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EISENHUT,D.G. Division of Licensing

SUBJECT: Forwards documentation of TMI action items requiring 810101 submittal,per 801031 request.Sixteen oversize drawings encl.
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NIAGARA MOHAWK POWER CORPORATION/300 ERIE BOULEVARD WEST, SYRACUSE, N.Y. 13202/TELEPHONE (315) 474-1511

December 31, 1980

Mr. Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Eisenhut:

Re: Nine Mile Point Unit 1
Docket No. 50-220
DPR-63

Attached is the requested documentation for those TMI items required to be submitted by January 1, 1981 in your October 31, 1980 letter. Also included is information for certain items which are planned to be implemented at Nine Mile Point Unit 1 before your schedule requires them.

Very truly yours,

NIAGARA MOHAWK POWER CORPORATION

Donald P. Dise

Donald P. Dise
Vice President Engineering

PEF:ja
Attachment

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TMI ACTION PLAN
ITEMS REQUIRING
A JANUARY 1, 1981
SUBMITTAL

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT UNIT 1

DOCKET NO. 50-220

DPR-63

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TMI ACTION PLAN ITEM NO. I.A.1.1

SHIFT TECHNICAL ADVISOR

NRC POSITION

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor (STA) may serve more than one unit at a multiunit site if qualified to perform the advisor function for the various units.

The STA shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The STA shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the STAs that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

RESPONSE

Since January 7, 1980, an Assistant Shift Supervisor (Shift Technical Advisor) has been added to the normal Control Room shift composition to be a non-shift technical advisor to the shift supervisor. The Assistant Shift Supervisors have a bachelor's degree or equivalent in a scientific or engineering discipline. The operations experience assessment function is performed by special meetings of the Site Operations Review Committee which are held at least once every two months. These meetings are attended by the Assistant Station Shift Supervisor as available.

Training which meets the lessons learned requirements has been completed (i.e. training in the response and analysis of the plant for transients and accidents and in plant design and layout, including the capabilities of instrumentation and controls in the control room).

Attachment 1 included at the end of this report provides a description of the current training program for the Nine Mile Point Unit 1 Shift Technical Advisors and the long term training requirements. Also included is a comparison of our training program to the draft INPO document entitled "Nuclear Power Plant Shift Technical Advisor - Recommendations for Position Description, Qualifications, Education and Training."

TMI ACTION PLAN ITEM NO. I.C.1

GUIDANCE FOR THE EVALUATION AND DEVELOPMENT OF PROCEDURES FOR TRANSIENTS AND ACCIDENTS

NRC's POSITION

In letters of September 13 and 27, October 10 and 30, and November 9, 1979, NRR required licensees of operating plants, operating license applicants, and licensees of plants under construction, to perform analyses of transients and accidents, prepare emergency procedure guidelines, upgrade emergency procedures, including procedures for operating with natural circulation conditions, and to conduct operator retraining (see also Item I.A.2.1). Emergency procedures are required to be consistent with the actions necessary to cope with the transients and accidents analyzed. Analyses of transients and accidents were to be completed in early 1980 and implementation of procedures and retraining were to be completed three months after emergency procedure guidelines were established; however, some difficulty in completing these requirements has been experienced. Clarification of the scope of the task and appropriate schedule revisions are being developed. In the course of review of these matters on B&W designed plants, the staff will follow up on the Bulletin and Orders matters relating to analysis methods and results, as listed in NUREG-0660 Appendix C. See Table C.1, Items 3, 4, 16, 18, 24, 25, 26, 27; Table C.2, Items 4, 12, 17, 18, 19, 20; and Table C.3, Items 6, 35, 37, 38, 39, 41, 42, 47, 55, 57.

RESPONSE

Niagara Mohawk has met the requirements of Item I.C.1 to perform revised analyses and prepare emergency procedure guidelines by January 1, 1981 through the BWR Owners Group by submittal of the following documents:

1. "Additional Information Required for NRC Staff Generic Report On Boiling Water Reactors", NEDO-24708, August 1979.

2. Section 3.2.1 (Revised) of NEDO-24708, "Analysis of Loss of Feedwater Events" transmitted by R.H. Buchholz's letter to D.F. Ross dated March 31, 1980.

3. BWR Emergency Procedure Guidelines Revision 0 (Prepublication form), transmitted by R.H. Buccholz's letter to D.G. Eisenhut dated June 30, 1980.

4. Section 3.2.2 of NEDO-24708, "Other Operational Transients" transmitted by R.H. Buccholz's letter to D.G. Eisenhut dated August 22, 1980.

5. Section 3.5.2.1 (Revised) of NEDO-24708, "Analysis to Demonstrate Adequate Core Cooling" and Section 3.5.2.4 of NEDO-24708, "Justification of Analysis Methods", transmitted by R.H. Buccholz's letter to D.G. Eisenhut dated September 16, 1980.

Upgrading of emergency procedures and operator training will be completed as required following the Nuclear Regulatory Commission's staff review of the BWR Emergency Procedure Guidelines.

TMI ACTION PLAN ITEM NO. I.C.5

PROCEDURES FOR FEEDBACK OF OPERATING EXPERIENCE TO PLANT STAFF

NRC POSITION

In accordance with Task Action Plan I.C.5, Procedures for Feedback of Operating Experience to Plant Staff (NUREG-0660), each applicant for an operating license shall prepare procedures to assure that operating information pertinent to plant safety originating both within and outside the utility organization is continually supplied to operators and other personnel and is incorporated into training and retraining programs. These procedures shall:

(1) Clearly identify organizational responsibilities for review of operating experience, the feedback of pertinent information to operators and other personnel, and the incorporation of such information into training and retraining programs;

(2) Identify the administrative and technical review steps necessary in translating recommendations by the operating experience assessment group into plant actions (e.g., changes to procedures; operating orders);

(3) Identify the recipients of various categories of information from operating experience (i.e., supervisory personnel, shift technical advisors, operators, maintenance personnel, health physics technicians) or otherwise provide means through which such information can be readily related to the job functions of the recipients;

(4) Provide means to assure that affected personnel become aware of and understand information of sufficient importance that should not wait for emphasis through routine training and retraining programs;

(5) Assure that plant personnel do not routinely receive extraneous and unimportant information on operating experience in such volume that it would obscure priority information or otherwise detract from overall job performance and proficiency;

(6) Provide suitable checks to assure that conflicting or contradictory information is not conveyed to operators and other personnel until resolution is reached; and,

(7) Provide periodic internal audit to assure that the feedback program functions effectively at all levels.

Response

A procedure has been issued and is available for review by the Nuclear Regulatory Commission at the Nine Mile Point Unit 1 site.

TMI ACTION PLAN ITEM NO. I.C.6

GUIDANCE ON PROCEDURES FOR VERIFYING CORRECT PERFORMANCE OF OPERATING ACTIVITIES

NRC POSITION

It is required (from NUREG-0660) that licensees' procedures be reviewed and revised, as necessary, to assure that an effective system of verifying the correct performance of operating activities is provided as a means of reducing human errors and improving the quality of normal operations. This will reduce the frequency of occurrence of situations that could result in or contribute to accidents. Such a verification system may include automatic system status monitoring, human verification of operations and maintenance activities independent of the people performing the activity (see NUREG-0585, Recommendation 5), or both.

Implementation of automatic status monitoring if required will reduce the extent of human verification of operations and maintenance activities but will not eliminate the need for such verification in all instances. The procedures adopted by the licensees may consist of two phases--one before and one after installation of automatic status monitoring equipment, if required, in accordance with item I.D.3.

Response

Procedures have been revised and are available for review by the Nuclear Regulatory Commission at the Nine Mile Point Unit 1 site.

TMI ACTION PLAN ITEM NO. II.B.2

DESIGN REVIEW OF PLANT SHIELDING AND ENVIRONMENTAL QUALIFICATION OF EQUIPMENT FOR SPACES/SYSTEMS WHICH MAY BE USED IN POST ACCIDENT OPERATIONS

NRC POSITION

With the assumption of a postaccident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core radioiodine, 100% of the core noble gas inventory, and 1% of the core solids are contained in the primary coolant), each licensee shall perform a radiation and shielding-design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during postaccident operations of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increases permanent or temporary shielding, or postaccident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

RESPONSE

Niagara Mohawk's submittals of December 31, 1979, January 31, 1980, June 20, 1980 and September 17, 1980 provided the results of the plant shielding design review and proposed modifications for Nine Mile Point Unit 1.

The shielding design review for Nine Mile Point Unit 1 did not include analyses of LOCA events in which the primary system remains pressurized. This is because the plant design and the emergency procedures would lead to depressurization of the system and injection of low pressure cooling water before fuel failures and resulting fission product releases would occur.

TMI ACTION PLAN ITEM NO. II.B.4

TRAINING FOR MITIGATING CORE DAMAGE

NRC POSITION

Licensees are required to develop a training program to teach the use of installed equipment and systems to control or mitigate accidents in which the core is severely damaged. They must then implement the training program.

RESPONSE

As indicated in our letter of December 17, 1980, the training program will not be available for submittal to the Nuclear Regulatory staff until April 1, 1981. Due to the spring 1981 refueling outage implementation of the training program will not begin until after the outage. The training will be completed by December 31, 1981.

The shift technical advisors and all operating personnel who held senior reactor operator or reactor operator licenses from the plant manager through the operations chain to the shift operators will participate in this training program. Managers and technicians in the Instrumentation and Control (I&C), health physics and chemistry departments will receive training commensurate with their responsibilities.

TMI ACTION PLAN ITEM NO. II.E.4.2

CONTAINMENT ISOLATION DEPENDABAILITY

NRC POSITION

(1) Containment isolation system designs shall comply with the recommendations of Standard Review Plan Section 6.2.4 (i.e., that there be diversity in the parameters senses for the initiation of containment isolation).

(2) All plant personnel shall give careful consideration to the definition of essential and nonessential systems, identify each system determined to be essential, identify each system determined to be nonessential, describe the basis for selection of each essential system, modify their containment isolation designs accordingly, and report the results of the reevaluation to the NRC.

(3) All nonessential systems shall be automatically isolated by the containment isolation signal.

(4) The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

(5) The containment setpoint pressure that initiates containment isolation for nonessential penetrations must be reduced to the minimum compatible with normal operating conditions.

(6) Containment purge valves that do not satisfy the operability criteria set forth in Branch Technical Position CSB 6-4 or the Staff Interim Position of October 23, 1979 must be sealed closed as defined in SRP 6.2.4, item II.3.f during operational conditions 1, 2, 3, and 4. Furthermore, these valves must be verified to be closed at least every 31 days. (A copy of the Staff Interim Position is enclosed as Attachment 1).

(7) Containment purge and vent isolation valves must close on a high radiation signal.

RESPONSE

Items 1, 2, 3 and 4 of the Nuclear Regulatory Commission's POSITION above were addressed in Niagara Mohawk's December 31, 1979 submittal which documented our review of the containment isolation provisions of Nine Mile Point Unit 1. Deviations from the NRC's documented position were identified and justified. Based upon this justification, no modifications were proposed for the Nine Mile Point Unit 1. The deviations and the justification are identified below:

1. The Main Steam (including warm-up and emergency cooling vents), Reactor Cleanup and Shutdown Cooling lines isolate on the low low reactor vessel water level containment isolation signal. Isolation of these systems is not initiated on high drywell pressure because they are closed systems capable of handling radioactivity levels associated with normal operation. Abnormally high levels of radioactivity could result from fuel damage caused by the reactor water level dropping below the top of the fuel. Isolation of these systems would occur at low low water level setpoint which is approximately 7 feet 6 3/4 inches above the top of fuel. Therefore, isolation would occur before any fuel failures.

2. The Drywell and Suppression Chamber N_2 Make up and H_2-O_2 sampling and Containment Airborne Activity Monitor Systems isolate on a containment isolation signal (i.e. low low water level or high drywell pressure). These systems are provided with overrides so that they can be manually reopened for controlled venting and purging and monitoring purposes.

3. The Reactor Building Closed Loop Cooling to the recirculation pump coolers and drywell coolers are non-essential systems which do not automatically isolate on containment isolation signals. These are closed systems inside the drywell and are not connected to the reactor coolant pressure boundary or open to the free space of containment. They provide cooling to the non-safety related pump motors and drywell coolers which, although not required to mitigate the consequences of an accident, are beneficial if they continued to operate. The supply lines are provided with a self-actuating check valve and the return lines are provided with a blocking valve which can be remotely closed from the control room. In addition, the supply line to the recirculation pump coolers has a blocking valve inside containment which can be remotely operated in the Reactor Building. Therefore, these systems can be isolated if high radiation leakage into the systems occurs. Remote manual isolation of the return line is acceptable based on General Design Criterion 57 - Closed System Isolation Valves which states that "Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic or locked closed or capable of remote manual operation." The use of check valves as the automatic isolation valves outside containment as in the supply lines discussed above has been previously justified in Niagara Mohawk's Technical Supplement to Petition for Conversion from Provisional Operating License to Full-Term Operating License.

4. The Atmosphere to Suppression Chamber Vacuum Relief line contains an air operated/DC solenoid valve and a self-activating check valve. The air operated/DC solenoid valve is a normally closed valve which will open on a negative pressure relative to atmosphere. The air operated/DC solenoid valve does not receive the automatic containment isolation signals. This is considered acceptable since this valve would not be normally opened and the self-activating check valve will prevent flow from the torus to atmosphere.

   The Suppression Chamber Water Makeup line has a diaphragm operated DC solenoid valve and a self-actuating check valve. The diaphragm operated DC solenoid valve is a normally closed valve which is remotely operable from the Control Room. Although not identified as a deviation from the NRC's position in our December 31, 1979 submittal, this isolation valve does not receive an automatic containment isolation signal. This is considered acceptable since this valve would not be normally opened but may be required to be opened to provide make up to the torus during an accident in which the containment is isolated. The self-actuating check valve will prevent flow from the torus out the make-up line.

In addition to the above a review of the isolation of lines penetrating the primary containment at Nine Mile Point Unit 1 has been performed in accordance with the requirements of General Design Criteria (GDC) 55, 56 and 57. Table 1 attached at the end of this report is a penetration by penetration listing of all lines penetrating containment. The last column in the table documents the compliance with GDC 55, 56 or 57. As a result of this review, the following modifications/changes have been recommended.

1. The service water and breathing air connections for the drywell (penetrations X-122 and 121 respectively) will have the inside manual valve changed to a normally locked closed valve. This will be performed prior to start up from the spring 1981 refueling outage.

2. The recirculation system sample line and the containment spray test line to waste disposal will be provided with automatic isolation valves. As indicated in our response to I. E. Bulletin 79-08, these modifications will be performed during the spring 1981 refueling outage.

The existing containment isolation pressure setpoint of 3.5 psig will continue to be used for initiating containment isolation. This is 1.3 psi higher than the maximum observed pressure inside containment over the past year during normal operation. The existing containment isolation pressure setpoint is not significantly different from the setpoint recommended by Nuclear Regulatory Commission to warrant a Technical Specification change. Therefore, Technical Specification changes reflecting a change in containment isolation pressure setpoint will not be submitted.

