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RICHARD B. ABBOTT
Vice President
Nuclear Generation

November 29, 1995
NMP2L 1596

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

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

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

Gentlemen:

Subject: ***SUBMITTAL OF REVISION 8 TO THE NINE MILE POINT NUCLEAR STATION UNIT 2 UPDATED SAFETY ANALYSIS REPORT AND THE ANNUAL 10 C.F.R. §50.59 SAFETY EVALUATION SUMMARY REPORT***

Pursuant to the requirements of 10 C.F.R. §50.71(e) and 10 C.F.R. §50.59(b), Niagara Mohawk Power Corporation hereby submits Revision 8 to the Nine Mile Point Nuclear Station Unit 2 Updated Safety Analysis Report (USAR) and the annual Safety Evaluation Summary Report.

One (1) signed original and ten (10) copies of the USAR, Revision 8, are enclosed. Copies are also being sent directly to the Regional Administrator, Region I, and the NRC Resident Inspector at Nine Mile Point. The USAR revision contains changes made since the submittal of Revision 7 in October 1994. In addition, numerous USAR sections have been edited to eliminate blank and partial pages. The elimination of blank and partial pages is editorial in nature and does not update or change substantive information previously described in the USAR. Changes to the Niagara Mohawk Quality Assurance Topical Report (NMPC-QATR-1) that were previously submitted with Unit 1 UFSAR (Updated) Revision 13, dated June 1995, have been incorporated in Unit 2 USAR Appendix B. The certification required by 10 C.F.R. §50.71(e)(2) is attached.

The enclosed annual Safety Evaluation Summary Report contains a brief description of changes, tests, and experiments, and includes a summary of the safety evaluation of each. None of the safety evaluations involved an unreviewed safety question as defined in 10 C.F.R. §50.59(a)(2).

Very truly yours,

Richard B Abbott

Richard B. Abbott
Vice President - Nuclear Generation

A063
1/11

RBA/JJL/kap
Enclosures

pc: Mr. T. T. Martin, Regional Administrator, Region I
Mr. L. B. Marsh, Director, Project Directorate I-1, NRR
Mr. G. E. Edison, Senior Project Manager, NRR
Mr. B. S. Norris, Senior Resident Inspector
Records Management

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

CERTIFICATION

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

By: Richard B. Abbott

Richard B. Abbott
Vice President - Nuclear Generation

Subscribed and sworn to before me

this 29 day of November, 1995.

Beverly W Ripka
Notary Public

BEVERLY W. RIPKA
Notary Public State of New York
Qual. in Oswego Co. No. 4614879
My Commission Exp. ~~Mar. 31.1~~ 2/28/96

Enclosure to
NMP2L 1596

NINE MILE POINT - UNIT 2

SAFETY EVALUATION SUMMARY REPORT

1995

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

Safety Evaluation No.: 87-046

Implementation Document No.: Mod. PN2Y87MX063

USAR Affected Pages: Figure 10.1-5b

System: Condensate (CND)

Title of Change: An Addition of Cyclone Separator to Condensate Booster Pump Seal Water Injection Lines

Description of Change:

This modification installed two new cyclone separators on the seal injection water lines of each of the condensate booster pumps. Also, a new flow restriction orifice was installed upstream of each cyclone separator and associated valves.

Safety Evaluation Summary:

This modification is in accordance with ANSI B31.1-1973. This new equipment interfaces only with the CND system and has no impact to any other systems. This modification will ensure condensate booster pump reliability and prevent costly pump downtime for maintenance on mechanical seals.

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

Safety Evaluation No.: 89-075 Rev. 7 & 8

Implementation Document No.: Mod. PN2Y87MX038

USAR Affected Pages: Figures 9.5-8 Sh 1 & 2, 9.5-10 Sh 1, 9.5-24, 9.5-29

System: Communications (COJ, COP, COS)

Title of Change: Addition of Communication Equipment

Description of Change:

Modifications to the Gaitronics communications system were previously reported with USAR revisions dated October 30, 1991, October 29, 1992, and October 29, 1993, under Safety Evaluation 89-075, Revisions 3, 4, 5, 6 and 7.

Additional modifications to add/improve speakers, jacks, and associated equipment to the Gaitronics communications system have been made as evaluated under Safety Evaluation 89-075, Revisions 7 and 8.

Safety Evaluation Summary:

This modification will add/improve communication capabilities to meet the requirements in USAR Section 9.5.2. These changes will improve communication capabilities required for surveillance testing, personnel to respond to alarms in areas with high noise levels, and add communication equipment in areas that have been identified as needing communication capabilities.

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

Safety Evaluation No.: 91-003 Rev. 7, 8, & 9

Implementation Document No.: Calculation ES-269

USAR Affected Pages: N/A

System: Secondary Containment

Title of Change: Secondary Containment Drawdown Analysis

Description of Change:

Revisions 7, 8, and 9 of the safety evaluation evaluated plant operation for the remainder of the fourth fuel cycle. The following parameters were changed for the ΔT requirements for the fourth cycle as compared with the previous cycle.

1. <u>Spent fuel heat loads:</u>

   A spent fuel pool heat load of 4.49×10^6 Btu/hr corresponding to 50 days after reactor shutdown (DARS) was used to define the ΔT requirements for the fourth fuel cycle.

   In order to reduce the ΔT requirements and, hence, heating of the building during the summer months, a lower spent heat load of 2.31×10^6 Btu/hr corresponding to 180 DARS was used to define the ΔT requirements for the remainder of the fourth refueling cycle.

2. <u>Unit cooler performance</u>:

   Based on the performance tests performed during the 1992-93 time period, a 2% degradation of unit coolers 2HVR*UC413A & B and an average degradation of 30% (same as previous cycle) for the remaining drawdown related unit coolers was used for defining the ΔT requirements for the entire fourth operating cycle. This provides sufficient margin to account for any further degradation that may occur over the next operating cycle.

3. <u>Piping heat load reductions</u>:

   To reduce the ΔT requirements, piping heat loads have been reduced assuming a minimum temperature of 80°F (Curve 2) and 90°F (Curves 3 and 4) in the building.

Safety Evaluation No.: 91-003 Rev. 7, 8, & 9 (cont'd.)

Description of Change: (cont'd.)

4. <u>Cubicle ΔT</u>:

For additional flexibility, drawdown analysis is done assuming emergency core cooling system, residual heat removal heat exchangers and reactor core isolation cooling cubicles in secondary containment are maintained at 8°F above the service water temperature.

The secondary containment in-leakage test performed on October 27, 1993, indicated that in-leakage is less than 90% of the value used in the drawdown analysis. This provides a margin of about 10% (same as the previous cycle) for any potential degradation of in-leakage over the next cycle.

NOTE: These changes were superseded by Safety Evaluation 94-049 and associated License Amendment No. 56.

Safety Evaluation Summary:

The drawdown analysis (Calculation ES-269 and subsequent dispositions) provides four curves that define ΔT requirements for the entire fourth fuel cycle.

Based on the evaluation performed, it is concluded that the use of new ΔT curves does not involve an unreviewed safety question.

Safety Evaluation No.: 91-068

Implementation Document No.: Mod. PN2Y89MX080

USAR Affected Pages: Figures 9.3-12h, 9.3-12k, 10.1-8b

System: Turbine Building Miscellaneous Drains

Title of Change: Reboiler Steam Line Drain Valve Interlock

Description of Change:

The turbine plant miscellaneous drain system removes condensate buildup from the steam supply lines either through the drain valves 2DTM-AOV128 and 2DTM-AOV144 or through bypass lines around these drain valves through restricting orifices sized to pass condensate.

The original design required the drain valves to open whenever the auxiliary steam supply valves 2ASS-STV112 and/or 2ASS-STV143 close, or whenever turbine first-stage pressure indicated insufficient extraction steam was available.

The interlocks between the auxiliary steam supply valves and their corresponding drain valve have been removed. This modification allows operator control of the drain valves irrespective of the steam supply to the clean steam reboilers and/or the building heating intermediate heat exchangers within the boundaries allowed by the turbine first-stage pressure sensor.

Safety Evaluation Summary:

The drain valves are nonsafety related and are not required for safe operation or shutdown of the plant.

This modification provides additional increased operator control which will result in an enhancement to plant efficiency that will not impact the safe operation or shutdown capabilities of the plant.

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

Safety Evaluation No.:	91-080
Implementation Document No.:	Mod. PN2Y88MX193
USAR Affected Pages:	8.3-10
System:	Low-Voltage Molded-Case Circuit Breakers for Power Distribution
Title of Change:	Replacement of Obsolete ITE Molded-Case Circuit Breakers

Description of Change:

This modification replaced six distribution panels in their entirety and various obsolete molded-case circuit breakers in motor control centers and other distribution panels. These breakers provide circuit protection for the low-voltage power distribution at Unit 2.

Safety Evaluation Summary:

This modification does not increase the probability of an accident or malfunction of equipment important to safety previously evaluated in the USAR. Replacement breakers are properly coordinated and adequately sized to their application in accordance with the standard ratings for the molded-case circuit breakers. Replacement of obsolete breakers and panels will preclude system outages, LCOs, and plant outage due to unavailable spares should any of these components fail in-service or surveillance testing.

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

Safety Evaluation No.: 91-089

Implementation Document No.: Simple Design Change SC2-0140-90

USAR Affected Pages: Figures 11.2-1d, 11.2-1g

System: Liquid Radwaste Management (LWS)

Title of Change: Retire Non-functional Conductivity Monitors

Description of Change:

This simple design change retired in place conductivity elements and conductivity indicating transmitters from the floor drain collector subsystem and the regenerant waste subsystem. These instruments provide display input only and have no logic function. Sparing the conductivity monitoring equipment eliminates repetitive maintenance and calibration. Grab samples are used for determining effluent conductivity in those areas where electronic monitoring is disabled.

Safety Evaluation Summary:

The LWS system provides diverse options for the processing of waste depending on the quality of the waste. However, in no case can the waste bypass a filtration or evaporation process. The conductivity of the waste is used to aid in the selection of a process method. Although grab samples will have to be used in lieu of electronic monitoring, no sacrifice to the integrity or function of the LWS will occur from the proposed change.

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

Safety Evaluation No.: 92-006 Rev. 1, 2, 4 & 5

Implementation Document No.: Mod. PN2Y89MX077

USAR Affected Pages: 4A-2, 5.2-21, 5.2-21a; Tables 3B-3 Sh 2, 6.2-56 Sh 7, 9.4-1 Sh 4; Figures 1.2-7 Sh 2, 1.2-11 Sh 3, 5.4-2b, 5.4-16a, 9.3-5g, 12.3-7, 12.3-40

System: Crack Arrest Verification

Title of Change: Installation of the Crack Arrest Verification System and RWCU Extension Tie-In

Description of Change:

This modification implemented the following changes:

1. Installation of the crack arrest verification system (CAVS) included a crack length monitor, water chemistry station, electrochemical potential monitor and tubing. The suction side of the CAVS was connected to the reactor recirculation system (RCS) sample line, downstream from the outboard isolation valve, 2RCS*SOV105, beyond the Class 2 line classification (i.e., connection will be made where the line is designated as Class 4). The return line of the CAVS was connected to the reactor water cleanup (RWCU) extension tie-in which is downstream from outboard isolation valve 2WCS*MOV112.

