer operational. The affected section of WTH is currently not in use; however, if system manipulation is required, manual valves located downstream of the affected air-operated valves can be used for process control. In order to provide isolation for this section of IAS, it is desirable to have the normal operating position of the above valve changed to closed.

Changing 2IAS-V329 to a normally closed valve in the instrument air distribution piping network will not affect the system's continuous operation. System operation is based upon the requirements of plant instrumentation and control air. Automatic loading/unloading controls maintain common air header pressure limits. Since system operation is not affected by changing 2IAS-V329 from open to closed, testing criteria of the IAS system with regard to Regulatory Guide 1.68 are not affected by this change. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.: 97-059

Implementation Document No.: DDC 2F01663

USAR Affected Pages: Figure 9.2-9a

System: Sanitary Plumbing (PBS)

Title of Change: Lift Station Sump Pump Control and Indication

Description of Change:

USAR Figure 9.2-9a for the sanitary waste and disposal system showed remote manual control, automatic control, and indication for the control of the lift station sump pump. The installed configuration consists of only automated control, as reflected by the original purchase specification and vendor configuration documentation.

This change evaluated operation of the system with only automatic float switch control for the sump.

Safety Evaluation Summary:

The factory-installed float switch on the sump pump provides reliable control of the pump and is in accordance with the original design intent, despite the control features erroneously shown by the USAR figure. The effects of the change are limited to the sanitary disposal system; therefore, it is concluded that potential 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.: 97-060 Rev. 1 & 2

Implementation Document No.: Temporary Mod. 97-009

USAR Affected Pages: N/A

System: MSS, DSR, FWS, CNM, ESS, TMS, CRS, HDL, HDH

Title of Change: Long-Term Plant Operation with MSR Reheating Steam Out of Service

Description of Change:

Due to repairs required to moisture separator reheater (MSR) 2MSS-E1B, both MSRs 2MSS-E1A and E1B are operating with reheating steam isolated until a repair is performed. This evaluation addressed acceptability of temporary, long-term operation of the plant under these conditions (i.e., MSRs with reheating steam out of service) until a repair is performed.

Safety Evaluation Summary:

This temporary condition, as installed, does not affect the design basis or the safety functions of the systems involved as evaluated in this safety evaluation. Probability of missile generation is not affected by this change. No accident conditions will be created since there are no accident initiators or precursors associated with the scope of change identified. Increased flow to the high-pressure turbine will result in higher first stage pressure; however, a review of the setpoints associated with turbine first stage pressure and main steam flow (and the corresponding adjustment of the RSCS setpoint) have indicated no adverse impact.

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

Safety Evaluation No.: 97-061

Implementation Document No.: Procedure N2-OP-78

USAR Affected Pages: 9B.8-3; Table 9B.8-3 Sh 7, 8, 9, 12

System: Residual Heat Removal (RHS)

Title of Change: Manual Operation of LPCI Injection Valves for Alternate Shutdown Cooling

Description of Change:

During a Control Room fire, control of equipment required for safe shutdown is transferred to the remote shutdown panels. The shutdown cooling mode of residual heat removal (RHR) is initiated to achieve and maintain cold shutdown. Also, for the Appendix R Safe Shutdown, it is assumed that a loss of offsite power occurs coincident with the Control Room fire, such that the recirculation system valves (pump suction and discharge) could remain open or reposition closed at the time of the event. With the recirculation valves open, the shutdown cooling path would be short circuited with flow to the discharge side of the recirculation pump (normal path) and flow to the suction side of the recirculation pump (shutdown cooling suction). Since this flow lineup is not analyzed, another flow path for shutdown cooling has been made available via the alternate shutdown cooling method. Water is pumped from the suppression pool through the RHR heat exchanger into the reactor pressure vessel (RPV) via the normal low-pressure coolant injection (LPCI) path. The cooled suppression pool water flows through the RPV, out the automatic depressurization system valves, and back to the suppression pool.

Safety Evaluation Summary:

Alternate shutdown cooling is a response to an event involving the loss of normal shutdown cooling capability. This event can occur only during low pressure when the RHS system is required to operate in the shutdown cooling mode. Loss of shutdown cooling can cause RPV water temperature and, therefore, RPV pressure to increase. Alternate shutdown cooling has the capability to achieve and maintain cold shutdown in the event of a failure of a single recirculation suction valve to the redundant RHR shutdown cooling loops. Manually operating the LPCI injection valve and the minimum flow valve during an Appendix R safe shutdown event will allow alternate shutdown cooling mode of operation, which has been analyzed to be within design capability. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.: 97-062

Implementation Document No.: DER 2-96-3381

USAR Affected Pages: Figures 9.4-2a, 9.4-2b, 9.4-2c

System: Yard Structures Ventilation (HVY), Auxiliary Boiler Room Ventilation (HVI)

Title of Change: Various HVAC System Updates, SAR Figures 9.4-2a, 2b, and 2c

Description of Change:

This change revised USAR Figure 9.4-2a to remove balancing damper 2HVY-DMPV21 and to show proper air distribution to the Screenwell Building general area.

USAR Figure 9.4-2b was revised to show dampers 2HVY-DMP16B and 2HVY-DMP17B from the closed to open position in the CST Building HVAC system.

USAR Figure 9.4-2c has been revised to show dampers 2HVI-DMPV4 and 2HVI-DMPV5 in the exhaust fans ducting in the Auxiliary Building HVAC system.

Safety Evaluation Summary:

Changes made to the nonsafety-related portion of the HVY and HVI systems will make the applicable documents agree with the original design intent and the current practice of the operating procedure. This is a safe and effective mode of operation and has no adverse effects on any associated systems or equipment.

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

Safety Evaluation No.: 97-064

Implementation Document No.: DDC 2M11123

USAR Affected Pages: Figure 9.5-3a

System: Fire Protection - Low-Pressure Carbon Dioxide (FPL)

Title of Change: Cardox Instrumentation Configuration Change

Description of Change:

This change updated engineering design documents and the USAR to reflect the plant as-built configuration. Pressure and level instrumentation and relief valves on the carbon dioxide storage tanks refrigeration compartments (2FPL-TK1 and 2FPL-TK2) were incorrectly shown on plant drawings. This evaluation ensures that the changes satisfy the FPL system design requirements.

Safety Evaluation Summary:

The changes conform with the applicable requirements of NFPA Standard 12 and GDC 3.

This change does not affect the operation of the FPL system nor does it introduce any new failure mode that has not been 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.: 97-065

Implementation Document No.: Design Change N2-97-059

USAR Affected Pages: 9.2-3

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

Title of Change: Inlet Service Water Temperature Control For Control Building Chillers

Description of Change:

This design change lowered the control point for the HVK chiller temperature control valves (2SWP*TV35A/B) to 69.5°F. This, in combination with raising the start/stop setpoint of the chiller condensing water pump to 69.5°F, reduces the gap between the two control devices. Minimizing the gap between the control points ensures a smooth condenser flow rate transition during pump shutdown. This eliminates low flow transients during the transition, which precludes a potential sustained low service water flow condition while still maintaining design heat removal capability.

Safety Evaluation Summary:

The design analysis determined that lowering the control point for the chiller service water outlet temperature control valves, combined with the increase in condensing pump start/stop setpoint, maintains the inlet service water to the chiller above the safety limit specified by the vendor, and maintains the design rating of the chillers.

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

Safety Evaluation No.: 97-066

Implementation Document No.: NFPA 24, FPEE 2-97-001

USAR Affected Pages: 9A.3-46, 9A.3-50; Table 9.5-3 Sh 3

System: Fire Protection Water (FPW)

Title of Change: Deviation from NFPA 24 Installation Requirements of Fire Hydrant 2FPW-FHY10

Description of Change:

The Unit 2 USAR documented NFPA 24 as the design basis standard for the yard main system and fire hydrants, the NMP2 yard main design and installation deviations to this standard, and the surveillance requirements to maintain the NMP2 hydrants and outside hose installations operable. This safety evaluation documents a deviation from the NFPA 24 requirements for fire hydrant 2FPW-FHY10. These USAR changes allow for the presence of water in the barrel of fire hydrant 2FPW-FHY10 up to the groundwater level but below the frost line.

Safety Evaluation Summary:

This safety evaluation and FPEE 2-97-001 demonstrate that fire hydrant 2FPW-FHY10 is operable and available for service at all times; therefore, the general design intent of NFPA 24 is satisfied and a deviation to the installation and inspection requirements is acceptable. This safety evaluation determined that the Safe Shutdown Analysis and Fire Hazards Analysis are not affected by this change.

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

Safety Evaluation No.: 97-068 Rev. 0 & 1

Implementation Document No.: Design Change N2-96-028

USAR Affected Pages: 8.3-33

System: Reactor Protection (RPS), Nuclear Steam Supply Shutoff (NS^4)

Title of Change: Electrical Protection Assembly (EPA) Upgrade

Description of Change:

This design change replaced eight EPA units, 2VBS*ACB1A&B, 2VBS*ACB2A&B, 2RPM*ACB1A&B, and 2RPM*ACB2A&B, with new upgraded units. The new units utilize solid state relays to sense undervoltage, overvoltage, and underfrequency conditions on the RPS power supplies as opposed to the previous units which utilized General Electric logic cards for sensing. The previous units had a history of persistent problems with spurious trips of the EPA breakers. The source of the problems was attributed to failure of the logic cards. In addition, the previous units could not be functionally tested online without initiating a half scram condition. The replacement units are highly reliable and can be functionally tested online without initiating a half scram.

Safety Evaluation Summary:

The replacement EPA units have specifications that meet the performance requirements of the existing EPAs to include Class 1E and seismic Category 1 qualifications. The new units will provide the same protection to the safety-related RPS and NS^4 circuits from damages that may be caused by abnormalities in the nonsafety-related power supplies as the existing units. All interfacing equipment will function as assumed in the accident analyses.

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

Safety Evaluation No.: 97-069

Implementation Document No.: LSK-23-12D

USAR Affected Pages: Figure 9.3-16 Sh 6

System: Miscellaneous Drains (DFM)

Title of Change: Revise USAR Figure for 2DFM-LS123 Logic Condition

Description of Change:

This change revised engineering design documents and the USAR figure for the Control Building sump 4 "low level" level switch 2DFM-LS123 to reflect that the switch contact opens on low level (decreasing). The purpose of this level switch is to provide an input to an alarm to identify a condition of excess flow into Control Building sump 4. This alarm indicates high leakage for Control Building floor drains. This change properly identified the level switch condition which will initiate the alarm.

Safety Evaluation Summary:

This change will not have any impact on the level switch or on any functions associated with the sump. This change will not alter the function of any components in the DFM system.

The change does not alter the ability of the level switch, sump, or any associated components to perform their intended functions. With this change, the level switch condition will be correctly identified as "Not Low Level" to provide an input to indication for high leakage. The proposed change maintains all other applicable design criteria associated with the miscellaneous buildings equipment and floor drains systems. Neither the sump, level switch panel nor any associated equipment is required for safe shutdown or decay heat removal.

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

Safety Evaluation No.: 97-070

Implementation Document No.: Procedure NTP-TQS-503

USAR Affected Pages: 12.5-16, 12.5-17, 13.2-10, 13.2-11

System: N/A

Title of Change: Radiation Protection Training Program Examination Approval

Description of Change:

The USAR requires the Manager Radiation Protection to approve written examinations for all personnel who attend the Radiation Protection Training Program. This requirement has been revised to reflect that all exam questions used in the Radiation Protection Training Program shall be approved by the Manager Radiation Protection or his designee. This change streamlines the examination process.

Additionally, this safety evaluation evaluated revision of the wording in the General Employee Training description in the USAR to reflect current terminology.

Safety Evaluation Summary:

The changes to the USAR description of Manager Radiation Protection approval of exam questions rather than individual exams in no way decrease the effectiveness of the Radiation Protection Training Program. The changes are consistent with Regulatory Guide 8.27. Examination review and approval requirements are contained in Administrative Procedures which are applicable to all training programs for which the Nuclear Training Department is responsible.

