Niagara Mohawk

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October 31, 2001
NMP1L 1621

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

10 C.F.R. §50.71(e)
10 C.F.R. §50.54(a)(3)
10 C.F.R. §50.59(d)(2)

RE: Nine Mile Point Unit 1
Docket No. 50-220
DPR-63
TAC No. MB3109

Subject: ***Submittal of Revision 17 to the Nine Mile Point Nuclear Station Unit 1 Final Safety Analysis Report (Updated), Including Changes to the Quality Assurance Program Description, and the 10 C.F.R. §50.59 Evaluation Summary Report***

Gentlemen:

Pursuant to the requirements of 10 C.F.R. §50.71(e), 10 C.F.R. §50.54(a)(3), and 10 C.F.R. §50.59(d)(2), Niagara Mohawk Power Corporation hereby submits Revision 17 to the Nine Mile Point Nuclear Station Unit 1 Final Safety Analysis Report (Updated), including changes to the Niagara Mohawk Power Corporation Quality Assurance Topical Report, and the 10 C.F.R. §50.59 Evaluation Summary Report.

One (1) signed original and ten (10) copies of the Unit 1 FSAR (Updated), Revision 17, are enclosed. Copies are also being sent directly to the Regional Administrator, Region I, and the Senior Resident Inspector at Nine Mile Point. The Unit 1 FSAR (Updated) revision contains changes made since the submittal of Revision 16 in November 1999. The revision reflects all changes up to and including May 8, 2001. The certification required by 10 C.F.R. §50.71(e) is attached.

Enclosure A provides the identification, reason, and basis for each change to the quality assurance program description, Unit 1 FSAR (Updated) Appendix B, in accordance with 10 C.F.R. §50.54(a)(3)(ii).

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The enclosed 10 C.F.R. §50.59 Evaluation Summary Report (Enclosure B) contains brief descriptions of changes to the facility design, procedures, tests, and experiments. None of the 10 C.F.R. §50.59 Evaluations involved obtaining a license amendment as defined in 10 C.F.R. §50.59(c)(1).

Very truly yours,

Richard B. Abbott

Richard B. Abbott
Vice President Nuclear Engineering

RBA/LWB/cld
Enclosures

cc: Mr. H. J. Miller, NRC Regional Administrator, Region I
Mr. G. K. Hunegs, NRC Senior Resident Inspector
Mr. P. S. Tam, Senior Project Manager, NRR (2 copies)
Records Management

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Niagara Mohawk Power Corporation	)	Docket No. 50-220
	)	
(Nine Mile Point Unit 1)	)	

CERTIFICATION

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

Richard B Abbott

Richard B. Abbott
Vice President Nuclear Engineering

Subscribed and sworn to before me, a Notary Public in and for the State of New York

and County of Oswego, this 31st day of October, 2001.

Notary Public in and for

Oswego County, New York

My Commission Expires:
8/9/2005 Lisa M. Clark

LISA M. CLARK
Notary Public in the State of New York
Oswego County Reg. No. 01CL6029220
My Commission Expires 8/9/2005

Enclosure A to
NMP1L 1621

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

ENCLOSURE A

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

UFSAR Appendix B Page/Section	Identification of Change	Reason for Change	Basis for Concluding that the Revised Program Continues to Satisfy 10CFR50 Appendix B and Commitments Previously Approved by the NRC
Page B-ii, List of Tables	Deleted Table B-3.	The information in Tables B-2 and B-3 has been incorporated into the revised Table B-2.	This change is strictly editorial and has no material impact on features of the QA plan or its implementation.
Page B.0-1, Before Introduction	Added two paragraphs to explain the method of using generic position titles within the body of the QA plan, and to explain the process of implementing changes to the plan to incorporate generic position titles.	Implement the use of generic position titles within the QA plan, to reduce the number of future plan changes necessary for nominal organizational changes.	The change to the QA plan to use generic position titles is specifically identified in the applicable regulation [10CFR50.54(a)(3)(iii)] as a change that does not reduce QA Program effectiveness and does not require prior NRC approval.
Page B.0-2, Section B.0 Last paragraph Page B.1-1, Section B.1.2.1 Page B.1-4, Section B.1.2.1.1, Item 5. Page B.1-5, Section B.1.2.1.1, Items a. through d. Page B.2-5, Section B.2.2.15, Item 2. Page B.2-10, Section B.2.2.17, Item 2.	"Manager Quality Assurance" was changed to "manager quality assurance."	Implement the use of generic position title of manager quality assurance.	The change to the QA plan to use generic position titles is specifically identified in the applicable regulation [10CFR50.54(a)(3)(iii)] as a change that does not reduce QA Program effectiveness and does not require prior NRC approval.
Page B.1-1, Section B.1.1, Third paragraph Page B.1-3, Section B.1.2.1.1, Item 3.	Moved cross-reference to Table B-1 to a new location and eliminated unnecessary cross-reference to the UFSAR and USAR.	Other organizational changes within the QA plan make revised references more appropriate.	These changes are strictly editorial and have no material impact on features of the QA plan or its implementation.

Page B.1-2, Section B.1.2.1.1 First paragraph Second Paragraph Page B.1-2, Section B.1.2.1.1, Item 1. Page B.1-3, Section B.1.2.1.1, After Item g. Item 2. Item 3. Page B.1-4, Section B.1.2.1.1, Item 4.	Added "Quality Assurance" after "(NSAS)." Added "Quality Assurance" after "Engineering." Added "has responsibility for Training/Emergency Preparedness, and" after "Generation" in the second sentence, and added "Manager Training" after "Plant Managers" in the third sentence. Added "Manager Training" after "Plant Managers." Added "and is responsible for Engineering, Licensing, and Security" after "Officer." Deleted "Quality Assurance, Licensing, Training/Emergency Preparedness, Security, and" after "responsible for." Changed existing item 4. to item 5. Added new item 4. as follows: "The Vice President Quality Assurance - Nuclear reports to the Chief Nuclear Officer and is responsible for Quality Assurance. Quality Assurance responsibilities are described in the Unit 1 UFSAR Section XIII and Unit 2 USAR Section 13. See Table B-1 for QA Program element responsibilities. "	Organizational change - responsibilities and functions realigned. The principle organizational change is the creation of the position of Vice President Quality Assurance - Nuclear, reporting directly to the Chief Nuclear Officer. As a result of this change, other functions previously assigned to the NSAS organization have been reassigned to the Engineering and Generation organizations, and the position of Manager Quality Assurance has been eliminated. The generic title of manager quality assurance is being implemented contemporaneously with this organization change to minimize the impact of any future title changes that do not involve a change in functional responsibility.	The only change affecting the QA organization is the creation of the position of Vice President Quality Assurance – Nuclear and the elimination of the position of Manager Quality Assurance. The QA Program functional groups, previously under the Manager Quality Assurance, now report to the Chief Nuclear Officer directly, rather than through an intervening corporate executive. The other QATR changes resulting from the organizational change do not involve a change in QA Program content or requirements; therefore, none of these changes reflect a reduction in effectiveness.
Page B.1-4, Section B.1.2.1.1, Item 5. First paragraph Second paragraph	Deleted all of paragraph after "QA Program." Replaced "The Manager Quality Assurance's responsibilities" with "The responsibilities of the individual assigned the manager quality assurance's function."	Implement the use of generic position title of manager quality assurance.	The change to the QA plan to use generic position titles is specifically identified in the applicable regulation [10CFR50.54(a)(3)(iii)] as a change that does not reduce QA Program effectiveness and does not require prior NRC approval.
Page B.1-5, Section B.1.2.1.2	Changed Unit 1 UFSAR reference to Section XIII.A.1.	Correction of previous typographical error; i.e., omission of "X."	This change is strictly editorial and has no material impact on features of the QA plan or its implementation.

Page B.2-1, Section B.2.2.2	Changed "at least annually" to "as required by 10CFR50.71(e)."	Revised to incorporate the requirement of the controlling regulation.	The regulations do not require certain changes to be submitted annually, but do require these changes to be submitted with the FSAR update submittal. This change makes the QATR consistent with current regulations, does not affect implementation of the QA Program, and does not constitute a reduction in effectiveness.
Page B.2-2, Section B.2.2.5	Replaced "Manager Quality Assurance" with "individual assigned the manager quality assurance function."	Implements the use of generic position title of manager quality assurance.	The change to the QA plan to use generic position titles is specifically identified in the applicable regulation [10CFR50.54(a)(3)(iii)] as a change that does not reduce QA Program effectiveness and does not require prior NRC approval.
Page B.2-3, Section B.2.2.10	Deleted reference to Table B-3.	The information in Tables B-2 and B-3 has been incorporated into the revised Table B-2.	This change is strictly editorial and has no material impact on features of the QA plan or its implementation.
Page B.2-12, Section B.2.2.18	Replaced "Quality First Program (Q1P)" with "Employee Concerns Program."	Program name change.	There are no changes to the program described in this section of the QATR, other than the change in program title. The program title change does not affect the QA Program content or requirements and, therefore, does not constitute a reduction in effectiveness.
Page B.3-1, Section B.3.2.4	Deleted last sentence and replaced with "An acceptable method of design verification for materials, parts, and processes is the use of qualification testing."	Clarification.	This change makes the QATR consistent with applicable QA standards, and clarifies that qualification testing is an acceptable method of design verification, rather than an alternative to design verification. This change does not affect the QA Program content or requirements and, therefore, does not constitute a reduction in effectiveness.
Page B.4-1, Section B.4.2.5	Added "and/or service(s)" after "item(s)" in first sentence.	Clarification.	This change makes this subsection consistent with the remainder of Section B.4, which recognizes that procurement requirements apply to both items and services. This change does not affect the QA Program content or requirements and, therefore, does not constitute a reduction in effectiveness.

Page B.6-1, Section B.6.1, Second paragraph	After "controlled documents," added the following: ", other than those identified as minor changes,". At end of paragraph, added: "The criteria for establishing and defining minor changes are provided in approved implementing documents such as directives, procedures, and instructions."	Clarification.	This change clarifies that minor changes, as defined by appropriate documents, do not need to be approved by the original review and approval organization. This is consistent with the intent of the last portion of the original wording; i.e., "or by the organizations designated in accordance with the procedures governing these documents." This change does not affect the QA Program content or requirements and, therefore, does not constitute a reduction in effectiveness.
Page B.11-1, Section B.11.1	Replaced "These parameters" with "Requirements and acceptance criteria."	Clarification.	The existing phrasing does not make logical sense as no 'parameters' are previously identified. The revised wording more clearly explains the intent of this section, and is consistent with the remaining context of this section. This change does not affect the QA Program content or requirements and, therefore, does not constitute a reduction in effectiveness.
Page B.11-2, Section B.11.2.5, Item 5.	Inserted the word "to" after "as."	Correction of previous typographical error; i.e., omission of "to."	This change is strictly editorial and has no material impact on features of the QA plan or its implementation.
Table B-1, Sheets 1 and 2 Table B-1, Sheet 2	Deleted columns for "VP-NSAS," "NL" and "NT." Added an "X" to Column "NG" for row "V" on Sheet 1. Deleted footnote 2. and revised footnote titled "NMPC Organizations" to reflect the organization described in the revised text of Section B.1.2.1.1.	Organizational change – responsibilities and functions realigned.	NSAS is no longer listed because its only responsibility is for ISEG, and it has no responsibilities relative to the areas addressed in this table. The other changes to the table reflect the new alignment of responsibilities resulting from the organizational changes identified in Section B.1.2.1.1. No previously existing QA Program responsibilities have been changed or eliminated; therefore, there is no reduction of effectiveness.
Table B-2, Sheets 1 through 9	Every line item and all material information for the previous Table B-3 has been incorporated into the revised Table B-2.	Improve ability to locate and understand the information contained within the previously existing tables.	Every line item and all material information for the previous Table B-3 have been incorporated into the revised Table B-2. The line items describing interpretations and exceptions are now grouped immediately following the applicable regulatory commitment reference document. Additional internal cross-references have been provided within the table to assist users in identifying appropriate information for related topics. A footnote has been added to direct users to appropriate FSAR sections for those portions of the table involving procedures

			and/or procedural control processes. No previous regulatory commitment, interpretation, or exception has been eliminated or materially modified. All changes were made to improve the ability of users to locate and understand the information contained within the previously existing tables; therefore, this change does not constitute a reduction in the effectiveness of the QA Program.

Enclosure B to
NMP1L 1621

NINE MILE POINT – UNIT 1

10CFR50.59 EVALUATION SUMMARY REPORT

2001

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

Safety Evaluation No.: 96-022

Implementation Document No.: Mod. N1-95-007

UFSAR Affected Pages: IV-5, IV-14, VIII-8, VIII-16, VIII-20, VIII-25, VIII-26, XV-9, XV-82; Table XV-2; Figure VIII-14

System: Neutron Monitoring

Title of Change: Thermal Hydraulic Stability

Description of Change:

This modification installed new average power range monitor (APRM) flow control trip reference (FCTR) cards in place of the previous APRM trip bias units. Also, this change:

- Revised the APRM flow-biased neutron flux scram and control rod block to provide automatic protection to assure that coupled neutronic/thermal-hydraulic instabilities can not compromise established fuel safety limits;

- Evaluated the APRM flow-biased neutron flux scram and control rod block to provide an increase above their current values in the high flow, high power region of the power/flow operating map; and

  NOTE: These changes were addressed in License Amendment 168, dated October 15, 1999.

- Eliminated the first and second level control rod block alarms.

Safety Evaluation Summary:

An assessment was performed of the impacts of the change on the UFSAR Section XV transient events results, the cycle-specific reload analysis, and criteria applicable to thermal-hydraulic instabilities. The assessment results indicate that:

- Revising the setpoints in the lower flow regions of the power/flow operating map results in more conservative setpoints than the existing setpoints and will suppress postulated reactor instabilities;

Safety Evaluation No.: 96-022 (cont'd.)

Safety Evaluation Summary: (cont'd.)

- Inadvertently selecting increased flow-biased APRM setpoints in the high flow, high power region of the power/flow operating map has negligible impact on the transient events results, and does not change the margin of safety; and

- Eliminating the first and second level control rod block alarms has no impact on transient events results.

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

Safety Evaluation No.: 98-003

Implementation Document No.: ASME Section XI, 1983 with Summer 1983 Addenda, Para. IWA-7210

UFSAR Affected Pages: XVI-60a

System: Various

Title of Change: Update Nine Mile Point Unit 1 UFSAR Regarding Code Reconciliation - LDCR 1-98-UFS-003

Description of Change:

This change updated the UFSAR to address the fact that system modifications, repairs, and replacements may have been performed in accordance with different codes and standards which have been reconciled against the original code. This safety evaluation reviewed the practice whereby modifications, repairs, replacements, and other work may be done on safety-related and nonsafety-related systems, and the work may be done in accordance with later editions of the original codes, or different codes altogether which have been duly reconciled against the original code, ensuring system integrity and quality standards as good as or better than the original code.

Safety Evaluation Summary:

Performing code reconciliation, and performing work or procuring equipment designed and built in accordance with more recent codes and standards, ensures that structures, systems, and components are designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the original code requirements, and commensurate with the importance of the safety function to be performed. The NRC has endorsed the use of ASME XI for repair and replacement which specifies the need for code reconciliation.

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

Safety Evaluation No.: 98-007

Implementation Document No.: DCR N1-93-900LG036

UFSAR Affected Pages: VIII-9, XV-19; Table VIII-4 Sh 1 of 3

System: Main Turbine

Title of Change: Setpoint Change for Turbine Vacuum Trip Incorporation into UFSAR

Description of Change:

This safety evaluation analyzed a change to the low-low condenser vacuum setpoint for closure of the turbine stop valves.