As indicated in our letter of December 17, 1979, the containment vent and purge lines at Nine Mile Point Unit 1 meets the Staff's Interim Position of October 23, 1979 by limiting the outboard isolation valves to 50 degrees maximum opening. Therefore, no further action is required on this item at this time.

Isolation of the containment purge and vent valves on high radiation will be provided by January 1, 1982. The isolation signal will be provided by the containment high range radiation monitors which are to be installed by January 1, 1982. Technical Specification changes reflecting this addition will be submitted by June 30, 1981 for Nuclear Regulatory Commission staff approval.

[illegible]

TMI ACTION PLAN ITEM NO. II.F.1.2

SAMPLING AND ANALYSIS

OF PLANT EFFLUENTS

NRC POSITION

Because iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by adsorption on charcoal or other media, followed by onsite laboratory analysis.

RESPONSE

Niagara Mohawk's submittal of December 17, 1980 indicated that procedures for the removal and analysis of samples would be reviewed and revised by January 1, 1981. Although procedures have been drafted, the procedures will not be approved and issued until January 31, 1981.

The sampling of radioiodine and particulate is performed by removing the charcoal canister (for iodine) and particulate filter located in the sample line to the stack monitor followed by analysis in the lab. This method of sampling is considered to be a continuous sampling method as a new charcoal canister and particulate filter will be installed when the others are removed for analysis.

By January 1, 1982 the sample holder for the charcoal canister and particulate filter will be modified so that highly radioactive samples can be removed.

TMI ACTION ITEM NO. II.F.1.3

CONTAINMENT HIGH-RANGE RADIATION MONITOR

NRC POSITION

In containment radiation-level monitors with a maximum range of 10^8 rad/hr shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be developed and qualified to function in an accident environment.

RESPONSE

Niagara Mohawk plans to install two independent containment high-range radiation monitors during the spring 1981 refueling outage, but no later than January 1, 1982. The monitors will be installed in existing spare penetration sleeves which will extend into the free space of the containment such that the entire active portion of the detector will be inside the containment. The penetrations maximum thickness is approximately 1/4 inch. Locating the monitors in penetration sleeves will increase the reliability of the monitors making them accessible for replacement, maintenance and calibration, while providing assessment of area radiation conditions inside containment.

The purpose of these monitors is to detect gross fuel failure. The proposed in-sleeve arrangement will perform this function since early indicators of fuel damage have high energy gammas (such as Xe-138 (2 MEV), Kr-87 (2.5 MEV), Kr-88 (2.4 MEV)). Although NUREG 0737 indicates that these monitors should respond to energies as low as 0.060 MEV, low energy isotopes (such as Xe-133) are not significant as outlined below.

1. Higher energy isotopes will dominate the fission product mix until they have decayed significantly. Low energy isotopes will not be a major constituent in the fission product mix until this occurs (i.e., several days after an accident).

2. Since low energy isotopes will not require monitoring until several days after an accident drywell sampling can be utilized.

3. Low energy isotopes such as Xe-133 are of little biological consequence because of their low energies and low gamma abundance.

The monitors to be installed at Nine Mile Pont Unit 1 are Model No. RD-23 Gamma Detector and RP-2C high-range radiation readout module signal processor supplied by the General Atomic Company.

The gamma detector has a range of 10^0 to 10^8 R/hr. It has been environmentally qualified to withstand 350^oF, 70 psig and 0% to 100% humidity and seismically qualified to IEEE 344-1975.

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The high-range radiation readout module signal processor has:

1) an accuracy of 1-3% of equivalent linear full scale output

2) an electronics temperature coefficient less than 0.1% of full scale per degree celsius

3) electronics calibration via internal current source corresponding to 10^5 R/hr

4) a time constant (RC) less than 25 milliseconds

5) two level trip alarms which are independently adjustable over the full range

6) environmentally qualified to withstand 130°F, 0% to 95% humidity

7) seismically qualified to IEEE 344-1975

The calibration required by Table II.F.1.3 will be performed prior to initial use of the monitors.

Figure 1 attached is a drawing showing the location of the monitors.

TMI ACTION PLAN ITEM NO. II.F.1.4

CONTAINMENT PRESSURE MONITOR

NRC POSITION

A continuous indication of containment pressure shall be provided in the control room of each operating reactor. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and -5 psig for all containments.

RESPONSE

During the Spring, 1981 refueling outage at Nine Mile Point Unit 1, but no later than January 1, 1982, two pressure transmitters will be installed on separate existing drywell pressure manifold taps. The measurement and indication capability of this instrumentation is from -5 psig to 250 psig, which is greater than four times the containment design pressure of 62 psig.

The pressure transmitters and instrument cable were procured in accordance with the quality assurance requirements of Appendix B to 10CFR50 for safety related equipment. Except as indicated below, the requirements of the draft of Regulatory Guide 1.97, Revision 2, were specified. The pressure transmitters are seismically qualified in accordance with IEEE 344-1971, and environmentally qualified in accordance with IEEE 323-1971. This was the best available transmitter at the time the equipment was purchased. The instrument cable is qualified in accordance with the latest edition of IEEE 383.

Continuous indication and recording over the range specified above will be provided in the control room for each pressure transmitter. Non-safety related indicating meters and safety related recorders with seismic qualification to IEEE 344-1971 were procured for this modification. This was the best available equipment which could be obtained at the time our equipment was purchased.

The pressure transmitters have an accuracy of $\pm$.25% of the calibrated span (-5 to 250 psig) and a response time of 0.2 seconds at 100°F.

The instrumentation will be powered from a vital instrument bus which will provide an emergency power supply for the containment pressure transmitters, indicators and recorders.

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TMI ACTION PLAN ITEM NO. II.F.1.5

CONTAINMENT WATER LEVEL MONITOR

NRC POSITION

A continuous indication of containment water level shall be provided in the control room for all plants. A narrow range instrument shall be provided for PWR's and cover the range from the bottom to the top of the containment sump. A wide range instrument shall also be provided for PWR's and shall cover the range from the bottom of the containment to the elevation equivalent to a 600,000 gallon capacity. For BWR's, a wide range instrument shall be provided and cover the range from the bottom to 5 feet above the normal water level of the suppression pool.

RESPONSE

During the Spring, 1981 refueling outage at Nine Mile Point Unit 1, but no later than January 1, 1982, redundant differential pressure transmitters will be installed on the existing separate instrument penetration on the torus. This instrumentation will provide measurement and indication of water level in the suppression pool from 15 inches above the bottom of the torus to 3 feet, 8.5 inches above the normal water level of the suppression pool. The lower limit is considered adequate, as this is 3 feet, 3 inches below the lowest emergency core cooling system suction line inlet. This is consistent with Mr. D. G. Eisenhut's clarification letter of October 31, 1980. The upper limit is considered adequate since it provides indication above the minimum normal water level of the suppression pool so that the operators know they have sufficient water inventory in the torus.

The differential pressure transmitters and instrument cable were procured in accordance with the quality assurance requirements of Appendix B to 10CFR50 for safety related equipment. Except as indicated below, the requirements of the draft of Regulatory Guide 1.97, Revision 2, were specified. The differential pressure transmitters are seismically qualified in accordance with IEEE 344-1971, and environmentally qualified in accordance with IEEE 323-1971. This was the best available transmitter at the time the equipment was purchased. The instrument cable is qualified in accordance with the latest edition of IEEE-383.

TMI Action Plan Item No. II.F.1.5 (Continued)

RESPONSE (Continued)

Continuous indication and recording of the suppression pool water level over the range specified above will be provided in the control room for each transmitter. Non-safety related indicating meters and safety related recorders with seismic qualification to IEEE-344 - 1971 were procured for this modification. This was the best available equipment at the time our equipment was purchased.

The instrumentation will be powered from a vital instrument bus which will provide an emergency power supply for the containment differential pressure transmitters, indicators and recorders.

The suppression pool water level transmitters have an accuracy of $\pm$.25% of the calibrated span and a response time of 0.2 seconds at $100^{o}F$.

TMI ACTION PLAN ITEM NO. II.K.3.3

REPORTING SAFETY AND RELIEF VALVE FAILURES AND CHALLENGES

NRC POSITION

(a) Future failures of a relief valve to close should be reported promptly to the NRC.

(b) Future challenges to the relief valves should be documented in the annual report.

(c) Future failures of a safety valve to close should be reported promptly to the NRC.

(d) Future challenges to the safety valves should be documented in the annual report.

RESPONSE

As indicated in our letters of June 20, 1980 and December 17, 1980, Niagara Mohawk will promptly report to the Nuclear Regulatory Commission all failures of safety and relief valves. The requirement for promptly reporting failures of relief/safety valves to close is documented in a procedure which is available for review at the Nine Mile Point Unit 1 site. Challenges to safety and relief valves will be documented in the monthly operating reports since annual reports are no longer required per changes to the Nuclear Regulatory Commission reporting requirements.

TMI ACTION PLAN ITEM NO. II.K.3.13

SEPARATION OF HIGH-PRESSURE COOLANT INJECTION AND REACTOR CORE ISOLATION COOLING SYSTEM INITIATION LEVELS - ANALYSIS AND IMPLEMENTATION

NRC POSITION

Currently, the reactor core isolation cooling (RCIC) system and the high-pressure coolant injection (HPCI) system both initiate on the same low-water-level signal and both isolate on the same high-water-level signal. The HPCI system will restart on low water level but the RCIC system will not. The RCIC system is a low-flow system when compared to the HPCI system. The initiation levels of the HPCI and RCIC systems should be separated so that the RCIC system initiates at a higher water level than the HPCI system. Further, the initiation logic of the RCIC system should be modified so that the RCIC system will restart on low water level. These changes have the potential to reduce the number of challenges to the HPCI system and could result in less stress on the vessel from cold water injection. Analyses should be performed to evaluate these changes. The analyses should be submitted to the NRC staff and changes should be implemented if justified by the analyses.

RESPONSE

As indicated in our letters of June 20, 1980 and December 17, 1980, this item is not directly applicable to Nine Mile Point Unit 1 because it pertains to boiling water reactors with RCIC and HPCI systems. Nine Mile Point Unit 1 does not have a RCIC system. However, the Nine Mile Point Unit 1 design does include a comparable Emergency Condenser and a HPCI system which utilizes the Feedwater system. These two systems currently initiate on different water level signals. The HPCI system initiates on low water level or automatic turbine trip, and the Emergency Condenser system initiates on low-low water level or high reactor pressure. Once placed in operation, the Emergency Condenser system will remain in operation until it is manually stopped. If manually stopped, the Emergency Condenser system will restart on receipt of one of its initiation signals. Therefore, this item requires no further action for Nine Mile Point Unit 1.

TMI ACTION PLAN ITEM NO. II.K.3.14

ISOLATION OF ISOLATION CONDENSERS ON HIGH RADIATION

NRC POSITION

Isolation condensers have radiation monitors on their vents. These monitors provide alarms in the control room but do not isolate the isolation condenser. The isolation condensers are currently isolated on a high-radiation signal in the steam line leading to the isolation condensers. The design should be modified such that the isolation condensers are automatically isolated upon receipt of a high-radiation signal at the vent rather than at the steam line. The purpose of the change is to increase the availability of the isolation condensers as heat sinks.

RESPONSE

As indicated in our letters of June 20, 1980 and December 17, 1980, the original design of the Emergency (Isolation) Condensers of Nine Mile Point Unit 1 provides automatic isolation of the system on a high radiation signal in the shell side vents. The Emergency Condenser system at Nine Mile Point Unit 1 does not automatically isolate on a high radiation signal in the steam line leading to the emergency (isolation) condensers. Therefore, this item requires no further action for Nine Mile Point Unit 1.

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TMI ACTION PLAN ITEM NO. II.K.3.15

MODIFY BREAK DETECTION LOGIC TO PREVENT SPURIOUS ISOLATION OF HIGH PRESSURE COOLANT INJECTION AND REACTOR CORE ISOLATION COOLING

NRC POSITION

The high-pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems use differential pressure sensors on elbow taps in the steam lines to their turbine drives to detect and isolate pipe breaks in the systems. The pipe-break-detection circuitry has resulted in spurious isolation of the HPCI and RCIC systems due to the pressure spike which accompanies startup of the systems. The pipe-break-detection circuitry should be modified so that pressure spikes resulting from HPCI and RCIC system initiation will not cause inadvertent system isolation.

RESPONSE

As indicated in our letters of June 20, 1980 and December 17, 1980, this item is not directly applicable to Nine Mile Point Unit 1 because it pertains to boiling water reactors with steam driven turbines in RCIC and HPCI systems. The Nine Mile Point Unit 1 design does not have RCIC and HPCI systems with steam driven turbines. Therefore, this item requires no further action for Nine Mile Point Unit 1.

TMI ACTION PLAN ITEM NO. II.K.3.17

REPORT ON OUTAGES OF EMERGENCY CORE - COOLING SYSTEMS LICENSEE REPORT AND PROPOSED TECHNICAL SPECIFICATION CHANGES

NRC POSITION

Several Components of the emergency core-cooling (ECC) systems are permitted by technical specifications to have substantial outage times (e.g., 72 hours for one diesel-generator; 14 days for the HPCI system). In addition, there are no cumulative outage time limitations for ECC systems. Licensees should submit a report detailing outage dates and lengths of outages for all ECC systems for the last 5 years of operation. The report should also include the causes of the outages (i.e., controller failure, spurious isolation).

Response

This requirement was met by our letter dated October 8, 1980 from Mr. D. P. Dise to Mr. D. G. Eisenhut. This letter transmitted a report with the information requested above. Included in the report were outage dates and duration of the outage, cause of the outage, system and component involved and the corrective action taken.

Niagara Mohawk did not propose any changes to improve the availability of emergency core cooling equipment since the number of outages was not significant enough to warrant changes.

TMI ACTION PLAN ITEM NO. II.K.3.19

INTERLOCK ON RECIRCULATION PUMP LOOPS

NRC POSITION

Interlocks should be installed on nonjet pump plants (other than Humboldt Bay) to assure that at least two recirculation loops are open for recirculation flow for modes other than cold shutdown. This is to assure that the level measurements in the downcomer region are representative of the level in the core region.

RESPONSE

As indicated in our letters of June 20, 1980 and December 17, 1980, Niagara Mohawk currently has administrative controls and Technical Specification requirements at Nine Mile Point Unit 1 to assure that at least two recirculation loops are open for recirculation flow for all operating modes other than cold shutdown. Therefore, no further action on this item is required at Nine Mile Point Unit 1.

TMI ACTION PLAN ITEM NO. II.K.3.21

RESTART OF CORE SPRAY AND LOW-PRESSURE

COOLANT INJECTION SYSTEMS

NRC POSITION

The core spray and low pressure coolant injection (LPCI) system flow may be stopped by the operator. These systems will not restart automatically on loss of water level if an initiation signal is still present. The core spray and LPCI system logic should be modified so that these systems will restart, if required, to assure adequate core cooling. Because this design modification affects several core cooling modes under accident conditions, a preliminary design should be submitted for staff review and approval prior to making the actual modification.