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

Safety Evaluation No.:	97-071
Implementation Document No.:	Procedure N2-OP-35
USAR Affected Pages:	5.4-15; Tables 7.4-1, 7.6-3 Sh 2
System:	Reactor Core Isolation Cooling (ICS)
Title of Change:	Clarify Reactor Core Isolation Cooling Instrumentation Data

Description of Change:

The ICS system consists of a turbine, pump, piping, valves, and instrumentation to maintain sufficient water in the reactor vessel to prevent the reactor fuel from overheating in the event of a reactor isolation accompanied by a loss of coolant flow from the reactor feedwater system.

This change revised the USAR to resolve discrepancies between the USAR tables for several ICS system instruments and ICS system design basis documents. Specifically, pressure transmitters 2ICS*PT106, 2ICS*PT103, 2ICS*PT167Y, 2ICS*PT168Y, 2ICS*PT167X, and 2ICS*PT168X have discrepancies between the documents.

Safety Evaluation Summary:

The proposed changes to the instrumentation do not diminish the capacity of the instruments evaluated in this safety evaluation to monitor and respond as required while in operation. The changes do not impact ICS operability because no equipment changes are being proposed. Therefore, the requirements of General Design Criterion 13 are satisfied.

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

Safety Evaluation No.: 97-072

Implementation Document No.: DDC 2M11154

USAR Affected Pages: Figures 10.1-8b, 10.1-8f, 10.1-8g

System: Gland Seal & Exhaust (TME), Auxiliary Steam (ASS)

Title of Change: Discrepancies Between P&IDs and Plant Configuration

Description of Change:

This change corrected a valve position from open to close (2TME-MOV20A) to reflect the original design and plant configuration. Additionally, the normal operating position of two valves (2TME-MOV22B, 2ASS-MOV8B) showing isolation valves for off-line components was changed from open to close.

Safety Evaluation Summary:

The valve lineups shown on the USAR figures reflect the configuration of the system during normal plant operation. Contrary to this, three valves were found on the figures that did not reflect actual plant configuration. No physical changes are being made to the valves or any changes to their function or how they are operated.

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

Safety Evaluation No.:	97-073
Implementation Document No.:	Design Change N2-97-047
USAR Affected Pages:	15.7-10, 15.7-13, 15.7-14; Tables 3.9B-2n Sh 3, 3-3 Sh 3 (Appendix 9C); Figures 9.1-2, 9.1-7, 9.1-8, 5-2 (Appendix 9C)
System:	Fuel Nuclear Refueling (FNR)
Title of Change:	Install Round Mast & Camera System on Refuel Bridge

Description of Change:

This modification replaced the triangular refueling mast and grapple head with a new round mast and new grapple head equipped with a camera/TV system. The round mast is structurally stronger than the triangular shape and less susceptible to damage during fuel moves. In addition, the round mast provides more shielding for the fuel handling personnel. The camera/TV system enhances visibility for spotting fuel bundles.

License Amendment 81 revised surveillance requirements in Technical Specification Section 4.9.6 to reflect appropriate setpoints for the modification.

Safety Evaluation Summary:

The round mast with a camera/TV system is essentially a direct replacement for the existing triangular mast. The round mast weighs approximately 400 lbs. more than the triangular mast; however, all existing standards, codes and design criteria will be met and the new mast will function exactly like the existing one.

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

Safety Evaluation No.: 97-074

Implementation Document No.: DDC 2M11167

USAR Affected Pages: Tables 9.5-1 Sh 7, 9.5-2 Sh 6, 9A.3-9 Sh 9, 3-2 Sh 1 (Appendix 9C); Figures 1.2-13 Sh 3, 12.3-26, 12.3-59

System: Radioactive Solid Waste (WSS)

Title of Change: Use of Radwaste 245' Elevation as an SRV Test and Storage Facility

Description of Change:

This safety evaluation evaluated testing, maintenance and storage of the main steam safety relief valves (SRV) in the low level radioactive waste (LLRW) storage area in Radwaste Building el. 245'. The Radwaste el. 245' LLRW storage area consists of three empty bays equipped with floor drains, ventilation, normal lighting, emergency lighting, radiation monitoring, fire detectors, communications equipment and a dedicated forklift. Additional tools and equipment were located in the area to facilitate SRV testing and maintenance activities.

The SRV test facility tools and equipment are capable of being removed in a short period of time should the need to store LLRW in the area arise during that time. As such, all criteria for use of Radwaste el. 245' as a LLRW storage facility are maintained.

Safety Evaluation Summary:

There are no accidents or transients evaluated in the USAR which will be affected by the use of Radwaste el. 245' as an SRV test and storage facility. Physical separation between Radwaste el. 245' and components and hardware associated with the accidents and transients in the USAR prohibits influence by the proposed activity. Testing and maintenance of the SRVs in Radwaste Building el. 245' does not affect nuclear safety.

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

Safety Evaluation No.:	97-077 Rev. 0 & 1
Implementation Document No.:	Design Change N2-95-035
USAR Affected Pages:	8.2-16, 8.2-18
System:	13.8 kV Normal Power (NPS)
Title of Change:	Changes to 13.8 kV Buses 2NPS-SWG001 & 003 Ground Overcurrent Fault Protection Schematic

Description of Change:

This design change provided each reserve station service transformer feed to the 13.8 kV buses with a dedicated directional ground overcurrent relay (67N). Each normal station service transformer feed to the 13.8 kV buses was provided with a dedicated ground overcurrent relay (51N). The previous configuration of paralleled current transformer circuits for normal and reserve station transformer neutrals was contributing to an event leading to the simultaneous loss of the 13.8 kV buses.

Safety Evaluation Summary:

The addition of dedicated protection for source feeders to the buses will meet or exceed the same design criteria and will function to perform the same tripping as the existing common relay schemes. The performance of interfacing equipment and systems will be as currently designed without change. This change will enhance the availability of onsite normal power by selective tripping actions for source feeders to the 13.8 kV buses. The independence of the two offsite power systems is unaltered and there is no effect on the site emergency power system. The existing fast/slow transfer capability in the event of loss of normal ac power from normal to reserve transformer is still maintained with this change.

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

Safety Evaluation No.: 97-078

Implementation Document No.: Calculation EC-032

USAR Affected Pages: Tables 8.3-2 Sh 13, 8.3-4 Sh 13

System: Reactor Building Closed Loop Cooling Water (CCP), Motor Control Center Emergency System (EHS)

Title of Change: Revise USAR Tables for 2CCP*MOV16A Motor Data

Description of Change:

This change revised USAR Tables 8.3-2 and 8.3-4 to properly depict the electrical data for valve 2CCP*MOV16A. The actual installed data for this valve was different from that listed in the tables because the data for valve 2CCP*MOV18A was inadvertently substituted.

Safety Evaluation Summary:

The revision of the USAR tables for the motor electrical data for 2CCP*MOV16A will not alter the ability of the motor, valve, or any associated components to perform their intended functions. With this change, the valve motor data will be correctly identified. The proposed change maintains all other applicable USAR criteria associated with the CCP system, the reactor recirculation system, and the diesel generator systems.

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

Safety Evaluation No.: 97-079

Implementation Document No.: DDC 2E11377

USAR Affected Pages: 9.3-14, 9.3-15, 9.3-16

System: Instrument Nitrogen (GSN)

Title of Change: 2GSN-PCV11A, B Overpressure Vent Regulator Setpoint

Description of Change:

The setpoints on the liquid nitrogen storage tank overpressure vent regulators 2GSN-PCV11A and 2GSN-PCV11B have been lowered by 45 psig from 240 psig to 195 psig. This has the effect of lowering the operating pressure of the liquid nitrogren storage tanks.

Additionally, the liquid nitrogen storage tank high pressure alarm 2GSN-PAH8A and 2GSN-PAH8B setpoints were raised by 15 psig from 195 psig to 210 psig. This has the effect of producing the high pressure alarm only if the overpressure vent regulator fails to control the storage tank pressure.

Safety Evaluation Summary:

The setpoint changes do not affect the function or the method of performing the function of the liquid nitrogen storage tanks or the GSN system. The liquid nitrogen storage tanks, their associated piping, heaters, and pressure reducers are not relied upon to provide a safety-related function.

Since the operating pressure is being reduced, this change does not increase the probability of an accident or malfunction resulting from a failure of either storage tank or the associated GSN piping. Since the new operating pressure will still be significantly (75 psig or greater) above the setpoints for all three pressure-reducing stations, the components supplied by the liquid nitrogen storage tanks will be unaffected by the proposed change. The portion of the GSN system affected by this setpoint change is not relied upon to function during or following an analyzed accident.

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

Safety Evaluation No.: 97-080

Implementation Document No.: Calculations A10.1-AA-24, A10.1-J-019, ES-276

USAR Affected Pages: Tables 9.2-1A Sh 3, 9.2-1B Sh 3

System: Spent Fuel Pool Cooling and Cleanup (SFC)

Title of Change: Post-Accident Initiation of the SFC System

Description of Change:

This change revised Calculations A10.1-AA-24, A10.1-J-019, and ES-276 to establish that service water cooling to the SFC system heat exchanger is required within three hours rather than seven hours previously calculated on a loss of CCP cooling to the SFC heat exchangers. The calculated spent fuel pool heat load has increased, and the calculated heatup rate (as the result of a loss of normal cooling) has increased, which results in a decreased time to establish service water cooling. The time required to re-establish cooling to the SFC heat exchanger following a loss of CCP cooling has been revised to maintain the spent fuel pool cooling temperature below its design temperature of 150°F. The new operating conditions are a result of 24 months fuel cycle change and the reduction in refueling outage duration from 30 days to 20 days.

Safety Evaluation Summary:

Calculation A10.1-J-019, Revision 5, calculated the spent fuel pool cooling heatup rate following loss of spent fuel pool cooling at an initial pool temperature of 125°F. The spent fuel pool cooling heatup rate is calculated to be 5.2°F/hr. At this rate, the spent fuel pool cooling temperature will be at 140.6°F three hours into the event. This is below the SFC system design temperature of 150°F.

The secondary containment drawdown calculation (ES-276) was dispositioned to incorporate the fuel pool heat load (from Calculation A10.1-AA-24). Results of the disposition indicate that with the proposed change, the secondary containment negative pressure (-0.25 in. WG) can be established in less than 60 minutes.

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

Safety Evaluation No.: 97-082

Implementation Document No.: Design Change N2-97-063

USAR Affected Pages: Figure 5.4-2c

System: Reactor Recirculation (RCS)

Title of Change: Eliminate Flex Hose 2RCS*HOSE40 on the Valve Body Side Drain on Flow Control Valve 2RCS*HYV17B

Description of Change:

This modification eliminated flex hose 2RCS*HOSE40 and permanently plugged the valve body side drain on 2RCS*HYV17B. A custom designed plug was installed which fills the drain connection cavity from the outside of the valve body. The associated drain line has been capped.

Safety Evaluation Summary:

The RCS valve body drain performs a passive function and its elimination will not impact RCS operation or the ability to perform its required function. The potential for draining of the reactor pressure vessel (RPV) is minimal because the RPV will be isolated from this valve by safety-related boundary valves.

This change will be implemented with the plant in cold shutdown and valve 2RCS*HYV17B will be isolated from the RPV. Leakage past the isolation valves will be made up via the control rod drive system or the condensate system as required to maintain RPV level.

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

Safety Evaluation No.: 97-083

Implementation Document No.: Calculation EC-032

USAR Affected Pages: Tables 8.3-1 Sh 32, 8.3-2 Sh 31

System: Control Rod Drive (RDS), Reactor Building Closed Loop Cooling Water (CCP), Emergency Diesel Generator (EGS)

Title of Change: Revise USAR Tables for Pump Motor Data

Description of Change:

This change revised USAR Tables 8.3-1 and 8.3-2 to properly depict the electrical motor data associated with the RDS system pumps and the CCP system pumps. The calculated KVA data was previously used in place of the nameplate HP data without changing the units of measure. The tables inadvertently showed the loading values for these pumps in units of "HP" rather than "KVA".

Safety Evaluation Summary:

The revision of the USAR diesel loading tables for the motor electrical data for the control rod drive pumps and the reactor building closed loop cooling water pumps will not alter the ability of the motors, valves, or any associated components to perform their intended functions. With this change, the pump motor data will be correctly identified. The proposed change maintains all other applicable USAR criteria associated with the RDS system, the CCP system, and the EGS systems.