2. The RWCU extension tie-in begins at line 2-WCS-008-88-3, which is downstream of outboard containment isolation valve 2WCS*MOV112 in the RWCU valve cubicle located on elevation 240′, secondary containment. Existing valves 2WCS-V45 and 2WCS-V46 were replaced with 3/4-inch pipe. The test connection, which is used during the leak rate testing of valve 2WCS*MOV112, was maintained by adding a threaded cap and two new valves, 2WCS-V431 and 2WCS-V432. The RWCU extension tie-in also included the 3/4-inch pipe run, including one isolation valve, 2WCS-V390, and a check valve, 2WCS-V392, which are inside the RWCU valve cubicle, a penetration (i.e., W-7512-C) through the cubicle wall, and an isolation valve, 2WCS-V391, outside the cubicle, which were added via Temporary Modification 90-054 and are now permanent per this modification.

Safety Evaluation No.: **92-006 Rev. 1, 2, 4 & 5 (cont'd.)**

Safety Evaluation Summary:

The CAVS is not a safety-related system nor does it perform any safety-related function, and its addition to the Unit 2 design does not affect the safety and reliability of Unit 2. The system's function is to collect data to provide an indication of the performance of plant materials in the boiling water reactor environment. The Class 3 section of RWCU extension tie-in is considered Q and the Class 4 section of the RWCU extension tie-in is nonsafety related. Both of these sections are properly designed and will not affect the safe operation or safe shutdown capability of the plant.

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

Safety Evaluation No.: 92-033 Rev. 2

Implementation Document No.: Procedure N2-OSP-RHS-R@009

USAR Affected Pages: N/A

System: Residual Heat Removal (RHS)

Title of Change: Procedure N2-OSP-RHS-R@009

Description of Change:

This safety evaluation evaluated changes to procedure N2-OSP-RHS-R@009, which allows the testing of the pressure isolation valves in the RHS system which isolate the RHS heat exchanger from the reactor core cooling injection system (ICS). The steps of the procedure delineate the methodology for testing the system in order to comply with Technical Specifications 4.0.5 and 4.4.3.2.2. Testing of the valves in the RHS system was conducted during refuel outages.

Safety Evaluation Summary:

This procedure and method of testing will have no impact on the safe operation or capability to keep the plant in the safe shutdown condition because the ICS functions are not required in operational conditions 4 or 5 and the RHS system safety functions are unaffected.

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

Safety Evaluation No.: 93-004

Implementation Document No.: Temporary Mod. 93-008

USAR Affected Pages: N/A

System: 4.16-kV and 600-V Normal Ac Distribution

Title of Change: Alternate Feed to Transformer 2NJS-X1F

Description of Change:

The 600-V unit subdistribution transformer 2NJS-X1F (feeder to unit sub 2NJS-US5) was temporarily powered from 4.16-kV stub bus 2NNS-SWG015 instead of its normal source, 2NNS-SWG014, which was out of service for repair of a cracked bushing in cubicle 14-6.

Safety Evaluation Summary:

This temporary modification does not affect or deal with any safety-related equipment in the plant. An Engineering review of the USAR, Technical Specifications and related documents indicates that this temporary modification is acceptable with procedural controls and limitations. With this change in place, the normal ac distribution will continue performing its intended function.

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

Safety Evaluation No.: 93-008

Implementation Document No.: Simple Design Change SC2-0008-93

USAR Affected Pages: Figure 10.1-6c

System: Feedwater (FWP)

Title of Change: Seal Water Injection Strainer Drain Valve and Pressure Gage

Description of Change:

This simple design change added a differential pressure indicator and drain valves to the seal injection duplex strainers to facilitate drainage for periodic maintenance of the strainers.

Safety Evaluation Summary:

This change is specific to the FWP system. No significant effects on any other plant systems and/or interlocks are being introduced.

The addition of the differential pressure gage and drain valves will improve the performance of maintenance on the subject strainers. Therefore, the system maintainability/availability is improved due to the ease in periodic changeout of the strainer baskets.

The system structural integrity will not be significantly affected by this change because the weight of the valves and the differential pressure gage are negligible relative to the piping size and schedule.

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

Safety Evaluation No.: 93-011

Implementation Document No.: Temporary Mod. 93-015

USAR Affected Pages: N/A

System: Auxiliary Boiler

Title of Change: Defeat Seal Cooling Low Flow Trip for Auxiliary Boiler Recirculation Pumps

Description of Change:

This temporary modification jumpered the auxiliary boiler recirculation pump seal cooling water flow switches in order to allow the pumps to run when seal water flow is throttled back. Throttling the seal water is done to reduce the frequency of required boiler blowdowns.

Safety Evaluation Summary:

Although this temporary change may result in damage to the pump seals, there is no nuclear safety significance to the proposed change since the auxiliary boilers are not required for the safe shutdown of the reactor. The risk associated with operating the boiler with the low flow trip defeated is acceptable since plant impact will be limited to boiler operability. Implementation of the proposed change does not constitute an unreviewed safety question.

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

Safety Evaluation No.:	93-017 Rev. 2
Implementation Document No.:	Simple Design Change SC2-0375-91
USAR Affected Pages:	N/A
System:	Standby Liquid Control (SLS)
Title of Change:	RRCS Logic Change to Preclude SLS Inoperability

Description of Change:

This modification changed the storage tank level zero logic in the redundant reactivity control system (RRCS) panels from deenergize-to-trip to energize-to-trip. Previously, if a RRCS panel was taken out of service, the respective SLS loop would become inoperable because SLS identified the RRCS out-of-service signal as a SLS tank level zero. The SLS tank level zero interlock disables the SLS pumps to protect them from damage due to running them dry. Previously, temporary jumpers needed to be installed if the RRCS panels were taken out of service to maintain SLS operability. This change eliminates the need for these jumpers and provides annunciation in the main control room to alert the operators if the storage tank level zero alarm is activated. This modification was incorporated into SLS and RRCS by changing the logic in RRCS panels 2CEC*P001 and 2CEC*P002 from deenergize-to-trip to energize-to-trip. This was done by minor panel wiring changes to the ac load driver printed circuit board in which it will no longer invert the alarm signal. Because of this logic change, new programmable read only memory integrated circuits were installed in the self test circuitry of RRCS. Minor wiring changes to the storage tank level zero interlock circuitry were made in panels 2CEC*PNL618 and 2CEC*PNL629 to accommodate this logic change in the RRCS panels to energize-to-trip.

Safety Evaluation Summary:

This modification will keep SLS operable regardless of the status of the RRCS panels and without the need for temporary jumpers. This modification has no impact on the safe operation or shutdown of the plant. Nuclear safety is enhanced in that temporary jumpers now do not need to be installed when a RRCS panel is out of service to keep SLS operable.

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

Safety Evaluation No.: 93-018 Rev. 0, 1, 2 & 3

Implementation Document No.: Mod. PN2Y91MX054

USAR Affected Pages: 5.4-44, 5.4-45; Figure 5.4-16f

System: Reactor Water Cleanup (WCS)

Title of Change: RWCU F/D Improvements

Description of Change:

This modification changed the reactor water cleanup (RWCU) filter demineralizer (F/D) system as follows:

1. Replaced septa in F/D vessels A, B, C, D with a new design.
2. Revised resin feed system to include replacement of the metering feed pumps with an eductor arrangement.

Safety Evaluation Summary:

The changes to the RWCU F/D system will enhance the system by making it easier for the operator to control and provide more operator options thereby increasing flexibility, and improve precoating of the F/D vessels. Ultimately, the system run cycles will increase and better utilization of precoat material will be achieved.

The proposed changes are nonsafety related and will have no impact on the safe operation or shutdown of the plant. Reactor water chemistry limits outlined in Regulatory Guide 1.56 Rev. 1, Table 1, and specified in Technical Specifications Table 3.4.4-1, will be maintained.

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

Safety Evaluation No.: 93-020

Implementation Document No.: Simple Design Change SC2-0049-93

USAR Affected Pages: Figure 9.4-10e

System: Radwaste Building Ventilation

Title of Change: Radwaste Control Room Noise Improvement

Description of Change:

Safety Evaluation 93-020 was previously reported in October 1994 when the Unit 2 USAR was revised to reflect replacement of the 7.5 hp return/exhaust air fans with new 3.0 hp fans.

This revision to the USAR revises the flow diagram to show a reduced flow of 10,700 cfm.

Safety Evaluation Summary:

This design change will improve environmental and working conditions in the Radwaste Control Room by reducing noise levels. The proposed change does not affect or involve any safety-related equipment.

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

Safety Evaluation No.: 93-027 Rev. 2

Implementation Document No.: N/A

USAR Affected Pages: 9.1-25, 9.1-44

System: FNR

Title of Change: Fuel-Preparation Machine Full-Up-Stop Settings

Description of Change:

This safety evaluation addresses changing the position of the west fuel preparation machine (FPM) full-up-stops. This change will reduce the time/exposure spent during the transfer of new fuel to the spent fuel pool. Additionally, this will reduce the potential for personal contamination and plant contamination.

The west (2FNR*TL1B) FPM will be changed so that its normal configuration will be:

- Full-up-stops permanently removed
- Motive power removed (air-supply line disconnected and blocked)
- To be activated and used only with new nonirradiated fuel under administrative controls and then deactivated after completion of nonirradiated fuel handling

The west (2FNR*TL1B) FPM will have its full-up-stops removed such that a new fuel assembly loaded into its carriage will have its bail handle above the spent fuel pool water level. Positive stopping of the FPM carriage is performed by the end stops on roller chain mechanism. After the crane is disconnected from the new fuel assembly, which is sitting in the FPM, the assembly will be transferred by the refueling platform to its temporary storage location in the spent fuel storage rack.

Safety Evaluation Summary:

The function of the full-up-stops is to provide enough water shielding when using a FPM to handle irradiated fuel assemblies. When a FPM is used to transfer a nonirradiated fuel assembly into the spent fuel pool, a specific full-up limit is not required because its specific function (i.e., provide water shielding) is not required.

Safety Evaluation No.: **93-027 Rev. 2 (cont'd.)**

Safety Evaluation Summary: (cont'd.)

Therefore, this safety evaluation is intended to allow the west (2FNR*TL1B) FPM to be configured to support the application appropriate for new fuel receipt/transfer activities and does not involve an unreviewed safety question.

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

Safety Evaluation No.: 93-037

Implementation Document No.: Simple Design Change SC2-0251-92

USAR Affected Pages: 7A.1-5

System: Reactor Building Heating and Ventilation (HVR), Containment Isolation (ISC), Main Steam (MSS), Residual Heat Removal (RHS), Reactor Protection (RPS), Standby Liquid Control (SLC), Service Water (SWP)

Title of Change: Replace P&B MDR Relays

Description of Change:

This simple design change replaced existing Potter and Brumfield (P&B) Model MDR relays that have been used as an isolation device to isolate nonsafety-related circuits from safety-related circuits, or to isolate redundant safety-related circuits.

Safety Evaluation Summary:

This change enhances the functionality of P&B MDR relays used as an isolation device in the systems listed above because the new P&B MDR relays are designed to preclude the failure modes of these relays.