RESPONSE

The Nine Mile Point Unit 1 design includes two Low Pressure Core Spray systems, but not a separate Low Pressure Coolant Injection System. As indicated in our letters of June 20, 1980 and December 17, 1980, the core spray pumps will automatically restart following a manual stop upon receipt of a low-low water level or high drywell pressure signal (LOCA) or if one or both of the signals is still present. Although it is possible to place the core spray pump switches in the locked out mode, Niagara Mohawk does not believe that modification of the core spray system logic is required at Nine Mile Point Unit 1.

This position is also set forth in the study performed by General Electric for the BWR Owner's Group. This report titled NUREG 0737 Item II.K.3.21 Core Spray and Low Pressure Coolant Inspection Systems Level Initiation was transmitted to the NRC by a letter dated December 29, 1980 from Mr. D. B. Waters, Chairman TMI BWR Owners Group to Mr. D. G. Eisenhut. Although this report is generic, the discussions regarding non-jet pump plants and the low pressure core spray system are applicable to Nine Mile Point Unit 1.

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TMI ACTION PLAN ITEM NO. II.K.3.22

AUTOMATIC SWITCHOVER OF REACTOR CORE ISOLATION

COOLING SYSTEM SUCTION - VERIFY PROCEDURES AND MODIFY DESIGN

NRC POSITION

The reactor core isolation cooling (RCIC) system takes suction from the condensate storage tank with manual switchover to the suppression pool when the condensate storage tank level is low. This switchover should be made automatically. Until the automatic switchover is implemented, licenses should verify that clear and cogent procedures exist for the manual switchover of the RCIC system suction from the condensate storage tank to the suppression pool.

RESPONSE

As indicated in Niagara Mohawk's letters of June 20, 1980 and December 17, 1980, this item is not applicable to Nine Mile Point Unit 1. Nine Mile Point Unit 1 has a gravity fed closed loop Emergency Condenser system instead of a reactor core isolation cooling system.

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TMI ACTION PLAN ITEM NO. II.K.3.24

CONFIRM ADEQUACY OF SPACE COOLING FOR HIGH-PRESSURE COOLANT INJECTION AND REACTOR CORE ISOLATION COOLING SYSTEMS

NRC POSITION

Long-term operation of the reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI) systems may require space cooling to maintain the pump-room temperatures within allowable limits. Licensees should verify the acceptability of the consequences of a complete loss of alternating-current power. The RCIC and HPCI systems should be designed to withstand a complete loss of offsite alternating-current power to their support systems, including coolers, for at least 2 hours.

RESPONSE

As indicated in Niagara Mohawk's letter of June 20, 1980 and December 17, 1980, no action on this item is required for Nine Mile Point Unit 1. The Nine Mile Point Unit 1 design does not include HPCI and RCIC systems with pump rooms which require space cooling to maintain temperatures within allowable limits.

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TMI ACTION PLAN ITEM NO. II.K.3.25

EFFECT OF LOSS OF ALTERNATING CURRENT POWER ON PUMP SEALS

NRC POSITION

The licensees should determine, on a plant-specific basis, by analysis or experiment, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The pump seals should be designed to withstand a complete loss of alternating-current (ac) power for at least 2 hours. Adequacy of the seal design should be demonstrated.

RESPONSE

As indicated in our letter of December 17, 1980, the original design of Nine Mile Point Unit 1 includes supplying emergency power to the components which provide cooling water to the reactor recirculation pump seal coolers, thus precluding damage to the seals as a result of a loss of offsite AC power. Therefore, no analysis or further action is required on this item for Nine Mile Point Unit 1.

TMI ACTION PLAN ITEM NO. II.K.3.27

PROVIDE COMMON REFERENCE LEVEL

FOR VESSEL LEVEL INSTRUMENTATION

NRC POSITION

Different reference points of the various reactor vessel water level instruments may cause operator confusion. Therefore, all level instruments should be referenced to the same point. Either the bottom of the vessel or the top of the active fuel are reasonable reference points.

RESPONSE

Technical Specification changes which reference the existing reactor vessel water level instruments to the same point (65 inches below the minimum normal water level at elevation 302 feet 9 inches) were submitted on August 5, 1980 for NRC review and approval. The water level instrumentation being installed to meet the requirements of TMI Action Plan Item No. II.F.2 "Instrumentation for Detection of Inadequate Core Cooling" has its reference zero at the top of the upper grid plate at elevation 291' 4-3/8".

The different reference points are justified because of the different utilizations of the instrumentation. The existing instrumentation would be used by the operators during normal operation and transients, while the inadequate core cooling water level instrumentation would be utilized by the operators during accidents in which inadequate core cooling may be present. Additional justification is also provided in a TMI BWR Owners Group report titled NUREG 0737 II.K.3.27 Common Water Level Reference which was transmitted to the Nuclear Regulatory Commission by a letter dated December 29, 1980 from Mr. D. B. Waters, Chairman TMI BWR Owners Group to Mr. D. G. Eisenhut.

TMI ACTION PLAN ITEM NO. II.K.3.28

VERIFY QUALIFICATION

OF ACCUMULATORS ON AUTOMATIC

DEPRESSURIZATION SYSTEM VALVES

NRC POSITION

Safety analysis reports claim that air or nitrogen accumulators for the automatic depressurization system (ADS) valves are provided with sufficient capacity to cycle the valves open five times at design pressures. GE has also stated that the emergency core cooling (ECC) systems are designed to withstand a hostile environment and still perform their function for 100 days following an accident. Licensee should verify that the accumulators on the ADS valves meet these requirements, even considering normal leakage. If this cannot be demonstrated, the licensee must show that the accumulator design is still acceptable.

RESPONSE

As indicated in our letters of June 20, 1980 and December 17, 1980, this item is not applicable to Nine Mile Point Unit 1. The Nine Mile Point Unit 1 design includes electromatic relief valves for it's Automatic Depressurization System (ADS) and not valves operated by accumulators. Therefore, no further action on this item is required for Nine Mile Point Unit 1.

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TMI ACTION PLAN ITEM NO. II.K.3.29

STUDY TO DEMONSTRATE PERFORMANCE OF

ISOLATION CONDENSERS WITH NONCONDENSABLES

NRC POSITION

If natural circulation plays an important role in depressurizing the system (e.g., in the use of isolation condensers), then the various modes of two-phase flow natural circulation, including noncondensables, which may play a significant role in plant response following a small-break loss-of-coolant accident (LOCA) should be demonstrated.

RESPONSE

As indicated in our letters of November 7, 1980, and December 17, 1980, the emergency (isolation) condensers at Nine Mile Point Unit 1 are being modified so that the tube side of the condensers can be vented to the torus under accident conditions. Therefore, a study to demonstrate the performance of the Nine Mile Point Unit 1 emergency condensers with noncondensables is not required. No further action is required on this item for Nine Mile Point Unit 1.

TMI ACTION PLAN ITEM NO. II.K.3.44

EVALUATION OF ANTICIPATED TRANSIENTS WITH SINGLE FAILURE TO VERIFY NO FUEL FAILURE

NRC POSITION

For anticipated transients combined with the worst single failure and assuming proper operator actions, licensees should demonstrate that the core remains covered or provide analysis to show that no significant fuel damage results from core uncovery. Transients which result from a stuck-open relief valve should be included in this category.

RESPONSE

As indicated in our December 17, 1980 letter, this was to be addressed through a generic submittal by the BWR Owners Group. The report titled NUREG-0737 Item II.K.3.44 - Adequate Core Cooling for Transients with a Single Failure was transmitted to the Nuclear Regulatory Commission by a letter dated December 29, 1980 from Mr. D. B. Waters, Chairman TMI BWR Owners Group to Mr. D. G. Eisenhut.

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TMI ACTION PLAN ITEM NO. II.K.3.45

EVALUATION OF DEPRESSURIZATION WITH OTHER THAN AUTOMATIC DEPRESSURIZATION SYSTEM

NRC POSITION

Analyses to support depressurization modes other than full actuation of the automatic depressurization system (ADS) (e.g., early blowdown with one or two safety relief valves (SRVs)) should be provided. Slower depressurization would reduce the possibility of exceeding vessel integrity limits by rapid cooldown.

RESPONSE

As indicated in our December 17, 1980 letter, this item was to be addressed through a generic submittal by the BWR Owners Group. The report titled NUREG-0737 Item II.K.3.45 - Alternate Modes of Depressurization was transmitted to the Nuclear Regulatory Commission by a letter dated December 29, 1980 from Mr. D. B. Waters, Chairman TMI BWR Owners Group to Mr. D. G. Eisenhut.

TMI ACTION PLAN ITEM NO. II.K.3.46

RESPONSE TO LIST OF CONCERNS FROM ACRS CONSULTANT

NRC POSITION

GE should provide a response to the Michelson concerns as they relate to BWRs. Licensees should access applicability and adequacy of this response to their plants.

RESPONSE

Niagara Mohawk's submittal of August 1, 1980 addressed this item.

TMI ACTION PLAN ITEM NO. III.A.2

IMPROVING LICENSEE EMERGENCY PREPAREDNESS - LONG TERM

NRC POSITION

Each nuclear facility shall upgrade its emergency plans to provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. Specific criteria to meet this requirement is delineated in NUREG-0654 (FEMA-REP-1), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparation in Support of Nuclear Power Plants."

RESPONSE

Niagara Mohawk has upgraded the Nine Mile Point Unit 1 Emergency Plan. The upgraded Nine Mile Point Unit 1 Emergency Plan has been submitted under separate cover by a letter dated December 30, 1980 from Mr. D. P. Dise to Mr. H. R. Denton.

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TMI ACTION PLAN ITEM NO. III.D.3.3

IMPROVED INPLANT IODINE INSTRUMENTATION UNDER ACCIDENT CONDITIONS

NRC POSITION

(1) Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

(2) Each applicant for a fuel-loading license to be issued prior to January 1, 1981 shall provide the equipment, training, and procedures necessary to accurately determine the presence of airborne radioiodine in areas within the plant where plant personnel may be present during an accident.

RESPONSE

Niagara Mohawk's submittal of December 17, 1980 indicated that a new type of charcoal cartridge and the analysis procedures were being evaluated. As indicated in a separate letter dated December 31, 1980 from Mr. D. P. Dise to Mr. D. G. Eisenhut, this evaluation and issuance of approved procedures will not be completed until January 31, 1981.

As indicated in our letter of December 17, 1981, the modification required to accurately measure iodine will be to provide a dedicated source of outside air. The laboratory area ventilation air supply is being modified to transfer from the discharge of the Turbine Building ventilation supply fan to a direct source of outside air on a loss of all offsite power. This modification will be completed during the spring 1981 refueling outage as indicated in a separate letter dated December 31, 1980 from Mr. D. P. Dise to Mr. D. G. Eisenhut.

TMI ACTION PLAN ITEM NO. III.D.3.4
CONTROL ROOM HABITABILITY, REQUIREMENTS

Position

In accordance with action item III D.3.4, Control Room Habitability, licensees shall assure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shut down under design basis accident conditions (Criterion 19, "Control Room", of Appendix A, "General Design Criteria for Nuclear Power Plants", to 10CFR Part 50).

Response

Niagara Mohawk has reviewed the Nine Mile Point Unit 1 control room for conformance with sections 2.2.1-2.2.2, 2.2.3 and 6.4 of the Standard Review Plan. The results of the review are presented herein.

Potential accidents involving releases of toxic substances from off-site and on-site locations were evaluated. Procedures outlined in Appendix B to Reg. Guide 1.78 were used to perform the evaluation. The results of the evaluation indicate that the release of any of the identified toxic materials within a 5 mile radius would have virtually no impact on the control room atmosphere. Off-site accidents were conservatively considered to be ground level, directly upwind, puff releases. On site accidents were assumed to be instantaneous, stack releases except for nitrogen which is stored outdoors. In all cases, the distances and toxicity limits involved precluded the concentrations in the control room from approaching dangerous limits.

The control room habitability was also evaluated for onsite radioactive releases. The following assumptions were used in the evaluation:

a) Reg. Guide 1.3 source terms and meteorology for an elevated release (130m).
b) MSIV leakage equal to Technical Specification limits and containment leakage equal to design limit.
c) No credit for control room filtration system
d) A delay in releases from the main steam lines to the turbines of approximately 25 hrs.

After 100 days, the total whole body integrated dose has been calculated to be approximately 0.85R. This value includes shine effects which may contribute to the dosage in the control room and is less than the limit, as outlined in General Design Criteria 19. The integrated thyroid dose has also been determined not to exceed the General Design Criteria 19 limit of 30R.

The above information demonstrates that the control room at Nine Mile Point Unit 1 is adequately protected from potential accidents involving toxic or radioactive releases in the vicinity of the plant. Therefore, no modifications are proposed to the habitability systems of the control room.

Information needed for an independent analysis is provided below.

1) Control room mode of operation - zone isolation with incoming air filtered and a positive pressure maintained by ventilation fans during accident conditions.

2) Control Room Characteristics:

a. Air volume of emergency zone - approximately 133,200 cf.
b. Control room emergency zone - main control room, auxiliary control room, toilet, kitchen, instrument shop, shift supervisor's office.
c. Control room ventilation schematic - see Figure 1.
d. Infiltation leakage rate - assumed 10 cfm. Positive pressure is maintained in the control room.
e. HEPA efficiency - 99 percent DOP aerosol
charcoal adsorber efficiency - 90 percent methyl iodide removal
f. Closest distance between containment and air intake - approximately 290 feet.
g. Layout of control room - see Figures 2, 3.
h. Control room shielding - 12" solid concrete blocks on north and west wall
8" concrete blocks on south wall
8 1/2" poured slab above
5 3/4" concrete below
i. Automatic isolation capability - none. Isolation is by operator action.
Damper Closing Time - approximately 45 sec.
Damper Area - 14" butterfly valve
Damper Leakage - 0 at 40 psf differential pressure
j. Chlorine detectors or toxic gas detectors - none
k. Self contained breathing apparatus - 2 masks with at least 8 hours oxygen supply, 2 Scott air packs at 30 minutes each, and 2 escape packs at 5 minutes each are provided in the control room.
l. Bottled air supply - 2 tanks, each with at least an 8 hour supply of oxygen (see k above).
m. Emergency food and water - 5 day supply for 5 men readily available.
n. Personnel capacity - at least 5 men for 5 days
o. Potassium iodide drug supply - supply of potassium iodide tablets available from the Supervisor of Radiation Chemistry

3) On-site storage of chlorine and other hazardous chemicals - see Attachment 1.

4) Off-site manufacturing, storage, or transportation facilities of hazardous chemicals - see Attachment 1.

5) Technical Specifications:

a. Chlorine detection system - none.
b. Control room emergency filtration system - see Attachment 2.

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IDENTIFICATION OF OFF-SITE HAZARDOUS CHEMICAL SOURCES

1) Name and Address of Company: Alcan Sheet and Plate, Oswego, N.Y.