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

Safety Evaluation No.: 97-084

Implementation Document No.: Simple Design Change SC2-0007-95

USAR Affected Pages: 9.1-36, 9.1-49, 9.1-50, 12.4-2; Tables 3.2-1 Sh 7, 9.1-4 Sh 1 & 2, 3-3 Sh 3 (Appendix 9C); Figures 1.2-10 Sh 2, 9.1-24, 9.1-25, 5-2 (Appendix 9C), 12.3-12, 12.3-45

System: Fire Handling Equipment (FHS)

Title of Change: Conversion of Decon Platform to an Auxiliary Service Platform

Description of Change:

This design change converted the decon platform (2DCS-PLAT1), located on Reactor Building El. 353', to an auxiliary service platform. The auxiliary service platform functions as a work area platform when positioned over the vessel for in-vessel work activities or other necessary tasks during refuel outages. In the past, these in-vessel work activities were performed from the Refuel Bridge, after fuel moves were completed. Use of this new work platform has the potential to shorten activities associated with refueling by one or two days, as in-vessel servicing can be done in parallel to fuel moves. The decon platform was qualified as a Q4 structure (Category II/I). Based on its new function as an auxiliary service platform, which may be used over the open vessel with or without core offload during refueling, its classification will remain Q4 (Category II/I).

Safety Evaluation Summary:

The conversion of the decon platform to an auxiliary service platform to provide an additional access area from which personnel can perform necessary tasks during refuel outages was done in accordance with Q4 (Category II/I) requirements as was the original platform. The auxiliary service platform has been designed such that it would not collapse or leave the Refuel Bridge rails during a safe shutdown earthquake (SSE). The platform's travel limits from the reactor internals storage pool to the cavity area remain the same.

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

Safety Evaluation No.:	97-087 Rev. 0 & 1
Implementation Document No.:	Plant Change Requests PC2-0005-97, PC2-0008-97
USAR Affected Pages:	10.4-20; Figures 10.4-7d, 10.4-8 Sh 3
System:	Circulating Water (CWS)
Title of Change:	Removal and Abandonment CWS Components

Description of Change:

This change removed the environmental interval sampler, pressure control valve 2CWS-PCV108, and relief valve 2CWS-RV141 for the automatic function of grab samples. Valve 2CWS-V293 is now normally closed and capped and its use will be procedurally controlled as a sampling point. In addition, the removal of flow elements 2CWS-FE47A & B, valves 2CWS-V134, 135, 136, 137, and flow indicators 2CWS-FI47A & B for monitoring return flow to the circulating pumps has been evaluated. Flow transmitters 2CWS-FT47A/B and computer points CWSFA01 and CWSFA02 and associated wiring have been abandoned.

Safety Evaluation Summary:

The CWS system does not perform a safety-related function. CWS is not required to effect or support safe shutdown of the reactor or to perform in the operation of reactor safety features. In addition, the changes being made to CWS will not affect an initiator or a precursor to accidents in the USAR.

The above-mentioned equipment is no longer required. Alternate means exist to provide return flow performance indication. Therefore, this equipment is not functionally needed for the CWS system to operate within its design basis.

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

Safety Evaluation No.: 97-088

Implementation Document No.: Temporary Mod. 97-016

USAR Affected Pages: N/A

System: Reactor Recirculation (RCS), Control Rod Drive (RDS)

Title of Change: Temporary Modification, Temporary Relief Valve For 2RCS-RV46B

Description of Change:

Relief valve 2RCS-RV46B, which is installed in the RDS supply line to RCS pump 2RCS*P1B mechanical seals, was leaking but failed to reseat. This temporary modification installed an identical relief valve with the same setpoint and capacity on an existing test connection (valve 2RCS-V76B) until such time as the supply could be depressurized to allow repair or replacement of the relief valve. After installation of the new valve, the original valve was manually adjusted to facilitate closure. Discharge from the temporary relief valve was through a limited length of hose (approximately 20 feet) of a size at least equal to the 3/4" relief valve discharge port, and directed to the same floor drain that the original valve discharges to.

Safety Evaluation Summary:

This temporary condition, which utilizes an identical relief valve in place of the current one, will maintain the current design in accordance with the appropriate codes and standards. The change does not affect the design basis or the safety functions of any systems, including reactor coolant pressure boundary/primary containment isolation as evaluated in this safety evaluation. No accident conditions will be created as there are no accident initiators (or precursors) associated with the scope of change identified.

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

Safety Evaluation No.: 97-089 Rev. 0 & 1

Implementation Document No.: Simple Design Change SC2-0178-93

USAR Affected Pages: 9.4-70

System: Plant Hot Water Heating System (HVH)

Title of Change: Alternate Hot Water Connections to HVH System

Description of Change:

This simple design change revised the USAR to state that the HVH system can be connected to a temporary hot water heating plant (non-radioactive) when this system is not capable of supplying hot water to the Reactor Building glycol heat exchanger.

Safety Evaluation Summary:

The HVH system is neither a precursor nor an initiator to any transients or accidents evaluated in the USAR. The new piping connections were designed to comply with the existing design conditions of the HVH system. The temporary hot water heating plant supplies hot water within these design limits. Additionally, the new piping connections are isolated from the HVH during normal operation. Therefore, this change will not increase the probability of occurrence of an accident previously evaluated in the USAR.

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

Safety Evaluation No.: 97-095

Implementation Document No.: Design Changes N2-97-026 and N2-97-027

USAR Affected Pages: Table 6.2-56 Sh 1

System: Main Steam (MSS)

Title of Change: Gear Set Change for Valves 2MSS*MOV111 and 2MSS*MOV112

Description of Change:

Motor-operated valves (MOV) 2MSS*MOV111 and 2MSS*MOV112 are 6-inch, normally closed, inside and outside containment isolation valves on the main steam drain line. Their safety function is to close upon an automatic isolation signal to provide containment isolation during a design basis event. The valves are only opened during plant startup and cooldown. As a result of changes to the NMP2 MOV sizing calculation methodology required for closure of NRC Generic Letter 89-10, the sizing calculations for 2MSS*MOV111 and 2MSS*MOV112 were 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 has shifted, resulting in the as-left torque switch settings of 2MSS*MOV111 and 2MSS*MOV112 being within the new design window, but with insufficient margin to allow setup and subsequent testing.

Safety Evaluation Summary:

In order to comply with the revised sizing calculation, gear set changes were required for 2MSS*MOV111 and 2MSS*MOV112 to produce sufficient thrust to function under design basis conditions (considering additional design margin) to meet the required stroke time of less than 60 seconds and to provide margin for setup and testing of the MOVs. The overall gear ratio changed from 162.95 to 185.47. The calculated valve opening and closing stroke time increased from approximately 34 seconds to approximately 40 seconds. The increased output thrust/torque of the modified actuator is sufficient to close the MOV under design basis conditions to meet the required stroke time, and to provide margin for setup and testing of the MOV without adversely affecting the safety function of either the valves or the MSS system. These valves are assumed to be closed in plant accident analyses and this change does not alter that assumption for any operational modes for these valves. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.: 97-096 Rev. 0 & 1

Implementation Document No.: DDC 2E11438

USAR Affected Pages: Figure 9.3-16 Sh 6

System: Miscellaneous Drains (DFM)

Title of Change: Revise Control Building Sump High Leakage Rate Alarm Circuit Logic

Description of Change:

The Control Building sump 4 leakage detection circuit includes a seal-in contact from an interlocking relay. The interlocking relay provides input to an alarm for high leakage. The seal-in contact serves only to hold the relay energized if sump level is above low; this function is already provided by a parallel instantaneous level timer contact. However, if the level condition clears before the timer times out, the interlocking relay could only be reset by removal of circuit control power because of the presence of the seal-in contact. Because this seal-in contact could hold the interlocking relay energized after the timer has been properly reset, a false high leakage alarm could be initiated on manual pump start. This change removed the seal-in contact from this circuit, preventing the alarm from coming in falsely on a manual pump start with the level reset.

Safety Evaluation Summary:

This change will not have any impact on the sump or on any functions associated with the sump, and will not alter the function of any components in the DFM system. The conditions which are indicative of a high leakage rate into the sump will be correctly identified and alarmed. The proposed change maintains all other applicable design criteria associated with the miscellaneous buildings equipment and floor drains systems. Neither the sump nor any associated equipment is required for safe shutdown or decay heat removal. The change does not present any new mode of malfunction or accident, does not reduce the margin of safety, and has no environmental impact.

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

Safety Evaluation No.: 97-099 Rev. 0, 1 & 2

Implementation Document No.: Temporary Mod. 97-019

USAR Affected Pages: 9.4-63; Figure 9.4-1a

System: Control Building Chilled Water (HVK)

Title of Change: Temporary Modification to Isolate the Makeup Supply to the HVK System

Description of Change:

Simple Design Change SC2-0053-93 changed the makeup water supply for the Control Building HVK system from the raw water (WTS) to makeup water (MWS). The raw water was responsible for blockage in the piping due to the formation of hard calcium scale caused by hardness in the raw water supply and by the buildup of corrosion product due to microbiologically-influenced corrosion (MIC).

This temporary modification isolated the HVK system from the normal MWS system makeup supply by manually closing safety-related isolation valves 2HVK*V325 and V326. Makeup to the HVK system was performed manually by opening the isolation valves. While opening these valves, the pressure downstream was verified to be below 100 psig. Thus, the makeup function was manual instead of automatic.

Safety Evaluation Summary:

This change to manual makeup will not impact the ability of the HVK system to perform its intended function to provide chilled water to the Control Building air conditioning units. Makeup will be performed, as needed, when the level in the expansion tank indicates low. At that time, the operator will manually open the makeup water supply isolation valve and close the valve after the level in the tank is restored. While opening the isolation valve, the pressure downstream will be verified to be below 100 psig to prevent the HVK system from being exposed to pressure higher than design pressure.

By isolating the HVK system from the makeup supply, it will be assured that the HVK system pressure and makeup supply pressure are within the HVK system design pressure.

Safety Evaluation No.: 97-099 Rev. 0, 1 & 2 (cont'd.)

Safety Evaluation Summary: (cont'd.)

The normal MWS system makeup to the HVK system is nonsafety related and, therefore, is postulated to be unavailable during an accident, whether the makeup function is manual or automatic. With makeup from the MWS system unavailable due to the failed open nonsafety-related pressure control valve, and makeup required for proper system operation, the safety-related SWP system is available for makeup during all modes of plant operation.

The HVK system will continue to function to maintain proper temperatures in the Control Building areas as required. Thus, it is assured that the safety function of equipment located in the Control Building and necessary to mitigate the consequences of an accident will not be impacted due to high room heat loads.

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

Safety Evaluation No.: 97-106

Implementation Document No.: DDC 2S11022

USAR Affected Pages: Figure 1.2-1

System: N/A

Title of Change: Removal of Substation "Q"

Description of Change:

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

Safety Evaluation Summary:

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

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

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

Safety Evaluation No.: 97-107

Implementation Document No.: Nuclear Division Policy (POL) Rev. 10, NSAS-POL-01 Rev. 10

USAR Affected Pages: 13.1-3, 13.1-5, 13.1-6; Figures 13.1-1, 13.1-5

System: N/A

Title of Change: Organization of Q1P, Labor Relations, HRD and Occupational Safety and Health Under the Newly Created Position of Director Human Resource Development

Description of Change:

The Nuclear Division Policy (POL) and NSAS-POL-01 have been revised to reorganize the functions of Employee/Labor Relations, Leadership/Career Development, Occupational Safety and Health, Quality First Program (Q1P) administrative issues, and the Fitness-for-Duty Program under the newly-created position of Director Human Resource Development.

Safety Evaluation Summary:

The proposed organizational changes establish responsibilities and lines of authority and communications for the newly-created position of Director Human Resource Development. The proposed organizational structure satisfies the criteria of SRP 13.1.1 and conforms with the requirements of Section 6.2.1.a of the Plant Technical Specifications. The proposed changes do not impact accident or malfunction initiation, or radiological consequences.

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

Safety Evaluation No.: 97-132

Implementation Document No.: DDC 2S11038

USAR Affected Pages: Figure 1.2-1

System: N/A

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

Description of Change:

The Paint Storage Silos and M&TE Buildings, located east of the NMP2 plant structures, have been demolished.

These buildings were built for use as temporary buildings during the construction of NMP2. They had fallen into a state of disrepair and were removed as part of the Facilities Improvement Program.