Replacement relays will be qualified to the same requirements as the old relays. The new relays will be of the same form and fit such that they can replace the old relays as one for one replacement without requiring any major modifications during the installation.

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

Safety Evaluation No.: 93-044

Implementation Document No.: Simple Design Change SC2-0102-91

USAR Affected Pages: N/A

System: Neutron Monitoring System (NMS)

Title of Change: APRM Upscale Alarm and Rod Block (SDC SC2-0102-91)

Description of Change:

This change replaced the neutron flux input signal to the average power range monitor (APRM) upscale alarm and rod block circuit with the filtered simulated thermal power signal. The purpose was to filter out and reduce the noise levels of the neutron flux signal, which in turn allows Unit 2 operational entry into the Extended Load Line Limit Analysis (ELLLA) region of the power flow map along with a reduction in nuisance upscale alarms and rod withdrawal blocks.

Safety Evaluation Summary:

This modification allows operational entry into the ELLLA region of the power flow map which was prohibited by nuisance rod blocks. This modification will have no impact on the safe operation or shutdown of the plant.

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

Safety Evaluation No.: 93-055

Implementation Document No.: Simple Design Change SC2-0342-92

USAR Affected Pages: 9.5-4, 9A.3-46; Figure 9.5-1b

System: Fire Protection Water (FPW)

Title of Change: Install Curb Boxes for 2FPW-V1060 and 2FPW-V1061

Description of Change:

This change installed curb boxes (valve boxes) for two underground sectional isolation valves in the fire main.

Safety Evaluation Summary:

The subject valves were added during the construction of the site cafeteria building. During final construction activities, the valves were inadvertently covered prior to the installation of curb boxes as was intended. This change does not affect the piping and will allow for use of the two valves as key-operated sectional isolation valves in accordance with 10CFR50 Appendix R, Section III.B. Normal construction activities involving excavation and fill are required for this installation. While this change will disrupt normal traffic flow in the area of installation, no impact to system or safe plant operation will result, and the ability to safely shut down the plant in the event of a fire is not impacted.

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

Safety Evaluation No.: 93-056 Rev. 1, 2 & 3

Implementation Document No.: Simple Design Change SC2-0328-92

USAR Affected Pages: Figures 1.2-1, 2.4-1

System: N/A

Title of Change: Construct a Spare Transformer Facility

Description of Change:

The spare transformer facility was constructed southwest of the Unit 2 345-kV switchyard. This facility will be used for the storage of the additional spare transformer for Unit 2.

Safety Evaluation Summary:

The construction of the spare transformer facility does not impact the pertinent licensing issues that are associated with hydrological engineering; i.e., flooding, local intense precipitation (probable maximum precipitation), and the impact on the air intake accident X/Q (Chi/Q), the atmospheric dispersion coefficient.

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

Safety Evaluation No.: 93-058 Rev. 2

Implementation Document No.: Calculation H21C-027

USAR Affected Pages: 9.1-39

System: FHS

Title of Change: Removal of Reactor Cavity Shield Plugs A, B, C and D at 40% or Less Reactor Power

Description of Change:

This safety evaluation evaluated the removal of reactor cavity shield plugs A, B, C and D at 40 percent or less reactor power.

Safety Evaluation Summary:

The removal of the reactor cavity shield plugs A, B, C and D at 40 percent or less reactor power does not affect the structural integrity of the shield plug barrier. The radiological effects of the proposed change have been calculated and determined to be negligible for radiological consequences to the refueling operators during normal refueling operations.

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

Safety Evaluation No.: 93-060

Implementation Document No.: Temporary Mod. 93-038

USAR Affected Pages: N/A

System: Reactor Building Ventilation (HVR)

Title of Change: Temporary Cooling for RWCU Pump Rooms

Description of Change:

This modification installed a temporary air conditioning unit outside the reactor water cleanup (RWCU) pump rooms to provide additional cooling to help alleviate high temperature conditions in the rooms. The air conditioning unit is powered from a welding receptacle fed from distribution panel 2WPS-PNL200.

Safety Evaluation Summary:

This modification does not affect any safety-related equipment, system, building or structure required to perform its safety function during normal operation or following a loss-of-coolant accident. An analysis of calculations indicates that a slight increase in the general area temperature is insignificant enough to cause any effect on the performance or the response time of a safety-related equipment or system to perform its intended function.

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

Safety Evaluation No.: 93-065

Implementation Document No.: Temporary Mod. 93-043

USAR Affected Pages: N/A

System: Ventilation Chilled Water (HVN)

Title of Change: Temporary Removal of 2HVN-TC17C

Description of Change:

The thermocouple well pipe connection for the thermocouple bulb from transmitter 2HVN-TC17C was leaking. The thermowell connection was temporarily removed and replaced with an isolation valve and pipe components until permanent replacement and maintenance was performed.

Safety Evaluation Summary:

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

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

Safety Evaluation No.: 93-075 Rev. 0 & 1

Implementation Document No.: Simple Design Change SC2-0014-93

USAR Affected Pages: 8.3-72; Figure 8.3-10

System: Safety-Related 125-V dc Battery System

Title of Change: Battery Charger Output Bifurcation

Description of Change:

This simple design change facilitates periodic testing of the Division I and II battery chargers, as required by Technical Specifications Section 4.8.2.1, with minimal impact to plant operations. This simple design change relocated the battery charger electrical connections to separate cubicles within their associated switchgear. This bifurcation was done utilizing the existing electrical power cabling between the battery chargers and the 125-V dc switchgear, and reterminating the cabling to individual cubicle load stabs within the switchgear. Only one charger was connected to the 125-V dc switchgear bus at a time. This was accomplished by using a breaker alternately between the breaker charger switchgear cubicles or installing a breaker in both of the battery charger switchgear breaker cubicles. In the event the 125-V dc switchgear breakers are installed for both chargers (of the same Division), one of the breakers shall be placed in A/C "Disconnect" position and locked out while the other breaker is closed. Although there will be an additional switchgear cubicle/breaker interface for battery charger connections to the switchgear, alarms and off-normal status displays will be maintained at those locations which currently provide such indications.

Safety Evaluation Summary:

This simple design change enhances the testability of the battery chargers by eliminating the need for lifting leads to perform the surveillance testing. This design change will have no impact on the safe operation or shutdown of the plant.

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

Safety Evaluation No.: 93-081 Rev. 0, 1, 2 & 3

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

USAR Affected Pages: 4.6-14; Figures 4.6-5c, 9.3-9a, 9.3-9b

System: Control Rod Drive (RDS), Reactor Building Equipment Drain (DER), Residual Heat (RHS), Reactor Building Ventilation (HVR), Reactor Core Isolation Cooling (ICS)

Title of Change: Elimination of Steam Emission from the Reactor Building Equipment Drains and RDS Scram Discharge Volume Collection Tank Installation

Description of Change:

This simple design change involved the following changes:

1. Isolated the hot pressurized drain lines from the cool gravity drains.

2. Added a pressure relief device in each of the Reactor Building drain loops to prevent overpressurization of the drain header in the event that the drain cooler inlet valves are inadvertently closed.

3. Rerouted the RDS scram discharge header vent line to the HVR system via 2DER-TK2B. This bypasses drain cooler 2DER-E2B and eliminates a possible blockage of the vent which would inhibit the scram discharge volume (SDV) drain flow.

4. Separated the RDS SDV drain line from the RHS and ICS pressurized steam-condensing header, and rerouted the drain to a new vented collection tank. SDV water entering the new collection tank post-scram is cooled by mixing with the existing water in the tank. The new tank then drains, via an overflow line, into the gravity drain header to the equipment drain tank, 2DER-TK2A.

The hot, pressurized drain lines within the "A" loop (i.e., ICS, RHS, and SDV drains) have been separated from the cool gravity drains, solving the ALARA concern. The "B" loop, hot pressurized drains from RHS and the main

Safety Evaluation No.: 93-081 Rev. 0, 1, 2 & 3 (cont'd.)

Description of Change: (cont'd.)

steam system were separated from the DER system, the RDS vent was rerouted, and the pressure relief devices were installed during Refuel Outage 4.

Safety Evaluation Summary:

All work associated with this change will be performed in the secondary containment elevations 175'-0" and 196'-0", in accordance with approved site Work Control and Radiation Protection procedures. The constructibility aspects of this change have been reviewed, and appropriate work sequencing instructions included within the applicable Work Orders. The use of construction aids, i.e., tank level tygon tube, pipe bladder, catch containments, flexible hose, etc., to facilitate installation of permanent piping have been reviewed and found adequate for system pressure retention and structural integrity. Temporary removal of pipe spools is required; replacement back to the original design, as required, will be controlled within the work order package. Temporary diversion of Reactor Building equipment drain effluent to the Reactor Building floor drain system has been approved and will be monitored by the Radwaste Department.

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

Safety Evaluation No.: 93-097

Implementation Document No.: Simple Design Change SC2-0078-93

USAR Affected Pages: Tables 9.3-1, 11.5-2 Sh 2; Figures 9.3-5c, 10.1-9e

System: Process Sampling

Title of Change: Deletion of Process Sample Points for URC Effluents

Description of Change:

This simple design change deleted process sample points for the condensate demineralizer system (CND) at the ultrasonic resin cleaner (URC) effluent, the URC resin effluent, and the URC resin receiver tank effluent. These process sample lines and associated valves are nonsafety related.

Safety Evaluation Summary:

Each of the above process sample points begins at a root valve in the CND system and proceeds via 1/4" tubing to sample sink SAS4. The 1/4" tubing was removed (prior to issuance of the plant Operating License) by modification PN2Y86MX044 in order to replace it with 1/2" tubing to alleviate plugging of the smaller diameter tubing. The modification was subsequently canceled and closed out before installing the 1/2" tubing.

The Standard Review Plan describes sample points for performance monitoring at the inlet and outlet of the condensate polishing system and sample points for radiological analysis of URC waste liquid effluent and by resin capacity analysis at panels located between the demineralizer and the URC process.

Sample points are provided for the common influent and common effluent of the CND system. In addition, Chemistry monitors URC performance by conductivity analysis at the resin mix and hold tank effluent and by resin capacity analysis at panels located between the demineralizers and the URC process.

The waste water from the URC is sent to the low conductivity waste tank, along with other liquid effluent from the CND system. The discharge from the low conductivity waste tank is provided with a sample point before being sent to either the anion regeneration tank or liquid radwaste. Therefore, a sample point exists for radiological analysis of common CND waste effluent, including URC waste water,

Safety Evaluation No.: 93-097 (cont'd.)

Safety Evaluation Summary: (cont'd.)

before discharge from the low conductivity waste tank. In addition, the sample root valves still remain so that temporary connections could be made to monitor the URC process.

Therefore, elimination of the URC effluent, URC resin effluent and the URC resin receiver tank effluent sample points does not violate any design bases or plant requirements.