The setpoint has been changed from 20 inches of Hg vacuum to 22.1 inches of Hg vacuum decreasing. The change was recommended as part of GE TIL 772-2 to reduce stress on the last stages of the turbine buckets during periods of low condenser vacuum.

Safety Evaluation Summary:

This low-low condenser vacuum setpoint change will not adversely affect minimum critical power ratio limits due to the fact that the turbine will trip earlier and produce a reactor scram, which will terminate the transient. This is true for any turbine transient with a first-stage bowl pressure >365 psig. For transients with first-stage bowl pressures <365 psig, the safety limit critical power ratio limits will not be exceeded.

The selected setpoint is more conservative than the analytical value of 20 inches of Hg vacuum described in UFSAR Sections VIII and XV.

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

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

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

UFSAR Affected Pages: Figure III-1

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

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

Description of Change:

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

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

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

Safety Evaluation Summary:

A temporary modification will supply fuel oil to the day tank through the use of a 2710-gallon diesel truck, which will ensure that fuel oil is available to replace the removed underground tank at all times. The removal and replacement of the diesel fire pump underground tank and piping will be accomplished after the temporary modification is implemented; therefore, the function of the diesel fire pump and the fire water system will be unaffected. The vehicle fueling station underground tanks and equipment are outside the restricted area and are not connected to any plant systems. The analysis of the NMP2 probable maximum precipitation (PMP) flooding impact on the NMP site has been reviewed, and it has been determined that the installation of the concrete containment pad will have no impact on the NMP2 PMP analysis. Cutting of the tanks will be performed outside the protected area in an area away from overhead lines and safety-related buildings. Precautions will be implemented to ensure personnel safety and to eliminate the possibility of an explosion accident.

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

Safety Evaluation No.: 98-024

Implementation Document No.: LDCR No. 1-98-UFS-036

UFSAR Affected Pages: IX-4

System: KV115

Title of Change: 115-kV Transmission Line Malfunction Alarm Location Change due to Niagara Mohawk Control Center Consolidation

Description of Change:

On April 7, 1998, Niagara Mohawk announced that it would consolidate the operation of its Northern Regional Control Center (NRCC) in Watertown, NY, into an existing Central Regional Control Center (CRCC) located at the Energy Management System (EMS) facility on Henry Clay Boulevard, Liverpool, NY, effective August 1, 1998. Niagara Mohawk announced that it is technically capable of controlling North Country operations from its CRCC using the same procedures; i.e., giving control of distribution to the Line Department during an ice storm, and directing switching on transmission from the CRCC.

The UFSAR has been revised to reflect the revised facility reference.

Safety Evaluation Summary:

This change involves revising the description of a Niagara Mohawk EMS facility and location, as described in the NMP1 UFSAR, from NRCC in Watertown, NY, to CRCC in Liverpool, NY. Like the NRCC, the CRCC receives the same 115-kV malfunction alarm and can respond equally to that alarm. The 115-kV transmission line malfunction sensing mechanism, and the relay scheme at Lighthouse Hill which initiates an alarm, are unaffected by this change. The change to the UFSAR resulting from the consolidation of operations does not adversely affect any failure mode of the 115-kV reserve bus and 4.16-kV power boards evaluated in the failure modes and effects analysis (FMEA).

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

Safety Evaluation No.: 99-005

Implementation Document No.: Mod. N1-98-008

UFSAR Affected Pages: V-6, IX-17, X-4, X-5, X-51 thru 56, 10A-111, 10A-111a, 10A-121, XI-6, XI-9, XI-11, XI-11a; Figures X-2, XI-3, XI-5, XI-7

System: Hydrogen Water Chemistry (HWC)

Title of Change: Hydrogen Water Chemistry Injection System

Description of Change:

This modification added a permanent HWC system at NMP1. The HWC program reduces rates of intergranular stress corrosion cracking in the recirculation piping and reactor vessel internals. The HWC system includes the flow monitoring and control equipment for both hydrogen and oxygen, system control panels, and an offgas oxygen monitor panel. The Control Room interface includes a shutdown switch, annunciation and status lights.

Gaseous hydrogen was injected into the feedwater booster pump suction. A series of manual maintenance valves and a companion check valve were installed near the hydrogen injection point at the feedwater booster pumps. An emergency shutoff valve was installed at the hydrogen piping entrance to the Offgas (OFG) Building to allow automatic isolation of the hydrogen supply within the OFG Building and Turbine Building. Combustible gas monitors were installed in the vicinity of mechanical joints to monitor for hydrogen leakage and initiate isolation of the supply upon detection. The addition of hydrogen reduces oxygen production in the reactor core, resulting in an offgas stream that does not contain sufficient oxygen to recombine all of the hydrogen. To balance the gas mixture, the HWC system adds oxygen upstream of the recombiners. The bulk supply piping has been routed through the OFG Building and Turbine Building to a new oxygen injection panel inside the Turbine Building. Gaseous oxygen has been injected downstream of the steam jet air ejector in the offgas train. New oxygen analyzers were installed in the OFG system to ensure that oxygen and hydrogen flows are properly balanced, and that satisfactory recombination has taken place. After the addition of the oxygen, the OFG system recombiner operates at essentially the same gas loading as encountered during normal water chemistry. The oxygen supply has also been tied into the existing feedwater/condensate oxygen injection system piping. This provides an alternate bulk supply to replace

Safety Evaluation No.: 99-005 (cont'd.)

Description of Change: (cont'd.)

or supplement the oxygen bottle located on El. 261 of the Turbine Building. Instrument air will be used for system control and valve actuation.

Safety Evaluation Summary:

This safety evaluation evaluated operation of the HWC system only up to injection levels that do not impact the steam area radiation levels. In this manner, there will be no impact on occupational dose rates, or on the main steam line radiation monitors, condenser air removal radiation monitors, or any area radiation monitors. Based on system startup testing, the hydrogen injection rate will be adjusted to achieve differing electrochemical corrosion potential (ECP) levels, with the restriction that the adjusted injection rate does not impact steam dose rates. Injection rates will be administratively controlled thereafter to maintain them at these established levels. If it is necessary to increase injection rates beyond this (up to the system design flow rate), further evaluations and a revised or new safety evaluation will be required.

In addition to HWC system design and operation, this safety evaluation evaluated the physical installation of the HWC system equipment, piping and controls located inside the plant. The installation of the common gas supply facilities and associated aboveground and underground exterior piping, including analysis, will be addressed in a separate safety evaluation prior to operating the system. The impact of the compounding effect of Noble Metal Chemistry Application (NMCA) on plant dose rates and associated aspects will be addressed by a separate safety evaluation.

Based on a review of the HWC system design, in accordance with EPRI Report NP-5283-SR-A (and associated NRC Safety Evaluation) and administrative controls specified for system operation, HWC operation will not adversely impact any materials or dose rates (normal or accident), will not adversely impact the function of any systems (including condensate, feedwater/HPCI and offgas), and will not increase the potential for a hydrogen explosion.

Based on the evaluation performed, it is concluded that adding a HWC system, and operating it up to injection rates that do not increase plant operating doses, does not involve an unreviewed safety question.

Safety Evaluation No.: 99-014

Implementation Document No.: P&ID C18030C, Sh 5

UFSAR Affected Pages: 10A-69

System: Fire Protection Water

Title of Change: NMP1 Track Bay Fire Protection Water Deluge Block Valve

Description of Change:

The UFSAR describes the manually-initiated deluge water spray system that provides fire suppression capability to the Turbine Building Track Bay, and includes a description of the manual shutoff valve provided to isolate the hazard. The UFSAR was in error because the physical location of the manual shutoff valve is not between the deluge valve and the nozzles, but upstream of the deluge valve. Additionally, the UFSAR description stated that the manual blocking valve is kept closed "to ensure that only a deliberate action will result in the system's operation." The normal position of the manual blocking valve to the Turbine Building Track Bay water deluge system was changed from normally closed to normally open, and the UFSAR description has been updated to reflect this change.

Safety Evaluation Summary:

The UFSAR description is in error since the physical location of the manual shutoff valve is not between the deluge valve and the nozzles, but upstream of the deluge valve. The physical plant location is appropriate, in accordance with the plant design documentation and with normal design practice. Placing the deluge system into remote manual operation is acceptable for the following reasons: 1) There is no safety-related equipment in or around the Turbine Building Track Bay; 2) The deluge valve is a Viking preaction/deluge valve that does not utilize a mechanical latch mechanism; the valve is a diaphragm type with sufficient area difference between the over-the-diaphragm and under-the-diaphragm areas that the line pressure fluctuations will not cause inadvertent tripping; 3) The control circuit for the deluge valve is designed such that a single open or a ground in the control circuit will not cause an inadvertent deluge; 4) The deluge pilot valve is motor operated and it fails "as is" with a loss of power. The deluge sprinkler system heads are directed towards the center of the Track Bay; and 5) The motor generator sets and control cabinets have been designed to be drip-proof. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.: 99-016

Implementation Document No.: Mod. N1-97-039

UFSAR Affected Pages: IX-22, IX-23, IX-24

System: 125VDC

Title of Change: Battery 14 Replacement

Description of Change:

This modification changed the Battery 14 configuration. Battery 14 consisted of two parallel connected quality-related (Q-related) 125V DC station batteries (14B-1 and 14B-2). These Q-related 125V DC station battery banks consisted of 60 C&D LCR-21 lead calcium cells, each with an eight-hour discharge rating of 1500 ampere-hours. These two parallel connected battery banks have been replaced with one bank of new battery cells. The replacement battery cells are C&D XT1LP-33 lead calcium cells with an eight-hour discharge rating of 1552 ampere-hours.

The Q-related 125V DC station batteries supply power to several nonsafety-related loads. The major loads on the Q-related batteries are the turbine generator DC emergency bearing oil pump and the emergency hydrogen seal oil pump. Batteries 14B-1 and 14B-2 were installed by Modification N1-90-126 to restore margin to Battery 12.

Safety Evaluation Summary:

The new single battery will operate within the design and testing limits of the existing system. No new loads were added. Re-sizing of the existing battery feeder cables was not required. The new battery is within the design limits of the existing overcurrent protection for trip setting, coordination, and the short circuit rating. The new design utilizes a vendor-recommended replacement cell, as this model has a better positive plate material retention system which is considered more suitable for the vibration conditions which exist in the battery room. The calculation dispositions to the battery sizing calculations conclude that replacement cells are sufficient to support the Battery 14 load profiles.

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

Safety Evaluation No.: 99-017

Implementation Document No.: DDC 1M00729

UFSAR Affected Pages: 10A-79, XII-2, XII-6, XII-18, XII-19; Table XII-2 Sh 1; Figures III-4, III-12, 10A-4, 10A-4A thru 10A-4D, XII-1

System: Radwaste

Title of Change: Shower Facility

Description of Change:

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

Safety Evaluation Summary:

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

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

Safety Evaluation No.: 99-021

Implementation Document No.: Mod. N1-92-013

UFSAR Affected Pages: X-30, X-31, X-39, X-40, X-40a, X-51; Figures X-7, X-8

System: Fuel Pool Cooling Filtration & Drain

Title of Change: Spent Fuel Pool Rerack Project Installation

Description of Change:

Technical Specification Amendment 167 (dated 6/17/99) allows for 4086 spent fuel storage locations using the neutron absorber material Boral, with 1840 storage locations in the north half of the pool and 2246 locations in the south half. The non-poison racks (flux trap racks) in the north half of the pool were replaced with new Boral racks following the 1999 refuel outage. This safety evaluation evaluated the installation activities associated with removing the 8 flux trap racks in the north half of the spent fuel pool (SFP) and replacing them with 8 new Boral racks.

The SFP reracking installation activities occurred over a period of several months. The fuel stored in the pool was removed from an existing rack prior to removing the empty rack from the pool, and new racks were installed in its place. The spent fuel is now stored in both existing and new racks. Technical Specification Amendment 167, received for the addition of Boral racks to the SFP, provided for both flux trap and Boral racks in the north half of the SFP. The removal of the flux trap racks and installation of the Boral racks required the lifting of heavy loads (loads greater than 1000 lbs.), the removal of interferences within the SFP, and the placement of a work bridge over the SFP for personnel access.

In order to maintain the rack to wall gaps and maximize the storage capacity of the SFP, there are several SFP liner attachments which will be removed, along with a portion of the spent fuel pool cooling (SFPC) system sparger piping in the north half of the pool. The performance of these removal activities will require the use of underwater divers in the SFP.

Safety Evaluation No.: 99-021 (cont'd.)

Safety Evaluation Summary:

Technical Specification Amendment 167 provides for increasing the number of fuel assemblies stored in the SFP.

The requirements of NUREG-0612 will be met for the reracking activities, including the use of the Reactor Building 125-ton main hoist for lifting the new and existing racks, safe load paths, NUREG-0612 qualified lifting devices, and training for crane operators. During work in the fuel pool in which irradiated fuel is handled, the high range monitor on the refueling platform will be operating to ensure isolation of the reactor building ventilation system and initiation of reactor building emergency ventilation system. The SFP water temperature will be maintained below 120°F and a design basis temperature of 140°F with cooling flow into the SFP through only the south sparger. The fuel movements required for the removal of the existing racks are bounded by UFSAR Chapter XV, "Refueling Accident," analysis. The placement of a work bridge over the SFP and irradiated fuel, for personnel access, is acceptable per calculation.

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

Safety Evaluation No.: 99-023

Implementation Document No.: DDC 1M00776

UFSAR Affected Pages: Figure III-14

System: Control Room Air Treatment

Title of Change: Clarification of Control Room Air Treatment System Branch Air Flows

Description of Change:

The control room air treatment (CRAT) system branch air flow rates described on UFSAR Figure III-14 were revised to clarify their design basis. The 5604 cfm to the Main Control Room, and the 7739 cfm to the Auxiliary Control Room were revised to reflect that these values are minimum required flows. The 1156 cfm to the Instrument Shop was revised to 725 cfm minimum for personnel comfort per ASHRAE standards, and it was noted that this flow was not used in any design basis heat removal transients or heatup analyses.

Safety Evaluation Summary:

The CRAT system is not an initiator of any accidents previously evaluated in the UFSAR. The CRAT system is utilized in maintaining Control Room habitability during and after a design basis accident. The changes to the CRAT branch flow rates will not cause a change to any system interface in a way that would increase the likelihood of an accident.

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

Safety Evaluation No.: 99-030

Implementation Document No.: DDC 1S00424

UFSAR Affected Pages: III-27; Figure III-1

System: Sewage Treatment

Title of Change: Sewage Treatment Plant Sludge Drying Beds

Description of Change:

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

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

Water is drained from the sludge via the sand drain and the sun evaporates water from the sludge. The remaining product is a 70% to 80% dried sludge by weight. Its consistency is similar to sand and is suitable for shipping. The water will be pumped from the sump back to the Wastewater Treatment Plant.

Safety Evaluation Summary:

The addition of sludge drying beds at the Wastewater Treatment Plant does not require a Technical Specification change for either Unit 1 or Unit 2. The possible effects of the sludge drying beds on equipment important to safety are no different than those of the Wastewater Treatment Plant. The sludge drying beds have no connections with any plant system or component. The installation of the drying bed facility has no effect or impact on the probable maximum precipitation flooding analysis. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.: 99-032

Implementation Document No.: Calculation S0.0-FPE-001

UFSAR Affected Pages: 10A-106, 10A-111, 10A-120, 10A-122, 10A-126

System: Fire Protection

Title of Change: Combustible Loading for NMP1 Truck Bays and Track Bays

Description of Change:

Chapter 10A of the UFSAR provides for an allowance for transient combustible loading. The Fire Hazards Analysis (FHA) (in-situ and transient loading) did not account for the combustible loading associated with a truck and forklift in the following areas: Reactor Building Track Bay, Turbine Building Track Bay and Truck Bay, Waste Building Truck Bay, and the Administrative Building Storeroom Truck Dock and Radwaste Solidification and Storage Building (RSSB) Truck Loading Area North and West.