2) Type of Industry: Melting, Casting and Rolling of Aluminum

3) Chemicals Stored or Used at Plant:

Chemical Name	Maximum Amt. Stored at One Time	Pressure and Temperature at Which Chemical is Stor
Carbon Dioxide (liq)	114,000 lbs	40 atm @ 20^0 C
Chlorine (liq)	20-1 ton cylinders	4.8 atm @ 20^o C
Propane (liq)	100,000 gal	6.8 atm @ 20^o C
Nitrogen (liq)	13,000 gal	13.6 atm @ -180^0 C
Sulfuric Acid	6,000 lb	1 atm @ 20^o C

4) Transportation of Chemicals:

Chemical Name	Mode of Transport	Route*	Maximum Quantity/ Shipment	Frequency of Shipment	Press. & Temp. During Shipment
Carbon Dioxide	Tank Trailer	Buffalo to Alcan	40,000 lb	Monthly	--------
Chlorine	Flat Bed Trailer	Buffalo to Alcan	12-1 ton cylinders	Weekly	4.8 atm @ 20^0 C
Propane	Tank Trailer	Syracuse to Alcan	9000 gal	Biweekly	6.8 atm " " "
Nitrogen	Tank Trailer	Buffalo to Alcan	5000 gal	Biweekly	--------
Sulfuric Acid	Trucks w/Caboloy	Syracuse to Alcan	~6000 gal	Biweekly	--------

*Normally via Rt. 104, Kocher Road, and County Route 1.

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IDENTIFICATION OF OFF-SITE HAZARDOUS CHEMICAL SOURCES

1) Name and Address of Company: J. A. FitzPatrick Nuclear Power Plant, Scriba, N.Y.

2) Type of Industry: Nuclear Power Plant

3) Chemicals Stored or Used at Plant:

Chemical Name	Maximum Amt. Stored at One Time	Pressure and Temperature at Which Chemical is Stor
Nitrogen (liq)	10,000 gal	217 psi @ -200° F
Hydrogen	28,800 ft^3	2400 psi
Sulfuric Acid	5,000 gal	Ambient
Sodium Hydroxide	5,000 gal	Ambient
Carbon Dioxide	26,000 lbs	340 psi @ 0°F
Propane	1,000 gal	$<$ 50 psi

4) Transportation of Chemicals:

Chemical Name	Mode of Transport	Route*	Maximum Quantity/ Shipment	Frequency of Shipment	Press. & Temp. During Shipment
Nitrogen	Truck	Local	6900 gal	Monthly	250 psi @ -200° F
Hydrogen	"	"	128,000 ft^3	2/Month	2400 psi
Sulfuric Acid	"	"	3000 lbs	Monthly	NA
Sodium Hydroxide	"	"	3000 lbs	Monthly	NA
Carbon Dioxide	"	"	6900 gal	--	340 psi @ 0° F
Propane	"	"	22,500 lbs	Monthly	$<$ 50 psi

*Deliveries made from Syracuse-Oswego-Rochester Area. Major routes are Route 104 and Interstate 81.

IDENTIFICATION OF ON-SITE HAZARDOUS CHEMICAL SOURCES

1) Name and Address of Company: Nine Mile Point Unit 1, Scriba, NY

2) Type of Industry: Nuclear Power Plant

3) Chemicals Stored or Used at Plant:

Chemical Name	Maximum Amt. Stored at One Time	Pressure and Temperature at Which Chemical is Stor
Nitrogen	15,000 gal	220 psi @ -200° F (1)
Carbon Dioxide	20,000 lb	300 psi @ 0° F
Sulfuric Acid	3,500 gal	Ambient
Sodium Hydroxide	3,500 gal	Ambient

4) Transportation of Chemicals:

Chemical Name	Mode of Transport	Route*	Maximum Quantity/ Shipment	Frequency of Shipment	Press. & Temp. During Shipment
Nitrogen	Truck	Local	6000 gal	Once per month	220 psi @ -200° F (
Carbon Dioxide	"	"	3 ton	" " "	300 psi @ 0° F
Sulfuric Acid	"	"	3000 gal	Once per quarter	Ambient
Sodium Hydroxide	"	"	40,000 lbs	" " "	Ambient

*Major routes are Route 104 and Interstate 81

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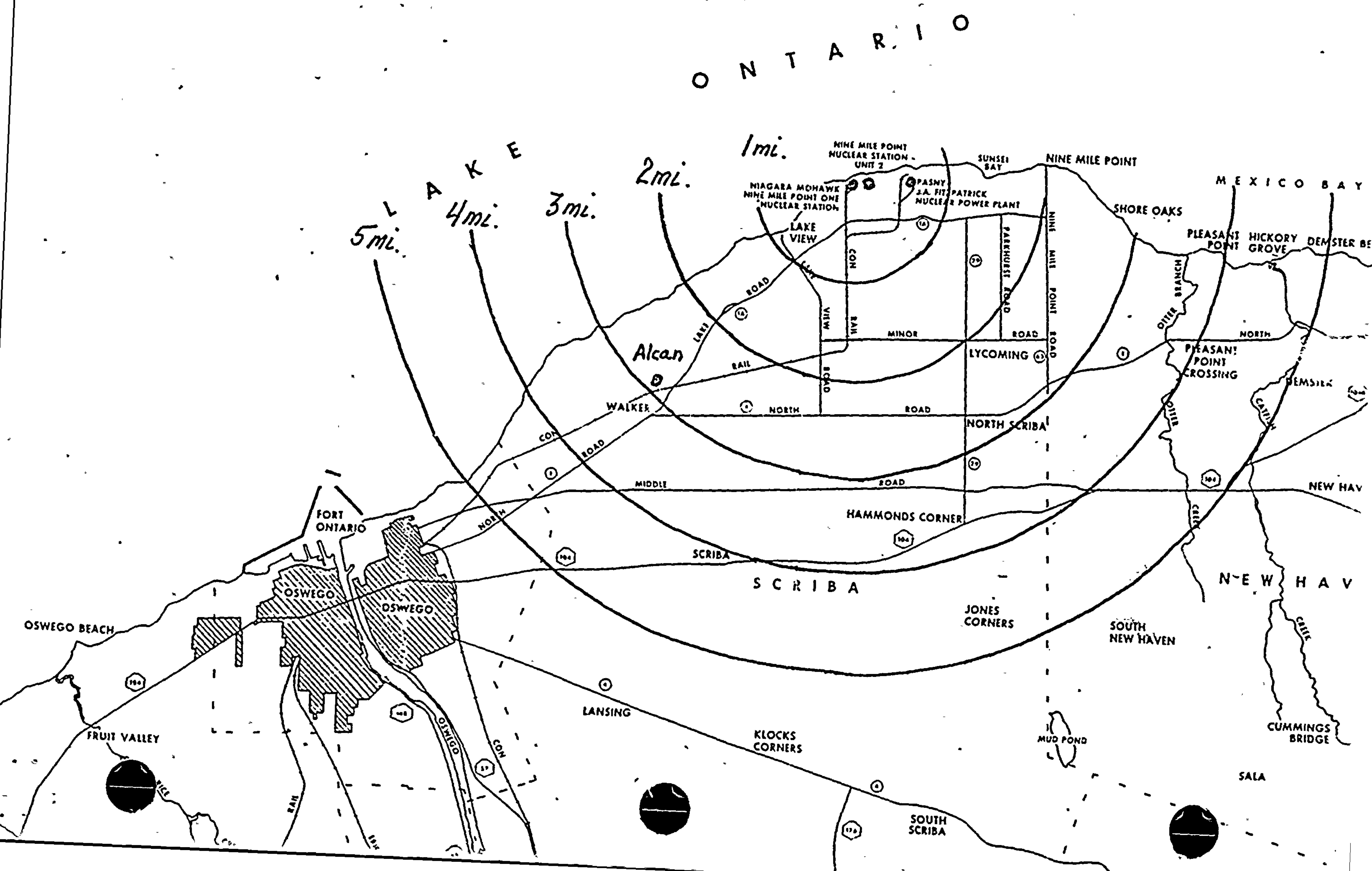
(1) Estimated


LAKE ONTARIO
1mi.
2mi.
3mi.
4mi.
5mi.
NINE MILE POINT NUCLEAR STATION - UNIT 2
NIAGARA MOHAWK NINE MILE POINT ONE NUCLEAR STATION
PASNY J.A. FITZPATRICK NUCLEAR POWER PLANT
SUNSET BAY
NINE MILE POINT
MEXICO BAY
SHORE OAKS
PLEASANT POINT
HICKORY GROVE
LAKE VIEW
Alcan
LYCOMING
WALKER
NORTH SCRIBA
PLEASANT POINT CROSSING
MINOR ROAD
NORTH ROAD
MIDDLE ROAD
NEW HAV
HAMMONDS CORNER
FORT ONTARIO
SCRIBA
OSWEGO
OSWEGO BEACH
JONES CORNERS
SOUTH NEW HAVEN
NEW HAV
LANSING
KLOCKS CORNERS
MUD POND
CUMMINGS BRIDGE
SALA
FRUIT VALLEY
SOUTH SCRIBA

.5 CONTROL ROOM AIR TREATMENT SYSTEM

Applicability:

Applies to the operating status of the control room air treatment system.

Objective:

To assure the capability of the control room air treatment system to minimize the amount of radioactivity or other gases entering the control room in the event of an incident.

Specification:

a. Except as specified in Specification 3.4.5e below, the control room air treatment system and the diesel generators required for operation of this system shall be operable at all times when containment integrity is required.

b. The results of the in-place cold DOP and halogenated hydrocarbon test design flows on HEPA filters and charcoal adsorber banks shall show > 99% DOP removal and > 99% halogenated hydrocarbon removal when tested in accordance with ANSI N.510-1975.

4.4.5 CONTROL ROOM AIR TREATMENT SYSTEM

Applicability:

Applies to the testing of the control room air treatment system.

Objective:

To assure the operability of the control room air treatment system.

Specification:

a. At least once per operating cycle, or once every 18 months, whichever occurs first, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate (± 10%).

b. The tests and sample analysis of Specification 3.4.5b, c and d shall be performed at least once per operating cycle or once every 18 months, or after 720 hours of system operation, whichever occurs first or following significant painting, fire or chemical release in any ventilation zone communicating with the system.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

c. The results of laboratory carbon sample analysis shall show > 90% radioactive methyl iodide removal when tested in accordance with ANSI N.510-1975 at 130C and 95% R.H. R.H. = relative humidity

d. Fans shall be shown to operate within ± 10% design flow when tested in accordance with ANSI N.510-1975.

e. From and after the date that the control room air treatment system is made or found to be inoperable for any reason, reactor operation or refueling operations is permissible only during the succeeding seven days unless the system is sooner made operable.

f. If these conditions cannot be met, reactor shutdown shall be initiated and the reactor shall be in cold shutdown within 36 hours for reactor operations and refueling operations shall be terminated within 2 hours.

c. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

d. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

e. The system shall be operated at least 10 hours every month.

ATTACHMENT 1 TO

TMI ACTION PLAN ITEM NO. I.A.1.1
SHIFT TECHNICAL ADVISOR

I. CURRENT TRAINING PROGRAM

Assistant Station Shift Supervisors (Shift Technical Advisors) trainees were enrolled in the Nine Mile Point Senior Reactor Operator License Preparation Program (outline attached). This program was modified to include training in the Shift Technical Advisor Function and to include those areas of training that were anticipated to be new requirements (i.e. thermodynamics, fluid flow, and increased training in transients and accidents). In addition, the Assistant Station Shift Supervisors were given an individual Nine Mile Point Training Manual to direct and document their on-the-job training received while covering shift responsibilities.

The Senior Reactor Operator License Preparation Program is a 17 week program that has been used at Nine Mile Point for the past several years. It includes 15 weeks of formal classroom training in plant specific items (40 hours per week), one (1) week of Simulator training and a one (1) week review period. At the conclusion of the training program, a Reactor Operator written examination, Senior Reactor Operator written examination, and a walk-through oral evaluation is performed by a consultant organization, to assure candidate readiness to be presented to the NRC for licensing.

One (1) Assistant Station Shift Supervisor, who met the NRC eligibility requirements, was presented to the Operator Licensing Branch examiner in May 1980. This individual passed the Senior Reactor Operator Licensing examination and received his SRO License.

It should be noted that the simulator experience provided to the five (5) Assistant Station Shift Supervisors who were not presented to the NRC for licensing was directed toward more emphasis on Accidents and Transients versus the requirements for Hot Licensing outlined in NUREG - 0094, Appendix F, Paragraph D. Other aspects of the program were identical.

At the conclusion of the training, the training documentation, including schedules, lesson plans, quizzes, exams, simulator experience reports, RO Written Exams, SRO Written Exams and Walk-through evaluations, were reviewed by the Vice President Nuclear Generation. Based on this review, certification was issued attesting to qualification to fill the Accident Assessment Function of the Assistant Station Shift Supervisor.

The current Assistant Shift Supervisors have received the training outlined above. A new Assistant Shift Supervisor trainee would receive this same or equivalent training.

II. REQUALIFICATION

All Assistant Station Shift Supervisors will attend the Requalification Simulator Program as part of the Licensed Operator Requalification Program. This three day, 8 hour/day, program is designed to meet the requirements of Enclosure 4, Control Manipulations of Mr. H. R. Denton's March 28th, 1980 letter. Assistant Station Shift Supervisors will as a minimum perform the role of an Accident Assessment Technical Advisor.

It is anticipated that the Assistant Station Shift Supervisor will be enrolled in the NRC approved Licensed Operator Requalification Program. Participation in this program would be identical to that of an NRC Senior Reactor Operator License holder.

III. LONG TERM TRAINING FOR ASSISTANT STATION SHIFT SUPERVISORS

It is intended that all Assistant Station Shift Supervisors obtain NRC Senior Reactor Operator Licenses as they meet the eligibility requirements.

IV. NIAGARA MOHAWK'S TRAINING PROGRAM AS IT PERTAINS TO THE INPO DOCUMENT ENTITLED "NUCLEAR POWER PLANT SHIFT TECHNICAL ADVISOR RECOMMENDATIONS FOR POSITION DESCRIPTION QUALIFICATIONS, EDUCATION AND TRAINING" REV. 0, APRIL 30, 1980

INPO SECTION	COMPARISON
5.1 Education & Training	See Section 6 below
5.2 Experience	All Assistant Station Shift Supervisors have a minimum of one year Nuclear Power Plant Experience, including training at Nine Mile Point Unit #1.
5.3 Absences from Duties	Absences from Assistant Station Shift Supervisor's duties are in accordance with the approved requalification program.
6.1.1 Education	All Assistant Station Shift Supervisors have a bachelor's degree in a scientific or engineering discipline.
6.1.2 College Level Fundamental Education	All Assistant Station Shift Supervisors have a bachelor's degree in a scientific or engineering discipline, plus demonstrated knowledge level equivalent to a SRO in Reactor Theory, Reactor Chemistry, Nuclear Materials, Thermal Sciences, Electrical Sciences, Nuclear Instrumentation, and Nuclear Radiation Protection and Health Physics.