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

Safety Evaluation No.:	93-113
Implementation Document No.:	Unit 1 Simple Design Change SC1-0173-91
USAR Affected Pages:	Table 2.3-4a
System:	Weather Station
Title of Change:	Replacement of 30' Level Dewpoint Monitoring System at the Main Meteorological Tower

Description of Change:

This simple design change replaced the 30' level dewpoint monitoring system at the main meteorological tower. A General Eastern Model 'E1' monitor with a 1211 HMP sensor and 175' of interconnecting cable were procured for this change. The dewpoint temperature measurement is made with a direct-measuring sensor utilizing a Peltier-cooled mirror, automatically held at the dewpoint temperature by a photo-sensing, condensate-detecting, optical system incorporating a solid state LED light source and direct and bias photo detectors. The mirror temperature, if above freezing, measures the true dewpoint temperature and, if below, measures the frost point temperature. The temperature is measured by an embedded linear thermistor sensor.

Safety Evaluation Summary:

The frequency of the repairs on the old model dewpoint has caused the need to replace the model. The new model dewpoint system is as accurate as the old system and more reliable. The location of the new dewpoint sensor is independent of the 30-ft. level boom and was determined to be located on the southeast leg of the tower. This location was chosen because of existing bolt holes in the tower steel. The relocation will not affect the accuracy or validity of the data provided. The holes are located at the same level as the boom. Putting the dewpoint sensor at the same level as the boom instruments is required for consistency in instrument readings. Maintaining the surge protection factor is required to protect the new controller/monitor. Therefore, new surge protection boards were procured and will be installed in the monitoring system circuit.

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

Safety Evaluation No.: 93-129

Implementation Document No.: N/A

USAR Affected Pages: Figures 1.2-1, 2.4-1, 9A.3-1

System: N/A

Title of Change: Construction of the New Engineering Services Building

Description of Change:

The Engineering Services Building has been constructed outside the protected area, north of the P-Building where the R-Building and North Olympic Building stand.

The Engineering Services Building is a two-story, nonsafety-related structure with a slab on grade. This facility provides additional space requirements for departments relocated from the Salina Meadows facility. This building has a total area of approximately 45,000 square feet and provides office space for about 250 personnel.

Safety Evaluation Summary:

Based on the evaluation performed, it is concluded that construction of the Engineering Services Building does not involve an unreviewed safety question.

Safety Evaluation No.: 94-001

Implementation Document No.: Simple Design Change SC2-0255-91

USAR Affected Pages: Figure 9.2-5e

System: Makeup Water Treating System (WTS)

Title of Change: Ecolochem Filtered and Purge Water Connections

Description of Change:

To continue the use of the Ecolochem portable demineralized trailer, permanent filtered and purge water connections were added to the existing WTS system piping. Temporary Modification 91-093 was employed providing a connection for the purge water from the Ecolochem to the makeup waste neutralizing tank (2WTS-TK1). This change made the connection permanent as installed. In addition, a new connection was installed from the water treating filter drain line, 2-WTS-002-134-4, to supply the Ecolochem trailer. Makeup water from the Ecolochem demineralized trailer is controlled in accordance with procedure N2-OP-15.

Safety Evaluation Summary:

An engineering review of the change found that installing additional connections to facilitate the Ecolochem demineralized water process will improve the system performance without causing any safety or operability issues.

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

Safety Evaluation No.: 94-006

Implementation Document No.: DER 2-91Q-1718

USAR Affected Pages: 11.4-1 through 11.4-6; Table 11.4-4 Sh 1 & 2

System: Solid Radwaste

Title of Change: Abandonment In-Place of Asphalt Solidification Equipment

Description of Change:

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

Safety Evaluation Summary:

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

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

Safety Evaluation No.:	94-009 Rev. 0, 1, 2 & 3
Implementation Document No.:	Simple Design Change SC2-0015-94
USAR Affected Pages:	10.4-33, 10.4-34; Figure 10.1-6c
System:	Feedwater (FWP)
Title of Change:	Install Throttle Valves in Feed Pumps Seal Water Injection Lines

Description of Change:

This change installed throttle valves in feed pumps seal water injection lines. The addition of the throttle valves allows each seal water inboard and outboard injection line to be equally balanced, providing greater reliability of the feed pump seals.

Safety Evaluation Summary:

An engineering review of this change has been performed. This review, which included the effects of the change on the system's operability, reliability, maintainability, structural integrity, and system interactions, has found that the implementation of this change will enhance system reliability/maintainability without causing any significant safety or operability issues.

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

Safety Evaluation No.: 94-012

Implementation Document No.: N/A

USAR Affected Pages: Table 9A.3-15 Sh 3; Figures 1.2-1, 2.4-1, 9.5-1b, 9A.3-1

System: N/A

Title of Change: New Unit 2 Maintenance Building

Description of Change:

The Unit 2 Maintenance Building has been constructed inside the protected area, south of the Unit 2 Access Control Building and north of the new Operations Building. This building consolidates maintenance facilities into a new single structure which is located closer to existing plant accessways, enhancing the Maintenance Department's overall efficiency.

The building is a two-story, nonsafety-related structure with approximately 42,000 square feet of floor area. The structure has a slab on grade and provides shop areas for Electrical, Mechanical, and Instrumentation and Controls Maintenance Groups. Additional areas for locker rooms, material issue, and office spaces for Maintenance Management and Support personnel are provided. Also, a portion of the building provides high bay vehicular access equipped with overhead cranes. The new Maintenance Building and Access Control Building are connected, and an elevated walkway between the Maintenance Building and the Operations Building has been constructed.

Safety Evaluation Summary:

The pertinent safety issues identified in this Safety Evaluation are flooding and the impact on the Control Room fresh air intake radiological atmospheric dispersion coefficient. The Maintenance Building location provides adequate separation from safety-related systems and structures to preclude any adverse impact from any compressed gases or chemicals stored in the building.

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

Safety Evaluation No.: 94-013

Implementation Document No.: Simple Design Change SC2-0035-94

USAR Affected Pages: Figures 10.1-9a, 10.1-9b, 10.1-9c, 10.4-9 Sh 7, 8, 9

System: Condensate Demineralizer (CND)

Title of Change: Condensate Demineralizer Flow Recorders Upgrade

Description of Change:

This change replaced five condensate demineralizer flow recorders and the resin strainer differential pressure meters associated with each condensate demineralizer with new recorders that are designed for improved reliability. This change also replaced a sixth recorder which monitors the total differential pressure across all of the condensate demineralizers, and an additional meter which monitors resin recycle strainer differential pressure with a new recorder that performs these combined functions.

Safety Evaluation Summary:

Upon implementation of this simple design change, new recorders will have been installed that provide improved reliability of monitoring of flow through the condensate demineralizers as well as adequate monitoring of the strainers differential pressure. The CND system is not required to effect or support safe shutdown of the reactor or to perform in the operation of reactor safety features.

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

Safety Evaluation No.:	94-017
Implementation Document No.:	Simple Design Change SC2-0004-91
USAR Affected Pages:	Tables 6.2-56 Sh 7, 9A.3-15 Sh 5; Figures 7.3-10 Sh 1, 9.5-1g
System:	Fire Protection Monitoring (FPM), Fire Protection Water (FPW)
Title of Change:	Remove Abandoned FPW Pipe Systems from FPM Monitoring

Description of Change:

Two abandoned water deluge piping systems originally designed to suppress fires at the reactor recirculation pumps were removed from the FPM system. This change removed nuisance spurious alarms, trouble signals, horns, annunciations, and computer inputs from two piping systems which were never functional and not required. This change also disconnected cabling to spared devices in the plant, removed fuses and relays in the local fire control panel, and included the removal of deactivated switches and indication lights in the Main Control Room.

Safety Evaluation Summary:

Since the two affected water deluge piping systems have been inactive and capped prior to plant operation, the associated components perform no useful function. The primary containment does not require fire protection systems during normal operation since it is inerted.

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

Safety Evaluation No.: 94-018

Implementation Document No.: Temporary Mod. 94-020

USAR Affected Pages: N/A

System: High-Pressure Core Spray (CSH)

Title of Change: Jumper Control Signal for 2CSH*MOV118

Description of Change:

A temporary jumper was installed in the control circuit of the high-pressure core spray (HPCS) suppression pool suction valve, 2CSH*MOV118, to simulate a closed valve signal from HPCS test return valve 2CSH*MOV112. This provided a permissive signal for 2CSH*MOV118 to open even though valve 2CSH*MOV112 was deenergized and/or being stroked open (not closed). With 2CSH*MOV118 capable of opening, the HPCS was capable of transferring water from the suppression pool to the reactor vessel and met the requirements of Technical Specification 3/4.5.1.c. The HPCS was declared operable without 2CSH*MOV112 functioning, which allowed it to receive maintenance and be VOTES tested prior to the refueling outage.

Safety Evaluation Summary:

The HPCS system can be considered operable since this jumper installation will allow it to perform its designed functions without any impact from 2CSH*MOV112 on the system's flow rates, pressures, response times, flow paths, or setpoints. The jumper will not affect any other components or systems. The repairs and testing of 2CSH*MOV112 can be performed safely prior to the refueling outage.

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

Safety Evaluation No.:	94-020
Implementation Document No.:	Simple Design Change SC2-0405-91
USAR Affected Pages:	Figure 10.1-5b
System:	Condensate (CNM)
Title of Change:	Condensate Booster Pump Mechanical Seal Cavity Drains

Description of Change:

This simple design change added drain lines to the existing inboard and outboard mechanical seal cavity connections. In addition, the existing drain from the skid was removed and the connection capped.

Safety Evaluation Summary:

All drain lines, whether new or existing, are nonsafety related and will not impact the safe operation of the plant. The change does not affect the operation of the CNM system, Turbine Building equipment and floor drain systems (DET, DFT), nor does it affect the safe shutdown of the plant. Both systems are designed to handle influent from oily or nonoily waste from radioactive and potentially radioactive sources. Both systems pump waste from their respective collection tanks or sumps to radwaste for processing. The condensate pumps will continue to function as designed because this change involves routing water that may pass through the mechanical seals to the DFT system without impacting pump performance characteristics. The existing skid drain lines 2-CNM-150-330-4, 2CNM-150-331-4 and 2CNM-150-332-4 will be removed and a short nipple and cap will be installed. Any water or oil that may collect on the skid may be drained through the capped connection or wiped away.

Should a water leak develop around the condensate booster pumps and said flow was sufficient to overflow the pump skid containment, the DFT system would collect the added volume.

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

Safety Evaluation No.:	94-021 Rev. 0 & 1
Implementation Document No.:	Simple Design Change SC2-0031-94
USAR Affected Pages:	Table 3.9A-12 Sh 12; Figure 9.2-1f
System:	Service Water (SWP)
Title of Change:	IST-SWP Check Valve Internals Removal

Description of Change:

This simple design change removed the internals from check valves 2SWP*V800A, B and V802A, B. Removal of the internals will preclude sedimentation within the value and preclude test failures during in-service testing.

Safety Evaluation Summary:

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

Deletion of the check valve internals will not prevent the SWP system from performing its intended safety function, nor will the system pressure integrity be degraded during any mode of system or plant operation.