This safety evaluation evaluated a revision to the FHA combustible loading tables. The change added the combustibles associated with a truck and forklift in the truck and track bays at NMP1 to the permanent loading.

Safety Evaluation Summary:

The presence of trucks and forklifts in the Reactor Building Track Bay, Turbine Building Truck and Track Bays, Waste Building Truck Bay, Administration Building Storeroom Truck Dock, and RSSB Truck Loading Areas North and West does not exceed the capability of the barriers to contain a fire involving these vehicles. The locations are being utilized as intended and the fire protection features in areas that protect safe shutdown capability are not being compromised. All the locations analyzed were found to be acceptable.

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

Safety Evaluation No.: 99-033

Implementation Document No.: Procedures N1-ISP-002-003, N1-ISP-002-013

UFSAR Affected Pages: Figure VIII-2

System: Reactor Protection, Main Turbine

Title of Change: Main Turbine 1st Stage Bowl Pressure Scram Bypass Analytical Limit

Description of Change:

This safety evaluation evaluated changing the analytical limit for the turbine 1st stage bowl pressure scram bypass function. This change established an analytical limit upon which allowable values and instrument setpoints can be derived, with the assurance that Technical Specification limits for that function will not be exceeded.

Turbine 1st stage bowl pressure switches are used as a bypass in the turbine trip coincident logic input for reactor trip (scram) when the turbine 1st stage bowl pressure is below a pressure corresponding to 45% of reactor thermal power or 833 MWt (i.e., 45% of 1850 MWt). If the pressure switches sense pressure below this value, then continuity is maintained to keep the associated coincident logic relays energized, thereby bypassing a turbine trip input signal to the reactor trip circuit. When turbine 1st stage steam pressure is above this value, the reactor trip (scram) input is initiated from generator load rejection or turbine stop valve closure. The Technical Specification limit of 833 MWt is based on maintaining the minimum critical power ratio (MCPR) above the SLCPR even with failure to bypass steam to the condenser. A reactor scram is required when at or above 45% power.

The pressure corresponding to 45% of reactor thermal power, as calculated, is 344 psig. This pressure is, therefore, the analytical limit on which the setpoint for pressure switches PS-02-13A through 13D is derived. Sufficient margin below the analytical limit of 344 psig is calculated for the pressure switch setpoint to allow for instrument uncertainties, ensuring the Technical Specification limit for function bypass is not exceeded. The calculated setpoint is established as 311 psig. The actual instrument setpoint in the field may be below this value, providing increased conservatism and providing consistency

Safety Evaluation No.: 99-033 (cont'd.)

Description of Change: (cont'd.)

with the setpoint used in the past. Since the current setpoint of 310 psig meets the requirements of the Technical Specification, a setpoint change does not result from this analytical limit change.

Safety Evaluation Summary:

Decreasing the turbine 1st stage bowl pressure scram bypass function analytical limit from 365 psig to 344 psig implements the Technical Specification requirement. Previously accepted values for the analytical limit were shown not to represent the Technical Specification requirement of 45% reactor thermal power (833 MWt). Use of this value to derive the appropriate instrument setpoint for the pressure switch will ensure that instrument uncertainties will not create a bypass of the scram above that prescribed by the Technical Specification.

Changing the analytical limit for the pressure switches will only have the potential for decreasing the pressure band for which bypass of the scram function is possible. It will have no bearing on the operation or functionality of the limit switches. The turbine 1st stage pressure switches serve to maintain continuity of the logic signal for a pressure band prescribed by the Technical Specification, without affecting how the rest of the logic functions. Therefore, this change will have no effect on the scram instrumentation function. Since the new analytical limit is less than the value that has been previously cited, it is more conservative.

Based on the evaluation performed, it is concluded that the change of the analytical limit for the turbine 1st stage bowl pressure scram bypass function does not involve an unreviewed safety question.

Safety Evaluation No.: 99-037

Implementation Document No.: Mod. N1-98-008

UFSAR Affected Pages: V-6, IX-17, X-4, X-5, X-51 thru X-56, 10A-111, 10A-111a, 10A-121, XI-6, XI-9, XI-11, XI-11a; Figures XI-3, XI-5, XI-7

System: Reactor Vessel, Reactor Water Cleanup

Title of Change: HWC and Noble Metal Chemistry Monitoring

Description of Change:

The design change for hydrogen water chemistry (HWC) and noble chemical addition (NobleChem) includes monitoring of the effectiveness for intergranular stress corrosion cracking mitigation and longevity of the noble metal coating. This safety evaluation evaluated the monitoring components, which include: two electrochemical corrosion potential (ECP) monitoring locations, material coupons (durability monitor) for monitoring the noble chemistry coating density, a R&D (MPM) skid for crack growth rate monitoring and ECP monitoring, and a reactor coolant sample supply to the General Electric Noble Metal Chemistry application temporary sampling equipment.

Reactor coolant supply to the monitoring systems is supplied from small bore piping (3/4-in.) tie-in connections installed from the reactor water cleanup (RWCU) 6-in. line, downstream of the outboard RWCU isolation valve and upstream of the RWCU regenerative heat exchanger. The 3/4-in. piping terminates in the spent fuel pool heat exchanger room, and a transition is made to 3/4-in. stainless steel tubing for supply to the ECP/durability/MPM-R&D crack growth monitoring stations. The flow is returned to the RWCU system using the existing hydrogen water chemistry stress corrosion test facility reactor water H_2 chemistry equipment cooler (110-164), and return flow path and components through PCV 110-163 and blocking valve BV 110-167. The NobleChem injection equipment and the ECP and crack growth monitoring systems requiring power are provided from two new transformers and associated distribution panels. These new distribution transformers are supplied from existing nonsafety-related 600V PB-155 and 480V PB-H10.

Safety Evaluation No.: 99-037 (cont'd.)

Safety Evaluation Summary:

The HWC/NobleChem monitoring system will monitor the effectiveness of the HWC and noble metal mitigation measures. The system consists of small bore 3/4-in. and 1/2-in. piping and tubing connections to the RWCU system. The piping is less than 1 in. in diameter and, therefore, pipe breaks do not need to be postulated. The sample piping diverts reactor coolant from a nonsafety-related portion of the RWCU system outside of the system isolation valves and, therefore, any pipe rupture can be isolated consistent with the current isolation capability for the existing cleanup system. These design attributes ensure that the system does not affect any accident precursors. The system interface connections are designed, inspected, and tested in accordance with codes and standards that meet or exceed the original system requirements.

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

Safety Evaluation No.: 99-038

Implementation Document No.: Procedure GAP-POL-01

UFSAR Affected Pages: Figure XIII-2

System: N/A

Title of Change: Radiation Protection Organizational Change

Description of Change:

The description of the Unit 1 Radiation Protection branch on UFSAR Figure XIII-2 has been changed to separate the ALARA and equipment sections under two Supervisors Radiation Protection. This separation provides a better distribution of responsibilities and workload among the Supervisors Radiation Protection. This enabled the remaining two Supervisors Radiation Protection to be assigned to Radiation Protection operations duties.

Additionally, the description of the direct reports to the Supervisor Radiation Protection in UFSAR Figure XIII-2 has been changed from "Generation Spec/Engs" to "Radiation Specialists/Technicians." The use of Radiation Specialists reflect a position title change for Generation Specialists and Engineers in the Radiation Protection branch. The inclusion of Technicians recognizes that Technicians are assigned to the Supervisor Radiation Protection for ALARA-related duties.

Safety Evaluation Summary:

The changes to the Supervisors Radiation Protection in the Unit 1 Radiation Protection branch meet the organizational criteria specified in the Technical Specifications, as well as ANSI N18.7-1972 and the Standard Review Plan (NUREG-0800). Clear lines of authority from the Plant Manager to the Supervisors Radiation Protection are maintained. Responsibilities for activities important to safe operation are maintained, and the district functional area of Radiation Protection remains separately supervised and managed.

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

Safety Evaluation No.: 99-039

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

UFSAR Affected Pages: 10A-12, XIII-1 thru XIII-4, XIII-8, XIII-10; Figures XIII-1 thru XIII-4

System: N/A

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

Description of Change:

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

Safety Evaluation Summary:

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

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

Safety Evaluation No.: 99-040

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

UFSAR Affected Pages: X-41, X-43

System: N/A

Title of Change: Removal of the Reactor Building 125-ton Main Crane 1000-pound Hoist

Description of Change:

This safety evaluation evaluated removal of the 1000-lb. hoist as a means for moving new fuel to the spent fuel pool, as described in the UFSAR. As an alternative, the overhead crane 25-ton hoist may be used.

Safety Evaluation Summary:

Removal of the 1000-lb. hoist was evaluated against the applicable criteria, and operation of the fuel handling equipment will remain within the assumptions of the fuel handling analysis. Conformance with the UFSAR criteria for new fuels movements has been demonstrated.

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

Safety Evaluation No.: 99-041

Implementation Document No.: DCR N1-89-000LS773

UFSAR Affected Pages: VII-38, VII-39

System: Reactor Building Ventilation

Title of Change: Retiring Humidity Instrumentation In-Place

Description of Change:

This change retired, in-place, humidity indicating transmitters 202-70 and 71 and their associated alarms. These instruments were considered obsolete and not required for compliance to either industry codes and standards or NRC regulations.

Safety Evaluation Summary:

These instruments and alarms performed no safety function, their retirement in-place impacts no equipment important to safety, and their function is not required by any regulations, codes, or standards.

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

Safety Evaluation No.: 99-042

Implementation Document No.: Procedure N1-OP-43A

UFSAR Affected Pages: VIII-14, VIII-14a

System: Various

Title of Change: Pressure Regulator Design Deficiency and Mitigating Requirements

Description of Change:

NMP1 was experiencing pressure oscillations between 80% and 92% power that were caused by a design deficiency in the turbine controls pressure regulator. Evaluation of the pressure oscillations in the region between 80% and 87% secondary speed relay (turbine control valve position) determined that a high flow gain existed in this region of operation. The pressure regulator oscillation was caused by the excessive gain in this region and not related to a control system malfunction or degraded component. Engineering concluded that these oscillations were not indicative of unstable pressure control. The pressure regulator was stable (decay ratio of less than 1.0) in the transition region but was subject to limit cycle behavior on the order of 2 psi pressure oscillations. Mitigating measures were required to ensure compliance with design criteria for the pressure regulator.

Safety Evaluation Summary:

Engineering has established operating requirements restricting steady state operation outside of a region defined as 82% to 85% secondary speed relay, ensuring that the initial conditions assumed in the transient and accident analysis are not affected. Additionally, transitions through this region using a continuous rate of change in power/steam flow are required to provide additional dampening of the oscillations. The guidance provided for this rate of change (1% to 4% steam flow change/min) is consistent with the normal operating requirements to maintain adequate control of all plant systems, including turbine, recirculation and level control systems, and ensures that operation through the region is consistent with that previously evaluated in the UFSAR. NMP1 can be safely operated with the pressure regulator design deficiency and the proposed operating requirements. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.: 99-043

Implementation Document No.: Procedure NEP-POL-01

UFSAR Affected Pages: XIII-2, XIII-3; Figure XIII-3

System: N/A

Title of Change: Nuclear Engineering Organization Change

Description of Change:

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

Safety Evaluation Summary:

The organization changes remain within the acceptance criteria utilized for the basis of the current Engineering organization, including NUREG-0800 (SRP). Implementation of the proposed organization change requires revision to existing policies and procedures that govern the applicable activities. Primary areas affected by this change are the administrative procedures that describe functional responsibility and activities. These changes do not alter the requirements that govern the procedures or the activities described in the UFSAR. The changes will maintain the required (Technical Specification 6.0) lines of authority, reporting requirements, procedural controls, administrative and recordkeeping functions of the current Engineering organization.

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

Safety Evaluation No.: 99-081

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

UFSAR Affected Pages: N/A

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

Title of Change: Replacement of the Sulfuric Acid Storage Tank

Description of Change:

This modification replaced tank 2WTA-TK3 in the WTA system for the CWS system. The new tank is slightly smaller and constructed of a different material, stainless steel ASTM A240-316L, versus high-density cross-linked polyethylene.

Safety Evaluation Summary:

This change will replace acid storage tank 2WTA-TK3 and associated components which are used for chemistry control in the CWS system. There is no effect on any previously analyzed accidents or transients, as this change does not change the method of operation of any system and does not result in operating the WTA or CWS system outside their design basis. Neither CWS nor WTA are accident precursors or are relied upon to prevent any accidents. A line 5 outage is governed by Technical Specification 3/4 3.8.1 and has previously been evaluated in the context of the Technical Specification LCO. A Line 4 outage does not affect NMP2. A Line 4 outage has been evaluated in the context of Unit 1 Technical Specification 3.6.3.b and Unit 2 USAR Section 1.2. Therefore, this change does not increase the probability of occurrence of an accident.

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

Safety Evaluation No.: 99-085

Implementation Document No.: Procedure EPMP-EPP-02

UFSAR Affected Pages: XII-26

System: N/A

Title of Change: Deleting Use of Portable Gamma Scan Equipment

Description of Change:

The Unit 1 UFSAR referred to a portable multichannel analyzer with a NaI (TI) detector, used and maintained usable in support of the Site Emergency Plan. The unit was used to perform gamma spectroscopy of samples collected for analysis by the offsite survey teams. There is no requirement to have and maintain the capability of performing onsite gamma spectroscopy analysis of samples collected by the offsite survey team. The Site Emergency Plan Procedures have been revised to permit sample analysis by an offsite facility. This change removed the procedure description to have and maintain a portable multichannel analyzer with a NaI (TI) detector gamma scanning unit.

Safety Evaluation Summary:

The portable gamma scanning equipment is not used for testing or experimentation and does not connect to, nor interact with, installed plant equipment. Thus, eliminating use of portable gamma scanning equipment does not constitute a test or experiment not described in the UFSAR, and could not degrade the margin of safety during normal operations or anticipated transients.

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

Safety Evaluation No.: 00-001 Rev. 1

Implementation Document No.: Mod. N1-98-008

UFSAR Affected Pages: V-6, IX-17, X-4, X-5, X-51 thru X-56, 10A-111, 10A-111a, 10A-121, XI-6, XI-9, XI-11, XI-11a; Figures XI-3, XI-5, XI-7

System: Reactor Vessel

Title of Change: Noble Metal Chemical Addition

Description of Change:

The noble metal chemical addition (NMCA or NobleChem) process deposits noble metal compounds (platinum [Pt] and rhodium [Rh]) onto the internal reactor vessel and associated piping wetted surfaces (in conjunction with hydrogen water chemistry) to prevent crack initiation and to mitigate any existing crack growth in the reactor vessel surfaces, internal components, and piping due to intergranular stress corrosion cracking (IGSCC). The catalytic behavior of noble metals provides an opportunity to efficiently achieve a dramatic reduction in the electrochemical corrosion potential (ECP) by catalytically reacting hydrogen with all oxidants at the catalytic surface. This reduction is achieved at low hydrogen injection rates, thereby significantly reducing the magnitude of the increase in main steam line radiation levels normally incurred when operating at the moderate to high hydrogen injection rates needed to protect reactor internals. In addition, with the implementation of low hydrogen injection and NobleChem application, the potentially significant increases in shutdown dose rates associated with moderate HWC injection rates are not expected to occur.