6.2	Applied Fundamentals	All Assistant Station Shift Supervisors have a bachelor's degree in a scientific or engineering discipline and demonstrated a knowledge level equivalent to a Senior Reactor Operator.
6.3	Management/Supervisory Skills	All Assistant Station Shift Supervisors are enrolled in a Corporate Management Development Program. Additional courses in Stress Management and Command Responsibilities and Limits have been attended or are scheduled to be attended by the Assistant Station Shift Supervisors.
6.4	Plant Systems	All Assistant Station Shift Supervisors have demonstrated a knowledge level equivalent to a Senior Reactor Operator.
6.5	Administrative Controls	All Assistant Station Shift Supervisors have demonstrated a knowledge level equivalent to a Senior Reactor Operator.
6.6	General Operating Procedures	All Assistant Station Shift Supervisors have demonstrated a knowledge level equivalent to a Senior Reactor Operator.
6.7	Transient/Accident Analysis and Emergency Procedures	All Assistant Station Shift Supervisors have demonstrated a knowledge level equivalent to a Senior Reactor Operator.
6.8	Simulator Training	All Assistant Station Shift Supervisors have participated in Simulator Training as part of initial training. In addition, all Assistant Station Shift Supervisors have attended or are scheduled to attend the Requalification Simulator Program designed to meet the requirements of Enclosure 4 of Mr. H. R. Denton's March 28, 1980 letter.
6.9	Annual Requalification Training	It is expected that all Assistant Station Shift Supervisors will be enrolled in the Operator Requalification Program and will participate as Senior Reactor Operator License holders.

All demonstrated knowledge is documented by quizzes and examinations. The final examination is similar in scope and content to a NRC Senior Operator Licensing Examination.

NIAGARA MOHAWK POWER CORPORATION
NINE MILE POINT
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WEEK # 1

INSTRUCTOR(S)______________

	MORNING	AFTERNOON
MONDAY	Introduction to STA Function (accident assessment) o Introduction to Course o Scope of Course o Policies o Lessons Learned Task Force - NUREG 0578	Introduction to BWR's and STEAM POWER PLN o Reactor Core o Steam System o Turbine Generator o Feedwater o Reactivity Control o Feedwater Control
TUESDAY	Station Reference Materials in Control Room o FSAR o Technical Specs. o Operating Procedures o Special Procedures o Admin. Procedures o P. ID's	Station Reference Materials (Continued) o Electrical Drawings o Print Reading o Symbols o Logic Diagrams
WEDNESDAY	NRC Function o Atomic Energy Act o AEC o NRC Responsibilities o Commissioners o NRR o OLB o IE	Other Advisory and Regulatory Bodies o ANSI, ASME, ANS o ACRS, INPO, NSAC Codes and Standards o CFR o ANSI Standards o IE Bulletins o NUREGS, Reg. Guides
THURSDAY	Conduct of Station Operations o Station Organization o Shift Organization. o Watch Relief Proc. o Selected Admin. Proc.	Conduct of Station Operations (Continued) o Selected Admin. Proc. o Selected SOP's
FRIDAY	TMI-2 Lessons Learned o B & W Plant o Scenario of TMI-2 Accident o Davis-Besse Transient	TMI-2 Lessons Learned (Continued) o Lessons Learned o NUREG 0578 o NUREG 0585 o Kemeny Report o Rogovin Report o NUREG 0660

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WEEK # 2

INSTRUCTOR(S)__________

	MORNING	AFTERNOON
MONDAY	Examination Review and Critique of Examination	o Atomic and Nuclear Structure
TUESDAY	o Radioactive Decay and Nuclear Reaction o Cross Sections, Flux and Reaction Rates	o Binding Energy and the Fission Process o Neutron Travel and Neutron Sources
WEDNESDAY	o Neutron Mult. and the 6-Factor Formula o Reactivity, Shutdown Margin and Excess Reactivity	o Subcritical Mult. o Prompt and Delayed Neutron Fraction
THURSDAY	o Reactor Period o Reactivity Coefficients	o Control Rod Worth o Fission Product Poisons and Samarium
FRIDAY	o Xenon	o Time-in-Life Effects o Exam Review

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WEEK # 3 INSTRUCTOR(S) Lukens

	MORNING	AFTERNOON
MONDAY	o Examination o Review and Critique of Examination o Reactor Vessel and Internals	o Reactor Vessel and Internals (cont'd) o Vessel Instrumentation
TUESDAY	o Nuclear Fuel and Core Components o Control Rods	o CRD Mechanism o CRD Hydraulics
WEDNESDAY	o RMCS o RWM o Liquid Poison	o Reactor Recirculation System o Recirculation Flow Control
THURSDAY	o Main Steam and Extraction Systems	o Principles of Detector Operation o SRM
FRIDAY	o IRM o LPRM o APRM o TIP	o RPS o Exam Review

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WEEK # 4, C_T Systems and ECCS

INSTRUCTOR(S)__________

	MORNING	AFTERNOON
MONDAY	Examination Review of Examination Primary C_T	C_T Isolation Secondary C_T
TUESDAY	Emergency Vent Reactor Building Vent	LOCA Scenario
WEDNESDAY	ADS Core Spray	C_T Spray
THURSDAY	Emergency Cooling	Shutdown Cooling
FRIDAY	Head Spray RWCU	Review and Operational Summary

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WEEK # 5, BOP Systems INSTRUCTOR(S) Klose

	MORNING	AFTERNOON
MONDAY	Examination Review of Examination RBCLC	Review of MS OPI
TUESDAY	Auxiliary Steam	Condensate
WEDNESDAY	Feedwater Feedwater Control	HPCI
THURSDAY	Circulating Water Steam Cycle Summary	Condenser Air Removal Off Gas
FRIDAY	Service Water	Operational Summary Review

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WEEK # 6 Electricity Fundamentals INSTRUCTOR(S) Eresian

	MORNING	AFTERNOON
MONDAY	Examination Review of Exam DC Theory (Lesson 1)	DC Theory (Lessons 1 & 2)
TUESDAY	Elect. Measurements (Lesson 3)	AC (Lesson 4)
WEDNESDAY	AC (Lesson 4)	SemiCond. Basics (Lesson 5)
THURSDAY	Diodes & Power Supplies (Lesson 6)	Temp. Measurements and X ducers (Lesson 11)
FRIDAY	Temp. Measurements and X ducers (Lesson 11) Logic & Logic Gates (Lesson 12)	Logic & Logic Gates (Lesson 12) Review

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WEEK # 7 Main Turbine and Electrical Systems INSTRUCTOR(S) Eresian

	MORNING	AFTERNOON
MONDAY	Examination Review Generators	Transformers Motors 3φ power
TUESDAY	Main Gen. H_2 Cooling	H_2 Seal Oil Stator Cooling
WEDNESDAY	Main Turb. Turbine Aux.	Main Turb. Control Turb. Control Oil
THURSDAY	RBCLC Service Water	Diesel Gen.
FRIDAY	Plant Electrical	Plant Electrical Review

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WEEK # 8 BOP Systems and Operations

INSTRUCTOR(S) Zerbo

	MORNING	AFTERNOON
MONDAY	Examination Review of Examination Fuel Pool	Fuel Pool Cooling Fuel Pool Clean-up OP6 and 20
TUESDAY	Plant Air Systems	Startup Procedure OP43
WEDNESDAY	Shutdown Procedure OP43	Control Room Familiarization
THURSDAY	Integrated Operations Fuel Handling	ARM PRM OP50
FRIDAY	Technical Specifications and BASES	Selected Review Exam Review

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WEEK # 9 Heat Transfer Fundamentals

INSTRUCTOR(S) Reardon

	MORNING	AFTERNOON
MONDAY	Examination and Review Fundamental Concepts PE, KE, U, H Definitions Properties of a Substance Temperature and the Ideal Gas	Work and Heat Introduction to PROBLEMS
TUESDAY	The 1st Law of Thermo P-V diagrams Energy as a property 1st Law for Closed System	Internal Energy 1st Law for an Open System Enthalpy Conservation of mass and the Continuity Equation Steady Flow PROBLEMS
WEDNESDAY	The 2nd Law of Thermo Heat Engines and Heat Pumps Statement of 2nd Law Reversible Processes Carnot Cycle	Entropy as a property T-S diagrams Entropy changes in reversible processes Entropy and Lost Work PROBLEMS
THURSDAY	Vapor Power Cycle Rankine Cycle Effect of P and T on Rankine Cycle Reheat Cycle	The Regenerative Cycle Deviation of Real Cycles from Ideal Cycles PROBLEMS
FRIDAY	Fluid Flow 1-D Flow Continuity Equation Bernoulli's Equation	PROBLEMS REVIEW

WEEK # 10 Fluids and Thermal Hydraulics

INSTRUCTOR(S) Klose

	MORNING	AFTERNOON
MONDAY	Examination Examination Review Heat Transfer Fundamentals	Fluid Flow BWR Heat Transfer
TUESDAY	BWR Thermal Hydraulics Friction Pressure Drop Acceleration Pressure Drop	Orificing Quality and Void Fraction
WEDNESDAY	Critical Power Transition Boiling Critical Quality	GEXL Correlation LHGR
THURSDAY	Peaking Factors APLHGR	MAPRAT Heat Balance BWR Heat Balance
FRIDAY	Problems in Heat Transfer and Thermal Hydraulics	Review

WEEK # 11 Materials and Process Instrumentation

INSTRUCTOR(S) Benge/Po[illegible]

	MORNING	AFTERNOON
MONDAY	Examination Examination Review Reactor Materials Corrosion	Fracture Modes Neutron Embrittlement
TUESDAY	M-W Reactions Reactor Water Chemistry	Temperature Thermocouples
WEDNESDAY	Temperature RTD's Thermistors	Pressure Manometers Elastic Deformation Elements
THURSDAY	Level Direct Methods Inferred Methods	Level Compensation Techniques
FRIDAY	Level Compensation Techniques Flow Heat Type Area Type	Flow Heat Type Area Type √ Extraction

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WEEK # 12

INSTRUCTOR(S)__________

	MORNING	AFTERNOON
MONDAY 5/12/80	Introduction to Transient **Transients:** • S/U of Cold Recirc Loop • Recirc Pump Trips	Transients (cont'd) • Recirc Pump Stall • Flow Controller Malfunctions
TUESDAY 5/13/80	Transients (cont'd) • Inadvertant Relief Valve Actuation • Safety Valve Actuation	Transients (cont'd) • MSIV Closure • Feedwater Controller Malfunction
WEDNESDAY 5/14/80	Transients (cont'd) • Turbine Trips 1) Low Power/High Power 2) Bypass/No Bypass	Transients (cont'd) • Inadvertant Opening of one bypass valve • Pressure Regulator Malfunction
THURSDAY 5/15/80	Accidents : • Intro. to Accident Analysis • Main Steam Line Break	Accidents (cont'd) • Rod Drop • Refueling Accident • LOCA's
FRIDAY 5/16/80	Accidents (cont'd) • Containment DBA • TMI Summary	Examination

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WEEK #: 13 Health Physics & Rad. Prot. INSTRUCTOR(S) Zerb.

	MORNING	AFTERNOON
MONDAY 5/19/80	Principles of Radiation Detection	Health Physics Fundamentals
TUESDAY 5/20/80	Health Physics Fundamentals	Health Physics Fundamentals
WEDNESDAY 5/21/80	Health Physics Fundamentals	RPP's 10 CFR 20 10 CFR 100
THURSDAY 5/22/80	Plume Models Gaussian Plume	EPA 520: Correlation of Concentrations with Dose Rates
FRIDAY 5/23/80	Derivation of Dose Rate from Activity	Review

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WEEK # 14 SIMULATOR TRAINING INSTRUCTOR(S) [illegible]

	MORNING	AFTERNOON
MONDAY 5/26/80	T.C. Orientation Control Room Tour Rx Start-up Heat-up Rate Control Moderator Temp. Effects	Power Escalation Turbine Roll Turbine Trip less than 30% power Turbine Trip greater than 30% power
TUESDAY 5/27/80	Rod Worth Considerations during Start-up Pressure Regulator Failure Power Increase to 50%	Power Increase to Rated Conditions Power Range Transient OPs Hot,Peak Xenon Start-up
WEDNESDAY 5/28/80	<u>Plant Transients for Recognition & Diagnosis</u> • Turbine Trip • Load Reject • Loss of Vacuum	• MSIV Failure • Pressure Regulator Failure • Loss of Off-site Power
THURSDAY 5/29/80	<u>Transients for Recognition & Diagnosis</u> (cont'd) • Loss of Feedwater Heating • Inadvertant Pump Start	• Continuous Rod Withdrawal 1) at power 2) during start-up • Recirc Pump Trip • Loss of all Recirc. Pumps
FRIDAY 5/30/80	Failure of Feedwater Control System LOCA: 1) small 2) intermediate 3) large	Main Steam Line Break TMI Scenario for BWR's

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WEEK # 15 EMERGENCY PLAN & PROCEDURES INSTRUCTOR(S) KLOSE

	MORNING	AFTERNOON
MONDAY 6/2/80	Emergency Plan Appendix A, B, C, D	Reactor Theory Review Fundamentals of Reactor Operation
TUESDAY 6/3/80	EPP-1 Radiation Emerg. EPP-2 Fire Fighting	Problems in Rx Theory Reactor Kinetics
WEDNESDAY 6/4/80	EPP-3 Search & Rescue EPP-4 Contaminated Injury EPP-5 Personnel Accountability	Source Neutrons and Subcritical Multiplication
THURSDAY 6/5/80	EPP 6 & 7 Surveys EPP-8 Off-site Dose Estimation	Reactivity and Neutron Multiplication Fission Product Poisons
FRIDAY 6/6/80	Emergency Procedures Review of BWR Heat Transfer	Examination

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WEEK # 16

INSTRUCTOR(S)__________

	MORNING	AFTERNOON
MONDAY 6/9/80	Simulator Review Press. Reg. Fail. Heatup Rate Control POAH Mod. Temp. Effects	Simulator Review Hot Pk Xe S/U MSIV Failure LOCA
TUESDAY 6/10/80	IE Bulletins 79-12 79-13 79-26	IE Bulletins 79-15 80-01 79-16 70-13
WEDNESDAY 6/11/80	IE Info Notices 79-13 79-37 80-01	IE Info. Bulletins 80-02 80-04 80-06 80-22
THURSDAY 6/12/80	IE Circulars 79-07 79-24 80-08	Tech. Spec. Review
FRIDAY 6/13/80	Tech. Spec. Review	Review Exam

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WEEK # 17 Certification

INSTRUCTOR(S)__________

	MORNING	AFTERNOON
MONDAY 6/16/80	SELECTED REVIEW	SELECTED REVIEW
TUESDAY 6/17/80	SELECTED REVIEW	SELECTED REVIEW
WEDNESDAY 6/18/80	SELECTED REVIEW	SELECTED REVIEW
THURSDAY 6/19/80	EXAMINATION REACTOR OPERATOR	
FRIDAY 6/20/80	EXAMINATION SENIOR REACTOR OPERATOR	