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

Safety Evaluation No.: 94-023

Implementation Document No.: Simple Design Change SC2-0174-93

USAR Affected Pages: Table 7.6-6

System: Neutron Monitoring System (NMS)

Title of Change: Revise APRM Flow-Biased Rod Block Setpoint SDC SC2-0174-93

Description of Change:

This design change revised the average power range monitor (APRM) flow-biased simulated thermal power (STP) scram setpoint from 0.66(W-ΔW) + 51% to 0.58(W-ΔW) + 59% (the APRM flow-biased STP upscale scram setpoint was analyzed under Technical Specification Amendment No. 51), and the APRM flow-biased rod block setpoint from 0.66(W-ΔW) + 42% to 0.58(W-ΔW) + 50%. This change allows Unit 2 to better utilize the extended load line limit analysis (ELLLA) region of the power/flow map.

Safety Evaluation Summary:

This change allows Operations to enter the ELLLA region of the power/flow map. Operation in the ELLLA region is restricted because at lower flows the APRM flow-biased scram and the APRM flow-biased rod block encroach on the ELLLA region. This change will have no impact on the safe operation or shutdown of the plant.

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

Safety Evaluation No.:	94-024 Rev. 1
Implementation Document No.:	Simple Design Change SC2-0361-91
USAR Affected Pages:	Tables 1.8-1 Sh 52, 7.5-2 Sh 1 & 8
System:	Regulatory Guide 1.97 Monitoring and Display Instrumentation
Title of Change:	Identification/Marking of Regulatory Guide 1.97 Display Instrumentation on Panels in the Control Room

Description of Change:

This change added to the panels in the Main Control Room a red plastic trim plate around the component identification label of the Regulatory Guide (RG) 1.97 Category 1 and Category 2 display devices for Type A, B, and C variables.

This change will assist the Control Room operators and supervisors in quickly locating the most important RG 1.97 display instruments (i.e., those expected to be the most useful for monitoring, assessing, and responding to postaccident conditions).

This change implements and conforms to a recommendation specified in RG 1.97 with the following exceptions: (1) the position indicating lights for the primary containment isolation valves were not marked with the red trim plate, and (2) the method used to identify the RG 1.97 display devices is the same as that used to identify several other important system control switches and display instruments.

Safety Evaluation Summary:

This change does not modify in any way the operation or performance of any plant systems or structures, nor does it require that any changes be made to any instructions currently specified in any plant operating, maintenance, or calibration procedures. This change does not require changing the currently specified safety classification or qualification criteria of any system component, and has no adverse impact on the safe operation or shutdown of the plant.

Also, the structural integrity of the reactor coolant system pressure boundary, the primary containment pressure boundary, and the secondary containment pressure boundary is in no way affected by the installation of the proposed change.

Safety Evaluation No.: 94-024 Rev. 1 (cont'd.)

Safety Evaluation Summary: (cont'd.)

The two noted deviations from full conformance with the subject recommendations of RG 1.97 have each been evaluated, and both have been determined to be acceptable on a plant-specific basis.

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

Safety Evaluation No.: 94-026

Implementation Document No.: Simple Design Change SC2-0134-93

USAR Affected Pages: Figure 9.2-8b

System: Domestic Water (DWS)

Title of Change: Domestic Hot Water Recirculation Pump Abandonment

Description of Change:

This change abandoned in place domestic water system recirculation pump 2DWS-P1 and associated motor and electrical equipment. The associated annunciator in the control room was also removed. Continuing problems concerning pump leakage and motor overloading were resulting in annunciator indication in the control room. Pump abandonment included necessary changes to associated equipment.

Safety Evaluation Summary:

The DWS system is not safety related, is not connected to any potentially radioactive process systems, and is nonseismic except in the Control Building, where appropriate design measures have been implemented.

System water pressure is provided by the normal source of domestic water, Oswego City Water, and is not affected by the recirculation pump. By abandoning the pump/motor and associated equipment in-place, an unnecessary annunciator will be removed and the need for pump repair/maintenance, which has proven to be quite extensive in the past, will be eliminated while not adversely affecting system operation.

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

Safety Evaluation No.: 94-027

Implementation Document No.: DER 2-94-0157

USAR Affected Pages: 9.3-20

System: Turbine Plant Sampling (SST), Reactor Plant Sampling (SSR)

Title of Change: Tolerance Change for Isobath Temperature in Sample Panels

Description of Change:

The temperature of the constant isothermal baths is maintained at 77°F ± 1°. The tight tolerance for the temperature requires constant change in refrigeration mode and this results in excessive wear and tear on the refrigeration units. This change provided for a wider temperature range (77°F ± 5°) to be maintained at the sample sink constant baths, thereby reducing the constant switching of refrigeration modes and the wear and tear on the units.

Safety Evaluation Summary:

The proposed change would reduce the wear and tear on the refrigeration units by expanding the tolerance of the allowable constant bath temperature and not significantly affect the accuracy of the conductivity measuring instrumentation.

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

Safety Evaluation No.:	94-028
Implementation Document No.:	Simple Design Change SC2-0020-94
USAR Affected Pages:	9.5-84; Figures 9.5-52a, 9.5-52c
System:	Auxiliary Boiler Systems (ABD, ABF, and ABH)
Title of Change:	Addition of Auxiliary Boiler Chemical Injection Piping and Boiler Feed and Blowdown Sample Connections

Description of Change:

This change provided a means of adding sodium sulfite directly to the auxiliary boilers when the boilers are in a hot standby condition. In addition, water chemistry sample connections were added to facilitate boiler feedwater and blowdown analysis. All sample piping has been routed to a new sample sink for convenience. Restricting orifice 2ABF-RO128 bore dimension has been decreased to eliminate excessive steam loss from the auxiliary boiler deaerator.

Safety Evaluation Summary:

This change upgrades the auxiliary boiler system to improve system reliability and its capability to support plant operations. The auxiliary boiler system and the impacted boiler subsystems are classified as nonsafety related. These changes will have no impact on the safe operation or shutdown of the plant since the hardware changes have been designed in accordance with the original plant design basis, and have no effect on the functional capability of the auxiliary boiler systems.

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

Safety Evaluation No.: 94-029

Implementation Document No.: Temporary Mod. 94-022

USAR Affected Pages: N/A

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

Title of Change: Temporary Makeup Water to the HVN System

Description of Change:

This temporary change provided an alternate source of makeup water to the chilled water ventilation system. The new makeup water is from the MWS system in lieu of the water treatment (WTS) system. The WTS system is experiencing a reduction of flow due to piping degradation and is not able to supply the required demand. MWS water will be routed from/to existing connections via temporary hose and associated components.

Safety Evaluation Summary:

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

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

Safety Evaluation No.: 94-032

Implementation Document No.: NUREG-0123

USAR Affected Pages: 9A.3-31, 9A.3-53, 9A.3-56, 9A.3-58

System: N/A

Title of Change: Changes to the UFSAR/USAR Actions Required for Inoperable Fire Protection Systems

Description of Change:

This change modified the UFSAR/USAR action statements for inoperable fire barriers, water-based extinguishing systems, Halon systems and carbon dioxide systems. In addition, the definition of fire watch patrol was changed in the Unit 1 UFSAR to reflect the action statement changes.

Safety Evaluation Summary:

The safety evaluation analyzes the current action statements and augments the options for compensatory measures with additional options to account for areas where fire detection systems are installed and operable. Further, the expanded use of engineering evaluation for impairments, which is currently recognized within the Unit 2 USAR, is expanded for application within the Unit 1 and Unit 2 action statements. Such impairment provisions allow greater flexibility in dealing with system impairments without adversely affecting the Fire Protection Program. The existing action statement options also remain as potential compensatory measures.

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

Safety Evaluation No.: 94-033

Implementation Document No.: Simple Design Change SC2-0164-93

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

System: Main Steam (MSS)

Title of Change: Replace SRV Crane 2MHR-CRN66

Description of Change:

The following changes were implemented by this simple design change:

1. Retired crane 2MHR-CRN66 and provided a replacement crane. This replacement crane is an electrical trolley and chain hoist and is designated as crane 2MHR-CRN66X.

2. Reworked and repaired electrical trolley and bus-bar for replacement crane 2MHR-CRN66X.

3. Provided an additional weld (nonstructural) for SRV crane 2MHR-CRN65X monorail splice at azimuth 240° to improve crane trolley performance.

Safety Evaluation Summary:

Replacement crane 2MHR-CRN66X is being supplied nonseismic and will be removed from the primary containment during plant operations to meet commitments made under the Guidelines for the Control of Heavy Loads (NUREG-0612) and USAR Appendix 9C at Unit 2. The load path has not changed and has been previously evaluated such that the failure of the crane during a seismic event will not affect plant safety. Replacement crane 2MHR-CRN66X is considered and included in the Control of Heavy Loads Analysis. The replacement crane and installation conditions meet requirements for seismic evaluation of nonsafety-related components in safety-related areas (inside primary containment) and does not affect the safety and reliability of Unit 2.

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

Safety Evaluation No.: 94-034 Rev. 1

Implementation Document No.: N/A

USAR Affected Pages: 8.2-1, 8.2-28; Figures 8.1-1, 8.2-1, 8.2-1a, 8.2-1b, 8.2-9

System: 345-kV Transmission Output, 115-kV Offsite Power Sources

Title of Change: Independence/Scriba 345-kV Transmission Line

Description of Change:

This change added a sixth 345-kV transmission line to Scriba Station through two new 345-kV circuit breakers. The two new 345-kV circuit breakers are the same electrical rating as the other eight 345-kV circuit breakers. Construction work included the electrical interconnection of one of the two 345-kV circuit breakers in the Spring of 1994 while Unit 1 and Unit 2 were running. In addition to the energization of this breaker, relay testing was also performed. The electrical interconnection of the second breaker and associated relay testing took place during the Unit 2 refuel outage in the Spring of 1995.

Safety Evaluation Summary:

This safety evaluation addresses the impacts on Unit 1 and Unit 2 resulting from Scriba Substation construction activities. It also analyzes the effect on the transmission system due to increased generation.

Worst-case scenarios were identified and found to be bounded by previous accidents and transients analyzed in both the Unit 1 and Unit 2 UFSARs. A Probabilistic Risk Assessment was performed to quantify the risks associated with the line outages, construction activities and operation of the new transmission lines. The results show the relative change in core damage frequency is small and is considered acceptable.

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

Safety Evaluation No.: 94-035 Rev. 2

Implementation Document No.: N/A

USAR Affected Pages: 8.1-3, 8.2-2, 8.2-7, 8.2-24; Figures 8.2-1, 8.2-1b, 8.2-4a, 8.2-6d through 8.2-6u

System: 115-kV Offsite Power Source

Title of Change: Alternate 115-kV Transmission Supply NMP2

Description of Change:

This modification allows the Unit 2 115-kV transmission line, No. 5 or No. 6, to be energized from the 115-kV transmission system instead of Scriba Substation. Either 115-kV transmission line No. 5 or line No. 6 would be energized from NMPC's 115-kV transmission system's line No. 2. Either transmission line will be connected to the Scriba Station 115-kV main bus (C for 5 line, D for 6 line) and will provide 115-kV offsite supply through existing 115-kV feeder breakers R50 or R60. No changes to protective trip schemes at Unit 2 would be required. Since existing 115-kV circuit breakers would still be energized, relay protective trip signals at Unit 2 will be functional.