The noble metals compounds are deposited on reactor internal surfaces with the reactor in hot standby condition. The noble metal compounds will be distributed by circulating coolant using all five recirculation pumps. The noble metal deposition process is expected to last approximately 48 hours, with the coolant temperature maintained between 250°F and 350°F, as required by the General Electric-Nuclear Energy Application Procedure. The NobleChem process deposits Pt and Rh onto all surfaces that come into contact with the moving reactor coolant in the applicable temperature range. For example, at a nominal deposition of 1 $\mu g/cm^2$, the uniform coverage would be approximately one atom layer of 3 Å thickness (1 Å is 1 x 10^{-7} mm or 3.94 x 10^{-9} in.). On an atomic scale, the deposited noble metals are discontinuous. Even with agglomeration, the maximum thickness of Pt and Rh will be significantly less than 0.001 inch, which is less than the

Safety Evaluation No.: 00-001 Rev. 1 (cont'd.)

Description of Change: (cont'd.)

minimum manufacturing tolerances of the vessel components (e.g., the tolerance of the fuel Zircaloy tubes is ±0.003 inch and the Zircaloy channels is ±0.004 inch). This safety evaluation evaluated the application process, including all the temporary injection and analysis equipment required to support the application of noble metals during a hot shutdown condition. It also evaluated both the short-term (application period) and long-term (steady state operation) effect of noble metals on reactor internal components.

In addition, this safety evaluation covers operational issues during the initial startup post-NobleChem application, including the hydrogen injection ramp testing. The safety evaluation also covers long-term steady state operation at low hydrogen injection levels.

Safety Evaluation Summary:

This safety evaluation recognizes that a change to the Technical Specification 3.2.3 conductivity limit is required during the injection of noble metals since the existing Technical Specification limits will be exceeded.

Noble metal injection reduces the susceptibility of stainless steel to IGSCC initiation by reducing the ECP below -400 SHE and essentially stops the growth of existing IGSCC, provided the local catalytic conditions achieve a local ECP at the crack mouth of less than -400 SHE. This evaluation has demonstrated that the noble metal coating does not degrade or affect the operation of any component or system. The effect on plant chemistry and radiation exposure is well within the UFSAR requirements and is controlled under special plant procedures such that post-injection plant operations are maintained within the required UFSAR criteria. The probability and consequences of the abnormal events evaluated in the UFSAR and the safety-related equipment functional capabilities are not affected by noble metal injection. The UFSAR safety analyses remain valid and bounding. Noble metal injection does not create any new failure mode or abnormal sequence of events that can result in an accident or malfunction of a different type than previously evaluated in the UFSAR. With the temporary application considerations, noble metal injection equipment can be installed, used and removed without affecting plant safety.

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

Safety Evaluation No.:	00-005 Rev. 1
Implementation Document No.:	Procedures N1-LWPP-1, N1-LWPP-02, N1-LWPP-03, N1-LWPP-4, N1-LWPP-5, N1-LWPP-6, N1-LWPP-7, N1-LWPP-8, N1-LWPP-09, N1-LWPP-10, N1-LWPP-12, N1-LWPP-13, S-WHP-01, N1-WHP-12, N1-WHP-13, NIP-ENV-02, GAP-RMP-01
UFSAR Affected Pages:	XII-1 thru XII-9; Tables XII-2 Sh 1 & 2, XII-3, XII-4; Figure XII-1
System:	Radioactive Waste
Title of Change:	UFSAR Corrections to Reflect Current Procedures and Policies in Regards to the Handling and Processing of Liquid and Solid Waste Streams

Description of Change:

This change updated UFSAR system descriptions and waste handling processes to reflect current procedures and policies in regards to the handling and processing of the liquid and solid waste streams. These changes are necessary due to previous system modifications and improved vendor-processing technology. These discrepancies were identified as part of the UFSAR verification project.

The changes included removal of requirements for the onsite solidification and compaction of waste, the addition of references for the Radwaste Solidification and Storage Building and other equipment where appropriate, revision of waste process descriptions to reflect current configurations, update of waste volumes and activities, and additional references for the use of vendor services.

Safety Evaluation Summary:

The liquid and solid radioactive waste systems are nonsafety-related systems, which collect liquid and solid wastes from remote locations in the plant and process them in an efficient manner without limiting station operations or availability. Waste processing procedures are in place to ensure conformance to the Process Control Program, 10CFR50, and Technical

Safety Evaluation No.: 00-005 Rev. 1 (cont'd.)

Safety Evaluation Summary: (cont'd.)

Specifications. Liquid and solid radioactive wastes are processed within the confines of the Radwaste Facility, which is designed so liquid spills are retained within the building and can be readily captured. The Waste Disposal Building is operated under negative pressure and vented through a dual train, filtered and monitored exhaust to the main stack where effluents are again monitored. The systems described have been operated in the current configurations for more than a decade in conformance with applicable criteria. Correcting the flow paths and processing methods in the UFSAR will ensure future compliance.

Radioactive waste handling and shipping procedures are already in place which ensure conformance to applicable requirements. These procedures direct the processing of radioactive material, packaging of radioactive material for transport in accordance with the Department of Transportation, and other applicable regulations. The procedures also ensure that offsite vendors are properly licensed to receive radioactive material prior to shipment. The vendor's radioactive material license and procedures govern the processing activities at their location. Offsite vendor processing of radioactive laundry and compressible solids is already included in the UFSAR, as is the shipment of radioactive waste for burial. The addition to the UFSAR to allow other radioactive wastes to be shipped to a vendor for processing has no additional impact on the health and welfare of the general public.

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

Safety Evaluation No.: 00-006

Implementation Document No.: Mod. N1-98-023 and Configuration Change 1M00925

UFSAR Affected Pages: Figures X-4, X-5

System: Service Water (SW)

Title of Change: Service Water Modifications

Description of Change:

This modification added the following:

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- A manual isolation valve to the SW system downstream of temperature control valve (TCV) 72-146 to maintain secondary containment when TCV 72-146 is disassembled for routine maintenance or inspection. Previously, opening TCV 72-146 for maintenance was a breach of secondary containment by creating a release path from the Reactor Building through the open valve body to the lake via the SW discharge header. Breaching secondary containment when the plant is on-line requires entry into a four-hour Limiting Condition of Operation (LCO), which is undesirable. This modification eliminated the necessity to enter a LCO during TCV 72-146 maintenance.

- An orifice downstream of SW bypass valve 72-92R. The orifice prevents emergency SW pump runout and cavitation-induced erosion of the bypass valve body and piping. The orifice was sized to provide adequate SW flow for design basis conditions when bypass flow around 72-146 is required.

This modification also replaced the existing bypass valve (72-92R) and downstream piping with components constructed of erosion-resistant material. The 72-92R valve body and downstream piping were subject to cavitation-induced erosion when used as a throttling valve to pass SW flow around TCV 72-146. Weld overlay repairs were done to these components during Operating Cycle 14 to build up the 72-92R valve body and downstream piping to assure their structural integrity. This is a like-for-like replacement for the bypass valve and piping, with the exception of upgraded materials.

Safety Evaluation No.: 00-006 (cont'd.)

Description of Change: (cont'd.)

1M00925 - Turbine Building

- Added a flow-restricting orifice downstream of bypass valve 72-93R to maintain backpressure on the bypass valve. Maintaining backpressure at the bypass valve will prevent cavitation at the valve and erosion of the valve body. The orifice will be sized to assure SW flow for all design basis conditions.

This change also replaced bypass valve 72-93R and downstream piping with components constructed of erosion-resistant material to eliminate or reduce repeat maintenance/repairs of these components. The 72-93R valve body and downstream piping were subject to cavitation-induced erosion when used as a throttling valve to pass SW flow around TCV 72-147.

Safety Evaluation Summary:

The isolation valve, orifices in the bypass lines, new bypass valves and piping are passive components designed and installed to the same codes and standards as the original components to withstand design basis events. The isolation valve will be a maintenance valve and perform no control function in the SW system, and its position will be controlled by operating procedures. The orifice in the Reactor Building bypass line prevents emergency SW pump runout and cavitation at the bypass valve when bypassing TCV 72-146, yet provide adequate flow for design basis conditions. The orifice downstream of bypass valve 72-93R maintains backpressure on the bypass valve and will be sized to assure SW flow for all design basis conditions. The replacement bypass valves and piping are functionally identical to the existing components, upgraded with materials more resistant to erosion. Therefore, operation and performance of the SW system will not be affected by these modifications. As normal service water will not be available during installation of these changes, cooling water will be provided by alternate means. The alternate cooling system will be available to ensure adequate cooling to the reactor building closed loop cooling heat exchangers and will be controlled as a temporary modification and/or temporary operating procedure.

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

Safety Evaluation No.: 00-008

Implementation Document No.: Mod. N1-00-014

UFSAR Affected Pages: N/A

System: Electric Pressure Regulator (EPR)

Title of Change: Turbine Controls Oscillation (Wilmar Engineering Cams)

Description of Change:

This modification replaced the four existing turbine control valve (TCV) cams with new cams designed by Wilmar Engineering. The new cams reduce the high flow gain in the region from 82% to approximately 86% of secondary speed relay (SSR) position. Reducing the high flow gain in this region eliminates the potential for control system (pressure regulator) oscillatory behavior. The reduced gain in the 82%-86% SSR position is offset by increased gain in the region of approximately 40% to 70% of SSR position, which results in a more linear turbine steam flow versus SSR position. The Wilmar Engineering cam design differs from the existing General Electric cams in that during initial turbine roll and generator synchronization to the grid, turbine steam admission is employed through all four TCVs. With the existing cams, turbine roll and generator synchronization to the grid are accomplished by admitting steam to the turbine through TCV #1 only. The Wilmar Engineering cams use more degrees of rotation to stroke the control valve full open (less than 2 degrees of cam rotation of 1%).

Safety Evaluation Summary:

Applicable single failure transients, the RFO15 reload analysis, and minimum critical power ratio operating limits have been evaluated and are not impacted by this modification. The higher gain in the 40% to 70% SSR position, the changes in initial turbine roll and synchronization, and longer SSR stroke have been evaluated and will not introduce new instabilities to the turbine and pressure control system.

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

Safety Evaluation No.: 00-009 Rev. 1

Implementation Document No.: Calculation 125VDC-SYSTEM-CASEB

UFSAR Affected Pages: IX-22; Table IX-1 Sh 1 & 2

System: 125V DC Distribution (VDC125)

Title of Change: Revision to UFSAR Table IX-1 Battery Duty Cycle

Description of Change:

A new battery sizing calculation, 125VDC-SYSTEM-CASEB, was created to combine calculations 125VDC-BATT-CASEB, 125VDC-BATT11-CASEB and 125VDC-BATT12-CASEB for the UFSAR Case B event into one calculation. The new calculation supersedes these three calculations. The results of this new calculation are not consistent with the listed values for Case B battery loading in UFSAR Table IX-1. Therefore, UFSAR Table IX-1 has been revised to be consistent with the results of the new calculation. In addition, the information in the table associated with the Case A event (loss of offsite power (LOOP) with Technical Specification leakage, no unit trip) was removed, as the Case A event is bounded by the Case B event (LOOP loss-of-coolant accident with unit trip) with respect to battery loading. The batteries still have adequate capacity to supply the Case B design basis loads with spare capacity remaining.

Safety Evaluation Summary:

The proposed changes involve the battery loading and duty cycle analyses associated with the Case B event. The UFSAR is being revised to reflect the results of the latest revision to the 125 VDC battery sizing calculation for the Case B event. This change does not impact the operation of the 125 VDC system or any loads supplied by the 125 VDC system.

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

Safety Evaluation No.: 00-012

Implementation Document No.: Mod. N1-00-010

UFSAR Affected Pages: V-3, XV-7, XV-8; Tables V-2, V-3, XVI-2 Sh 1 & 2, XVI-3

System: Reactor Recirculation

Title of Change: Design Basis Analysis for Reactor Recirculation System Thermal Transient

Description of Change:

This safety evaluation addressed design basis analysis of the recirculation piping, nozzles and safe end for thermal transients not included in the original design basis analyses. During investigation into restart of an isolated reactor recirculation loop, design basis analyses of the impact of relatively cold water impinging on the recirculation nozzles could not be recovered. In addition, analyses evaluating thermal transients associated with several other operational configurations of the recirculation piping could not be recovered. Without a formal evaluation of the impact of the thermal transients on the structural adequacy of the nozzles, piping and safe ends, the operability of the recirculation system was in question. Design basis analyses were performed to address the new transients. The results of these thermal transient stress evaluations impacted reactor fatigue and stress evaluations described in the UFSAR. Accordingly, the UFSAR has been revised.

Safety Evaluation Summary:

The major safety significance of performing design basis analyses to evaluate the new thermal transients is the potential for these new thermal transients to produce stresses in the nozzles, safe ends or piping which exceed design basis compliance limits. Other concerns include nozzle nil ductility transition temperature and the shroud repair. Evaluations of the new thermal transients determined that one of the new thermal transients, emergency condenser (EC) actuation into an isolated loop, bounds the other thermal transients not previously evaluated. Note that the EC actuation transient bounds the unisolation of an isolated loop transient, and the inadvertent actuation of a recirculation pump in an isolated loop transient, only when the isolated loop's minimum temperature is 150°F or higher. A minimum 155°F limit is controlled by plant procedure

Safety Evaluation No.: 00-012 (cont'd.)

Safety Evaluation Summary: (cont'd.)

(includes 5°F margin for instrument uncertainty). Design basis analyses for this transient show that the recirculation system piping, supports, nozzles, safe ends, nozzle nil ductility transition temperature, and the shroud repair are all within design basis compliance limits.

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

Safety Evaluation No.: 00-013

Implementation Document No.: Mod. N1-99-027

UFSAR Affected Pages: Figures VI-22, X-2

System: Reactor Water Cleanup

Title of Change: Replace IV-33-02R and Add Manual Blocking Valve

Description of Change:

IV-33-02R is a 6-inch motor-operated gate valve, serving as the inboard isolation valve on the reactor water cleanup (RWCU) outlet line. This modification replaced this isolation valve because it has a poor history of failing local leak rate tests during refueling outages.

The existing valve remained in place with its operator de-terminated. Its function has become a manually-operated maintenance valve. The replacement valve (6-inch 900# CS gate valve) and its actuator were installed in the carbon steel piping between the existing valve and penetration X-9. The existing power and control wiring were connected to the new valve, which now performs all functions intended for the new valve (IV-33-02R). The old valve was given a new component identification number of BV-33-226. Appropriate vent and drain valves were added to provide for inservice testing.

Safety Evaluation Summary:

The possible failures and malfunctions of the new valve are identical to those of the existing isolation valve. All system design requirements are being maintained. All power and control wiring for the existing valve IV-33-02R will be transferred to the new IV-33-02R valve. No changes will be made to the control circuits; therefore, the new valve will function the same as the old valve.

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

Safety Evaluation No.: 00-014

Implementation Document No.: Mod. N1-99-025

UFSAR Affected Pages: VI-25; Table VI-3a Sh 2, XV-10; Figures VI-22, VII-1

System: Core Spray

Title of Change: Addition of Overpressure Protection for the Core Spray System Containment Penetrations X-13A and X-14

Description of Change:

This modification installed a relief path for the overpressure protection of core spray penetrations X-13A and X-14 under design basis accident conditions. This modification was required for compliance to NMP1 commitments to Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions."

Safety Evaluation Summary:

This change proposes to install pressure relief bypass lines with check valves on each of the two loops of the core spray system. These lines will be installed inside the drywell, across one of the two inboard isolation valves, and will provide overpressure protection by venting fluid from the isolated penetrations to the upstream side of the inboard core spray isolation valves. This modification ensures that proper thermal relief is provided as required by GL 96-06 while containment and reactor coolant isolation is maintained for the core spray system. The proposed size of the stainless steel lines and the check valves will be one inch or smaller. The modification to the core spray system will meet or exceed the design criteria for the existing system and will maintain the functional integrity of the core spray system.