Prim. Cont. Penetration Number	Line Size, In.	System	Is System an Engineered safety function	Figure	Process Fluid	Isolation Valves: Valve Number	Isolation Signal Code(s)	Location	Type	Actuator	Primary Actuation Mode	Secondary Actuation Mode	Full Closure Time, sec.	Power Source	Position Indication in Control Rm.	Positions: Normal	Shutdown	Post Accident	Power Failure	Compliance with GDC 55, 56 and 57
X-40	3/4	Drywell O_2 Samp.Ret.	N	2	A	201.2-66	N/A	0	GB	N/A	M	N/A	N/A	H	N	0	0	0	N/A	(6) *
X-40	3/4	Drywell O_2 Samp.Ret.	N	2	A	201.2-67	N/A	0	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-40	3/4	Drywell O_2 Samp.Ret.	N	2	A	201.2-68	N/A	0	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-48	1/2	Drywell O_2 Sample Supply	N	2	A	201.2-25	E	0	GT	SO	A	RMC	60	DC	D	0	0	C	C	(6)
X-48	1/2	Drywell O_2 Sample Supply	N	2	A	201.2-26	E	0	GT	SO	A	RMC	60	DC	D	0	0	C	C	
X-49	1/2	Drywell O_2 Sample Supply	N	2	A	201.2-29	E	0	GT	SO	A	RMC	60	DC	D	0	0	C	C	(6)
X-49	1/2	Drywell O_2 Sample-Supply	N	2	A	201.2-30	E	0	GT	SO	A	RMC	60	DC	D	0	0	C	C	
X-50	1/2	Drywell O_2 Sample Supply	N	2	A	201.2-27	E	0	GT	SO	A	RMC	60	DC	D	0	0	C	C	(6)
X-50	1/2	Drywell O_2 Sample Supply	N	2	A	201.2-28	E	0	GT	SO	A	RMC	60	DC	D	0	0	C	C	
X-51	1	EC Elbow Flow Meter	Y	4	W		N/A	0	GB	N/A	M	N/A	N/A	H	N	0	0	0	N/A	(8)
X-51	1	EC Elbow Flow Meter	Y	4	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	

Prim. Cont. Penetration Number	Line Size, In.	System	Is System an Engineered safety function	Figure	Process Fluid	Isolation Valves: Valve Number	Isolation Signal Code(s)	Location	Type	Actuator	Primary Actuation Mode	Secondary Actuation Mode	Full Closure Time, sec.	Power Source	Position Indication in Control Rm.	Positions: Normal	Shutdown	Post Accident	Power Failure	Compliance with GDC 55, 56 and 57
X-53	1	Reactor Vessel Inst	N	3	W		N/A	O	GB	N/A	M	N/A	N/A	H	N	O	O	O	O	(6)
X-53	1	Reactor Vessel Inst	N	3	W		N/A	O	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-54	1	EC Elbow Flow Meter	N	3	W		N/A	O	GB	N/A	M	N/A	N/A	H	N	O	O	O	N/A	(8)
X-54	1	EC Elbow Flow Meter	N	3	W		N/A	O	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-64	1	CAD System Sample	N_2	2	A	201.7-01	E	O	DCV	AO	A	RMC	60	A	D	O	O	C	C	(6)
X-64	1	CAD System Sample	N_2	2	A	201.7-02	E	O	DCV	AO	A	RMC	60	A	D	O	O	C	C	
X-71A	1	Reactor Vessel Inst	Y	3	W		N/A	O	GB	N/A	M	N/A	N/A	H	N	O	O	O	N/A	(8)
X-71A	1	Reactor Vessel Inst	Y	3	W		N/A	O	FCV	N/A	HF	N/A	N/A	N/A	N	N/A	N/A	N/A	N/A	
X-71B	1	Reactor Vessel Inst	Y	3	W		N/A	O	GB	N/A	M	N/A	N/A	H	N	O	O	O	N/A	(8)
X-71B	1	Reactor Vessel Inst	Y	3	W		N/A	O	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-71C	1	EC Elbow Flow Meter	Y	4	W		N/A	O	GB	N/A	M	N/A	N/A	H	N	O	O	O	N/A	(8)
X-71C	1	EC Elbow Flow Meter	Y	4	W		N/A	O	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-71D	1	Reactor Vessel Inst	Y	5	W		N/A	O	GB	N/A	M	N/A	N/A	H	N	O	O	O	N/A	(8)
X-71D	1	Reactor Vessel Inst	Y	5	W		N/A	O	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-71E	1	Reactor Vessel Inst	Y	5	W		N/A	O	GB	N/A	M	N/A	N/A	H	N	O	O	O	N/A	(8)
X-71E	1	Reactor Vessel Inst	Y	5	W		N/A	O	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-71F	1	Reactor Vessel Inst	Y	5	W		N/A	O	GB	N/A	M	N/A	N/A	H	N	O	O	O	N/A	(8)
X-71F	1	Reactor Vessel Inst	Y	5	W		N/A	O	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-72A	1	Reactor Vessel Inst	Y	5	W		N/A	O	GB	N/A	M	N/A	N/A	H	N	O	O	O	N/A	(8)
X-72A	1	Reactor Vessel Inst	Y	5	W		N/A	O	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	

Isolation Valves

Prim. Cont. Penetration Number	Line Size, In.	System	Is System an Engineered safety function	Figure	Process Fluid	Valve Number	Isolation Signal Code(s)	Location	Type	Actuator	Primary Actuation Mode	Secondary Actuation Mode	Full Closure Time, sec.	Power Source	Position Indication in Control Rm.	Positions: Normal	Shutdown	Post Accident	Power Failure	Compliance with GDC 55, 56 and 57
X-72B	1	Reactor Vessel Inst	Y	5	W		N/A	0	GB	N/A	M	N/A	N/A	H	N	0	0	0	N/A	(8)
X-72B	1	Reactor Vessel Inst	Y	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-72C	1	EC Elbow Flow Meter	Y	4	W		N/A	0	GB	N/A	M	N/A	N/A	H	N	0	0	0	N/A	(8)
X-72C	1	EC Elbow Flow Meter	Y	4	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-72D	1	Reactor Vessel Inst	Y	5	W		N/A	0	GB	N/A	M	N/A	N/A	H	N	0	0	0	N/A	(8)
X-72D	1	Reactor Vessel Inst	Y	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-72E	1	Reactor Vessel Inst	Y	5	W		N/A	0	GB	N/A	M	N/A	N/A	H	N	0	0	0	N/A	(8)
X-72E	1	Reactor Vessel Inst	Y	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-72F	1	Reactor Vessel Inst	Y	5	W		N/A	0	GB	N/A	M	N/A	N/A	H	N	0	0	0	N/A	(8)
X-72F	1	Reactor Vessel Inst	Y	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-74	3/4	ILRT Sample Pt.	N	2	A	201.2-41	N/A	0	GB	N/A	M	N/A	N/A	H	N	C	C	C	N/A	(8)
X-74	3/4	ILRT Sample Pt.	N	2	A	201.2-42	N/A	0	GB	N/A	M	N/A	N/A	H	N	C	C	C	N/A	
X-80	1/2	ILRT Sample Pt.	N	2	A		N/A	0	GB	N/A	M	N/A	N/A	H	N	C	C	C	N/A	(8)
X-80	1/2	ILRT Sample Pt.	N	2	A		N/A	0	GB	N/A	M	N/A	N/A	H	N	C	C	C	N/A	
X-80	1/2	ILRT Sample Pt.	N	2	A	201.6-14	N/A	0	GB	N/A	M	N/A	N/A	H	N	C	C	C	N/A	
X-82(5)	1	Reactor Vessel Inst.	Y	3	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	(8)
X-82	1	Reactor Vessel Inst	Y	3	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-82	1	Reactor Vessel Inst	Y	3	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	
X-82	1	Reactor Vessel Inst	Y	3	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-98	1	CAD System Sample	N	2	A	201.7-03	E	0	DCV	AO	A	RMC	60	A	D	0	0	C	C	(6)
X-98	1	CAD System Sample	N	2	A	201.7-04	E	0	DCV	AO	A	RMC	60	A	D	0	0	C	C	
X-75	1	Main Steam Flow Imp(5)	N	10	S		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	(8)
X-75	1	Main Steam Flow Imp	N	10	S		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-75	1	Main Steam Flow Imp	N	10	S		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	
X-75	1	Main Steam Flow Imp	N	10	S		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	

Nine [illegible] Point Unit 1

PRIMARY CONTAINMENT ISOLATION SYSTEM DATA

Prim. Cont. Penetration Number	Line Size, In.	System	Is System an Engineered safety function	Figure	Process Fluid	Valve Number	Isolation Signal Code(s)	Location	Type	Actuator	Primary Actuation Mode	Secondary Actuation Mode	Full Closure Time, sec.	Power Source	Position Indication in Control Rm.	Positions: Normal	Positions: Shutdown	Positions: Post Accident	Positions: Power Failure	Compliance with GDC 55, 56 and 57
						Isolation Valves														
X-122	3/4	Service Water	N	8	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	LC	0	LC	N/A	(10)
X-122	3/4	for Drywell	N	8	W		N/A	I	GT	N/A	M	N/A	N/A	H	N	C	0	C	N/A	
X-122	3/4	"	N	8	W		N/A	I	GT	N/A	M	N/A	N/A	H	N	C	0	C	N/A	
X-131	2	Liquid Poison	Y	1	B	42.1-01	N/A	I	GB	N/A	M	N/A	N/A	H	D	0	0	0	N/A	(2)
X-131	2	Liquid Poison	Y	1	B	42.1-02	N/A	I	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-131	2	Liquid Poison	Y	1	B	42.1-03	N/A	0	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-129	2	Reactor Head Spray	N	1	W	34-02	N/A	I	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	(1)
X-129	2	Reactor Head Spray	N	1	W	34-01	E	0	GT	MO	A	RMC	30	AC	D	C	0	C	AI	
X-133	1	Reactor Vessel Inst.	Y	3	W		N/A	0	GB	N/A	M	N/A	N/A	H	N	0	0	0	N/A	(8)
X-133	1	Reactor Vessel Inst.	Y	3	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-134	1	Drywell Airborne Monitor	N	2	A	201.7-08	E	0	DCV	AO	A	RMC	60	A	D	0	0	C	C	(6)
X-134	1	Drywell Airborne Monitor	N	2	A	201.7-09	E	0	DCV	AO	A	RMC	60	A	D	0	0	C	C	
X-137	12	Cont. Spray Inlet	Y	1	W	80-18	N/A	0	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	(2)
X-137	12	Cont. Spray Inlet	Y	1	W	80-16	Manual(6)	0	GT	AO	RMC	N/A	60	A	D	C	C	0	0	
X-139	1	Reactor Water Sample	N	6	W		N/A	I	GB	N/A	M	N/A	N/A	H	N	0	0	0	N/A	(9)
X-139	1	Recirculation Sys.	N	6	W		N/A	I	GB	N/A	M	N/A	N/A	H	N	0	0	0	N/A	
X-139	1	Recirculation Sys.	N	6	W		N/A	0	GB	N/A	M	N/A	N/A	H	N	C	C	C	N/A	
X-81	1	Main SteamFlowImp(5)	N	10	S		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	(8)
X-81	1	MainSteam FlowImp(5)	N	10	S		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-81	1	MainSteam FlowImp(5)	N	10	S		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	
X-81	1	MainSteam FlowImp(5)	N	10	S		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	

Isolation Valves

Prim. Cont. Penetration Number	Line Size, In.	System	Is System an Engineered safety function	Figure	Process Fluid	Valve Number	Isolation Signal Code(s)	Location	Type	Actuator	Primary Actuation Mode	Secondary Actuation Mode	Full Closure Time, sec.	Power Source	Position Indication in Control Rm.	Positions: Normal	Positions: Shutdown	Positions: Post Accident	Positions: Power Failure	Compliance with GDC 55, 56 and 57
X-139	1	Reactor Water Sample	N	6	W		N/A	O	FF	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	(8)
X-139	1	Recirculation Sys.	N	6	W		N/A	O	GB	N/A	M	N/A	N/A	H	N	C	C	C	N/A	
X-140	12	Cont. Spray Inlet	Y	1	W	80-17	N/A	O	CK	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	(2)
X-140	12	Cont. Spray Inlet	Y	1	W	80-15	Manual(6)	O	GT	AO	RMC	-	60	A	D	O	O	O	O	
X-149	12	Cont. Spray Inlet	Y	1	W	80-38	N/A	O	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	(2)
X-149	12	Cont. Spray Inlet	Y	1	W	80-36	Manual(6)	O	GT	AO	RMC	-	60	A	D	O	O	O	O	
X-150	12	Cont. Spray Inlet	Y	1	W	80-37	N/A	O	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	(2)
X-150	12	Cont. Spray Inlet	Y	1	W	80-35	Manual(6)	O	GT	AO	RMC	-	60	A	D	O	O	O	O	
X-156	4	Recirc. Pump Cooling	N	1	W		N/A	I	GB	N/A	M	N/A	N/A	H	N	O	O	O	N/A	(5)
X-156	4	Water Return	N	1	W	70-92	Manual	O	GB	MO	RMC	N/A	30	DC	D	O	O	C	AI	
X-157	4	Recirculation Pump	N	1	W		Manual	I	GT	SO	RM	N/A	N/A	DC	D	O	O	O	N/A	(4)
X-157	4	Cooling Water Sup.	N	1	W	70-93	N/A	O	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-168	3/4	Dry Level Imp. Line	N	2	A		Manual	O	GT	N/A	M	N/A	N/A	H	N	O	O	O	N/A	(8)
X-174	3	Control Rod Drive to Reactor	NY	1	W	301-114	N/A	I	GT	N/A	M	N/A	N/A	H	LO	O	O	O	N/A	(2)
X-174	3	Control Rod Drive to Reactor	NY	1	W	301-113	N/A	I	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-174	3	Control Rod Drive to Reactor	NY	1	W	301-112	N/A	O	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-230	3/4	N_2 Purge to TIP	N	1	N_2	201.2-65	N/A	O	GT	N/A	M	N/A	N/A	H	LO	O	O	O	N/A	(8)
X-230	3/4	N_2 Purge to TIP	N	1	N_2	201.2-39	N/A	O	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-230	3/4	N_2 Purge to TIP	N	1	N_2	201.2-40	N/A	O	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-121	3/4	Breathing Air	N	8	A		N/A	O	GT	N/A	M	N/A	N/A	H	N	LC	O	LC	N/A	(10)
X-121	3/4	For Drywell	N	8	A		N/A	I	GT	N/A	M	N/A	N/A	H	N	C	O	C	N/A	
X-121	3/4	" " "	N	8	A		N/A	I	GT	N/A	M	N/A	N/A	H	N	C	O	C	N/A	