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 run-off coefficient makes 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 this change does not involve an unreviewed safety question.

Safety Evaluation No.: 97-151

Implementation Document No.: Temporary Mod. 97-021

USAR Affected Pages: N/A

System: Auxiliary Steam (ASS), Extraction Steam (ESS), Hot Water Heating (HVH)

Title of Change: Jumper for 2ASS-STV143 Provided by Temp. Mod. 97-021

Description of Change:

Valve 2ASS-STV143 developed a packing gland steam leak which could not be properly sealed using Furmanite.

This temporary modification defeated the electrical interlocks which caused 2ASS-STV143 to be backseated open. This action minimized packing gland leakage. The interlocks were defeated by installation of a wire jumper at local control panel 2CES-IPNL203. Manual valve 2ASS-V18, located downstream of 2ASS-STV143, was closed during normal power operation.

Safety Evaluation Summary:

Installation of this temporary modification does not impact the system design bases. Originally specified codes and standards used for the design and construction of the system shall be utilized for incorporating the jumper in the control panel.

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

Safety Evaluation No.:	97-152
Implementation Document No.:	DDC 2M11251
USAR Affected Pages:	9.4-63; Table 9.2-2 Sh 2, Figure 9.2-1j
System:	Control Building Chilled Water (HVK)
Title of Change:	Operating Redundant Control Building Chilled Water Trains

Description of Change:

The HVK system supplies chilled water to the air conditioning units in the Control Room, Remote Shutdown Rooms, and the Computer Room. There are two redundant (divisional) chilled water trains, each being a closed loop. Each loop is capable of meeting the total chilled water demand utilizing one centrifugal water chiller, one chilled water circulating pump, one expansion tank, and associated controls and piping. Per the original design, one chilled water train is normally operated while the redundant train is in standby.

With one chilled water train operating, and the other chilled water train in standby, only one Remote Shutdown Room air conditioning unit (ACU) is supplied with chilled water. The Remote Shutdown Rooms are separated by a fire-rated wall such that the rooms are isolated. Thus, the ACU in one Remote Shutdown Room does not provide cooling air to the other Remote Shutdown Room. With this system design, an auto start of a remote shutdown ACU (2HVC*ACU3A/B), in the same division as the chiller which is in auto/standby, initiates a start of that chiller (2HVK*CHL1A/B). The chillers are sized such that the remote shutdown ACU that auto starts, based on the room temperature, does not present an adequate load for the train, and places the water chiller in jeopardy of a trip. It has become normal practice to operate both redundant chiller trains simultaneously with one major load (either a control room ACU or relay room ACU) assigned to each train, thereby allowing the Remote Shutdown Room ACUs to cycle on and off as required. This mode of operation minimizes the cycling of the chiller unit due to low water temperature as a result of inadequate heat load.

Safety Evaluation Summary:

In the mode of operating both chiller trains simultaneously, one major chilled water system load (control or relay room ACUs) is assigned to each of the two chiller trains to provide a balance of shared loads. This minimizes the cycling of the chiller unit due to low water temperature as a result of inadequate heat load.

Safety Evaluation No.: 97-152 (cont'd.)

Safety Evaluation Summary: (cont'd.)

Each redundant chilled water train is capable of meeting the total chilled water demand of the system. With both redundant chilled water trains operating simultaneously, if a single failure were to occur, the affected chilled water train could be shut down and all loads transferred to the running chilled water train, which is capable of meeting the load requirement. Thus, the single failure criteria is satisfied.

With adequate chilled water being supplied to the Control Building ACUs and the temperature controls for the ACUs not being impacted, it is assured that the proper room temperatures can be maintained during all modes of plant operation.

The service water steady state analyses take into account both chilled water trains operating simultaneously at full load and, therefore, are not impacted by the proposal.

In addition, based on statistical data from Probability Risk Assessment (PRA) studies, with the chiller compressor being the most limiting component, reliability is higher with both chiller trains operating simultaneously than with one chiller train in standby. The increase in reliability, i.e., decrease in probability of failure, is due to maintaining operation of the running standby chiller train as opposed to starting and maintaining operation of the standby chiller train.

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

Safety Evaluation No.: 97-153

Implementation Document No.: Procedure N2-PM-M@004

USAR Affected Pages: 10.2-8

System: Extraction Steam (ESS)

Title of Change: Extraction Steam System Check Valve Testing Requirements

Description of Change:

This change revised the method of testing ESS system nonreturn valves 2ESS-NRV113 and 2ESS-NRV114 to verify that the valve disc is free to move.

The methods of testing the valves, as described in the USAR, could not be accomplished with the current plant configuration. Steam flow to the clean steam boiler was reduced. The original testing method would not allow sufficient motion to confirm operability. The testing required observation of proper disc movement to verify freedom and disc attachment. This testing was consistent with the intent of GEK 72349, as the purpose was to verify that the valve was not stuck, and moving the counterweight arm would provide sufficient motion to confirm operability.

Safety Evaluation Summary:

An acceptable method of testing the critical nonreturn valves is to verify that the disc is free to move during power changes, and/or manually moving the counterweight arm because it serves as a method to verify the check valve disc is free to move. Either method will verify the functionality of the valve disc.

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

Safety Evaluation No.: 97-154

Implementation Document No.: Calculation A10.1-J-019

USAR Affected Pages: Table 9B.8-3 Sh 12

System: Spent Fuel Pool Cooling and Cleanup (SFC), Service Water (SWP)

Title of Change: RFO6 Cycle-Specific Fuel Pool Cooling Analysis

Description of Change:

The USAR addresses two bounding full core offload scenarios for the SFC system: (1) Case 2, an emergency core offload; and (2) Case 3, a normal full core offload during refueling. The emergency core offload scenario includes analysis of the impact of isolating the spent fuel pool from the reactor cavity. The analysis for the normal full core offload does not fully address this configuration. A review of the applicable criteria determined that the single failure considerations for Case 3 do not apply to Case 2. Therefore, the analysis for Case 2 does not envelope all aspects of Case 3. The outage plan for Refuel Outage 6 (RFO6) is based on completing a full core offload approximately 10 days after shutdown, at which time the fuel pool will be isolated from the reactor cavity. This is being done to reduce personnel exposure, decrease consequential risk and improve the effectiveness of planned in-vessel work. This safety evaluation evaluated the results of an RFO6 cycle-specific heat load analysis associated with isolating the spent fuel pool from the reactor cavity following a full core offload during RFO6.

Safety Evaluation Summary:

The proposed cycle-specific activity for RFO6 allows fuel transfer from the reactor vessel to the pool to begin 96 hours after shutdown, allows a full core offload during normal refueling outages, and takes credit for the cool Lake Ontario water temperature in May to cool the spent fuel pool. Based on a review of Chapters 6 and 15 of the USAR, failure of the SFC system has not been identified as an initiating event of the evaluated accidents. Additionally, it was determined that the proposed activity does not increase the probability of a fuel bundle drop accident as described in Section 15.7.4 of the USAR. No SFC system components will be operating beyond their current design basis capacity, pressure, or temperature limits. The maximum temperature limits of 125°F and 150°F will be maintained. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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

Implementation Document No.: Procedure NEP-POL-01, Rev. 3

USAR Affected Pages: 9A.3-2, 13.1-4, 13.1-14; Figure 13.1-3

System: N/A

Title of Change: Nuclear Engineering Organization Changes

Description of Change:

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

Safety Evaluation Summary:

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

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

Safety Evaluation No.:	98-006
Implementation Document No.:	Nuclear Division Policy (NDD-POL)
USAR Affected Pages:	13.1-3, 13.1-4, 13.1-8, 13.4-1; Figures 13.1-1, 13.1-2, 13.4-1
System:	N/A
Title of Change:	Reorganization: Change to Nuclear Division Policy to Reflect Establishment of Position, "Vice President Nuclear Generation"

Description of Change:

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

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

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

Safety Evaluation Summary:

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

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

Safety Evaluation No.: 98-029

Implementation Document No.: DDC 2F01713

USAR Affected Pages: 7A.1-8, 7A.1-10

System: BYS, CEC, CMS, CSH, ENS, GTS, HCS, HVC, HVP, IAS, LDS, MSS, RHS, RPS, SFC, SLS, SWP, WCS

Title of Change: Analog Input Optical Isolator Card Replacements

Description of Change:

Optical isolators are used at NMP2 as isolation devices in various instrumentation and control system circuits for the purpose of isolating nonsafety-related circuits from the safety-related circuits, or to isolate redundant safety-related circuits. USAR Appendix 7A lists specific model numbers for originally installed plant isolator cards. General Electric no longer supplies two of the listed models, but has developed updated versions of each card. These updated cards eliminate the potential for momentary voltage dropout randomly experienced on some of the obsolete models. This change revised the USAR to reflect installation of these updated models. Additionally, there was a need to replace other installed (obsolete) isolator cards (due to normal expected failures). This change authorized replacement (when necessary) of the obsolete analog input optical isolator cards with updated cards, and evaluated several of the updated models which have already been installed in the plant.

Safety Evaluation Summary:

General Electric has provided documentation stating that the replacement isolator cards conform to all qualification requirements applied to the earlier version of their cards. Specifically, the updated isolator cards do not alter characteristics related to the isolation features of the originally designed card (namely, the optical light quartz rod contained in a ceramic barrier block assembled in a cast aluminum housing). Additionally, the performance characteristics associated with adequate protection against the effects of EMI, short circuit failures, voltage faults and/or surges are not altered with the updated design, and the failure modes of the updated isolator cards are no different than the earlier versions. No new or different testing is required with the replacement cards and isolator calibrations. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.:	98-031
Implementation Document No.:	Mod. N2-97-044
USAR Affected Pages:	6.2-86; Tables 3.9B-2k Sh 1, 5, 3.9B-2L Sh 1, 6.2-56 Sh 18; Figures 5.4-2b, 5.4-2c, 5.4-9a, 5.4-13a, 5.4-16a, 6.3-7a, 9.3-9f
System:	RCS, RHS, WCS, ICS, CSL
Title of Change:	Flexible Hose Replacement and RCS Vent and Drain Valve Modification

Description of Change:

A failure of flexible metal hose 2RCS*HOSE40 caused increasing unidentified drywell leakage and a subsequent unscheduled shutdown of NMP2. As a result of a root cause analysis, it has been determined that 26 3/4" diameter flexible metal hoses installed at NMP2 are potentially subject to throughwall failure due to intergranular stress corrosion cracking (IGSCC). The replacement population was established as 26 based on the conditions required to support IGSCC, including temperature, chemistry, pressure and residual stresses. The population that has been replaced includes primarily instrumentation sensing lines connected to the reactor coolant pressure boundary (RCPB), as well as two in a warmup line for the ICS system, one of which also serves as a containment isolation barrier.

The corrective action for these hoses was to eliminate 7 and replace 19 with tubing designed to accommodate the thermal, seismic and hydrodynamic movements. Included in the modification scope was the replacement of four vent valve installations on the main RCS isolation valves which eliminated a maintenance issue requiring cutting of 3/4" pipe whenever the main RCS valve bonnets are removed.

Safety Evaluation Summary:

The replacement installations are designed to comply with NMP2 design, fabrication and installation requirements for safety-related Category I, ASME Class 2 piping and components. In addition, the new designs have been evaluated and modified to address resonance issues to assure that there are no long-term root weld fatigue concerns. The issues associated with tap line failures for RCS small bore lines have been recognized as nuclear plants have aged. The replacement designs have been analyzed and incorporate design features to avoid this issue. Finally, the majority of replacements involve instrument lines which penetrate

Safety Evaluation No.: 98-031 (cont'd.)

Safety Evaluation Summary: (cont'd.)

primary containment, and all provisions of Regulatory Guide 1.11 have been met. Thus, the new configurations are more reliable than the original design, and the RCPB, RCIC, instrumentation and containment isolation functions are not adversely impacted.