Safety Evaluation Summary:

This safety evaluation addresses the impacts on Unit 1 and Unit 2 resulting from providing an electrical offsite power supply to Unit 2 from the 115-kV transmission system. It also analyzes the effect on the transmission system due to the increase in electrical load.

Worst-case scenarios were identified and found to be bounded by previous accidents and transients analyzed in both the Unit 1 and Unit 2 USARs. The analysis performed shows that the 115-kV transmission line No. 2 can be used as an alternate supply to the 115-kV No. 5 or No. 6 line under worst-case loading conditions as long as certain administrative controls are maintained.

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

Safety Evaluation No.:	94-036
Implementation Document No.:	Simple Design Change SC2-0040-94
USAR Affected Pages:	Table 10.2-1 Sh 2; Figure 10.2-3 Sh 1
System:	Electro-Hydraulic Control (EHC)
Title of Change:	Keylock Switch Addition to the Turbine Backup Overspeed Test Circuit

Description of Change:

This change added a keylock switch to the turbine backup overspeed test circuit. Redundant switch contacts were necessary to prevent the original potentially faulty test push button from tripping the turbine during normal testing. The new switch, in the test position, disables the trip relay and serves as a permissive for the test. Initiation of the test continues to be controlled by the push button only after the new switch is placed in the test position. In the normal position the new switch has no impact on the backup overspeed trip circuit.

Safety Evaluation Summary:

The turbine generator, designed to minimize the possibility of a failure that could produce high-energy missiles, is not required to trip for nuclear steam supply upsets but does so to protect itself from conditions that may cause damage. The turbine generator is not required to effect or support safe shutdown of the reactor or to perform in the operation of reactor safety features.

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

Safety Evaluation No.: 94-039

Implementation Document No.: Simple Design Change SC2-0113-94

USAR Affected Pages: Figure 10.1-9c

System: Condensate Demineralizer (CND)

Title of Change: Condensate Demineralizer System Improvements: Replacement of Valve 2CND-PV188 and Removal of 2CND-RV278

Description of Change:

This simple design change replaced valve 2CND-PV188, a 1-1/2", 300# flanged Tufline plug valve, actuator and positioner with a Fisher Controls 2", 300# flanged globe control valve with actuator and positioner. The Tufline plug valve was not adequate for pressure control and controlled erratically. Valve 2CND-RV278 was removed from the system. The valve leaked, adversely impacting system performance. Valve RV278 is redundant and system overpressurization was provided by 2CND-RV352. In addition, pressure indicators 2CND-PI282, PI303 and PI304 were replaced with a larger scale gauge.

Safety Evaluation Summary:

This change does not affect any system, equipment or component of the plant which performs a safety-related function. Nuclear safety will not be affected as the change impacts the nonsafety-related CND system.

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

Safety Evaluation No.:	94-043 Rev. 2
Implementation Document No.:	NEP-POL-0101
USAR Affected Pages:	13.1-4, 13.1-5; Figure 13.1-3
System:	N/A
Title of Change:	Engineering Technical Support Organization Changes

Description of Change:

The following changes were made in the Unit 2 Engineering organization:

1. General Supervisor Nuclear Design - position abolished.

2. Site Engineering - name changed to Plant Support.

3. Supervisor Safety Analysis - name changed to Supervisor Analysis.

4. Supervisor Chemistry/RP Support - position abolished; RP Support function moved under Supervisor Analysis (see 3 above) and the chemistry function moved to Supervisor Environmental Protection.

5. Lead Engineer Inspection Program - position abolished and the function integrated into Mechanical Design.

6. Lead Engineer Special Programs - position abolished and function integrated under Supervisor Analysis (see 3 above).

7. General Supervisor Engineering Performance Services - function integrated under Manager Unit 2 Engineering.

8. Supervisor Engineering Performance - cost estimating and scheduling functions integrated under Supervisor Project Management Unit 2.

9. Supervisor Administrative Services - position abolished; each Engineering Supervisor will oversee their own administrative staff.

10. Associate Senior Staff Tech Building Services - position upgraded and moved under Manager Information Management as "Supervisor Building Services."

Safety Evaluation No.: 94-043 Rev. 2 (cont'd.)

Description of Change: (cont'd.)

11. Program Director Independent Safety Engineering Group - word "program" deleted from the position title. The function remains unchanged.

12. Supervisor Document Control (site), Supervisor Document Control (Salina), Supervisor Records Management and Supervisor Resource Centers - these positions have been abolished and their functions have been transferred into a new position, "Supervisor Document Control/ Records Management."

13. Supervisor Software Development - this is a new position reporting to Manager Information Management.

Safety Evaluation Summary:

After implementation of these changes, adequate resources will exist to provide Engineering support for safe operation and maintenance of the facility under both normal and off-normal conditions. Consequently, the safe operation and maintenance of the facility is not adversely affected.

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

Safety Evaluation No.: 94-044

Implementation Document No.: DER 2-94-0036

USAR Affected Pages: N/A

System: N/A

Title of Change: Boraflex Coupon Removal with NETCO Procedure SEP-093-01

Description of Change:

A visual inspection was made and measurements taken of the full-length surveillance assembly (SA) at Unit 2. The boraflex sheets, or coupons, from the short-length SA were removed and sent to a qualified laboratory (Penn State) for testing and analysis. This analysis was used to establish a baseline (the "original" data described in the USAR) to compare future coupon tests against. The coupons will also be compared to unirradiated coupons taken from the original lot of Boraflex used to manufacture Unit 2's SAs.

Safety Evaluation Summary:

Performing a baseline characterization of the Boraflex coupons installed in the spent fuel racks is necessary to develop a Boraflex Poison Surveillance Program to track boron depletion. The lack of preinstallation baseline characterization will not have a significant impact on the development of a long-term surveillance program and will not pose a safety concern.

Standard industry practice and statistical studies show that removing all of the short-length coupons from the spent fuel racks, for the time period required to take this baseline data, will not have a significant impact on the neutron and gamma exposures seen by these coupons, and future coupon surveillance will be accurate.

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

Safety Evaluation No.: 94-045

Implementation Document No.: N/A

USAR Affected Pages: 3.1-24, 6.2-55, 6.2-99, 6.3-20, 6.4-6, 7.3-26, 7.3-34, 8.3-2, 8.3-48, 8.3-75, 9.1-18, 9.2-6, 9.2-16, 9.3-8, 9.3-11, 9.3-16, 9.3-29, 9.4-8, 9.4-24, 9.4-48, 9.4-54, 9.4-64, 9.5-32, 9.5-49, 9.5-61, 9.5-73, 9.5-81, 11.5-13; FMEA Volumes 1 and 2

System: N/A

Title of Change: Removal of the Failure Modes and Effects Analysis (FMEA), Book 1 and 2, from the USAR

Description of Change:

The FMEA was originally submitted to the NRC in 1983 as part of the Operating License application, and it documented the single-failure analyses for safety-related systems at that time. The FMEA is a very detailed, component-level, computer-based fault tree analysis.

The two FMEA volumes have been removed from the USAR and are retained as a separate engineering document which is referenced in the USAR.

Safety Evaluation Summary:

This is a documentation-only change which has no effect on the plant, its systems or procedures, and does not affect the safe operation or shutdown of the plant, nor does it affect the requirement to consider single-failure criterion as a normal part of the design process. Removing the FMEA from the USAR eliminates the requirement to update this document annually with the USAR.

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

Safety Evaluation No.: 94-046 Rev. 1

Implementation Document No.: Simple Design Change SC2-0099-93

USAR Affected Pages: 10.2-4, 10.2-5; Figure 1.2-40

System: Generator Hydrogen Supply (GMH)

Title of Change: Bulk Hydrogen Control Cabinet

Description of Change:

This change replaced all piping, valves, and controls associated with the existing bulk hydrogen storage unit. Changes included the replacement of all cylinder isolation valves, fabrication of a new stainless steel discharge manifold, installation of a new tube trailer discharge station, installation of a vendor-supplied (Air Products) standard pressure control station, and replacement of the excess flow check valve with a properly-sized unit. In addition, the discharge height for safety relief vents was increased.

Safety Evaluation Summary:

This change was made to address leakage and safety concerns with the previous piping arrangement, and to modify the system to provide adequate makeup flow rate for generator replenishment without defeating the protective features of the excess flow check valve. The design flow rate of the excess flow check valve was not changed.

The main generator hydrogen supply system is a nonsafety-related system that is used to provide hydrogen to the main generator after an outage or on an as-needed basis to make up for hydrogen loss from the generator. The system consists of a vendor-supplied bulk hydrogen storage unit with pressure-reducing controls located in the yard area between Unit 2 and Unit 1, and a network of distribution piping and controls which convey the hydrogen into the turbine building where it is used for generator makeup. The system's purpose, function, method of performing its function, and design basis was not changed.

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

Safety Evaluation No.:	94-048
Implementation Document No.:	Simple Design Change SC2-0062-92
USAR Affected Pages:	7.7-33
System:	Plant Process Computer System (PMS)
Title of Change:	Remove Balance of Plant Performance Calculations (BOPCALC) and Vessel Temperature Rate of Change (VTC) Software from the Plant Process Computer System (PMS)

Description of Change:

This change disabled the current BOPCALC and VTC functions by removing the associated software programs from the PMS computer.

Safety Evaluation Summary:

Removing the BOPCALC and VTC software will have no impact on the safe operation or shutdown of the plant. The PMS computer will remain as a system to provide operators with the means to monitor nuclear steam supply system (NSSS) and BOP events.

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

Safety Evaluation No.: 94-049 Rev. 0 & 1

Implementation Document No.: Mod. PN2Y89MX146

USAR Affected Pages: 6.2-66 through 6.2-71; Table 6.2-54; Figures 6.2-77, 6.2-95A through 6.2-95D

System: Secondary Containment and Standby Gas Treatment (SGTS)

Title of Change: 1-Hour Drawdown Analysis

Description of Change:

This safety evaluation supports plant operation for 1-hour drawdown time. The following parameters have been changed for the reduced ΔT requirements as compared with the fourth operating cycle. These parameters are discussed below:

1. Spent Fuel Heat Loads: A design basis spent fuel pool heat load (16 batches of fuel with 12 days of cooling and power uprate) is used for the ΔT requirements. The use of the design basis heat load is conservative.

2. Unit Cooler Performance: Forty percent degradation for all unit coolers is assumed. Based on the performance tests performed during the 1992-93 time period, the overall degradation of all unit coolers including 2HVR*UC413A & B is 13%. This provides sufficient margin to account for any further degradation that may occur in the future.

3. Secondary Containment In-leakage Rate: The maximum allowable secondary containment in-leakage is 2,670 cfm to support 1-hour drawdown time and reduced ΔT requirement. This is 17% higher than the fourth operating cycle but still meets the SRP Section 6.2.3 guideline.

4. Decay Heat Removal Flow Reduction: The 2,670 cfm in-leakage selection will be made such that it will permit a flow diversion up to 300 cfm for the decay heat cooling after 5 hours into an accident. The 300 cfm flow division is more than the cooling flow requirement of 145 cfm.

5. Elimination of ΔT Annunciation: Existing four-hour surveillance program is sufficient to ensure that the ΔT requirement will be met. The ΔT annunciation is no longer required because of significantly lower ΔT requirement, and periodic surveillance is adequate. Therefore, the ΔT annunciation can be eliminated.