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

Safety Evaluation No.: 00-017 Rev. 0 & 1

Implementation Document No.: Mod. N1-00-015

UFSAR Affected Pages: VI-27, X-19

System: Reactor Building Closed Loop Cooling (RBCLC)

Title of Change: RBCLC Supply Temperature Setpoint Change/Drywell Minimum Temperature

Description of Change:

This modification raised the RBCLC supply temperature setpoint (TSP 70-23E) from 80°F to 88°F. The setpoint for drywell el. 319′ local area high temperature alarm was lowered from 195°F to the original alarm setpoint of 175°F. Also, a minimum drywell bulk average temperature design limit of 120°F has been imposed during steady state power operating conditions.

Safety Evaluation Summary:

Raising the RBCLC supply temperature setpoint returns the RBCLC system to its design operating range. Imposing a minimum drywell bulk average temperature design limit during steady state power operating conditions ensures the emergency operating procedures are maintained within the bounds of current analysis, and that drywell temperature effects on reactor head safety valves and vessel level instrumentation are minimized. Restoring the RBCLC supply temperature to its design operating range ensures that the initial conditions in the UFSAR Chapter XV analyses are not affected. Imposing a minimum drywell temperature during steady state power operating conditions ensures that protective actions associated with vessel level instrumentation and reactor head safety valves remain unaffected.

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

Safety Evaluation No.: 00-018 Rev. 0 & 1

Implementation Document No.: Mod. N1-00-022

UFSAR Affected Pages: VII-42

System: High Pressure Coolant Injection (HPCI)

Title of Change: HPCI Trips - Phase 2

Description of Change:

This modification changed the HPCI channel 11 logic circuit so that #11 FCV-29-141, for motor-driven feedwater pump FWP11, does not open if there is sufficient feedwater flow to the reactor by maintaining FCV11 closed until feedwater flow to the reactor drops below 4.5 mMlb/hr. This logic is bypassed if FWP12 is not running or locked out, allowing FCV11 to open. A separate low pressure timing function for FWP12 (preferred HPCI train) has been provided in a separate timer to increase the low suction trip time delay. This change also provided a clamp of the level setpoint setdown controller (ID66B) to a maximum 70% output for FCV-29-134. Control is transferred to ID66B upon a plant scram and a level drop below 52".

Safety Evaluation Summary:

Revising HPCI logic and limiting the maximum demand output of the level setpoint setdown controller (ID66B) eliminates the potential for feedwater pump low suction pressure trips during HPCI initiation and provides reliable HPCI operation for all plant transients.

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

Safety Evaluation No.: 00-024

Implementation Document No.: N/A

UFSAR Affected Pages: XV-65

System: Containment Spray

Title of Change: Emergency Operating Procedure Revision - Primary Containment Control

Description of Change:

Emergency Operating Procedure N1-EOP-4, Primary Containment Control, has been revised to change the direction to terminate drywell spray from "before drywell pressure drops to 0 psig" to "when drywell pressure drops below 3.5 psig." This safety evaluation addresses the change to the EOPs relative to containment spray operation currently described in the UFSAR. Accordingly, the UFSAR description has been revised to reflect the revisions to the EOPs.

Safety Evaluation Summary:

The proposed change incorporates a minor revision to the EOPs to update the pressure at which containment sprays are terminated. Verbatim compliance with current EOP instructions could result in continuous containment sprays to just above 0 psig. This action is not consistent with current design basis analyses for emergency core cooling system suction strainers, the loss-of-coolant accident analysis, the containment spray water seal, and diesel load shedding. Continuous spray to 0 psig provides some fission product scrubbing. This change will not adversely affect the capability of the containment spray system to satisfy the emergency procedure guidelines/severe accident guidelines (EPG/SAG). The EPG/SAGs recognize that the point at which containment sprays are terminated is event specific. Previous revisions of the EOPs utilized the 3.5 psig value. Those procedures were demonstrated to have been successfully utilized to mitigate adverse primary containment conditions through a validation program, and extensive use in simulator training and evaluated scenarios.

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

Safety Evaluation No.: 00-025

Implementation Document No.: Mod. N1-00-031

UFSAR Affected Pages: Figure VI-22

System: Waste Disposal

Title of Change: GL 96-06 Thermal Relief for Floor Drain Penetration X-25

Description of Change:

Overpressure protection is added for containment penetration X-25 in accordance with Generic Letter (GL) 96-06. Containment isolation can be concurrent with a design basis accident. The higher area temperatures during a design basis accident will cause thermal expansion of the water trapped between the containment isolation valves. This modification installed a relief valve on the 3/4" test line between the containment isolation valves, downstream of valve 83.1-20 in the Reactor Building outside containment. This valve was replaced with a valve rated for a higher pressure. This 3/4" valve is normally closed and is a containment isolation valve. As a part of this modification, this valve was locked open to allow the relief valve to perform its function. This modification also installed another test valve that is normally closed and is a containment isolation valve. This new valve performs the previous function of valve 83.1-20, which allows the piping in the penetration to be tested in accordance with the NMP1 Appendix J Testing Program Plan. The setpoint of the relief valve is 78 psig. The setpoint of relief valve 83.1-33 has been changed to 65 psig. The design pressure of the piping between valves 83.1-11 and 83.1-12 has been changed to 85 psig.

Safety Evaluation Summary:

This modification will be performed on the waste disposal system, the portion of the system that allows discharge of the drywell floor drain sump and the containment system. The modification will maintain the pressure boundary and the double isolation requirement for the containment boundary. It will have no effect on the operation of the drywell floor drain sump discharge. It will meet the requirements of GL 96-06 for overpressure protection due to thermal expansion. The modification will be designed, installed, inspected, and tested to requirements that meet or exceed the original requirements for this system.

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

Safety Evaluation No.: 00-071

Implementation Document No.: FPEE 0-00-003

UFSAR Affected Pages: 10A-40, 10A-41, 10A-43, 10A-46, 10A-47, 10A-51, 10A-54

System: Fire Protection Water, Carbon Dioxide, Halon 1301 Suppression

Title of Change: Surveillance Frequency Revisions for Fire Protection Systems

Description of Change:

Several surveillance frequencies at Unit 1 and Unit 2 have been revised based on Fire Protection Engineering Evaluation 0-00-003, NFPA 25-1998, and NFPA 12-2000. These frequency changes greatly reduce the cost of inspecting and testing several fire protection systems while still providing equivalent reliability as that specified in the NFPA codes.

Safety Evaluation Summary:

This safety evaluation demonstrates that the proposed changes will not degrade the reliability of any fire protection system installed in the plant. The changes to the surveillance frequencies are for fire protection systems in areas that contain safety-related equipment and are based on plant-specific failure rates or NFPA code specified frequencies. These changes do not increase the probability of the postulated fire in the Fire Hazards Analysis, nor do they increase or decrease the severity of a fire.

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

Safety Evaluation No.: 00-086

Implementation Document No.: Procedure NTP-TQS-101

UFSAR Affected Pages: N/A

System: N/A

Title of Change: NTP-TQS-101, Licensed Operator Candidate Training, Rev. 7

Description of Change:

This safety evaluation evaluated the following changes to Procedure NTP-TQS-101:

- Incorporated new/revised guidance contained in ACAD 00-003, Guidelines for Initial Training and Qualification of Licensed Operators, which superseded ACAD 91-012, Guidelines for Initial Training and Qualification of Licensed Operators.
- Replaced topical lists within the body of the procedure with attachments which identify the required training by lesson plan identification number and title.
- Changed "should" to "shall" in all action statements required to meet regulatory requirements.
- Revised the responsibilities of the Plant Manager to be consistent with those described in the Unit 1 UFSAR.
- Added a reference to Unit 2 USAR Section 8.3, which identifies training requirements for diesel generators.
- References to support procedures were added.
- Incorporated flowcharts to be used to determine and document eligibility (developed from ACAD 00-003, ANSI N18.1-1971 and ANSI/ANS 3.1-1978).
- Incorporated specific criteria to determine "significant" and "diverse" reactivity changes to meet the requirements of 10CFR55.31.
- Added action statements to ensure all position-specific training required to meet OSHA, Environmental, and Emergency Plan requirements is complete before assuming duties at the plant.
- Added action statements to ensure all certification documentation required by NIP-TQS-01 is complete.

Safety Evaluation No.: 00-086 (cont'd.)

Safety Evaluation Summary:

The Niagara Mohawk Nine Mile Point Licensed Operator Candidate Training Program, described in Procedure NTP-TQS-101, has been developed using a Systems Approach to Training, as stated in 10CFR55, INPO ACAD 00-003, and is accredited by the National Nuclear Accrediting Board. Based on this accreditation, the change satisfies 10CFR55 requirements for Licensed Operator Candidate Training.

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

Safety Evaluation No.: 01-001

Implementation Document No.: Procedures N1-MMP-GEN-903, N1-MMP-GEN-904, N1-OP-34

UFSAR Affected Pages: X-43, X-44

System: Reactor Vessel (RXVE)

Title of Change: Procedure Changes to Modify the Movement of Water in the Reactor Head Cavity During Disassembly for Refueling or for Reassembly

Description of Change:

This safety evaluation analyzed changes to Procedures N1-MMP-GEN-903, N1-MMP-GEN-904, and N1-OP-34, which control aspects of refueling regarding certain stages of reactor disassembly and reassembly and movement of water in the reactor head cavity. Implementation of the changes provides flexibility in the timing of flood-up of the reactor during disassembly for refueling and drain-down during reassembly.

The procedure changes allow the flood-up (up to the normal level of the spent fuel storage pool) to begin prior to the steam dryer assembly being removed. In a similar manner, during the vessel reassembly process, flexibility is provided in determining when water drain-down can occur. The changes permit transfer and reassembly of the separator and dryer assemblies prior to drain-down being initiated.

Safety Evaluation Summary:

The procedure changes in the sequence of reactor component disassembly and reassembly do not involve a change to the plant Technical Specifications. These changes do not increase the probability of occurrence of accidents or malfunctions, nor do they increase the consequences of accidents or malfunctions previously evaluated in the UFSAR. The changes do not create the possibility of an accident or malfunction of a different type than previously evaluated in the UFSAR, nor do they reduce the margin of safety as defined in the basis of the Technical Specifications.

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

Safety Evaluation No.: 01-002 Rev. 0 & 1

Implementation Document No.: N/A

UFSAR Affected Pages: I-16, I-19, I-21, IV-7, IV-11, IV-12, IV-32, V-21, VII-3, VII-6, VII-44, VIII-13, VIII-55, XV-3, XV-5, XV-6, XV-7, XV-15, XV-68; Table XV-9a; Figure VII-2

System: Various

Title of Change: Operation of NMP1 Reload 16/Cycle 15

Description of Change:

This change consisted of the addition of new fuel bundles and the establishment of a new core loading pattern for Reload 16/Cycle 15 operation of NMP1. One hundred forty-eight (148) new fuel bundles of the GE11 design were loaded. Various evaluations and analyses were performed to establish appropriate operating limits for the reload core. These cycle-specific limits were documented in the Core Operating Limits Report (COLR).

Revision 1 to this safety evaluation implements revised analyses of a pressure regulator out of service, including the associated changes to the COLR.

Safety Evaluation Summary:

The reload analyses and evaluations are performed based on the General Electric Standard Application for Reactor Fuel, NEDE 24011-P-A-14 and NEDE 24011-P-A-14-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 16, 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 evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.: 01-004

Implementation Document No.: DDC 1A00041

UFSAR Affected Pages: IV-18, IV-24, IV-32, VII-23

System: Reactor & Containment Instruments (RPVI), Reactor Vessel (RXVE), CRD Hydraulic After Strainer

Title of Change: NMP1 Marathon Control Blades

Description of Change:

This change allows the use of the General Electric Marathon control rods in the NMP1 reactor. The Marathon control rod is a one-for-one direct replacement for the designs previously used at NMP1. The control rod weight is slightly less than the original equipment control rods, but scram time remains unaffected. The initial control rod worth is within ±5% of the initial worth of the control rods previously used and, therefore, is considered to be the same.

Safety Evaluation Summary:

The Marathon control rods have the same initial worth as other control rods currently in use at NMP1. In addition, they fit in the same dimensional envelope and are slightly lighter than the control rods currently in use. The velocity limiter and the coupling to the control rod drive are the same as the original equipment control rods. Therefore, it is concluded that the use of Marathon control rods will not adversely affect the safe operation of NMP1.

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

U.S. NUCLEAR REGULATORY COMMISSION
DOCKET 50-220
LICENSE DPR-63

NINE MILE POINT NUCLEAR STATION UNIT 1

FINAL SAFETY ANALYSIS REPORT (UPDATED)

OCTOBER 2001

REVISION 17

NIAGARA MOHAWK POWER CORPORATION
SYRACUSE, NEW YORK

INSERTION INSTRUCTIONS

The following instructions are for the insertion of the current revision into the Nine Mile Point Unit 1 FSAR (Updated).

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

Vertical bars have been placed in the margins of inserted pages and tables to indicate revision locations.

INSERTION INSTRUCTIONS

LIST OF EFFECTIVE PAGES

REMOVE	INSERT
EP i	EP i
EP 1-1	EP 1-1
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EP 4-1	EP 4-1
EP 5-1	EP 5-1
EP 6-1	EP 6-1
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EP 8-1 thru EP 8-2	EP 8-1 thru EP 8-2
EP 9-1	EP 9-1
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EP 11-1	EP 11-1
EP 12-1	EP 12-1
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EP 17-1 thru EP 17-2	EP 17-1 thru EP 17-2
EP 18-1	EP 18-1
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INSERTION INSTRUCTIONS

VOLUME 1

REMOVE	INSERT
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---	xa/xb
xi/xii	xi/xii
xiii/xiv	xiii/xiv
xv/xvi	xv/xvi
xvii/xviii	xvii/xviia
---	xviib/xviii
xix/xx	xix/xx
---	xxa/xxb
xxix/xxx	xxix/xxx
xxxiii/xxxiv	xxxiii/xxxiv
---	xxxiva/xxxivb
xxxv/xxxvi	xxxv/xxxvi
xliii/xliv	xliii/xliv
xlv/xlvi	xlv/xlvi
xlvii/xlviii	xlvii/xlviia
---	xlviib/xlviii
I-1/2 thru I-15/-	I-1/2 thru I-21/-
---	T I-1/-
F I-1	F I-1
III-5/6 thru III-7/8	III-5/6 thru III-7/8
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VOLUME 1 (Cont'd.)

REMOVE	INSERT
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---	V-6a/6b
V-7/8	V-7/8
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T V-1 thru T V-4	T V-1 thru T V-4
F V-1	F V-1

INSERTION INSTRUCTIONS

VOLUME 2

REMOVE	INSERT
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ix/x	ix/x
---	xa/xb
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xiii/xiv	xiii/xiv
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---	xviib/xviii
xix/xx	xix/xx
---	xxa/xxb
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---	xxxiva/xxxivb
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T VI-4	T VI-4
F VI-20	F VI-20
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VII-1/2	VII-1/2
---	VII-2a/2b
VII-3/4 thru VII-13/14	VII-3/4 thru VII-13/14
---	VII-14a/14b thru VII-14g/14h
VII-23/24	VII-23/24
VII-33/34 thru VII-35/36	VII-33/34 thru VII-35/36
VII-37/37a	VII-37/38
VII-37b/38	---
VII-39/40 thru VII-41/42	VII-39/40 thru VII-41/42
---	VII-42a/42b

INSERTION INSTRUCTIONS

VOLUME 2 (Cont'd.)