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

Safety Evaluation No.: 98-033 Rev. 0, 1 & 2

Implementation Document No.: Design Change N2-97-067

USAR Affected Pages: 3.7A-29, 6.2-47, 6.2-48, 6.2-49, 6.2-50, 6.2-53, 6.3-7, 6.3-8, 6.3-9, 6A.4-33, 6A.4-36, 6A.4-36a, 6C-2, 6D-12, 6D-13; Tables 3.9A-2 Sh 1, 3, 4, 7, 3.9A-4 Sh 1, 6.2-64, 6A.2-1 Sh 2, 6A.2-1a Sh 2, 6A.4-14 Sh 1, 2, 6A.4-15 Sh 1, 2; Figures 1.2-7 Sh 1, 6.2-39a, 12.3-6, 12.3-39

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

Title of Change: ECCS Suction Strainer Replacement

Description of Change:

The ECCS suction strainers on the RHS, CSL and CSH systems have been replaced with strainers that have a much larger surface area and are capable of handling more debris with a lower pressure drop. Modifications were required to the south hatch enclosure wall and mezzanine to enable strainer installation into the suppression pool.

Safety Evaluation Summary:

The replacement strainers will be designed to all applicable codes and quality requirements of the existing strainers and designed to withstand all applicable hydrodynamic and seismic loads. In addition, the new strainers will be designed to maintain adequate net positive suction head to the ECCS pumps with an increased debris loading. All aspects of the installation (including wall/mezzanine modifications and downcomer removal contingency) will be conducted in accordance with existing design requirements and approved procedures and controlled such that required ECCS functions will be maintained during installation. Activities required for installation of the strainer (including wall, mezzanine modification and downcomer contingency plan) will be controlled by approved procedures and meet all applicable design criteria including safety related, seismic, radiation shielding, and radiation zone requirements. Strainer movement (both old and new), as well as removed concrete sections and downcomer contingency removal, will be in accordance with the heavy loads criteria of NUREG-0612 and USAR Appendix 9C. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.:	98-034
Implementation Document No.:	Procedure N2-CSP-OFG-@336
USAR Affected Pages:	11.5-2, 11.5-12; Figure 11.3-1b
System:	Offgas (OFG)
Title of Change:	Offgas Sampling and Analysis

Description of Change:

In order to provide continuous isotopic monitoring of the OFG system during power suppression testing to identify a fuel leakage location, the supply and return to the offgas pretreatment monitor's tie to 2OFG-SMPV13 must be disconnected and a continuous isotopic analyzer installed via station procedure N2-CSP-OFG-@336. The function of 2OFG-SMPV13 is to obtain offgas grab samples, and SMPV13 will be disconnected from one of the offgas radiation monitors while the isotopic monitor is in service. This safety evaluation evaluated the operability of both radiation monitors 2OFG-CAB13A and 2OFG-CAB13B while the continuous isotopic monitor is connected to either offgas radiation monitor. It also evaluated the ability to draw offgas samples in the time required by Technical Specifications should both radiation monitors be declared inoperable.

Safety Evaluation Summary:

This change temporarily disconnects 2OFG-SMPV13 from either 2OFG-CAB13A or 2OFG-CAB13B and installs a continuous isotopic monitor that returns its sample flow to the monitor to which it is connected. Sample flows to 2OFG-CAB13A and 2OFG-CAB13B will be maintained at 2 scfm (nominal) with only minor flow variations. This does not affect either radiation monitor's operability.

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

Safety Evaluation No.: 98-035

Implementation Document No.: Design Change N2-97-087

USAR Affected Pages: 3.9B-41, 5.4-22, 5.4-23, 5.4-27; Figures 5.4-9d, 5.4-10 Sh 1 & 2

System: Reactor Core Isolation Cooling (ICS)

Title of Change: Increase RCIC Turbine/Pump Set Speed to 4700 rpm

Description of Change:

The reactor core isolation cooling (RCIC) system is a safety-related system that provides high-pressure reactor coolant. Since the RCIC system is safety related, it must be periodically tested in accordance with the Technical Specifications (TS) and In-service Testing (IST) requirements. During these tests it has been documented that the tested RCIC pump head meets the required TS and IST RCIC pump head total developed head with little margin.

This design change increased the rated rpm of the RCIC pump/turbine from 4600 to 4700 rpm and changed the RCIC turbine power control system instrument and control settings to support operation of the RCIC turbine at a maximum 4700 rpm. This provides more margin to the RCIC pump head at the required RCIC flow rate.

Safety Evaluation Summary:

Increasing the rated speed of the RCIC pump and turbine to 4700 rpm enhances the capability of the RCIC system to inject water into the reactor pressure vessel in accordance with the current licensing and design basis and, as such, does not affect the design basis or the safety functions of the systems involved. The pump and turbine are re-rated to the higher operating speed of 4700 rpm which is well below the mechanical turbine overspeed trip setpoint of 5460 rpm. Therefore, the probability of missile generation is not affected by this change. Additionally, the existing piping design pressure envelopes the pump head at 4700 rpm. Steam supply piping pressure and temperature are unchanged. Therefore, acceptance limits (design pressure and temperature) used to define the margin of safety are unchanged. Piping and component support stresses have been evaluated to ensure that stresses remain below acceptance limits as well.

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

Safety Evaluation No.: 98-036

Implementation Document No.: DDC 2F01775

USAR Affected Pages: 9.5-36; Table 9B.8-1 Sh 8, 10, 12; Figures 9.5-40b, 9.5-40c, 9.5-41 Sh 1, 2

System: Standby Diesel Generator Fuel Oil Storage and Transfer (EGF)

Title of Change: Revise the Division I, II & III Standby Diesel Generator Fuel Oil Storage Tank Level Transmitter Component Identification from "LIT" to "LT"

Description of Change:

This safety evaluation evaluated the removal of integral level indication from the Division I, II, and III system level indicating transmitters 2EGF-LIT10A, 2EGF-LIT10B and 2EGF-LIT101. The plant as-built configuration of the subject level transmitters does not include integral level indication as described by the "LIT" component identification. The level indication is provided remotely on the Diesel Generator Building exterior wall near each respective fuel oil fill station. Therefore, the existing "LIT" component identification has been revised to "LT". Additionally, this change revised the USAR location description for level indicators 2EGF-LI10A, 2EGF-LI10B and 2EGF-LI101.

Safety Evaluation Summary:

The subject components have no interaction with or impact on systems or components important to safety. The plant as-built configuration of the subject components conforms to the applicable design basis and regulatory requirements including ANS 59.51 and is, therefore, acceptable. The proposed change has no effect on the EGF system or other plant systems important to nuclear safety.

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

Safety Evaluation No.: 98-037

Implementation Document No.: Mod. N2-96-020

USAR Affected Pages: 5.4-21, 5.4-22, 7.4-8, 8.3-69; Tables 3.9A-12 Sh 5, 8.3-4 Sh 1, 2, 3, 8.3-8; 9B.8-1 Sh 27, 9B.8-3 Sh 6; Figures 5.4-9b, 5.4-9c, 5.4-9d, 7.4-1 Sh 1, 3, 4, 5, 6

System: Reactor Core Isolation Cooling (ICS)

Title of Change: 2ICS*MOV120 Replacement

Description of Change:

Main steam inlet valve 2ICS*MOV120 works in parallel with bypass valve 2ICS*MOV159. The bypass valve addresses industry problems associated with overspeed of the reactor core isolation cooling (RCIC) turbine.

However, the 2ICS*MOV159 valve has been experiencing chronic leakage problems since commencement of commercial operation of NMP2. It was suspected that this valve suffered from steam cutting of the seat and was not capable of providing leak-tight shutoff under normal operating conditions. In addition, both valves 2ICS*MOV159 and 2ICS*MOV120 required changes pursuant to Generic Letter 89-10 to correct insufficient torque problems which were not pertinent to the steam cutting problem.

While the bypass valve leakage in itself was not unacceptable, the leakage appeared to be the root cause for recurring RCIC turbine problems which resulted in system unavailability. The leakage was associated with governor valve sticking, overspeed trip tappet sticking, and turbine shaft pitting, as well as lube oil contamination due to condensation.

The chronic leakage of the bypass valve resulted in an industry-wide modification which eliminates the bypass valve (2ICS*MOV159) and replaces the main steam admit valve (2ICS*MOV120) with a modified globe valve. This modified valve has a combined Cv versus stroke curve that matches that of the current detail valve installation. That is, the Cv versus stroke for the new valve limits flow to approximately 5,000 lbm/hr (bypass flow) over the first 50% of the stroke and then gradually ramps to full open for the last 50% of the stroke.

Safety Evaluation No.: 98-037 (cont'd.)

Safety Evaluation Summary:

This change eliminated RCIC steam bypass valve 2ICS*MOV159 and replaced the 2ICS*MOV120 with a combined bypass and full flow valve. The modified system is in accordance with the design requirements of the original system and retains the actuation to full flow operating time of 30 seconds established for RCIC in the original design basis.

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

Safety Evaluation No.: 98-039

Implementation Document No.: DDCs 2E11484, 2E11485, 2E11486

USAR Affected Pages: Figure 9.4-9 Sh 21

System: Drywell Cooling (DRS), Main Steam (MSS)

Title of Change: Control Room Chart Recorder Replacement

Description of Change:

Control Room chart recorders 2MSS-TRSH1001 (2CEC*PNL632), 2DRS-TR10A (2CEC*PNL873) and 2DRS-TR10B (2CEC*PNL873) are 12-point Tracor Westronics Model M11E. The recorders are nonsafety-related components and are installed in safety-related panels. These chart recorders had become obsolete and were presenting an increasing burden with regard to procurement of replacement parts and increasing maintenance activities. Therefore, the recorders have been replaced with new 24-point Westronic 3200 Series chart recorders.

Safety Evaluation Summary:

The new recorders were selected based on technology changes and design features which allow essentially a one-for-one replacement without altering the existing Control Room panels. The new recorders also have similar power requirements when compared to the obsolete units. This change has no impact on the function of the recorders, nor on the output available from the recorders.

All applicable criteria are satisfied or exceeded by the design and features of the new recorders. In conclusion, the old and new recorders maintain equivalent form, fit and function.

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

Safety Evaluation No.: 98-040 Rev. 0, 1 & 2

Implementation Document No.: Procedures N2-TTP-DECON-@002T, N2-TTP-DECON-@001T

USAR Affected Pages: N/A

System: RCS, WCS, RHS, MWS, CNS, SAS, SWP, DFR, HVR

Title of Change: Chemical Decon of RCS

Description of Change:

This procedure change controls the chemical decontamination of the reactor recirculation system (RCS). Decontamination of the RCS has the potential for having a major impact on reducing personnel man-rem exposure during inspection, maintenance, repair or replacement of boiling water reactor components. The low oxidation state metal ion (LOMI) decontamination process reduces radiation fields on piping and components through the removal of radioactive corrosion products incorporated into, and deposited on, the established corrosion film. The chemicals reduce or dissolve the corrosion film from the interior of piping and components. The chemicals and dissolved corrosion film constituents will be removed from plant systems via an ion exchange process. In addition, this safety evaluation evaluated the movement and storage of up to 12 jet pump nozzle/inlet mixers on the walls of the spent fuel pool.

Safety Evaluation Summary:

This evaluation addresses the impact of the decontamination evolution with respect to the permanent plant equipment, and the decontamination process itself in relation to plant materials compatibility. Components in the RCS that will be in contact with the LOMI chemicals were researched for their material characteristics and compatibility. They were determined to have no unacceptable levels of corrosion. The corrosion rates anticipated for the process are such that the expected materials corrosion will be within the specified limits so as not to infringe upon the ASME pressure/temperature minimum wall thickness. The proceduralized decontamination process has not been determined to cause or accelerate intergranular stress corrosion cracking or intergranular attack.

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

Safety Evaluation No.: 98-044

Implementation Document No.: Design Change N2-98-001

USAR Affected Pages: Figure 9.2-1h

System: Service Water (SWP), Service Water Chemical Treatment (SCT)

Title of Change: Service Water Detoxicant Injection Quill Design

Description of Change:

Piping wall thickness associated with 2SWP*V1221 and 2SWP*V1222 is less than manufacturer minimum wall thickness, caused by manufacturer out-of-round tolerance and chemical corrosion. A corrective action was implemented to monitor piping wall thickness. Followup ultrasonic testing (UT) readings showed further piping deterioration since this condition was identified. This prompted a re-evaluation of the cause of piping wall thinning. Results of the wall thickness data provided by three sets of UT readings, from June 1997 through January 1998, show that piping is chemically corroding at a steady rate of several mils per month since the monitoring program started. The piping wall thinning is attributed to chemical corrosion.