Safety Evaluation No.: 94-049 Rev. 0 & 1 (cont'd.)

Description of Change: (cont'd.)

6. Surveillance Acceptance Criteria: The secondary containment and SGTS system surveillance acceptance criteria are revised to reflect 2,670 cfm in-leakage rate. The analysis methods are the same as those used previously.

7. Use of Electric Heaters: Up to 45 kW of electric heaters can be used to maintain proper emergency core cooling system room temperature (high-pressure core spray room excluded). This change does not adversely affect 60 minutes drawdown capacity. Use of additional heaters may be allowed, following Engineering evaluation and with Applicability Review.

NOTE: Revision 1 to the Safety Evaluation evaluated the use of the electric heaters as described in item 7. The electric heaters were prohibited from use in Revision 0.

Safety Evaluation Summary:

The drawdown analyses (Calc. ES-271, Rev. 0 and ES-259, Rev. 02) provide a curve that defines ΔT requirements based on 1-hour drawdown time for the remainder of the plant life. The ΔT requirement varies from 5 to 10°F during summer months. Because of low ΔT requirement, building heating is not anticipated. During winter months, the available ΔT will be more than the maximum ΔT requirement of 20°F.

The safety evaluation concludes that no safety concerns are involved and no unreviewed safety questions exist if the ΔT requirements of Figure 1 and other requirements as stipulated in the safety evaluation are adhered to.

Safety Evaluation No.: 94-050

Implementation Document No.: EDC 2E10933

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

System: Residual Heat System (RHS)

Title of Change: Revise Safety Class of Control Components for RHS Steam Condensing Pressure Reducing Valves 2RHS*PV21A/B from SR to Q

Description of Change:

This change revised the safety class from SR (Safety Related) to Q (Quality) for components of the RHS steam-condensing pressure-reducing valve instrument loops which perform no safety function. The safety classification was changed for the pressure indicating controller, pressure indicator and the current to pneumatic converter for 2RHS*PV21B and the current to pneumatic converter for 2RHS*PV21A.

Safety Evaluation Summary:

Changing the safety classification, from SR to Q, of components which perform no safety function, will have no impact on the safe operation or shutdown of the plant.

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

Safety Evaluation No.: 94-053 Rev. 1

Implementation Document No.: Simple Design Change SC2-0283-91

USAR Affected Pages: Figure 7.4-1 Sh 1

System: ICS - Reactor Core Isolation Cooling (RCIC)

Title of Change: Add Time Delay in RCIC Initiated Turbine Trip

Description of Change:

The RCIC system at Unit 2 is designed in such a way that RCIC initiation provides automatic signal to trip the main turbine instantaneously, regardless of the cause of RCIC actuation. Therefore, any inadvertent RCIC actuation due to human error or equipment malfunction will cause an unnecessary trip of the main turbine. If at that moment the reactor is running at 35% power or higher, the reactor scram will follow.

To resolve this discrepancy, a time delay was added to the turbine trip signal initiated by RCIC. This change allows the operator to verify the cause of starting RCIC prior to the turbine trip and take appropriate actions.

To provide this time delay, the nonsafety-related auxiliary Agastat relay was replaced with a nonsafety-related time delay Agastat relay.

Safety Evaluation Summary:

General Electric (GE) report GE-NE-E51-00171-01, dated June 1994, and GE letters, dated 8/29/94, 8/30/94, and 9/13/94, provided requested analysis of the proposed change based on calculated moisture level in the steam and steam piping configuration.

The GE report concluded that a time delay of up to five minutes will not compromise the turbine protection and can be introduced to the RCIC initiated turbine trip, providing the total accumulated time of RCIC operation is monitored and does not exceed eight minutes per year.

It is concluded that this change will not alter the design or function of the main steam system or main turbine performance in a way that adversely affects the turbine protection or system performance or plant nuclear safety. The addition of the time delay will not affect the RCIC performance.

Safety Evaluation No.: **94-053 Rev. 1 (cont'd.)**

Safety Evaluation Summary: (cont'd.)

Based on the analysis performed, it is concluded that the proposed change does not alter design, function, or method of performing the function of the safety-related system and is in compliance with NRC requirements.

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

Safety Evaluation No.: 94-055 Rev. 0 & 1

Implementation Document No.: Mod. PN2Y94MX004

USAR Affected Pages: Figure 6.2-71a

System: Containment Atmosphere Monitoring (CMS)

Title of Change: Eliminate Moisture From H_2/O_2 Analyzers

Description of Change:

Water intrusion in the sample lines has had a consistent deleterious effect on the performance of both Train A and Train B hydrogen/oxygen (H_2/O_2) analyzer panels (2CMS*PNL66A and 2CMS*PNL66B). Problems range from water in the analyzing components (which result in inaccurate outputs) to equipment failures (sample pumps, analyzers, etc.). The source of water intrusion was determined to be due to the high humidity (in the primary containment and the area above the suppression pool) being carried into the sample suction lines.

Surveillance durations have increased as a result of having to repair/replace various components which are prone to water incursion damage. In addition, a 7-day LCO is started whenever a surveillance/maintenance is initiated on either H_2/O_2 train.

Safety Evaluation Summary:

This modification installed a moisture collector in both the sample inlet and return lines for both Trains A and B.

While the moisture collectors are identified as safety related, they have no permissive or control functions. They function only to prevent water intrusion into the H_2/O_2 analyzer system components. The modification will increase the systems' reliability and availability and decrease maintenance.

Most of the modification was installed prior to Refueling Outage 4 (RFO-4) with final tie-ins to the system during RFO-4.

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

Safety Evaluation No.: 94-056

Implementation Document No.: DER 2-94-0217

USAR Affected Pages: 6.4-2

System: N/A

Title of Change: Change in NMP2 Control Room Supplies Requirement

Description of Change:

This change allows for the provision of food, sleeping facilities, and other personal comfort supplies from outside the Control Room vital area envelope on an as-needed basis. Providing food and sleeping facilities from either onsite or offsite sources can be readily performed when UFSAR-identified access routes are considered.

Safety Evaluation Summary:

Providing personal comfort supplies to the Control Room from outside the vital area envelope during design basis accident (DBA) conditions is in accordance with previously evaluated access routes described in the USAR. Habitability of the Control Room envelope without these supplies has been evaluated as consistent with the guidelines set by 10CFR50 Appendix A, General Design Criteria 19. This change does not affect any equipment important to safety previously evaluated in the USAR and has no impact on the safe operation or shutdown of the plant. This change has no impact on radiological effluents or nonradiological consequences to the environment.

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

Safety Evaluation No.: 94-058

Implementation Document No.: Drawing EY-8S

USAR Affected Pages: Figure 1.2-1

System: N/A

Title of Change: New Structures to Connect Unit 2 Access Control Building to the Plant

Description of Change:

The Unit 2 Access Control Building (Phase 1) was constructed in 1993. This structure was connected to the Reactor Building via temporary wooden structure. This wooden structure has been replaced with a permanent structure and an additional enclosed walkway from this structure to the Cardox Room/Auxiliary Services Building. These structures provide additional entry paths to both the radiologically-controlled areas and the nonradiologically controlled areas of Unit 2. The new passageway to the Cardox Room is outside the radiologically-controlled zone.

Safety Evaluation Summary:

The pertinent safety issues identified in this safety evaluation are impact on the Control Room fresh air intake, impact on the Flood Analysis, additional loads on Auxiliary Bay roof, Reactor Building and Control Building walls, impact on the CO_2 tank rupture analysis, and accessibility for the removal of the Auxiliary Bay roof plugs.

Based on the evaluation performed, it is concluded that construction of these interconnecting structures between the Access Control Building and the Unit 2 Plant structures does not involve an unreviewed safety question.

Safety Evaluation No.: 94-059 Rev. 0 & 1

Implementation Document No.: Mod. PN2Y93MX004

USAR Affected Pages: N/A

Systems: Common Electrical (CES), Moisture Separator Vents/Drains (DSM), Moisture Separator Reheater Vents/Drains (DSR), Feedwater Pump Recirculation (FWR), Feedwater (FWS), High Pressure Feedwater Heater Drains (HDH), Low Pressure Feedwater Heater Drains (HDL), Hot Reheat (HRS), Meteorological Monitoring (MMS), Main Steam (MSS), Reactor Water Cleanup (WCS)

Title of Change: Modified ASME PTC 6.1 Turbine Generator Performance Test Capability for Unit 2

Description of Change:

A modified ASME PTC 6.1 Turbine Generator Performance Test (ASME Test) was required in order for General Electric (GE) to warrantee recovering the electrical megawatts lost due to the removal of two L-1 stage wheels in low-pressure turbines B and C by replacing rotors for low-pressure turbines A, B, and C.

This modification installed 50 thermocouples, a data acquisition terminal, and 6 condenser (basket tip) backpressure sensing lines (2 per condenser). These items, in addition to existing plant instrumentation, enabled the ASME test to be performed. Through the outputs of the data acquisition terminal, the modification also enabled the temperatures (sensed by the thermocouples) to be monitored on the site meteorological computer (METVAX).

Safety Evaluation Summary:

This modification will add supplementary instrument/pressure inputs in order to enable a modified ASME Test to be performed at Unit 2. This test is required in order to evaluate the efficiency of the turbine generator before and after the installation of the new low-pressure turbine monoblock rotors and, subsequently, the efficiency due to implementing power uprate. In addition, the change will enable plant personnel to permanently monitor temperatures at the power cycle

Safety Evaluation No.: 94-059 Rev. 0 & 1 (cont'd.)

Safety Evaluation Summary: (cont'd.)

block valves (PCBV) close to the condensers. Significant increase in temperatures, when the valves are closed, would be indicative of PCBV leakage.

The changes will have no impact on safe operation or shutdown of the plant. The modified ASME test is not discussed in the Unit 2 USAR. The Unit 1 USAR, Section II.C., describes the meteorology requirements and will not be impacted by the data link inputted from Unit 2 recorder 2CES-TJR100. Unit 2 USAR Section 2.3.3.2.3 describes the METVAX as a weather data processing system and it will not be impacted.

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

Safety Evaluation No.: 94-061

Implementation Document No.: PN2Y94MX008

USAR Affected Pages: Figures 9.2-1c, 9.2-1e, 9.2-1f, 9.2-1g, 9.2-1j, 9.2-1p

System: Service Water (SWP)

Title of Change: Installation of Service Water System Chemical Cleaning Valve Tie-Ins

Description of Change:

This modification provided the isolation and interface tie-ins necessary to facilitate chemical cleaning of the small diameter service water piping (i.e., 3-inch NPS and smaller) in the Reactor and Control Buildings. The cleaning operation was implemented to remove corrosion product and silt deposition from the affected piping and associated unit cooler coils. The cleaning process was the first step in suppressing further aggressive corrosion attack of the pipe surface due to microbiologically influenced corrosion. This precludes future costly piping repairs due to excessive pitting. The scope of this safety evaluation addresses the isolation valve tie-ins necessary to accommodate future cleaning. The actual cleaning operation will be addressed in separate documentation.