REMOVE	INSERT
VII-43/44	VII-43/44
F VII-1	F VII-1
F VII-2	F VII-2
F VII-3	F VII-3
F VII-12	F VII-12
F VII-13	F VII-13
VIII-7/8	VIII-7/8
VIII-9/9a	VIII-9/9a
VIII-9b/10	VIII-9b/10
VIII-13/14	VIII-13/14
---	VIII-14a/14b
VIII-15/16 thru VIII-19/20	VIII-15/16 thru VIII-19/20
VIII-25/26 thru VIII-29/30	VIII-25/26 thru VIII-29/30
VIII-33/34	VIII-33/34
VIII-35/36	VIII-35/36
VIII-43/44 thru VIII-45/46	VIII-43/44 thru VIII-45/46
VIII-51/52	VIII-51/52
VIII-55/-	VIII-55/-
T VIII-4 Sh 1	T VIII-4 Sh 1
T VIII-4 Sh 3	T VIII-4 Sh 3
F VIII-2 thru F VIII-4	F VIII-2 thru F VIII-4
F VIII-6	F VIII-6
F VIII-12 thru F VIII-14	F VIII-12 thru F VIII-14
F VIII-22	F VIII-22
F VIII-24	F VIII-24
F VIII-26	F VIII-26
---	F VIII-26b
F VIII-29	F VIII-29
IX-3/4	IX-3/4
IX-11/12	IX-11/12
IX-15/16 thru IX-17/18	IX-15/16 thru IX-17b/18
IX-21/22 thru IX-23/24	IX-21/22 thru IX-23/24
T IX-1 Sh 1 thru 3	T IX-1 Sh 1 & 2
F IX-1	F IX-1
---	F IX-8

INSERTION INSTRUCTIONS

VOLUME 3

REMOVE	INSERT
i/ii	i/ii
v/vi	v/vi
vii/viii	vii/viii
ix/x	ix/x
---	xa/xb
xi/xii	xi/xii
xiii/xiv	xiii/xiv
xv/xvi	xv/xvi
xvii/xviii	xvii/xviia
---	xviib/xviii
xix/xx	xix/xx
---	xxa/xxb
xxix/xxx	xxix/xxx
xxxiii/xxxiv	xxxiii/xxxiv
---	xxxiva/xxxivb
xxxv/xxxvi	xxxv/xxxvi
xliii/xliv	xliii/xliv
xlv/xlvi	xlv/xlvi
xlvii/xlviii	xlvii/xlviia
---	xlviib/xlviii
X-1/2	X-1/1a
---	X-1b/2
X-3/4 thru X-5/6	X-3/4 thru X-5/5a
---	X-5b/6
X-9/10 thru X-23/24	X-9/10 thru X-23/24
X-29/30 thru X-35/36	X-29/30 thru X-35/36
X-39/40	X-39/40
---	X-40a/40b
X-41/42 thru X-43/44	X-41/42 thru X-43/44
X-51/-	X-51/52 thru X-55/56
F X-2	F X-2
F X-4	F X-4
F X-5	F X-5
F X-7	F X-7
F X-8	F X-8
10A-i/ii	10A-i/ii
10A-11/12 thru 10A-13/14	10A-11/12 thru 10A-13/14
10A-23/24 thru 10A-25/26	10A-23/24 thru 10A-25/25a
---	10A-25b/26
10A-39/40 thru 10A-43/44	10A-39/40 thru 10A-43/44
---	10A-44a/44b
10A-45/46 thru 10A-47/48	10A-45/46 thru 10A-47/48
10A-51/52 thru 10A-53/54	10A-51/52 thru 10A-53/54
10A-69/70	10A-69/70
10A-79/80	10A-79/80

INSERTION INSTRUCTIONS

VOLUME 3 (Cont'd.)

REMOVE	INSERT
10A-90/-	10A-90/-
10A-106/-	10A-106/-
10A-111/-	10A-111/-
10A-120/-	10A-120/-
10A-121/-	10A-121/-
10A-122/-	10A-122/-
10A-126/-	10A-126/-
F 10A-4	F 10A-4
F 10A-4A	F 10A-4A
F 10A-4B	F 10A-4B
F 10A-4C	F 10A-4C
F 10A-4D	F 10A-4D
F 10A-5	F 10A-5
F 10A-5A	F 10A-5A
F 10A-5B	F 10A-5B
F 10A-5C	F 10A-5C
F 10A-5D	F 10A-5D
F 10A-7	F 10A-7
F 10A-7A	F 10A-7A
F 10A-7B	F 10A-7B
F 10A-7C	F 10A-7C
F 10A-7D	F 10A-7D
10B-9/10	10B-9/10
10B-13/14	10B-13/14
---	10B-14a/14b
10B-24/-	10B-24/-
---	10B-24a/-
10B-25/- thru 10B-26/-	10B-25/- thru 10B-26/-
10B-28/-	10B-28/-
10B-30/-	10B-30/-
---	10B-30a/-
10B-31/-	10B-31/-
---	10B-31a/-
10B-60/-	10B-60/-
10B-103/104	10B-103/104
10B-111/-	10B-111/-
10B-113/-	10B-113/-
10B-207/-	10B-207/-
10B-223/-	10B-223/-
XI-5/6	XI-5/6
XI-9/10	XI-9/9a
---	XI-9b/10
XI-11/12	XI-11/11a
---	XI-11b/12
XI-13/14	XI-13/14

INSERTION INSTRUCTIONS

VOLUME 3 (Cont'd.)

REMOVE	INSERT
F XI-1	F XI-1
F XI-3	F XI-3
F XI-5	F XI-5
F XI-7	F XI-7
XII-1/2 thru XII-13/14	XII-1/2 thru XII-13/14
---	XII-14a/14b
XII-17/18 thru XII-19/20	XII-17/18 thru XII-19/20
XII-23/24 thru XII-27/28	XII-23/24 thru XII-27/28
T XII-2 Sh 1 & 2	T XII-2 Sh 1 & 2
T XII-3/4	T XII-3/4
T XII-8 Sh 1/2	T XII-8 Sh 1/2
---	T XII-8 Sh 3/-
F XII-1	F XII-1
XIII-1/2 thru XIII-3/4	XIII-1/2 thru XIII-3/4
XIII-7/8 thru XIII-15/16	XIII-7/8 thru XIII-15/16
T XIII-1	T XIII-1
F XIII-1 thru F XIII-4	F XIII-1 thru F XIII-4

INSERTION INSTRUCTIONS

VOLUME 4

REMOVE	INSERT
i/ii	i/ii
v/vi	v/vi
vii/viii	vii/viii
ix/x	ix/x
---	xa/xb
xi/xii	xi/xii
xiii/xiv	xiii/xiv
xv/xvi	xv/xvi
xvii/xviii	xvii/xviia
---	xviib/xviii
xix/xx	xix/xx
---	xxa/xxb
xxix/xxx	xxix/xxx
xxxiii/xxxiv	xxxiii/xxxiv
---	xxxiva/xxxivb
xxxv/xxxvi	xxxv/xxxvi
xliii/xliv	xliii/xliv
xlv/xlvi	xlv/xlvi
xlvii/xlviii	xlvii/xlviia
---	xlviib/xlviii
XV-1/2 thru XV-7/8	XV-1/2 thru XV-7/8
---	XV-8a/8b
XV-9/10	XV-9/10
XV-15/16	XV-15/16
XV-19/20 thru XV-21/22	XV-19/20 thru XV-21/22
XV-65/66 thru XV-67/68	XV-65/66 thru XV-67/68
XV-81/82	XV-81/82
T XV-1 Sh 1/2	T XV-1/T XV-2
T XV-2/-	---
T XV-9a/-	T XV-9a/-
T XV-10/-	T XV-10/-
F XV-1	F XV-1
XVI-23/24	XVI-23/24
XVI-31/32	XVI-31/32
---	XVI-32a/32b
XVI-49/50	XVI-49/50
XVI-52a/52b	XVI-52a/52b
XVI-53/54	XVI-53/54
XVI-55/56	XVI-55/55a
---	XVI-55b/56
---	XVI-60a/60b
XVI-65/66	XVI-65/65a
---	XVI-65b/66
XVI-71/72	XVI-71/72
T XVI-2 Sh 1 & 2	T XVI-2 Sh 1 & 2

INSERTION INSTRUCTIONS

VOLUME 4 (Cont'd.)

REMOVE	INSERT
T XVI-3	T XVI-3
B-i/ii	B-i/ii
B.0-1/2	B.0-1/2
B.1-1/2 thru B.1-5/-	B.1-1/2 thru B.1-5/-
B.2-1/2 thru B.2-5/6 B.2-9/10 thru B.2-11/12	B.2-1/2 thru B.2-5/6 B.2-9/10 thru B.2-11/12
B.3-1/2	B.3-1/2
B.4-1/2	B.4-1/2
B.6-1/2	B.6-1/2
B.11-1/2	B.11-1/2
T B-1 Sh 1 & 2 T B-2 T B-3 Sh 1 thru 8	T B-1 Sh 1 & 2 T B-2 Sh 1 thru 9 ---

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vii	17	xlv	16
viii	17	xlvi	17
ix	16	xlvii	17
x	17	xlviia	17
xa	17	xlviib	17
xb	17	xlviii	16
xi	17	xlix	16
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xix	17		
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NINE MILE POINT NUCLEAR STATION UNIT 1

FINAL SAFETY ANALYSIS REPORT (UPDATED)

VOLUME 1

OCTOBER 2001

REVISION 17

NIAGARA MOHAWK POWER CORPORATION
SYRACUSE, NEW YORK

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SECTION I

INTRODUCTION AND SUMMARY

This report is submitted in accordance with 10 CFR Part 50.71(e) entitled "Periodic Updating of Final Safety Analysis Reports" for Niagara Mohawk Power Corporation's (NMPC) Nine Mile Point Nuclear Station - Unit 1 (Unit 1). The Station is located on the southeast shore of Lake Ontario, in Oswego County, New York, 7 mi northeast of the city of Oswego.

A. PRINCIPAL DESIGN CRITERIA

The following paragraphs describing the principal design criteria are oriented toward the twenty-seven criteria issued by the United States Atomic Energy Commission (USAEC) on November 22, 1965.[1] The twenty-seven criteria represented proposed "General Design Criteria for Nuclear Power Plant Construction Permits." The twenty-seven criteria are presented here for historical reference and are followed by the Unit 1 principal design criteria.

Table I-1 provides historical information regarding an assessment of Unit 1 against criteria that were being used by the USAEC at the time of NMPC's application for a full-term operating license.

<u>Facility</u>

<u>Criterion 1</u>

Those features of reactor facilities which are essential to the prevention of accidents or to the mitigation of their consequences must be designed, fabricated, and erected to:

(a) Quality standards that reflect the importance of the safety function to be performed. It should be recognized, in this respect, that design codes commonly used for nonnuclear applications may not be adequate.

(b) Performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces imposed by the most severe earthquakes, flooding conditions, winds, ice, and other natural phenomena anticipated at the proposed site.

<u>Criterion 2</u>

Provisions must be included to limit the extent and the consequences of credible chemical reactions that could cause or materially augment the release of significant amounts of fission products from the facility.

<u>Criterion 3</u>

Protection must be provided against possibilities for damage of the safeguarding features of the facility by missiles generated through equipment failures inside the containment.

Reactor

Criterion 4

The reactor must be designed to accommodate, without fuel failure or primary system damage, deviations from steady state norm that might be occasioned by abnormal yet anticipated transient events such as tripping of the turbine generator and loss of power to the reactor recirculation system pumps.

Criterion 5

The reactor must be designed so that power or process variable oscillations or transients that could cause fuel failure or primary system damage are not possible or can be readily suppressed.

Criterion 6

Clad fuel must be designed to accommodate throughout its design lifetime all normal and abnormal modes of anticipated reactor operation, including the design overpower condition, without experiencing significant cladding failures. Unclad or vented fuels must be designed with the similar objective of providing control over fission products. For unclad and vented solid fuels, normal and abnormal modes of anticipated reactor operation must be achieved without exceeding design release rates of fission products from the fuel over core lifetime.

Criterion 7

The maximum reactivity worth of control rods or elements and the rates with which reactivity can be inserted must be held to values such that no single credible mechanical or electrical control system malfunction could cause a reactivity transient capable of damaging the primary system or causing significant fuel failure.

Criterion 8

Reactivity shutdown capability must be provided to make and hold the core subcritical from any credible operating condition with any one control element at its position of highest reactivity.

Criterion 9

Backup reactivity shutdown capability must be provided that is independent of normal reactivity control provisions. This system must have the capability to shut down the reactor from any operating condition.

Criterion 10

Heat removal systems must be provided which are capable of accommodating core decay heat under all anticipated abnormal and credible accident conditions, such as isolation from the main condenser and complete or partial loss of primary coolant from the reactor.

Criterion 11

Components of the primary coolant and containment systems must be designed and operated so that no substantial pressure or thermal stress will be imposed on the structural materials unless the temperatures are well above the nil ductility temperatures. For ferritic materials of the coolant envelope and the containment, minimum temperatures are NDT + 60°F and NDT + 30°F, respectively.

Criterion 12

Capability for control rod insertion under abnormal conditions must be provided.

Criterion 13

The reactor facility must be provided with a control room from which all actions can be controlled or monitored as necessary to maintain safe operational status of the plant at all times. The control room must be provided with adequate protection to permit occupancy under the conditions described in Criterion 17 below, and with the means to shut down the plant and maintain it in a safe condition if such accident were to be experienced.

Criterion 14

Means must be included in the control room to show the relative reactivity status of the reactor such as position indication of mechanical rods or concentrations of chemical poisons.

Criterion 15

A reliable reactor protection system must be provided to automatically initiate appropriate action to prevent safety limits from being exceeded. Capability must be provided for testing functional operability of the system and for determining that no component or circuit failure has occurred. For instruments and control systems in vital areas where the potential consequences of failure require redundancy, the redundant channels must be independent and must be capable of being tested to determine that they remain independent. Sufficient redundancy must be provided that failure or removal from service of a single component or channel will not inhibit necessary safety action when required. These criteria should, where applicable, be satisfied by the instrumentation associated with containment closure and isolation systems, afterheat removal and core cooling systems, systems to prevent cold-slug accidents, and other vital systems, as well as the reactor nuclear and process safety system.

Criterion 16

The vital instrumentation systems of Criterion 15 must be designed so that no credible combination of circumstances can interfere with the performance of a safety function when it is needed. In particular, the effect of influences common to redundant channels which are intended to be independent must not negate the operability of a safety system. The effects of gross disconnection of the system, loss of energy (electric power, instrument air), and adverse environment (heat from loss of instrument cooling, extreme cold, fire, steam, water, etc.) must cause the system to go into its safest state (fail-safe) or be demonstrably tolerable on some other basis.

Engineered Safeguards

Criterion 17

The containment structure, including access openings and penetrations, must be designed and fabricated to accommodate or dissipate without failure the pressures and temperatures associated with the largest credible energy release including the effects of credible metal-water or other chemical reactions uninhibited by active quenching systems. If part of the primary coolant system is outside the primary reactor containment, appropriate safeguards must be provided for that part if necessary, to protect the health and safety of the public, in case of an accidental rupture in that part of the system. The appropriateness of safeguards such as isolation valves, additional containment, etc., will depend on environmental and population conditions surrounding the site.

Criterion 18

Provisions must be made for the removal of heat from within the containment structure as necessary to maintain the integrity of the structure under the conditions described in Criterion 17 above. If engineered safeguards are needed to prevent containment vessel failure due to heat released under such conditions, at least two independent systems must be provided, preferably of different principles. Backup equipment (e.g., water and power systems) to such engineered safeguards must also be redundant.