This design change replaced the sodium bisulfite injection lines located below el. 261' in the Screenwell Building pipe tunnel with a quill design. The quill design uses a carbon steel sockolet, isolation valve, and gland flange together with a stainless steel injection nozzle, slip-on flange and a quill isolation valve. The quill isolation valve serves as the ASME Class 3 to nonsafety-related boundary isolation valve.

Safety Evaluation Summary:

This design change replaces the existing sodium bisulfite injection line with a quill design. The quill design uses stainless steel material which is chemically resistant to sodium bisulfite solution. The change will not impact the SWP system safety-related function. The SCT system detoxicant injection is not required for safe shutdown or accident mitigation. Adequate provisions are in place in the plant to preclude any potential flooding that could occur due to malfunction in the SCT system even though the SWP pipe tunnel is designed to be completely flooded.

The proposed change would not result in operating the SWP system outside of its design basis. This change would not cause a change to any system interface in a way that would increase the likelihood of an accident. Failure in the

Safety Evaluation No.: 98-044 (cont'd.)

Safety Evaluation Summary: (cont'd.)

nonsafety-related portion of the system (SWP to SCT) is unlikely to occur since all piping and components affected by this design change are seismically designed. In the unlikely event of SWP to SCT piping failure, it does not impact the SWP system safety-related function since service water has already performed its safety-related cooling function and it is being returned to Lake Ontario.

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

Safety Evaluation No.: 98-046

Implementation Document No.: Emergency Operating Procedures (EOP), Calculation A10.1-AC-001

USAR Affected Pages: 1.10-110, 7A.2-2; Table 7.5-2 Sh 5; Figure 5.2-4 Sh 1

System: Various

Title of Change: Cycle 7 Changes to the NMP2 Emergency Operating Procedures

Description of Change:

A revision to the EOPs has changed some operating parameters as a result of the complete replacement of GE9 fuel with GE11 fuel. The parameters which were revised are as follows:

- Top of active fuel (TAF)
- Minimum core flooding intervals
- Minimum steam cooling reactor pressure vessel (RPV) water level
- Minimum zero injection RPV water level

The Fuel Zone Correction Curve was revised to reflect the new levels of TAF, minimum steam cooling RPV water level, and minimum zero injection RPV water level. The curves for the following values were also affected by the calculation changes. However, the magnitude of the changes cannot be measured by control room instruments and, therefore, were not changed.

- Heat capacity temperature limit
- Heat capacity level limit
- Pressure suppression pressure

Safety Evaluation Summary:

These changes do not alter the philosophy, logic, or validity of the NMP2 EOPs, nor do they affect the capability of the operators to recover from an accident. The operator actions prescribed in this new revision are in accordance with the BWROG EPGs and, when applied to the licensing basis accidents and transients, the EOPs will not increase the probability or the consequences of these events as depicted in USAR Chapter 15. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.: 98-047

Implementation Document No.: ABB Fuel Sipping Procedure

USAR Affected Pages: 9.1-28, 9.1-30, 9.1-43

System: Fuel Nuclear Refueling (FNR)

Title of Change: Telescoping Sipping System

Description of Change:

This change temporarily installed fuel leak detection equipment on the mast and the deck of the refueling bridge. The purpose of this equipment is to detect a leaking fuel bundle while it is being transported from the reactor vessel to the spent fuel pool. The system can also be used to detect leaking fuel bundles by lifting the bundle 10 feet in either the reactor vessel or the spent fuel pool. The procedure provides controls to assure that fuel bundles will not be raised above a height which would be required during normal core off-load procedures.

Safety Evaluation Summary:

The telescoping sipping system is a nondestructive means of identifying a fuel leak. The installation and operation of the system will not change the fit, form or function of the existing refueling equipment and is bounded by the current fuel handling accident analysis contained in USAR Section 15.7.4. The use of this equipment at NMP2 was found to be acceptable when evaluated against applicable criteria.

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

Safety Evaluation No.: 98-048 Rev. 0 & 1

Implementation Document No.: NEDE-24011-P-A-13
NEDE-24011-P-A-13-US

USAR Affected Pages: 3.9B-70, 15.2-2, 15.2-32, 15.4-16, A.0-1, 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.15.1-1, A.15.1-8, A.15.2-5, A.15.2-6, A.15.2-9, A15.4-7, A.15B-1, A.15D-1; Tables 3.9B-2o, A.6-2, A.15.0-4 Sh 1

System: Various

Title of Change: Operation of NMP2 Reload 6/Cycle 7

Description of Change:

This change consisted of the addition of new fuel bundles and the establishment of a new core loading pattern for Reload 6/Cycle 7 operation of NMP2. Two hundred and seventy-two (272) new fuel bundles of the GE11 design were loaded. Eighty (80) GE9B-P8CWB320-9GZ1-100M-150-T bundles and 192 of the GE11-9CUB332-137GZ-120M-146T 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-13 and NEDE-24011-P-A-13-US (GESTAR II). This document describes the fuel licensing acceptance criteria; the fuel thermal-mechanical, nuclear, and thermal-hydraulic analyses bases; and the safety analysis methodology. For Reload 6, 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.

Based on the evaluations performed, it is concluded that NMP2 can be safely operated during Reload 6/Cycle 7, and that this change does not involve an unreviewed safety question.

Safety Evaluation No.: 98-049

Implementation Document No.: Procedure N2-OP-71A

USAR Affected Pages: 8.1-3, 8.2-1, 8.3-4, 8.3-4a

System: 13.8 kV Normal Power (NPS)

Title of Change: Paralleling of 13.8 kV Buses 2NPS-SWG001 & 003 During Plant Shutdown

Description of Change:

This change revised the bus transfer of the normal 13.8 kV buses to their alternate backup offsite power reserve station service transformers during maintenance activities. The temporary alteration required racking out of an available normal station service transformer 13.8 kV feed breaker and installing it in the 13.8 kV switchgear backup power spare breaker cubicle, and installation of temporary jumpering of early "b" contacts to allow simultaneous closing of normal backup source and alternate backup source breakers. The two reserve transformer feeds to 13.8 kV buses were paralleled manually and momentarily from the Power Generation Control Complex (PGCC) control panel 2CEC-PNL852. After completion of the manual transfer, the normal backup source 13.8 kV breaker was disconnected and one reserve transformer fed both 13.8 kV switchgear buses. The original plant configuration was restored prior to plant startup.

Safety Evaluation Summary:

This live bus transfer allows certain 13.8 kV loads and critical loads connected to the 4.16 kV stub buses (switchgear 2NNS-SWG014 and 015) to continue to operate. The transfer of the two buses is performed in accordance with approved procedures and installation specifications. The proposed change to the operating procedure permits momentary paralleling of the two offsite power sources only during plant shutdown or refueling condition. The offsite power sources and the onsite power sources are not considered as the initiator of any previously analyzed accident, but are required to mitigate the consequences of an accident. The proposed change would not violate any offsite source design basis criteria outlined in regulatory documents and industry standards. In addition, the proposed evolution does not affect the design functions of the offsite power sources and the associated equipment. Therefore, the proposed momentary paralleling of the two offsite power sources during plant shutdown or refueling operation 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.:	98-050
Implementation Document No.:	Design Change N2-97-064
USAR Affected Pages:	3.7A-17; Tables 3.9A-4 Sh 4, 5, 3.9A-12 Sh 7, 12, 3.10A-1 Sh 10
System:	Reactor Recirculation (RCS), Sampling - Reactor Building (SSR)
Title of Change:	Replacement of 2RCS*SOV104 and 2RCS*SOV105 with Direct-Acting Solenoid Valves

Description of Change:

This design change replaced Target Rock 3/4-inch valves 2RCS*SOV104 and 2RCS*SOV105, which are the inside and outside containment isolation valves in the normal reactor coolant sampling line, with Valcor direct-acting solenoid valves. These valves have a smaller seat area and, thus, do not depend on the balanced port feature. The sample line provides a sample point for continuous monitoring of reactor coolant conductivity, oxygen and zinc levels. In addition, it also provides a grab sample point for reactor coolant.

Previously these valves were balanced port pilot solenoid valves which depended on a very small pilot hole and small annular clearances to equalize the pressure across the seat in order to function efficiently. In the primary sampling application, this led to a history of stroking and position indication problems, as well as a trend of degraded leak rate test results over the installed life.

Safety Evaluation Summary:

Replacement of these valves meets all original quality assurance, code, design (including seismic and environmental) and installation requirements associated with the active safety-related ASME III nature of these functions. This will ensure that the replacement items will perform the identical pressure boundary and isolation function of the original installation.

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

Safety Evaluation No.: 98-051

Implementation Document No.: Procedure GAP-RPP-04

USAR Affected Pages: Table 12.5-4

System: N/A

Title of Change: Passive Whole Body Counting Bioassay Methodology Change

Description of Change:

Procedure GAP-RPP-04, Step 3.7.3, previously defined an in-vivo bioassay as a whole body count. The procedure has been revised to redefine the term in-vivo bioassay as being either a whole body count or a passive whole body monitor. This change permits in-vivo bioassay exams to be performed using passive whole body monitoring equipment.

Safety Evaluation Summary:

The proposed change in operation of whole body counting equipment, as described in the revision to GAP-RPP-04, satisfies applicable acceptance criteria and will not affect the design of systems, structures, or components; the operation of plant equipment or systems; nor maintenance or testing activities.

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

Safety Evaluation No.: 98-053

Implementation Document No.: Temporary Mod. 97-007

USAR Affected Pages: N/A

System: Neutron Monitoring (NMS)

Title of Change: Install.Temporary Modification To Facilitate Installation of GE NUMAC Power Range Neutron Monitoring System

Description of Change:

The power range monitoring (PRM) system was replaced with a General Electric nuclear measurement analysis and control (NUMAC) power range neutron monitoring (PRNM) system under Modification PN2Y93MX002. When the existing equipment was removed from panel 2CEC*PNL608, there were no inputs to the reactor protection system (RPS) and reactor manual control system (RMCS) from the PRM system. This results in a full scram signal in RPS and prevents RMCS from allowing any control rod movement. This would have greatly inhibited maintenance and surveillance activities on RPS and RMCS. To facilitate operation in the shutdown or refuel mode and installation of the PRNM system, jumpers were installed in RPS and RMCS to simulate the normal shutdown and refuel configuration of the PRM system.

Safety Evaluation Summary:

The PRNM system is downscale and is not required to be operable when the reactor is in the shutdown or refuel mode. Jumpers will be installed in RPS and RMCS to simulate the normal shutdown and refuel configuration of the PRNM system. The source range monitors and intermediate range monitors in the startup range neutron monitoring system will provide neutron flux information and any mitigating actions required.

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

Safety Evaluation No.: 98-054

Implementation Document No.: Procedure N2-OP-11

USAR Affected Pages: 9.2-3, 9.2-7

System: Service Water (SWP)

Title of Change: Service Water Lineup During Unit Outages

Description of Change:

During refueling outages, or during prolonged outages where major service water maintenance is required, the SWP system is placed in various operational lineups to establish divisional outage windows and support the outage schedule. These lineups are established in such a way as to ensure that Technical Specification requirements on service water operability are met. During these divisional outages, it is desirable to continue to provide service water flow to the nondivisional nonessential portions of the system, including the reactor building closed loop cooling system, the turbine building closed loop cooling system, and various turbine building unit coolers. This allows continued operation of various plant support systems such as instrument air, radwaste, and normal air conditioning/ventilation.

Due to the layout of the system, where the nonessential loads are fed only from the Division 1 (A Loop) end of the main pump discharge header, manual cross-tie valves (2SWP*V17 and 2SWP*V32) have been provided off the Division 2 (B Loop) end of the header to supply the nonessential loads. These manual valves bypass the automatic isolation valves which normally close on loss of offsite power (LOOP) or loss of flow in the pump discharge headers, so that flow is directed to the essential headers under these conditions. In addition, a manual valve (2SWP-V8) is provided between the nondivisional portions of the A Loop and B Loop in order to allow for the discharge of the nonessential flow through the desired discharge header.