Isolation Valves

Prim. Cont. Penetration Number	Line Size, In.	System	Is System an Engineered safety function	Figure	Process Fluid	Valve Number	Isolation Signal Code(s)	Location	Type	Actuator	Primary Actuation Mode	Secondary Actuation Mode	Full Closure Time, sec.	Power Source	Position Indication in Control Rm.	Positions: Normal	Positions: Shutdown	Positions: Post Accident	Positions: Power Failure	Compliance with GDC 55, 56 and 57
XS-316	3/4	D/P Points for ILRT	N	2	A		N/A	O	GT	N/A	M	N/A	N/A	H	N	O	O	O	N/A	(8)
XS-316	3/4	D/P Points for ILRT	N	2	A		N/A	O	GB	N/A	M	N/A	N/A	H	N	O	O	O	N/A	
XS-316	3/4	D/P Points for ILRT	N	2	A		N/A	O	GB	N/A	M	N/A	N/A	H	N	C	C	C	N/A	
XS-321	3/4	Supp. Atmos. Dil. Sample	N	2	A	201.2-23	E	O	GB	SO	A	RMC	60	DC	D	C	C	C	C	(6)
XS-321	3/4	Supp. Atmos. Dil. Sample	N	2	A	201.2-24	E	O	GB	SO	A	RMC	60	DC	D	C	C	C	C	
XS-321	3/4	Supp. Atmos. Dil. Sample	N	2	A	201.2-436	N/A	O	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
XS-321		Suppr. O2 Samp. Ret	N	2	A		N/A	O	GB	M	N/A	N/A	N/A	H	N	O	O	O	N/A	(6)
XS-321		Suppr. O2 Samp. Ret	N	2	A	201.2-70	N/A	O	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
XS-321		Suppr. O2 Samp. Ret	N	2	A	201.2-71	N/A	O	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
XS-328	3/4	Suppr. Sample Ret.	N	2	A	201.2-109	E	O	GB	AO	A	RMC	60	A	D	O	O	C	C	(6)
XS-328	3/4	Suppr. Sample Ret.	N	2	A	201.2-112	E	O	GB	AO	A	RMC	60	A	D	O	O	C	C	
XS-328	3/4	Suppr. Sample Supply	N	2	A	201.2-111	E	O	GB	AO	A	RMC	60	A	D	O	O	C	C	
XS-328	3/4	Suppr. Sample Supply	N	2	A	201.2-110	E	O	GB	AO	A	RMC	60	A	D	O	O	C	C	
XS-324	12	Cont. Spray Suction	Y	1	W	80-21	Manual	O	GT	MO	RMC	N/A	70	AC	D	O	O	O	AI	(2)
XS-325	12	Cont. Spray Suction	Y	1	W	80-22	Manual	O	GT	MO	RMC	N/A	70	AC	D	O	O	O	AI	
XS-326	3	Torus Spray	Y	1	W	80-67	N/A	O	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	(2)
XS-326	3	Torus Spray	Y	1	W	80-19	N/A	O	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
XS-326	3	Torus Spray	Y	1	W	80-39	N/A	C	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	

Prim. Cont. Penetration Number	Line Size, In.	System	Is System an Engineered safety function	Figure	Process Fluid	Isolation Valves: Valve Number	Isolation Signal Code(s)	Location	Type	Actuator	Primary Actuation Mode	Secondary Actuation Mode	Full Closure Time, sec.	Power Source	Position Indication in Control Rm.	Positions: Normal	Shutdown	Post Accident	Power Failure	Compliance with GDC 55, 56 and 57
XS-327	20	Suppr. Vent&Purge	N	1	N_2	201-16	E	O	B	AO	A	RMC	60	A	D	C	C	C	C	(6)
XS-327	20	Suppr. Vent&Purge	N	1	N_2	201-17	E	O	B	MO	A	RMC	60	AC	D	C	C	C	AI	
XS-327	3	N_2 Makeup & Bleed	N	1	N_2	201.2-33	E	O	DCV	AO	A	RMC	60	DC	D	C	C	C	C	(6)
XS-327	3	N_2 Makeup & Bleed	N	1	N_2	201.2-06	E	O	DCV	AO	A	RMC	60	DC	D	C	C	C	C	
XS-330	4	Torus Makeup	N	1	W	58.1-01	Manual	O	DCV	AO	RMC	N/A	60	A	D	C	C	C	C	(11)
XS-330	4	Torus Makeup	N	1	W	58.1-02	N/A	O	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
XS-332	12	Core Spray Suction	Y	1	W	81-01	Manual	O	GT	MO	RMC	N/A	90	AC	D	O	O	O	AI	(2)
XS-334	6	Core Spray Test Ret	Y	1	W	40-05	E	O	GT	MO	A	RMC	90	AC	D	C	C	C	AI	(12)
XS-335	6	Core Spray Test Ret	Y	1	W	40-06	E	O	GT	MO	A	RMC	90	AC	D	C	C	C	AI	
XS-336	12	Core Spray Suction	Y	1	W	81-22	Manual	O	GT	MO	RMC	N/A	90	AC	D	O	O	O	AI	(2)
XS-337	12	Core Spray Suction	Y	1	W	81-21	Manual	O	GT	MO	RMC	N/A	90	AC	D	O	O	O	AI	(2)
XS-333	12	Core Spray Suction	Y	1	W	81-02	Manual	O	GT	MO	RMC	N/A	90	AC	D	O	O	O	AI	(2)
XS-340	20	Torus Air Vent&Fill	N	1	A	201-08	E	O	B	AO	A	RMC	60	DC	D	C	C	C	C	(6)
XS-340	20	Torus Air Vent&Fill	N	1	A	201-07	E	O	B	MO	A	RMC	60	AC	D	C	C	C	AI	
XS-341	12	Cont. Spray Suction	Y	1	W	80-02	Manual	O	GT	MO	RMC	N/A	70	AC	D	O	O	O	AI	(2)
XS-342	12	Cont. Spray Suction	Y	1	W	80-01	Manual	O	GT	MO	RMC	N/A	70	AC	D	O	O	O	AI	(2)
XS-352	6	Cont.Spray TestLine	Y	9	W	80-43	N/A	O	GT	N/A	M	N/A	N/A	H	I	LC	LC	LC	N/A	(12)
XS-352	4	Cont.Spray Test Line to Waste Disposal	N	9	W	80-42	N/A	O	GT	N/A	M	N/A	N/A	H	I	LC	LC	LC	N/A	(9)

PRIMARY CONTAINMENT ISOLATION SYSTEM DATA

Isolation Valves

Prim. Cont. Penetration Number	Line Size, In.	System	Is System an Engineered safety function	Figure	Process Fluid	Valve Number	Isolation Signal Code(s)	Location	Type	Actuator	Primary Actuation Mode	Secondary Actuation Mode	Full Closure Time, sec.	Power Source	Position Indication in Control Rm.	Positions: Normal	Positions: Shutdown	Positions: Post Accident	Positions: Power Failure	Compliance with GDC 55, 56 and 57
XS-354	3	Torus Spray	Y	1	W	80-65	N/A	0	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	(2)
XS-354	3	Torus Spray	Y	1	W	80-66	N/A	0	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
XS-354	3	Torus Spray	Y	1	W	80-68	N/A	0	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
XS-365	20	Reactor Water Cleanup	N	1	W	63.1-02	N/A	0	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	(13)
XS-365	20	Relief to Torus	N	1	W	63.1-01	N/A	0	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
XS-362	4	Torus Drain	N	1	W	121-03	N/A	0	plg	N/A	M	N/A	N/A	H	N	C	C	C	N/A	(14)
XS-363	4	Torus Drain	N	1	W	121-02	N/A	0	plg	N/A	M	N/A	N/A	H	N	C	C	C	N/A	(14)
XS-364	4	Torus Drain	N	1	W	121-01	N/A	0	plg	N/A	M	N/A	N/A	H	N	C	C	C	N/A	(14)
XS-315 &XS-319	30	Vacuum Relief System	N	1	A	68-01	Manual	0	CK	AO	RF	N/A	N/A	A	D	C	C	C	C	(15)
XS-313 &XS-317	30	Vacuum Relief System	N	1	A	68-02	Manual	0	CK	AO	RF	N/A	N/A	A	D	C	C	C	C	(15)
	30		N	1	A	68-08	Manual	0	B	AO	DP	RMC	N/A	A	D	C	C	C	C	
	30		N	1	A	68-05	Manual	0	ck	AO	RF	N/A	N/A	A	D	C	C	C	C	
XS-314 &XS-318	30	Vacuum Relief System	N	1	A	68-03	Manual	0	ck	AO	RF	N/A	N/A	A	D	C	C	C	C	(15)
	30		N	1	A	68-09	Manual	0	B	AO	DP	RMC	N/A	A	D	C	C	C	C	
	30		N	1	A	68-06	Manual	0	ck	AO	RF	N/A	N/A	A	D	C	C	C	C	
XS-316 &XS-320	30	Vacuum Relief System	N	1	A	68-04	Manual	0	ck	AO	RF	N/A	N/A	A	D	C	C	C	C	(15)
	30		N	1	A	68-10	Manual	0	B	AO	DP	RMC	N/A	A	D	C	C	C	C	
	30		N	1	A	68-07	Manual	0	ck	AO	RF	N/A	N/A	A	D	C	C	C	C	

Footnotes for Isolation Valve Table

1. The valves in the emergency condenser steam supply line remain open during an accident unless there is a break in the emergency condenser line, indicated by high steam flow in the emergency condenser line or high radiation in the emergency condenser vents. These signals automatically close the valves.

2. The air operated valve in the emergency condenser remains closed during accident conditions. They are opened by high reactor pressure or low-low water level signals. The air operated valves will then remain opened unless a break in an emergency condenser line is indicated as discussed in (1) above.

3. The Core Spray System is considered to be an extension of containment, therefore core spray valves 40-01, 40-09, 40-10 and 40-11 do not automatically isolate. These valves open on a low reactor pressure signal in conjunction with a high drywell pressure or low-low reactor water level. If the Core Spray System is not needed to maintain reactor vessel water level, these valves can be isolated manually.

4. The drywell vent and fill line consists of a 20-inch line which penetrates primary containment. Once outside primary containment, it branches into a 24-inch vent line and a 4-inch nitrogen supply line with each line containing two isolation valves.

5. There are two lines per penetration for the reactor recirculation system instrumentation. Each line has a manual gate valve and a flow check valve outside containment.

6. The Containment Spray System is considered to be an extension of containment, therefore, the containment spray valves do not automatically isolate. If the Containment Spray System is not required to mitigate the consequences of an accident, it can be manually isolated from the control room.

ABBREVIATIONS

Engineered Safety Function

N = NO
Y = YES

Position Indication in Control Room

D = Direct
I = Indirect
N = None
Others stated in Table

Fluid

A = Air
B = Sodium Pentaborate
S = Steam
W = Water
N_2 = Nitrogen

Isolation Valve Location

I = Inside Containment
O = Outside Containment
Others stated in Table

Isolation Valve Actuation Mode

A = Automatic
HF = High Flow
M = Manual
OP = Overpressure
RF = Reverse Flow
RMC = Remote Manual Control Room
RM = Remote Manual (Local)
DP = Differential Pressure

Isolation Valve Positions

AI = As Is
C = Closed
O = Open

Isolation Valve Type

A = Angle
B = Butterfly
BCK = Ball Check
BL = Ball
CK = Check
DCV = Diaphragm Control Valve
FCV = Flow Check Valve
FF = Flow Fuse
GB = Globe
GT = Gate
RV = Relief
SCV = Stop Check
SV = Solenoid
VB = Vacuum Breaker
PLG = Plug

Isolation Valve Power Source

A = Air
AC = AC
DC = DC
H = Hand

Isolation Valve Actuator

AO = Air
MO = Motor
SO = Solenoid

Isolation Signal Codes

Code or Group	Parameter(s) Sensed for Isolation	Set Point (units)
A	High Steam Flow - Main Steam Line	≤ 105 psid
	High Radiation - Main Steam Line	≤ 5 times normal background
	Low Reactor Pressure	≥ 850 psig
	Low Low Low Condenser Vacuum	≥ 7 inches mercury vacuum
	High Temperature - Main Steam Line Tunnel	≤ 200F
	Low Low Reactor Water Level	≥ 5 inches Indicator Scale
B	High Steam Flow - Emergency Cooling System	≤ 19 psid
	High Radiation - Emergency Cooling System Vent	≤ 25 mr/hr

ABBREVIATIONS (Continued)

Isolation Signal Codes
(Continued)

Code or Group	Parameter(s) Sensed for Isolation	Set Point (units)
C	Manual	N/A
D	Low-Low Reactor Water Level	> 5 inches (Indicator Scale)
	High Area Temperature	≤ 190F for Cleanup System ≤ 170F for Shutdown Cooling System
E	Low-Low Reactor Water Level	> 5 inches (Indicator Scale)
	High Drywell Pressure	≤ 3.5 psig

Figure Codes

1 =	C-18006-C	Sheets 1 & 2
2 =	C-18014-C	Sheets 1 & 2
3 =	C-18015-C	
4 =	C-18017-C	
5 =	C-18020-C	
6 =	C-18041-C	Sheet 2
7 =	C-18022-C	Sheet 1
8 =	C-18578-C	
9 =	C-18012-C	Sheet 2
10 =	C-18002-C	Sheet 1

Footnotes for Compliance
with GDC 55, 56 and 57

(1) This line meets the requirements of GDC 55 regarding two isolation barriers.

(2) This is an essential system which is not required to isolate on a containment isolation signal.

(3) This does not meet the requirement of GDC 55 in that a check valve is used as the automatic isolation valve outside containment. However, the use of check valves as the automatic isolation valve outside containment has been previously justified in Niagara Mohawk's Technical Supplement to Petition for Conversion from Provisional Operating License to Full-Term Operating License. Therefore, no modifications are required.

(4) This line does not meet the requirements of GDC 57 in that a check valve is used as the automatic isolation valve outside containment. However, the use of check valves as the automatic isolation valve outside containment has been previously justified in Niagara Mohawk's Technical Supplement to Petition for Conversion from Provisional Operating License to Full-Term Operating License. Therefore, no modifications are required.

(5) This line meets the requirements of GDC 57 for a closed system penetrating containment.

(6) This line does not meet the requirements of GDC 56 in that both isolations are located outside containments. This is justified because these lines, although initially isolated following an accident, are required to be opened for controlled venting and purging and monitoring. Therefore, locating the valves in a less hostile environment (i.e. outside of the drywell during a LOCA) is considered more acceptable for safety.

(7) This line meets the requirements of GDC 56 regarding two isolation barriers.

(8) This is an instrument line which is exempt from the requirements of GDC 55, 56 and 57.

(9) This line does not meet the requirements of GDC 55 in that there is not an inside isolation valve which is locked closed or capable of automatic isolation. As indicated in our response to I. E. Bulletin 79-02, this line will be modified during the spring 1981 refueling outage to provide two automatic isolation valves.

(10) This line does not meet the requirements of GDC 56 in that the inside isolation valve is not locked closed or capable of automatic isolation. Therefore, the inside isolation valve on this line will be a locked closed valve before Nine Mile Point Unit 1 starts up from the spring 1981 refueling outage.

(11) The torus makeup line does not meet the requirements of GDC 56 in that both isolation valves are located outside containment and one of them is a check valve. Although the remotely operable valve is normally closed, makeup to the torus may be required during/following an accident. Therefore, locating the valves in a less hostile environment (i.e. outside of the drywell during a LOCA) is considered more acceptable for safety. The check valve will prevent any reverse flow from the drywell while the remotely operable valve is opened for torus makeup. Therefore, no modifications are required.