Criterion 19

The maximum integrated leakage from the containment structure under the conditions described in Criterion 17 above must meet the site exposure criteria set forth in 10CFR100. The containment structure must be designed so that the containment can be leak tested at least to design pressure conditions after completion and installation of all penetrations, and the leakage rate measured over a suitable period to verify its conformance with required performance. The plant must be designed for later tests at suitable pressures.

Criterion 20

All containment structure penetrations subject to failure such as resilient seals and expansion bellows must be designed and constructed so that leak-tightness can be demonstrated at design pressure at any time throughout operating life of the reactor.

Criterion 21

Sufficient normal and emergency sources of electrical power must be provided to assure a capability for prompt shutdown and continued maintenance of the reactor facility in a safe condition under all credible circumstances.

Criterion 22

Valves and their associated apparatus that are essential to the containment function must be redundant and so arranged that no credible combination of circumstances can interfere with their necessary functioning. Such redundant valves and associated apparatus must be independent of each other. Capability must be provided for testing functional operability of these valves and associated equipment to determine that no failure has occurred and that leakage is within acceptable limits. Redundant valves and auxiliaries must be independent. Containment closure valves must be actuated by instrumentation, control circuits and energy sources which satisfy Criterion 15 and 16 above.

Criterion 23

In determining the suitability of a facility for a proposed site the acceptance of the inherent and engineered safety afforded by the systems, materials and components, and the associated engineered safeguards built into the facility, will depend on their demonstrated performance capability and reliability and the extent to which the operability of such systems, materials, components, and engineered safeguards can be tested and inspected during the life of the plant.

Radioactivity Control

Criterion 24

All fuel storage and waste handling systems must be contained if necessary to prevent the accidental release of radioactivity in amounts which could affect the health and safety of the public.

Criterion 25

The fuel handling and storage facilities must be designed to prevent criticality and to maintain adequate shielding and cooling for spent fuel under all anticipated normal and abnormal conditions, and credible accident conditions. Variables upon which health and safety of the public depend must be monitored.

Criterion 26

Where unfavorable environmental conditions can be expected to require limitations upon the release of operational radioactive effluents to the environment, appropriate holdup capacity must be provided for retention of gaseous, liquid, or solid effluents.

Criterion 27

The plant must be provided with systems capable of monitoring the release of radioactivity under accident conditions.

1.0 General

The Station is intended as a high load factor generating facility to be operated as an integral part of the NMPC system. The recirculation flow control system described in Section VIII contributes to this objective by providing a relatively fast means for adjusting the Station output over a preselected power range. Overall reliability, routine and periodic test requirements, and other design considerations must also be compatible with this objective.

Careful attention has been given to fabrication procedures and adherence to Code requirements. The rigid requirements of specific portions of various codes have been arbitrarily applied to some safety-related systems to ensure quality construction in such cases where the complete Code does not apply.

For piping, the ASA B31.1-1955 Code was used and where exceptions were taken, safety evaluations were performed to document that an adequate margin of safety was maintained.

Periodic test programs have been developed for required engineered safeguards equipment. These tests cover component testing such as pumps and valves and full system tests, duplicating as closely as possible the accident conditions under which a given system must perform.

2.0 Buildings and Structures

The Station plot plan, design and arrangement of the various buildings and structures are described in Section III. Principal structures and equipment which may serve either to prevent accidents or to mitigate their consequences are designed, fabricated and erected in accordance with applicable codes to withstand the most severe earthquake, flooding condition, windstorm, ice condition, temperature and other deleterious natural phenomena which can be expected to occur at the site.

3.0 Reactor

1. A direct-cycle boiling water system reactor (BWR), described in Section IV, is employed to produce steam (1030 psig in reactor vessel, 950 psig turbine inlet) for use in a steam-driven turbine generator. The rated thermal output of the reactor is 1850 MWt.

2. The reactor is fueled with slightly enriched uranium dioxide contained in Zircaloy clad fuel rods described in Section IV. Selected fuel rods also incorporate small amounts of gadolinium as burnable poison.

3. To avoid fuel damage, the minimum critical power ratio (MCPR) is maintained greater than or equal to the safety limit CPR.

4. The fuel rod cladding is designed to maintain its integrity throughout the anticipated fuel life as described in Section IV. Fission gas release within the rods and other factors affecting design life are considered for the maximum expected burnup.

5. The reactor and associated systems are designed so that there is no inherent tendency for undamped oscillations. A stability analysis evaluation is given in Section IV.

6. Heat removal systems are provided which are capable of safely accommodating core decay heat under all credible circumstances, including isolation from the main condenser and loss of coolant from the reactor. Each different system so provided has appropriate redundant features.

   Independent auxiliary cooling means are provided to cool the reactor under a variety of conditions. The normal auxiliary cooling means during shutdown and refueling is the shutdown cooling system described in Section X-A. A redundant emergency cooling system, described in Section V-E, is provided to remove decay heat in the event the reactor is isolated from the main condenser while still under pressure. Additional cooling capability is also available from the high-pressure coolant injection (HPCI) system and the fire protection system.

Redundant and independent core spray systems are provided to cool the core in the event of a loss-of-coolant accident (LOCA). Automatic depressurization is included to rapidly reduce pressure to assist with core spray operation (see Section VII-A).

Operation of the core spray system assures that any metal-water reaction following a postulated LOCA will be limited to less than 1 percent of the Zircaloy clad.

7. Reactivity shutdown capability is provided to make and hold the core adequately subcritical, by control rod action, from any point in the operating cycle and at any temperature down to room temperature, assuming that any one control rod is fully withdrawn and unavailable for use.

   This capability is demonstrated in Section IV-B. A physical description of the movable control rods is given in Section IV-B. The control rod drive (CRD) hydraulic system is described in Section X-C.

   The force available to scram a control rod is approximately 3000 lb at the beginning of a scram stroke. This is well in excess of the 440-lb force required in the event of fuel channel pinching of the control rod blade during a LOCA, as discussed in Section XV. Even with scram accumulator failure, a force of at least 1100 lb from reactor pressure acting alone is available with reactor pressures in excess of 800 psig.

8. Redundant reactivity shutdown capability is provided independent of normal reactivity control provisions. This system has the capability, as shown in Section VII-C, to bring the reactor to a cold shutdown condition, $K_{eff} < 0.97$, at any time in the core life, independent of the control rod system capabilities.

9. A flow restrictor in the main steam line (MSL) limits coolant loss from the reactor vessel in the event of a MSL break (Section VII-F).

4.0 Reactor Vessel

1. The reactor core and vessel are designed to accommodate tripping of the turbine generator, loss of power to the reactor recirculation system and other transients, and maneuvers which can be expected without compromising safety and without fuel damage.

   A bypass system having a design capacity of approximately 40 percent of turbine steam flow for the throttle valves wide open (VWO) condition partially mitigates the effects of sudden load rejection. An actual bypass system test was performed and the results indicated a system bypass capacity of about 2,500,000 lb/hr. This and other transients and maneuvers which have been analyzed are detailed in Section XV.

2. Separate systems to prevent serious reactor coolant system (RCS) overpressure are incorporated in the design. These include an overpressure scram, solenoid-actuated relief valves, safety valves and the turbine bypass system. An analysis of the adequacy of RCS pressure relief devices is included in Section V-C.

3. Power excursions which could result from any credible reactivity addition accident will not cause damage, either by motion or rupture, to the pressure vessel, or impair operation of required safeguards systems.

   The magnitude of credible reactivity addition accidents is curtailed by control rod velocity limiters (Section VII-D), by a control rod housing support structure (Section VII-E), and by procedural controls supplemented by a rod worth minimizer (RWM) (Section VIII-C). Power excursion analyses for control rod dropout accidents are included in Section XV.

4. The reactor vessel will not be substantially pressurized until the vessel wall temperature is in excess of the nil ductility reference temperature (RT_{NDT}) + 60°F. The initial RT_{NDT} of the reactor vessel material is no greater than 40°F. The change of RT_{NDT} with radiation exposure has been evaluated in accordance with Regulatory Guide (RG) 1.99 Revision 2 to determine an adjusted reference temperature (ART) for the most limiting vessel material. Vessel material surveillance samples are located within the reactor vessel to permit periodic verification of material properties with exposure.

5.0 Containment

1. The primary containment, including the drywell, pressure suppression chamber, and associated access

openings and penetrations, is designed, fabricated and erected to accommodate, without failure, the pressures and temperatures resulting from or subsequent to the double-ended rupture (DER) or equivalent failure of any coolant pipe within the drywell.

The primary containment is designed to accommodate the pressures following a LOCA, including the generation of hydrogen from a metal-water reaction. Pressure transients, including hydrogen effects, are presented in Section XV.

The initial NDTT for the primary containment system is about -20°F and is not expected to increase during the lifetime of the Station.

These structures are described in Sections VI-A, B and C. Additional details, particularly those related to design and fabrication, are included in Section XVI.

2. Provisions are made for the removal of heat from within the primary containment, for reasonable protection of the containment from fluid jets or missiles, and such other measures as may be necessary to maintain the integrity of the containment system as long as necessary following a LOCA.

   Redundant containment spray systems, described in Section VII, pump water from the suppression chamber through independent heat exchangers to spray nozzles, which discharge into the drywell and suppression chamber. Water sprayed into the drywell is returned by gravity to the suppression chamber to complete the cooling cycle. Studies performed to verify the capability of the containment system to withstand potential fluid jets and missiles are summarized in Section XVI.

3. Provision is made for periodic integrated leakage rate tests (ILRT) to be performed in accordance with 10CFR50 Appendix J. Provision is also made for leak testing penetrations and access openings and for periodically demonstrating the integrity of the reactor building. These provisions are all described in Section VI-F.

4. The containment system and all other necessary engineered safeguards are designed and maintained such that offsite doses resulting from postulated accidents are below the values stated in 10CFR100. The analysis results are detailed in Section XV.

5. Double isolation valves are provided on most lines directly entering the primary containment freespace, or penetrating the primary containment and connected to

the RCS. Lines which are not equipped with double isolation valves have been determined to be acceptable based upon the fact that the system reliability is not compromised, the system is closed outside containment, and a single active failure can be accommodated with only one isolation valve in the line. Periodic testing of these valves will assure their capability to isolate at all times. The isolation valve system is discussed in detail in Section VI-D.

6. The reactor building provides secondary containment when the pressure suppression system is in service and serves as the primary containment barrier during refueling and other periods when the pressure suppression system is open or not required. This structure is described in Section VI-C. An emergency ventilation system (Section VII-H) provides a means for controlled release of halogens and particulates via filters from the reactor building to the stack under accident conditions.

6.0 Control and Instrumentation

1. The Station is provided with a control room (Section III-B) which has adequate shielding and other emergency features to permit occupancy during all credible accident situations.

2. Interlocks or other protective features are provided to augment the reliability of procedural controls in preventing serious accidents.

   Interlock systems are provided which block or prevent rod withdrawal from a multitude of abnormal conditions. The control rod block logic is shown on Figures VIII-6 and VIII-8, respectively, for the source range monitor (SRM) and intermediate range monitor (IRM) neutron instrumentation. In the power range, average power range monitor (APRM) instrumentation provides both control rod and recirculation flow control blocks, as shown on Figure VIII-14.

   Reactivity excursions involving the control rods are either prevented or their consequences substantially mitigated by a control RWM (Section VIII-C.4.0) which supplements procedural controls in avoiding patterns of high rod worths, a local power range monitor (LPRM) neutron monitoring and alarm system (Section VIII-C.1.1.3), and a control rod position indicating system (Section IV-B.6.0), both of which enable the Operator to observe rod movement, thus verifying his actions. A control rod overtravel position light verifies that the blade is coupled to a withdrawn CRD.

A refueling platform operation interlock is discussed in Section XV, Refueling Accident, which, along with other procedures and supplemented by automatic interlocks, serves to prevent criticality accidents in the refueling mode.

A cold water addition reactivity excursion is prevented by the procedures and interlocks described in Section XV, Startup of Cold Recirculation Loop (Transient Analysis).

Containment integrity is maintained through the use of strict procedural controls and is enforced by interlocking mechanisms at the airlock doors to the drywell and a local alarm system at the access openings of the reactor building.

3. A reliable, dual-logic channel reactor protection system (RPS), described in Section VIII-A, is provided to automatically initiate appropriate action whenever various parameters exceed preset limits. Each logic channel contains two subchannels with completely independent sensors, each capable of tripping the logic channel. A trip of one-of-two subchannels in each logic channel results in a reactor scram. The trip in each logic channel may occur from unrelated parameters, i.e., high neutron flux in one logic channel coupled with high pressure in the other logic channel will result in a scram. The RPS circuitry fails in a direction to cause a reactor scram in the event of loss of power or loss of air supply to the scram solenoid valves. Periodic testing and calibration of individual subchannels is performed to assure system reliability. The ability of the RPS to safely terminate a variety of Station malfunctions is demonstrated in Section XV.

4. Redundant sensors and circuitry are provided for the actuation of equipment required to function under post-accident conditions. This redundancy is described in the various sections of the text discussing system design.

7.0 Electrical Power

Sufficient normal and standby auxiliary sources of electrical power are provided to assure a capability for prompt shutdown and continued maintenance of the Station in a safe condition under all credible circumstances. These features are discussed in Section IX.

8.0 Radioactive Waste Disposal

1. Gaseous, liquid and solid waste disposal facilities are designed so that discharge of effluents is in

accordance with 10CFR20 and 10CFR50 Appendix I. The facility descriptions are given in Section XII-A while the development of appropriate limits is covered in Section II.

2. Gaseous discharge from the Station is appropriately monitored, as discussed in Section VIII, and automatic isolation features are incorporated to maintain releases below the limits of 10CFR20 and 10CFR50 Appendix I.

9.0 Shielding and Access Control

Radiation shielding and access control patterns are such that doses will be less than those specified in 10CFR20. These features are described in Section XII-B.

10.0 Fuel Handling and Storage

Appropriate fuel handling and storage facilities which preclude accidental criticality and provide adequate cooling for spent fuel are described in Section X.

B. CHARACTERISTICS

The following is a summary of design and operating characteristics.

1.0 Site

Location	Oswego County, New York State
Size of Site	900 Acres
Site and Station Ownership	Niagara Mohawk Power Corporation
Net Electrical Output	615 MW (Maximum)

2.0 Reactor

Reference Rated Thermal Output	1850 MW
Dome Pressure	1030 psig
Turbine Inlet Pressure	950 psig
Total Core Coolant Flow Rate	67.5×10^6 lb/hr
Steam Flow Rate	7.32×10^6 lb/hr

3.0 Core

Circumscribed Core Diameter	167.16 in
Active Core Height + Assembly	171.125 in

4.0 Fuel Assembly

Number of Fuel Assemblies	532
Fuel Rod Array	SRLR[(2)]
Fuel Rod Pitch	Reference 3
Cladding Material	Reference 3
Fuel Material	UO_2 and UO_2-Gd_2O3
Active Fuel Length	Reference 3
Cladding Outside Diameter	Reference 3
Cladding Thickness	Reference 3
Fuel Channel Material	Reference 3

5.0 Control System

Number of Movable Control Rods	129
Shape of Movable Control Rods	Cruciform
Pitch of Movable Control Rods	12.0 in
Control Material in Movable Control Rods	B_4C - 70% Theoretical Density; Hafnium
Type of Control Drives	Bottom Entry, Hydraulic Actuated

Control of Reactor Output	Movement of Control Rods and Variation of Coolant Flow Rate

6.0 Core Design and Operating Conditions

Maximum Linear Heat Generation Rate	Core Operating Limits Report
Heat Transfer Surface Area	*
Average Heat Flux - Rated Power	*
Minimum Critical Power Ratio for Most Limiting Transients	Core Operating Limits Report
Core Average Void Fraction - Coolant within Assemblies	*
Core Average Exit Quality - Coolant within Assemblies	*

7.0 Design Power Peaking Factor

Total Peaking Factor	GE11	- 2.94**
		- 2.62***

8.0 Nuclear Design Data

Average Initial Volume Metric Enrichment	Reference 3
Beginning of Cycle 15 - Core Effective Multiplication and Control System Worth - No Voids, 20C(2)	
Uncontrolled	1.093
Fully Controlled	0.948
Strongest Control Rod Out	0.977

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* These parameters are recalculated for each reload because of their dependency on core composition and exposure. These calculated values are intermediate quantities that do not represent design requirements or operating limits and thus are not separately reported in the SRLR(2).