It is recognized that certain criteria should continue to be met while these cross-tie valves are in service. In particular, it is important to stop flow at either the inlet or the outlet of the nonessential headers to ensure flow is redirected to the essential loads. In addition, prompt action should be taken to close the manual valves in the event of a LOOP or upon indication of a break in the nonessential header. This safety evaluation evaluated the use of the manual cross-tie valves.

Safety Evaluation No.: 98-054 (cont'd.)

Safety Evaluation Summary:

Service water primarily provides cooling water to systems and equipment that help mitigate the consequences of an accident, and does not act as a precursor to any accidents. The safety functions that service water supports, including cooling of "important to safety" equipment (directly and via room coolers), control building cooling, secondary containment drawdown, and decay heat removal, will continue to function at acceptable safety levels.

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

Safety Evaluation No.: 98-055

Implementation Document No.: EDCs 2F00824, 2F00763B

USAR Affected Pages: Table 6.1-2 Sh 1

System: High-Pressure Core Spray (CSH)

Title of Change: Replace Valve Stem Material for CSH Valves

Description of Change:

Anchor Darling Valve Company offered replacement valve stem material for valves 2CSH*MOV101, MOV105, MOV107, MOV110, MOV111, MOV112 and MOV118. The stem material change was from ASTM A479TP410 to ASTM A564TP630-1075, except for 2CSH*MOV111 where the replacement stem material is ASTM 276TP410. The new stem material equals or exceeds the requirements of the old material in strength, hardness and corrosion resistance.

Safety Evaluation Summary:

The replacement stem material will provide higher ASME allowable stresses for the stem. The chemical composition of the new stem material has better or equivalent corrosion resistance. Thus, the integrity of the stem is not impacted. The new stem material equals or exceeds the critical characteristics necessary for the valves to perform their safety function.

The new stem material neither affects the original design basis nor the operability of the valves. The CSH system, as well as the valves, will continue to operate as designed during or after an accident.

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

Safety Evaluation No.: 98-056

Implementation Document No.: Procedures N2-MPM-ESS-R@023, N2-PM-M@004

USAR Affected Pages: 10.2-10

System: Extraction Steam (ESS)

Title of Change: Clarification of Inspection/Maintenance in USAR for ESS Check Valves

Description of Change:

This safety evaluation evaluated an acceptable alternate method of administrative control with respect to the inspection and maintenance of ESS check valves (nonreturn valves).

The USAR stated that testing requirements for critical and noncritical ESS check valves would be determined based on performance established by the preventive maintenance program, and that internal inspection and maintenance is performed in accordance with the five refueling cycle turbine and valve maintenance and inspection program. However, internal inspection and maintenance of the ESS check valves (nonreturn valves) is not included in the five refueling cycle turbine and valve maintenance inspection program. Additionally, the five refueling cycle turbine and valve maintenance inspection program is complete. The USAR has been revised to identify an acceptable alternate method of administrative control with respect to the inspection/maintenance of ESS check valves. The alternate method of administrative control is the existing preventive maintenance program for these valves.

Safety Evaluation Summary:

The design basis of the ESS system is not impacted by this change. The existing administrative requirements with respect to inspection/maintenance of the nonreturn valves will continue to ensure that the ESS nonreturn valves will function as required per design. The change to the USAR with respect to the alternate method of administrative control regarding the inspection/maintenance of ESS check valves will not result in a reduction of the safety margins.

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

Safety Evaluation No.: 98-057

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

USAR Affected Pages: 12.1-8, 12.5-13, 13.1-12; Table 13.1-2; Figure 13.1-2

System: N/A

Title of Change: Radiation Protection Department Organization Change

Description of Change:

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

Safety Evaluation Summary:

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

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

Safety Evaluation No.: 98-060

Implementation Document No.: Temporary Mod. 98-009

USAR Affected Pages: N/A

System: Reactor Water Cleanup (WCS), Reactor Building Equipment Drains (DER), Reactor Coolant (RCS)

Title of Change: Temporary Re-route of WCS Drain Lines to the Suppression Pool

Description of Change:

This temporary modification cut the Non-ASME Class 4 nonsafety-related WCS drain pipes downstream of drain valves 2WCS*V366/V367 and 2WCS*V368/V369, which lead to the equipment drain funnel. A temporary hose was then attached and routed to a nearby drywell floor downcomer. Subsequent opening of the drain valves directed flow to the suppression pool. The pipes were cut downstream of the ASME Class 1/Class 4 break. The hose was attached with a temporary mechanical connection and proved acceptable for the maximum anticipated pressure in this gravity drain application.

Safety Evaluation Summary:

This change will be in effect during plant shutdown. The USAR Chapter 15 accidents applicable to this plant condition are a fuel bundle drop and a decrease in reactor coolant inventory (as it applies to the reactor coolant pressure boundary [RCPB]). The draining configuration is unrelated to fuel movement activities. In addition, the proposed change has no impact on the design and qualification of the RCPB. Positive controls will be in place after the drain evolution to ensure that the required reactor vessel/spent fuel pool levels are maintained.

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

Safety Evaluation No.: 98-061

Implementation Document No.: Design Change N2-98-007

USAR Affected Pages: Figure 9.3-9e

System: Reactor Building Equipment and Floor Drains (DFR)

Title of Change: 3/4" Test Line Configuration Change

Description of Change:

This change revised the configuration of the 3/4" test line off the 6" drain line. The DFR test line is located upstream of the inboard isolation valve. The changes involved replacement of outer test line valve 2DFR*V113 by a safety-related threaded cap, and rework on the existing weld.

These changes were required to offset the impact of higher weight of the valves on the DFR test line, and maintain the functional configuration that existed before the change.

Safety Evaluation Summary:

The proposed changes enhance the piping structural capacity by eliminating the isolation valve and upgrading portions of pipe from nonsafety related to safety related. In addition, these changes do not impact containment isolation capability, leak rate testing, primary containment leak rate, suppression pool bypass capability, etc.

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

Safety Evaluation No.: 98-062

Implementation Document No.: DDC 2M11372

USAR Affected Pages: 5.4-24

System: Reactor Core Isolation Cooling (ICS)

Title of Change: RCIC Pump Lubricating Oil Cooling Water Flow

Description of Change:

Cooling water to the RCIC turbine lube oil cooler comes from the discharge of the RCIC pump through a pressure control valve (PCV) (2ICS*PCV115) and a restricting orifice, and is recirculated back to the suction of the pump. The PCV establishes a desired pressure upstream of the lube oil cooler, and the driving differential is determined by the pump suction pressure. The orifice is sized to maintain 16 to 25 gpm to the lube oil cooler based upon pump suction pressure varying from 50 psig to minimum net positive suction head (NPSH) value. The orifice also limits flow in the event that the PCV fails open.

As a result of a modification to the PCV to make it more reliable, there was an associated change in valve flow coefficient and a resizing of the restricting orifice. Post-modification testing of the cooling water line and subsequent calculations to evaluate sensitivity to changing conditions revealed the upper end of the range could not be achieved under the highest differential conditions without falling below the lower end of the range under the lowest differential conditions.

This change increased the upper end of the cooling water band from 25 gpm to 40 gpm under worst-case conditions.

Safety Evaluation Summary:

The cooling water range of 16 to 25 gpm was established based on General Electric Specification 22A4967AA as a nominal design range. Subsequently, 16 gpm was used as the design flow rate used in sizing the lube oil cooler, as documented in the RCIC turbine vendor manual. Twenty-five gpm is a nominal number for the upper end of the band, and was used for initial sizing of the RCIC pump.

Hydraulic analysis has confirmed that, with the new PCV and restricting orifice, the minimum cooling water flow rate of 16 gpm will be maintained with large margin when subjected to a differential corresponding to the minimum PCV setpoint and the maximum RCIC pump suction pressure. The analysis also shows

Safety Evaluation No.: 98-062 (cont'd.)

Safety Evaluation Summary: (cont'd.)

that the maximum cooling water flow rate will increase to approximately 40 gpm with the maximum PCV setpoint and the minimum pump NPSH. Further analysis also demonstrates that the RCIC system will still deliver the required 600 gpm to the reactor at a vessel pressure of 1215 psia with adequate margin for future pump degradation allowed under the IST Program.

Evaluations have also shown that the increased maximum cooling water flow will not be detrimental to the RCIC lube oil cooler when considering low lube oil temperature limitations, tube vibration and flow-induced erosion. Finally, the change has no impact on the previously analyzed potential for a cooling water PCV failure or its associated relief valve.

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

Safety Evaluation No.: 98-063

Implementation Document No.: DDC 2F01871

USAR Affected Pages: N/A

System: Neutron Monitoring (NMS)

Title of Change: Install Temporary Power to Power Range Neutron Monitoring System During the Time the EPAs are Replaced

Description of Change:

To facilitate testing of the newly installed power range neutron monitoring (PRNM) system, the reactor protection system (RPS) power cables were lifted at the input terminals to PRNM panel 2CEC*PNL608 and temporary nonsafety-related power was connected to power up the PRNM system. The electrical protection assemblies for the RPS power were replaced and a bypass switch was installed for one of the power sources to uninterruptible power supply 2VBB-UPS3B.

Safety Evaluation Summary:

During the time the PRNM system is being replaced and tested, the APRMs are downscale and not required to be operable.

The PRNM system has switching power supplies to convert the 120 V ac incoming power to dc for use by the nuclear measurement analysis and control (NUMAC) electronics. The PRNM power supplies will buffer the temporary nonsafety-related power sources from the PRNM system electronics. The divisional independence of the PRNM electronics on the load side of the power supplies will be maintained. In addition, the PRNM system is designed to operate over a wider range of input voltage and frequency than what will be received from the temporary power.

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

Safety Evaluation No.: 98-066

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

USAR Affected Pages: N/A

System: Service Water (SWP)

Title of Change: Freeze Seals for SWP Supply and Return Branch Runs to 2HVT-UC210A and 2HVT-UC210B

Description of Change:

During refueling outage RF06, two freeze seals were required in the 3" NPS Sch 40 branch runs of the SWP supply and return piping for unit coolers 2HVT-UC210A/B. These freeze seals provided closure isolation between the piping in the Turbine Building and the piping and unit coolers located in the main steam tunnel (MST). The application of the freeze seals allowed the piping in the MST to be drained in order to repair unit cooler inlet and outlet isolation valves 2SWP-V809A, V809B, V810A and V810B in the MST. The freeze seals were located in piping runs 2SWP-003-646-4 and 2SWP-003-647-4 at el. 272 ft. in the Turbine Building.

Safety Evaluation Summary:

Contingency Plan and Recovery Plan Criteria in the Freeze Seal Data Sheets describe actions which are to be taken to minimize or preclude those conditions which may result in the loss of freeze seals, protection of personnel and plant equipment, or damage to the piping system after the freeze seals are thawed. Dimensional measurements of the piping and liquid penetrant examinations before and after freeze seal activities ensure that piping pressure-retaining and structural integrity is maintained.

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

Safety Evaluation No.: 98-067 Rev. 0, 1 & 2

Implementation Document No.: Design Change N2-98-008

USAR Affected Pages: 9.4-11, 9.4-12, 15.6-13, 15.7-14, 15.7-18, 15.7-19; Tables 6.2-56 Sh 7, 8.3-1 Sh 15, 16, 20, 8.3-2 Sh 14, 15, 20, 8.3-5 Sh 1, 3, 5, 6, 8.3-6 Sh 1, 3, 5, 6, 9B.8-1 Sh 13, 14, 16, 17, 40, 15.4-10, 15.4-12, 15.4-13, 15.6-1, 15.6-4, 15.6-6 Sh 1, 2, 15.6-9, 15.6-13 Sh 10, 11, 15.6-15b, 15.6-16b, 15.7-9 Sh 1, 2, 15.7-10, 15.7-11, 15.7-12, 15.7-16 Sh 1, 15.7-17; Figures 9.4-1b, 9.4-1c, 9.4-1d, 9.4-1e, 9.4-4 Sh 1, 2, 6, 9.4-5 Sh 1, 3, 9.4-7 Sh 1, 2, 9.4-20 Sh 4

System: Control Building Ventilation (HVC), Containment Purge (CPS)

Title of Change: Logic Changes for Control Building Ventilation Fans

Description of Change:

This modification changed the control circuits of the following pairs of redundant fans: Main Control Room (2HVC*ACU1A/B), Relay Room (2HVC*ACU2A/B), Battery Rooms (2HVC*FN4A/B), Control Room Special Filter Trains (2HVC*FN2A/B), and Control Building Corridors & Adjoining Spaces (2HVC*FN11A/B) to preclude parallel operation. Specifically, the start times for the redundant fans other than 2HVC*FN2A/B were staggered through the use of time delay relays. The new logic for 2HVC*ACU1A/B and 2HVC*ACU2A/B includes a permissive from the other fan to ensure that the second fan does not start until the first one is stopped. In this manner, parallel operation of these units will be precluded. The logic for Control Room special filter train booster fans 2HVC*FN2A/B was modified to ensure that at least one fan will start in the time required for Control Room envelope pressurization upon receipt of an initiation signal (LOCA or high radiation).