(12) These valves are not considered isolation valves from a containment isolation standpoint, since the core spray and containment spray systems are considered to be extensions of containment. These valves close on a containment isolation signal so that water is not diverted through the test lines when the core spray and containment spray systems are required to operate. Therefore, no modifications are required.

(13) The reactor water cleanup relief to torus does not meet GDC 56 in that both isolation valves are located outside containment. Since this is a submerged line, it is not practical to have a locked closed or automatic isolation valve inside containment. The use of check valves as the automatic isolation valve outside containment has been previously justified in Niagara Mohawk's Technical Supplement to Petition for Conversion from Provisional Operating License to Full-Term Operating License. Therefore, no modifications are required.

(14) These lines do not meet GDC 56 in that there is only one manual valve located outside the containment. However, the lines have a bolted flange at the end which provides a second isolation barrier. Access to these lines is also located in a locked compartment. Therefore, no modifications are required.

(15) These lines do not meet GDC 56 in that the isolation valves are all located outside containment. However, the function of these lines are to equalize pressure between the torus and drywell and the containment and atmosphere during normal operation. Therefore, these valves are required to be located outside containment to perform these intended functions. No modifications are required.

TABLE 1 to
TMI ACTION PLAN ITEM NO. II.E.4.2
CONTAINMENT ISOLATION DEPENDABILITY
Nine Mile Point Unit 1

PRIMARY CONTAINMENT ISOLATION SYSTEM DATA

Isolation Valves

Prim. Cont. Penetration Number	Line Size, In.	System	Is System an Engineered safety function	Figure	Process Fluid	Valve Number	Isolation Signal Code(s)	Location	Type	Actuator	Primary Actuation Mode	Secondary Actuation Mode	Full Closure Time, sec.	Power Source	Position Indication in Control Rm.	Positions: Normal	Positions: Shutdown	Positions: Post Accident	Positions: Power Failure	Compliance with GDC 55, 56 and 57
X-2A	24	Main Steam	N	1	S	01-01	A	I	GB	MO	A	RMC	10	AC	D	O	C	C	AI	(1)
X-2A	24	Main Steam	N	1	S	01-03	A	O	A	AO	A	RMC	10	A	D	O	C	C	C	
X-2B	24	Main Steam	N	1	S	01-02	A	I	GT	MO	A	RMC	10	AC	D	O	C	C	AI	(1)
X-2B	24	Main Steam	N	1	S	01-04	A	O	A	AO	A	RMC	10	A	D	O	C	C	C	
X-3A	10	Emergency Condenser	Y	1	S	39-09	B	O	GT	MO	A	RMC	38	AC	D	O	O	O(1)	AI	(2)
X-3A	10	Emergency Condenser	Y	1	S	39-07	B	O	GT	MO	A	RMC	38	DC	D	O	O	O(1)	AI	
X-3B	10	Emergency Condenser	Y	1	S	39-10	B	O	GT	MO	A	RMC	38	AC	D	O	O	O(1)	AI	(2)
X-3B	10	Emergency Condenser	Y	1	S	39-08	B	O	GT	MO	A	RMC	38	DC	D	O	O	O(1)	AI	
X-4A	18	Feedwater	N	1	W	31-04	C	O	GB	MO	RMC	N/A	60	AC	D	O	O	O	AI	
X-4A	18	Feedwater	N	1	W	31-02	N/A	O	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	(2)
X-4B	18	Feedwater	N	1	W	31-03	C	O	GB	MO	RMC	N/A	60	AC	D	O	O	O	AI	
X-4B	18	Feedwater	N	1	W	31-01	N/A	O	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	(2)
X-5A	10	Emergency Condenser	Y	1	W	39-04	N/A	I	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	(2)
X-5A	10	Emergency Condenser	Y	1	W	39-06	B	O	DCV	AO	A	RMC	60	A	D	C	C	C(2)	O	
X-5B	10	Emergency Condenser	Y	1	W	39-03	N/A	I	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	(2)
X-5B	10	Emergency Condenser	Y	1	W	39-05	B	O	DCV	AO	A	RMC	60	A	D	C	C	C(2)	O	
X-7	14	Shutdown Cool. Rtn.	N	1	W	38-13	D	I	GT	MO	A	RMC	40	AC	D	C	O	C	AI	(3)
X-7	14	Shutdown Cool. Rtn.	N	1	W	38-12	N/A	O	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-8	14	Shutdown Cool. Sup.	N	1	W	38-01	D	I	GT	MO	A	RMC	40	AC	D	C	O	C	AI	(1)
X-8	14	Shutdown Cool. Sup.	N	1	W	38-02	D	O	GT	MO	A	RMC	40	DC	D	C	O	C	AI	
X-9	6	React.Cleanup Supl.	N	1	W	33-02	D	I	GT	MO	A	RMC	18	AC	D	O	O	C	AI	(1)
X-9	6	React.Cleanup Supl.	N	1	W	33-04	D	O	GT	MO	A	RMC	18	DC	D	O	O	C	AI	
X-154	6	React.Cleanup Rtn	N	1	W	33-01	D	I	GT	MO	A	RMC	18	AC	D	O	O	C	AI	(3)
X-154	6	React.Cleanup Rtn	N	1	W	33-03	N/A	O	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	

Prim. Cont. Penetration Number	Line Size, In.	System	Is System an Engineered safety function	Figure	Process Fluid	Isolation Valves: Valve Number	Isolation Signal Code(s)	Location	Type	Actuator	Primary Actuation Mode	Secondary Actuation Mode	Full Closure Time, sec.	Power Source	Position Indication in Control Rm.	Positions: Normal	Shutdown	Post Accident	Power Failure	Compliance with GDC 55, 56 and 57
X-12B	8	RBCLC to Drywell Coolers	N	1	W	70-95	N/A	O	CK	N/A	RF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	(4)
X-13B	8	RBCLC from Dry. Clrs	N	1	W	70-94	Manual	O	GT	MO	RMC	N/A	30	DC	D	O	O	O	AI	(5)
X-13A	12	Core Spray Inlet	Y	1	W	40-01	Manual	I	GT	MO	A	RMC	90	AC	D	C	C	O(3)	AI	(2)
X-13A	12	Core Spray Inlet	Y	1	W	40-09	Manual	I	GT	MO	A	RMC	90	AC	D	C	C	O(3)	AI	
X-13A	12	Core Spray Inlet	Y	1	W	40-02	Manual	O	GT	MO	A	RMC	N/A	AC	D	LO	O	O(3)	AI	
X-14	12	Core Spray Inlet	Y	1	W	40-10	Manual	I	GT	MO	A	RMC	90	AC	D	C	C	O(3)	AI	(2)
X-14	12	Core Spray Inlet	Y	1	W	40-11	Manual	I	GT	MO	A	RMC	90	AC	D	C	C	O(3)	AI	
X-14	12	Core Spray Inlet	Y	1	W	40-12	Manual	O	GT	MO	A	RMC	N/A	AC	D	LO	O	O(3)	AI	
X-18	24	Drywell Vent & Purge	N	2	A	201-09	E	O	B	MO	A	RMC	60	AC	D	C	O	C	AI	(6)
X-18	24	Air Supply	N	2	A	201-10	E	O	B	AO	A	RMC	60	A	D	C	O	C	C	
X-19	20	Drywell N_2 Vent&Fill	N	2	N_2	201-31	E	O	B	MO	A	RMC	60	AC	D	C	O	C	AI	(6)
X-19	20	Drywell N_2 Vent&Fill	N	2	N_2	201-32	E	O	B	AO	A	RMC	60	A	D	C	O	C	C	
X-19	4	Drywell N_2 Vent&Fill	N	2	N_2	201.2-03	E	O	DCV	AO	A	RMC	60	A	D	O	O	C	C	
X-19	4	Drywell N_2 Vent&Fill	N	2	N_2	201.2-32	E	O	DCV	AO	A	RMC	60	A	D	O	O	C	C	
X-20	1	Cont. Atmos. Dil Sample Ret.	N	2	A	201.7-10	E	O	DCV	AO	A	RMC	60	A	D	O	O	C	C	(6)
X-20	1	Cont. Atmos. Dil Sample Ret.	N	2	A	201.7-11	E	O	DCV	AO	A	RMC	60	A	D	O	O	C	C	
X-25	4	Drywell Floor Dr. Sump Outlet	N	1	W	83.1-11	E	I	GT	MO	A	RMC	60	AC	D	O	O	C	AI	(7)
X-25	4	Drywell Floor Dr. Sump Outlet	N	1	W	83.1-12	E	O	DCV	AO	A	RMC	60	A	D	O	O	C	C	

PLANT Nine [illegible] Point Unit 1

PRIMARY CONTAINMENT ISOLATION SYSTEM DATA

Prim. Cont. Penetration Number	Line Size, In.	System	Is System an Engineered safety function	Figure	Process Fluid	Isolation Valves: Valve Number	Isolation Signal Code(s)	Location	Type	Actuator	Primary Actuation Mode	Secondary Actuation Mode	Full Closure Time, sec.	Power Source	Position Indication in Control Rm.	Positions: Normal	Shutdown	Post Accident	Power Failure	Compliance with GDC 55, 56 and 57
X-26	3	Drywell Equip. Dr. Sump Outlet	N	1	W	83.1-09	E	I	GT	MO	A	RMC	60	AC	D	O	O	C	AI	(7)
X-26	3	Drywell Equip. Dr. Sump Outlet	N	1	W	83.1-10	E	O	DCV	AO	A	RMC	60	A	D	O	O	C	C	
X-34(5)	1	Reactor Recir. Sys. Inst.	N	5	W		N/A	O	GT	N/A	M	N/A	N/A	H	N	O	O	O	N/A	(8)
X-34	1	Reactor Recir. Sys. Inst.	N	5	W		N/A	O	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-34	1	Reactor Recir. Sys. Inst.	N	5	W		N/A	O	GT	N/A	M	N/A	N/A	H	N	O	O	O	N/A	
X-34	1	Reactor Recir. Sys. Inst.	N	5	W		N/A	O	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-35(5)	1	Reactor Recir. Sys. Inst.	N	5	W		N/A	O	GT	N/A	M	N/A	N/A	H	N	O	O	O	N/A	(8)
X-35	1	Reactor Recir. Sys. Inst.	N	5	W		N/A	O	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-35	1	Reactor Recir. Sys. Inst.	N	5	W		N/A	O	GT	N/A	M	N/A	N/A	H	N	O	O	O	N/A	
X-35	1	Reactor Recir. Sys. Inst.	N	5	W		N/A	O	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-36(5)	1	React.Recir.SysInst	N	5	W		N/A	O	GT	N/A	M	N/A	N/A	H	N	O	O	O	N/A	(8)
X-36	1	React.Recir.SysInst	N	5	W		N/A	O	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-36	1	React.Recir.SysInst	N	5	W		N/A	O	GT	N/A	M	N/A	N/A	H	N	O	O	O	N/A	
X-36	1	React.Recir.SysInst	N	5	W		N/A	O	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	

Prim. Cont. Penetration Number	Line Size, In.	System	Is System an Engineered safety function	Figure	Process Fluid	Isolation Valves: Valve Number	Isolation Signal Code(s)	Location	Type	Actuator	Primary Actuation Mode	Secondary Actuation Mode	Full Closure Time, sec.	Power Source	Position Indication in Control Rm.	Positions: Normal	Shutdown	Post Accident	Power Failure	Compliance with GDC 55, 56 and 57
X-37(5)	1	React.Recir.SysInst	N	5	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	(8)
X-37	1	React.Recir.SysInst	N	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-37	1	React.Recir.SysInst	N	5	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	
X-37	1	React.Recir.SysInst	N	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-43(5)	1	React.Recir.SysInst	N	5	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	(8)
X-43	1	React.Recir.SysInst	N	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-43	1	React.Recir.SysInst	N	5	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	
X-43	1	React.Recir.SysInst	N	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-28(5)	1	React.Recir.SysInst	N	5	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	(8)
X-28	1	React.Recir.SysInst	N	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-28	1	React.Recir.SysInst	N	5	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	
X-28	1	React.Recir.SysInst	N	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-32(5)	1	React.Recir.SysInst	N	5	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	(8)
X-32	1	React.Recir.SysInst	N	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-32	1	React.Recir.SysInst	N	5	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	
X-32	1	React.Recir.SysInst	N	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-31(5)	1	React.Recir.SysInst	N	5	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	(8)
X-31	1	React.Recir.SysInst.	N	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-31	1	React.Recir.SysInst	N	5	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	
X-31	1	React.Recir.SysInst	N	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-30(5)	1	React.Recir.SysInst	N	5	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	(8)
X-30	1	React.Recir.SysInst	N	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-30	1	React.Recir.SysInst.	N	5	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	
X-30	1	React.Recir.SysInst	N	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	

PRIMARY CONTAINMENT ISOLATION SYSTEM DATA

Prim. Cont. Penetration Number	Line Size, In.	System	Is System an Engineered safety function	Figure	Process Fluid	Isolation Valves: Valve Number	Isolation Signal Code(s)	Location	Type	Actuator	Primary Actuation Mode	Secondary Actuation Mode	Full Closure Time, sec.	Power Source	Position Indication in Control Rm.	Positions: Normal	Shutdown	Post Accident	Power Failure	Compliance with GDC 55, 56 and 57
X-29(5)	1	React.Recir.SysInst	N	5	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	(8)
X-29	1	React.Recir.SysInst	N	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-29	1	React.Recir.SysInst	N	5	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	
X-29	1	React.Recir.SysInst	N	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-41(5)	1	React.Recir.SysInst	N	5	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	(8)
X-41	1	React.Recir.SysInst	N	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-41	1	React.Recir.SysInst	N	5	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	
X-41	1	React.Recir.SysInst	N	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-44(5)	1	React.Recir.SysInst	N	5	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	(8)
X-44	1	React.Recir.SysInst	N	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-44	1	React.Recir.SysInst	N	5	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	
X-44	1	React.Recir.SysInst	N	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-42(5)	1	React.Recir.SysInst	N	5	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	(8)
X-42	1	React.Recir.SysInst	N	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-42	1	React.Recir.SysInst	N	5	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	
X-42	1	React.Recir.SysInst	N	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-38(5)	1	React.Recir.SysInst	N	5	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	(8)
X-38	1	React.Recir.SysInst.	N	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-38	1	React.Recir.SysInst	N	5	W		N/A	0	GT	N/A	M	N/A	N/A.	H	N	0	0	0	N/A	
X-38	1	React.Recir.SysInst	N	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-47(5)	1	React.Recir.SysInst	N	5	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	(8)
X-47	1	React.Recir.SysInst	N	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
X-47	1	React.Recir.SysInst.	N	5	W		N/A	0	GT	N/A	M	N/A	N/A	H	N	0	0	0	N/A	
X-47	1	React.Recir.SysInst	N	5	W		N/A	0	FCV	N/A	HF	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	