** Maximum total peaking factor for the portion of the bundle containing part length rods.

*** Maximum total peaking factor for the region above the part length rods.

Standby Liquid Control System Capability:

	ppm	Shutdown Margin (Δk) (20C, Xenon Free)
	SRLR(2)	SRLR(2)

9.0 Reactor Vessel

Inside Diameter	17 ft - 9 in
Internal Height	63 ft - 10 in
Design Pressure	1250 psig at 575°F

10.0 Coolant Recirculation Loops

Location of Recirculation Loops	Containment Drywell
Number of Recirculation Loops and Pumps	5
Pipe Size	28 in

11.0 Primary Containment

Type	Pressure Suppression
Design Pressure of Drywell Vessel	62 psig
Design Pressure of Suppression Chamber Vessel	35 psig
Design Leakage Rate	0.5 weight percent per day at 35 psig

12.0 Secondary Containment

Type	Reinforced concrete and steel superstructure with metal siding
Internal Design Pressure	40 lb/ft^2
Design Leakage Rate	100% free volume per day discharged via stack while maintaining 0.25-in water negative pressure in the reactor building relative to atmosphere

13.0 Structural Design

Seismic Ground Acceleration	0.11g
Sustained Wind Loading	125 mph, 30 ft above ground level
Control Room Shielding	Normal Operation - Dose not to exceed hourly equivalent (based on 40-hr week) of maximum permissible quarterly dose specified in 10CFR20.

	Accident Conditions - Meets the design gamma dose for personnel in the control room such that the exposure guidelines of 10CFR50 Appendix A, General Design Criteria (GDC) 19, will not be exceeded in the course of the LOCA. In addition, the cumulative dose from any design basis accident (DBA) would also meet GDC 19 limits.

14.0 Station Electrical System

Incoming Power Sources	Two 115-kV transmission lines
Outgoing Power Lines	Two 345-kV transmission lines
Onsite Power Sources Provided	Two diesel generators Two safety-related Station batteries One Q-related 125-V dc battery system

15.0 Reactor Instrumentation System

Location of Neutron Monitor Sensors	In-core

Ranges of Nuclear Instrumentation:

Four Startup Range Monitors	Source to 0.01% rated power and to 8.3% with chamber retraction
Eight Intermediate Range Monitors	0.0003% to 40% rated power
120 Power Range Monitors	5% to 125% rated power

16.0 Reactor Protection System

Number of Channels in Reactor Protection System	2
Number of Channels Required to Scram or Effect Other Protective Functions	2
Number of Sensors per Monitored Variable in each Channel (Minimum for scram function)	2

C. IDENTIFICATION OF CONTRACTORS

The General Electric Company (GE) was engaged to design, fabricate and deliver the nuclear steam supply system (NSSS), turbine generator, and other major elements and systems. GE also furnished the complete core design and nuclear fuel supply for the initial core. Global Nuclear Fuel (GNF) is currently furnishing replacement cores.

NMPC, acting as its own architect-engineer, specified and procured the remaining systems and components, including the pressure suppression containment system, and coordinated the complete integrated Station. Stone and Webster Engineering Corporation (SWEC) was engaged by NMPC to manage field construction. Currently, NMPC utilizes various contractors to assist in continuous Station modifications.

D. GENERAL CONCLUSIONS

The favorable site characteristics, criteria and design requirements of all the systems related to safety, the potential consequences of postulated accidents, and the technical competence of the applicant and its contractors, assure that Unit 1 can be operated without endangering the health and safety of the public.

E. REFERENCES

1. USAEC Press Release H-252, "General Design Criteria for Nuclear Power Plant Construction Permits," November 22, 1965.

2. GNF-J11-03785SRLR, Revision 0, "Supplemental Reload Licensing Report for NMP1, Reload 16, Cycle 15," February 2001.

3. GE Fuel Bundle Designs, General Electric Company Proprietary, NEDE-31152P, Revision 5, June 1996.

TABLE I-1

COMPARISON TO STANDARDS-HISTORICAL
(PROVIDED WITH APPLICATION TO CONVERT TO
FULL-TERM OPERATING LICENSE)

As part of its application to convert to a full-term operating license, NMPC provided an assessment of Unit 1 against criteria being used by the Commission in evaluating new plants. Construction of Unit 1 was well along or already completed when many of these standards were developed. These assessments discussed the adequacy of Unit 1 in relation to Appendices A through J of 10CFR50, Safety Guides 1 through 21, IEEE Standards, and Regulatory Guides 1.22 through 1.59. This historical information is located in the permanent plant file under the following submittals:

- Technical Supplement to Petition for Conversion From Provisional Operating License to Full-Term Operating License, July 1972

- Amendment No. 1 to Application to Convert Provisional Operating License to Full-Term Operating License, November 1973

The information provided in the above submittals does not represent a commitment to the standards.

PIPING, INSTRUMENT AND EQUIPMENT SYMBOLS

INSTRUMENT FUNCTION / MEASURED VARIABLE		CONTROLLING							MEASURING														SWITCH			ALARMS					
		RECORDING	INDICATING	NON-INDICATING	CONTROL VALVES	SAFETY-RELIEF VALVES	ORIFICE RESTRICTING		HIGH PRESS. CONN.	LOW PRESS. CONN.	RECORDING	INDICATING	OBSERVATION GLASS	PRIMARY ELEMENT	TEST POINT OR WELL	SET POINT	TRANSMITTER	INTEGRATOR	AMPLIFIER	SAMPLER		INDICATING	NON-INDICATING	ELECTRONIC	OPEN	CLOSED	ALARM	ALARM HIGH	ALARM LOW	POWER SUPPLY
		-RC	-IC	-C	-CV	-SV	O_R				-R	-I	-G	-E	-X		-T	-Q	$-A_M$	$-S_M$		-IS	-S	$-S_I$	$-A_O$	$-A_C$	-A	$-A_H$	$-A_L$	$-P_W$
CONDUCTIVITY	C	CRC	CIC								CR	CI		CE	CX					CS_M		CIS	CS				CA	CA_H	CA_L	
DIFF. PRESS.	DP	DPRC	DPIC	DPC							DPR	DPI					DPT					DPIS	DPS				DPA	DPA_H	DPA_L	
FLOW	F	FRC	FIC	FC	FCV		FO_R		K1	K2	FR	FI	FG	FE	FX		FT	FQ				FIS	FS				FA	FA_H	FA_L	
HUMIDITY	H	HRC	HIC								HR	HI		HE	HX															
PH	PH	PHRC	PHIC	PHC							pHR	pHI		pHE	pHX				pHA_M	pHS_M							pHA	pHA_H	pHA_L	
LEVEL	L	LRC	LIC	LC	LCV						LR	LI	LG	LE			LT					LIS	LS	LS_E			LA	LA_H	LA_L	
PRESSURE	P	PRC	PIC	PC	PCV	PSV	PO_R				PR	PI			PX		PT					PIS	PS	PS_E			PA	PA_H	PA_L	
POSITION	P_O										P_OR	P_OI					P_OT					POIS	POS		P A	P A				
RADIATION	R				RCV						RR	RI		RE	RX				RA_M	RS_M			RS				RA	RA_H		RP_W
TEMPERATURE	T	TRC	TIC	TC	TCV						TR	TI		TE	TX		TT					TIS	TS				TA	TA_H	TA_L	
VIBRATION	V_B										V_BR						V_BT		V_BA_M				V_BS				V_BA			
VACUUM	V_C	V_CRC	V_CIC	V_CC		V_CSV						V_CI					V_CT						V_CS				V_CA	V_CAH	V_CAL	
VISCOSITY	V_1																													
SAMPLE	S														SX					SS_M										
SPEC. GRAVITY	S_G	S_GRC																												
NEUTRON FLUX	N		NIC								NR	NI		NE	NX		NT	NQ	NA_M			NIS	NS				NA	NA_H	NA_L	NP_W
OXYGEN	O_2										O_2R	O_2I		O_2E																
SPEED	S_P	S_PRC	S_PIC	S_PC							S_PR	S_PI		S_PE																
DENSITY	D			DC										DE		DSP	DT													
HYDROGEN	H_2	H_2RC												H_2E													H_2A			
FREON	F_{12}										$F_{12}R$			$F_{12}E$																

SUFFIX SUB 'H' OR 'L' (HIGH, LOW)
SUFFIX SUB 'O' OR 'C' (OPEN, CLOSED)

VALVE SYMBOLS

- BALL
- GATE
- GLOBE
- CHECK (USUALLY SWING TYPE)
- LIFT CHECK
- STOP CHECK
- BUTTERFLY (CLOSE)
- BUTTERFLY (OPEN)
- ANGLE
- SAFETY OR RELIEF
- THREE WAY
- FOUR WAY
- 2 PORT COCK, PLUG OR BALL VALVE
- THREE-PORT COCK
- FOUR-PORT COCK
- FLOW CHECK VALVE
- SAUNDERS TYPE
- HISTORIC SYMBOL FOR GLOBE

LINE SYMBOLS

- PRIMARY FLOW
- SECONDARY FLOW
- INSTRUMENT PROCESS AIR
- INSTRUMENT CONTROL AIR
- INSTRUMENT ELECT. LEADS
- INSTRUMENT CAPILLARY TUBING
- CONNECTION
- CROSSING
- DRYWELL PENETRATION
- OFF GAS
- FLOW DIRECTION
- HEATED LINES
- BALL JOINT
- DIAPHRAGM SEAL

SPECIAL NOTE

L.O. -LOCKED OPEN
L.C. -LOCKED CLOSED
R.M. -REMOTE MANUAL(HANDWHEEL EXTENSION)
L.R. -LANTERN RING
B.S. -BELLOW SEAL
K.O. -KEY OPERATED
R.P.S. -REACTOR PROTECTION SYSTEM
* SUFFIX SUB 'H' OR 'L' MAY BE ADDED (HIGH, LOW)
SUFFIX SUB 'O' OR 'C' MAY BE ADDED (OPEN, CLOSED)

INSTRUMENT SYMBOLS

PR — SERVICE (SEE TABLE)
LOCAL MOUNTED INST.

PR — SERVICE (SEE TABLE)
PANEL MOUNTED INST.

dPT — SERVICE (SEE TABLE)
LOCAL MOUNTED TRANS.

FT — SERVICE (SEE TABLE)
PANEL MOUNTED TRANS.

RA — SERVICE (SEE TABLE)
ANNUNCIATOR ALARM
AUDIBLE AND/OR VISIBLE

E/P — ELECTRIC — PNEUMATIC
CONVERTER

CONTROL COMPONENTS

SYS. -TOTAL CONTROL SYSTEM
√ -SQUARE ROOT EXTRACTOR
Σ -SUMMER
X -MULTIPLIER
÷ -DIVIDER
f -FUNCTION GENERATOR
▷ -PROPORTIONAL AMPLIFIER
MV/I -MILLIVOLT/CURRENT TRANSMITTER
⊣⊢ -RELAY
G -INTEGRATOR

TA_H — SERVICE (SEE TABLE)
DATA HANDLING SYSTEM

CONTROL VALVE OPERATION

- FLOAT OPERATED
- MO / AC: M.O. -MOTOR OPERATED(AC OR DC); A.O. -AIR PISTON OPER.; H.O. -HYDRAULIC PISTON OPER.; P.O. -POWER OPER.
- DIAPHRAGM OPERATED
- PRESSURE CONTROLLER (SHOW FLOW DIRECTION)
- DIFF. PRESS. CONTROLLER (SHOW FLOW DIRECTION)
- DIAPHRAGM OPERATED WITH POSITIONER
- SOLENOID VALVE
- RMC: REMOTE MANUAL CONTROL (CONTROL ROOM) EXCEPT AS NOTED OTHERWISE
- RMC: REMOTE MANUAL CONTROL (LOCAL)

FLOW DEVICES

- FE — EQUIPMENT PIECE NO. — FLOW ELEMENT ORFICE
- FI — EQUIPMENT PIECE NO. — IN-LINE FLOW DEVICE
- FE — EQUIPMENT PIECE NO. — FLOW ELEMENT NOZZLE
- PILOT TUBE

PUMPS AND FANS

- CENTRIFUGAL PUMP
- RECIPROCATING UNIT PUMP COMPRESSOR
- FAN OR BLOWER
- ROTARY UNIT PUMP BLOWER COMPRESSOR

PIPING SYMBOLS

- STRAINER
- FILTER
- TRAP
- SEPARATOR
- OPEN VENT
- OPEN DRAIN
- REDUCER
- PLUGGED OR FLANGED END CAP
- SCREWED CAP
- WELDED CAP
- HEAT EXCHANGER
- FLANGED SPOOLPIECE
- 25KW — ELECTRIC HEATER
- SPRAY NOZZLE OR MIXING JET

VENTILATION DAMPERS

- SHUT-OFF
- THROTTLING
- BACK DRAFT
- MULTI-DAMPERS

VALVE POSITIONS

DURING USUAL PLANT OPERATION

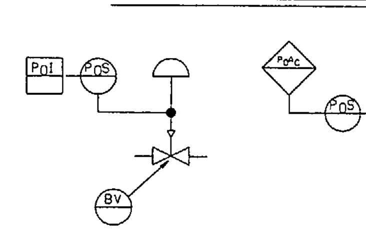
- OPEN
- CLOSED
- AIR SUPPLY / CLOSED PORT / NEXT TO CLOSE / VENT — 3 WAY SOLENOID VALVE, ALWAYS SHOWN DE-ENERGIZED

EXPANSION JOINTS

NOT SHOWN ON P&I WHEN USED ONLY FOR NORMAL FLEXIBILITY

- PIPE LOOP
- FLEXIBLE HOSE
- BELLOWS JOINT
- SLIP JOINT

CONTROL VALVE DESIGNATION



SERVICE (SEE TABLE OR LIST BELOW)

IV -DRYWELL ISOLATION VALVE
BV -BLOCKING VALVE

CONTROL VALVE OPERATION

USE APPLICATOR OPERATOR & VALVE SYMBOL

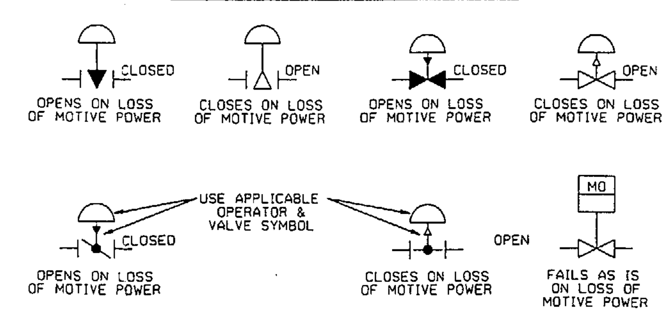


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FIGURE I - 1
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