In addition, the maximum stroke time for valves 2CPS*SOV119, 120, 121 and 122 were changed from 5 to 2 seconds to more appropriately reflect the standard fast-acting valve stroke time acceptance criteria.

Safety Evaluation No.: 98-067 Rev. 0, 1 & 2 (cont'd.)

Safety Evaluation Summary:

The analysis for this change concluded that the HVC system will perform its design function. Particularly, the Control Room envelope pressure and temperature will be maintained at all times when required by Technical Specification 3/4.7.3. The battery rooms and the corridor and adjoining spaces will continue to be provided with ventilating air.

Faster stroke times are more conservative for containment isolation valves, and the IST Program Plan will reflect the appropriate time. Since special booster fan operation is prompted by a radiological event that is not postulated concurrent with a chemical or toxic gas spill, and parallel ACU will be precluded by design even in the event of a single failure, outside air intake will not exceed 1500 cfm and the existing analysis remains bounded.

Physical and electrical separation requirements of independent and redundant electrical and control systems are being met. Analyses of the diesel loading sequence and UPS/battery loading profiles reveal that these systems will not be loaded beyond their design capabilities as a result of this proposed change.

Revision 1 and 2 provided additional consideration regarding conformance to 10CFR50 Appendix R requirements.

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

Safety Evaluation No.: 98-068

Implementation Document No.: BWRVIP-01

USAR Affected Pages: 3.9B-68

System: Main Steam (MSS)

Title of Change: Core Shroud Weld Cracking

Description of Change:

The inspection of the core shroud welds identified intergranular stress corrosion cracking of the horizontal welds. The inspections revealed fairly significant cracking of welds H4, H5, and H7, and relatively minor or no cracking of welds H1, H2, H3, H6 and H8. No evidence of cracking was noted on the vertical welds inspected.

Safety Evaluation Summary:

The NMP2 shroud meets the requirements of General Design Criterion (GDC) 1 and 10CFR50.55a with respect to designing the shroud to quality standards commensurate with the importance of the safety functions to be performed. BWRVIP-01, Rev. 1, describes the flaw evaluation methodology suggested for use in evaluating cracking in core shrouds. This document has been reviewed and approved by the NRC. The methodologies used in evaluating the core shroud weld cracking is consistent with ASME Section XI and the BWRVIP-01/NRC approved methodology. All welds meet the structural margin requirements, considering 17,000 hours of operation before the next required inspection. The NMP2 shroud meets the requirements of GDC 2, 4 and 10 with respect to designing components important to safety to withstand the effects of earthquakes and the effects of normal operation, maintenance, testing and postulated LOCA, with sufficient margin to assure that the capability to perform its safety functions is maintained and the specified acceptable fuel design limits are not exceeded. The alignment of the fuel and core spray system spargers is maintained such that the design basis required coolable geometry is maintained. This alignment is assured by maintaining the structural integrity of the core shroud welds. The shroud weld integrity assures that the deflections and deformation is limited such that the control rods and the emergency core cooling systems (core spray systems) can perform their safety functions during anticipated operational occurrences and accidents. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.: 98-069

Implementation Document No.: DDC 2M11382

USAR Affected Pages: Figures 5.4-16a, 9.3-9f

System: Reactor Water Cleanup (WCS), Drywell Equipment Drains (DER), Reactor Recirculation (RCS)

Title of Change: Modification of WCS Drain Lines to Support Alternate Draining Paths

Description of Change:

The termination points of two Non-ASME, Class 4 (Q) drain lines off the RCS system (via WCS) have been changed from drywell equipment drain funnels to the open space of containment just above the drywell floor. This allows the option of draining the RCS during maintenance/refueling outages to either the floor drains, equipment drains or the suppression pool without the limitation of flow rate that the funnel imparts. Future use of the drains involves temporary hose, as with any other maintenance drain, to direct drain flow to the chosen area.

Safety Evaluation Summary:

A review of the USAR reveals the accidents potentially associated with this change are the loss-of-coolant accident (small break) and instrument line break. The proposed change has no impact on the design and ASME qualification of the reactor coolant pressure boundary (RCPB). Thus, the proposed change will have no impact on the potential for line breaks in the RCPB. This drain path is used only when the plant is in a safe shutdown condition. All of the normal applicable Technical Specifications and procedural requirements for RCS draining will be in effect when this drain path is used. Procedural control of the existing valves will ensure there is not an uncontrolled draining of the RCS system or the reactor vessel. As such, the reroute of this drain path does not impact or increase the potential to drain the reactor vessel. Therefore, this change does not increase the probability of occurrence of an accident previously evaluated in the USAR. The new configuration does not adversely impact the ASME or seismic qualification of the Class 1 WCS piping or the Class 4 drain piping.

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

Safety Evaluation No.: 98-071

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

USAR Affected Pages: N/A

System: Turbine Building Closed Loop Cooling (CCS)

Title of Change: Freeze Seal for CCS Return from 2GMC-E1B

Description of Change:

During refueling outage RF06, a freeze seal was required in the 8" NPS Sch 40 return line of CCS return piping from generator stator water cooler 2GMC-E1B. The freeze seal provided closure isolation between generator stator water cooler inlet valve 2CCS-V46B and the common return line. The application of the freeze seal allowed the water cooler piping to be drained in order to repair generator stator water cooler outlet valve 2CCS-V48B.

The freeze seal was located in the generator stator water cooler outlet line 2CCS-008-172-4 at about el. 260 ft. in the Turbine Building. Freeze Seal Procedure S-MMP-GEN-014 controlled all freeze seal activities.

Safety Evaluation Summary:

Contingency Plan and Recovery Plan Criteria in the Freeze Seal Data Sheets describes actions which are to be taken to minimize or preclude those conditions which may result in the loss of freeze seal, protection of personnel and plant equipment, or damage to the piping system after the freeze seal is thawed. Dimensional measurements of the piping and liquid penetrant examinations before and after freeze seal activities ensures that piping pressure-retaining and structural integrity is maintained.

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

Safety Evaluation No.: 98-072 Rev. 0 & 1

Implementation Document No.: Design Change N2-98-008

USAR Affected Pages: N/A

System: Control Building Ventilation (HVC)

Title of Change: Field Installation of Logic Changes for Control Building Ventilation Fans

Description of Change:

The HVC system provides ventilation and conditioned air and, as necessary, space pressurization and filtration of airborne radioactive materials with the following pairs of redundant fans: Main Control Room (2HVC*ACU1A/B), Relay Room (2HVC*ACU2A/B), Battery Rooms (2HVC*FN4A/B), Control Room Special Filter Trains (2HVC*FN2A/B), and Control Building Corridors & Adjoining Spaces (2HVC*FN11A/B).

Revision 1 reflects a change in the start logic for fans 2HVC*FN2A/B from a staggered start design to a dual start configuration.

During surveillance testing of the Division I Diesel Generator, both divisions of Control Room air conditioning units (ACU) started due to the response characteristics of the flow switches that start the standby fan. As a result, the fans oscillated (surged) causing considerable flexing and noise in the ductwork. In addition, the low flow trip point for one fan was reached and the fan tripped. Since this trip causes the ACU to lock out, a postulated single failure of the other unit would render the Control Room without pressurization until manual action is taken to start one fan. A similar potential existed for all of the above-listed fans.

This modification changed the control circuits of the above-listed fans, except 2HVC*FN2A/B, to preclude parallel operation. Specifically, the start times for the ACUs and fans 2HVC*FN4A/B and 2HVC*FN11A/B were staggered through the use of time delay relays. The new logic for 2HVC*ACU1A/B and 2HVC*ACU2A/B includes a permissive from the other fan to ensure that the second fan does not start until the first one is stopped. In this manner, parallel operation of these units will be precluded. The logic for fans 2HVC*FN2A/B was changed to start both fans on a loss of offsite power (LOOP), LOOP/LOCA or high radiation initiation. This design minimizes the unfiltered air entering the Control Room envelope during radiological releases. Recent testing verified that the fans can be operated in parallel for prolonged durations with no impact to the fans, ductwork or filters. The Control Room habitability analysis has been updated to accommodate the

Safety Evaluation No.: 98-072 Rev. 0 & 1 (cont'd.)

Description of Change: (cont'd.)

increased outside air makeup and timing changes. Based on a review of the remaining fans, parallel operation is not desired but can be tolerated for short durations. Thus, these fans have been staggered, but parallel operation is not precluded in the event of extreme setpoint uncertainties or a postulated single failure.

Safety Evaluation Summary:

The analysis for this change concluded that the HVC system will perform its design function. Particularly, the Control Room envelope pressure and temperature will be maintained at all times when required by Technical Specification 3/4.7.3. The battery rooms and the corridor and adjoining spaces will continue to be provided with ventilating air.

Faster stroke times are more conservative for containment isolation valves and the IST Program Plan will reflect the appropriate time. Since special booster fan operation is prompted by a radiological event that is not postulated concurrent with a chemical or toxic gas spill, and parallel ACU will be precluded by design even in the event of a single failure, outside air intake will not exceed 1500 cfm and the existing analysis remains bounded.

Physical and electrical separation requirements of independent and redundant electrical and control systems are being met. Analyses of the diesel loading sequence and UPS/battery loading profiles reveal that these systems will not be loaded beyond their design capabilities as a result of this proposed change.

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

Safety Evaluation No.: 98-073

Implementation Document No.: DDC 2M11403

USAR Affected Pages: Figure 5.4-9a

System: Reactor Core Isolation Cooling (ICS)

Title of Change: ½" Test Line Configuration Change

Description of Change:

This change replaced the small bore test line valve, 2ICS*V225, and threaded cap mounted on the 12" ICS turbine exhaust line with a welded cap. The ICS test line is located upstream of the outboard containment isolation valve. Additionally, a threaded cap was replaced with a safety-related welded cap.

These changes were required to offset the impact of the higher weight of the test line valve so that structural integrity and pressure boundary of the test line and the exhaust line are maintained.

Safety Evaluation Summary:

The proposed change to the test line ensures that the piping stresses are within the allowable limits of ASME Code Section III. Therefore, structural integrity and pressure boundary of the turbine exhaust steam line, the test line, and affected components is restored to a level that is acceptable for the safe operation of the system and plant. In addition, the change does not impact containment isolation capability, leak rate testing capability, primary containment leak rate, ICS system operation, etc.

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

Safety Evaluation No.: 98-074

Implementation Document No.: Design Change N2-98-011

USAR Affected Pages: 6.2-67; Tables 8.3-1 Sh 3, 4, 8.3-5 Sh 1, 3, 5

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

Title of Change: Emergency Ventilation Unit Coolers 2HVR*413A Auto Start Timer Modification

Description of Change:

This design change increased the time delay setting of the start relay for unit cooler 2HVR*UC413A from 25 seconds to 50 seconds. The secondary containment drawdown analysis was revised to reflect the change (SGTS operation is delayed from 60 seconds to 90 seconds). This modification ensures that emergency recirculation unit cooler 2HVR*UC413A will not start unless lead unit cooler 2HVR*UC413B cannot meet the system air flow requirements.

Safety Evaluation Summary:

The proposed change to increase the time delay of 2HVR*UC413A will not impact the diesel generator 2EGS*EG1 load transient sequence or increase its electrical load. Further, the 30-second delay will not impact the secondary containment drawdown time requirement of less than 60 minutes.

Based on the analysis, the proposed change would not adversely impact the 2EGS*EG1 loading, secondary containment drawdown analysis, or the HVR system to perform its design function.

The constructability aspects of this change have been reviewed, and appropriate work sequencing instructions will be included with the applicable work and design documents.

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