Safety Evaluation Summary:

This design provides in-line isolation valves and fittings for use in a future chemical cleaning operation. The isolation valves segment the affected headers into six independent cleaning loops. During normal operation, the isolation valves are maintained in the full open position and perform no active throttling or isolation function. During the cleaning operation the valves are closed, isolating the affected loop from the main header. Consideration for component access was addressed in the design and placement of the new valves and fittings. The system changes have been reviewed against the existing system flow calculations and system pipe stress calculations. These changes do not change or impede the function of the original installation. The new design provides isolation capabilities not included in the original installation, enhancing the operability and maintainability of the system.

This installation does not change the operation of the system or its function. The cleaning valve tie-ins provide the capability to support a future chemical cleaning operation. The added isolation capability enhances system maintenance

Safety Evaluation No.: 94-061 (cont'd.)

Safety Evaluation Summary: (cont'd.)

capabilities. These additions to the system do not change any licensing or design bases requirements of the SWP system.

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

Safety Evaluation No.: 94-062

Implementation Document No.: PN2Y94MX009

USAR Affected Pages: Figures 9.2-1c, 9.2-1e, 9.2-1g, 9.2-1L, 9.2-1m, 9.2-1p

System: Service Water (SWP)

Title of Change: Resize/Reroute SWP Piping

Description of Change:

This modification upsized approximately 1,100 ft. of SWP piping in the Reactor Building and Auxiliary Bays to improve the hydraulic performance of selected unit coolers.

Safety Evaluation Summary:

This project will replace approximately 1,100 ft. of the safety-related pipe. This pipe was identified, through hydraulic analysis, to be undersized for the duty requirements of the system. This review identified portions of piping for unit coolers 2HVR*UC401A through F, UC406, UC407A, B and C, UC408A and B, UC410A, UC411C and UC414A and B as having marginal clean pipe hydraulic performance. This design changes the size of the currently installed piping and in most cases conforms to the original routing of the existing pipe. There are some sections of the piping that will require minor reroutes for greater accessibility. The piping changes have been reviewed against the existing system flow calculations and system pipe stress calculations. These changes do not affect the function of the original installation. The new design provides isolation capabilities not included in the original installation, enhancing the operability and maintainability of the system.

Portions of the new piping will be installed in parallel with the existing piping. The existing pipe will be abandoned in place following the system tie-ins. Most piping installation will take place during normal power operation. The remainder of the pipe and all the required system connections will be installed during Refueling Outage 4. Precautions will be taken to insure that installation activities will not interfere with plant operation or endanger the ability of plant systems to perform their necessary functions.

The piping size increase enhances the capability of the system to respond to plant needs. This is accomplished by increasing the capability of the system to supply

Safety Evaluation No.: 94-062 (cont'd.)

Safety Evaluation Summary: (cont'd.)

service water to the coolers. The increases in pipe size will not change or impact the function of the system. Considering the documents reviewed it has been determined that this modification complies with all of the design and licensing requirements applicable to the Unit 2 SWP system.

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

Safety Evaluation No.: 94-063

Implementation Document No.: Simple Design Change SC2-0029-94

USAR Affected Pages: Figure 9.4-9 Sh 1, 2 & 3

System: Reactor Building Ventilation (HVR)

Title of Change: HVR Fans Repeatedly Failed to Start

Description of Change:

Reactor Building normal ventilation system spare/standby supply and exhaust fans have repeatedly failed to auto start or manually start. The start circuits for the fans contain a permissive logic requiring respective discharge dampers to be greater than 40% open. When a fan start signal is received, a relay/timer is initiated. Also, the start signal initiates the opening of the associated discharge dampers. If the associated discharge damper takes longer than 10 seconds to open to 40% open position, the timer will time out and send a signal to close the discharge damper; the fan will not auto start.

Field testing demonstrated the start sequence was between 7.3 and 9.2 seconds with the relay/timer tripping in 10.1 seconds; relay/timer range was 1.5 to 15 seconds. The margin allowed between "Start" and "Fail to Start" was too restrictive and did not allow for any anomaly or variance of operation. Therefore, the relay/timer setpoint was increased to ≥12 seconds.

Safety Evaluation Summary:

This change will enhance the performance of the nonsafety-related portion of the Reactor Building normal heating, ventilating and air conditioning system and will not affect the operation of any systems important to the safe operation and shutdown of the plant.

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

Safety Evaluation No.: 94-064

Implementation Document No.: Simple Design Change SC2-0118-94

USAR Affected Pages: 9.5-33, 9A.3-16, 9A.3-17, 9A.3-63; Table 9A.3-7 Sh 2 & 3; Figure 9A.3-5

System: N/A

Title of Change: Deletion of Fire Barrier Rating - Diesel Generator Day Tank Rooms

Description of Change:

This change deleted the requirement for a three-hour rated fire barrier separating each diesel fuel day tank from its associated diesel generator.

Safety Evaluation Summary:

The current Fire Hazards Analysis, as presented in USAR Section 9.A, postulates a fire in each diesel generator area which includes the entire inventory of diesel fuel contained within the day tank. Since the diesel generator will not function without the day tank supplying fuel and the day tank has no value without the availability of a diesel generator, the provision of a fire barrier to isolate the fuel supply from the diesel generator is of little value. The postulated fire does not credit the fire barrier with isolating the day tank from the diesel generator. Elimination of the fire barrier does not place the area outside of compliance with applicable criteria since diking and spill containment, in accordance with BTP CMEB 9.5-1 Position C.7.i.(2), is maintained.

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

Safety Evaluation No.:	94-065
Implementation Document No.:	Simple Design Change SC2-0034-94
USAR Affected Pages:	Table 3.9A-12 Sh 12 & 13; Figures 9.2-1e, 9.2-1f, 9.2-1j, 9.2-1L
System:	Service Water (SWP)
Title of Change:	SWP Check Valve Removal, Relocation or Replacement

Description of Change:

The SWP system was reviewed by the project team in response to industry concerns and Unit 2 performance problems noted during SWP check valve and unit cooler tests. A study was performed which noted several areas for improvement of the system performance. The scope of this project addressed one aspect of the proposed system improvements recommended by the study. This project addressed check valve performance enhancements which included:

- Removal of the internals from check valves 2SWP*V75A and 75B.
- Relocation of check valves 2SWP*V1024 and V1025, and the installation of blocking valves and drain provisions.
- Replacement of lift check valves 2SWP*V201A and 201B with nozzle check valves, and rerouting the associated piping.
- Replacement of swing check valves 2SWP*V240A and 240B with nozzle check valves.

Safety Evaluation Summary:

The check valve changes enhance the capability of the system to respond to plant needs. This is accomplished by improving valve reliability, eliminating unnecessary maintenance and testing, and increasing the capability of the system to supply service water to the associated heat loads. These changes do not adversely change or impact the function of the system and comply with all of the design and licensing requirements applicable to the Unit 2 SWP system.

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

Safety Evaluation No.: 94-066

Implementation Document No.: Temporary Mod. 94-039

USAR Affected Pages: N/A

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

Title of Change: Defeat of Main Steam Line Rad Monitoring Trip Signal Channel B1

Description of Change:

This temporary modification installed a jumper in panel 2CEC*PNL633 Bay B in order to defeat a trip signal (Channel B1) which would normally be provided whenever detector 2MSS*RE46B is inoperable.

Safety Evaluation Summary:

This temporary modification allows for ample time for the replacement of the faulty detector without entering a LCO per Technical Specifications 3/4.3.1.a and 3/4.3.2.b.1.b. This change reduces the plant's vulnerability to a full scram by prohibiting the half scram signal to be present during the time period that the detector is being replaced. In the event of fuel damage, the remaining main steam line radiation monitors will function to detect the release of fission products and initiate the appropriate mitigating actions to limit the release and to shut down the plant. This change does not impact the remaining detectors from performing their safety functions as originally designed.

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

Safety Evaluation No.:	94-067
Implementation Document No.:	DER 2-94-0036
USAR Affected Pages:	9.1-9, 9.1-10
System:	N/A
Title of Change:	Update UFSAR Description of a Revised Boraflex Surveillance Program and Use of New Surveillance Assemblies

Description of Change:

US Tool & Die (UST&D) originally supplied a surveillance sample consisting of 2-inch square pieces of Boraflex for the Unit 2 spent fuel storage racks. This was shipped in 1984 and was the standard surveillance sample supplied by UST&D at that time. However, these samples were lost and therefore never installed in the Unit 2 spent fuel pool. Replacement Boraflex surveillance samples were purchased in 1990. These coupons were installed prior to the racks being put into use and the hottest bundles from each reload have been placed around them. This safety evaluation updated the description in the SAR of the Boraflex coupon surveillance assemblies to be in agreement with the actual coupons which were installed in 1990. In addition, the long-term Boraflex Surveillance Program has been modified to be in agreement with the most recent industry guidance for Boraflex surveillance, issued in a report by the Electric Power Research Institute (EPRI).

Safety Evaluation Summary:

Performing periodic testing of the Boraflex coupons installed in Unit 2's spent fuel racks is necessary to provide assurance that the Boraflex material in the spent fuel racks continues to perform acceptably over the service life of the racks. The replacement coupons and their associated surveillance program will provide adequate assurance that the Boraflex material is performing as intended and will preclude the occurrence of a criticality accident due to degradation of the Boraflex material. This revised surveillance program will continue to implement the requirements of GDC 61 and is a more conservative program than the original one described in the USAR. Neither the Boraflex coupons or their associated surveillance program present a safety concern for Unit 2.

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

Safety Evaluation No.: 94-068 Rev. 0 & 1

Implementation Document No.: Simple Design Change SC2-0040-93

USAR Affected Pages: 9.1-23, 9.1-24, 9.1-40, 9C.3-4, 9C.3-5, 9C.8-1, 9C.8-2, 9C.8-3; Tables 3.9A-4 Sh 6, 3-1 Sh 1, 3-3 Sh 1; Figure 5-2

System: MHR

Title of Change: Upgrade of 125 Ton Polar Crane to 132 Ton Capacity

Description of Change:

This change to the polar crane's use and function upgraded the main hoist from 125-ton to 132-ton capacity to allow for the full utilization of the reactor pressure vessel (RPV) head carousel strongback which was initially employed during Refueling Outage 3.

The upgrading of the polar crane main hoist involved modifications and recertification load testing of the hoist at 125% of the new rated capacity during plant operation. The constructibility review, the load drop assessment calculation, and the load test procedure ensured that plant equipment was not affected by the work and testing. The upgrading of the polar crane main hoist reduces personnel exposure and saves critical path refuel outage time by permitting the removal and reinstallation of the RPV head, the stud tensioners, the 76 studs, nuts, and washers in one lifting operation.

Safety Evaluation Summary:

The design changes and reanalysis of the polar crane main hoist for the upgrade to 132 tons meets the same single-failure proof criteria of NUREG-0612, as did the existing equipment.

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

Safety Evaluation No.: 94-069

Implementation Document No.: Procedure N2-CSP-2V

USAR Affected Pages: 10.4-2, 10.4-22 through 10.4-27

System: Condensate Demineralizer System

Title of Change: Condensate Demineralizer Water Purity Maintenance

Description of Change:

This change replaced condensate demineralizer resin at intervals which are based on inlet conductivity and its relationship to the composition of the circulating water, and condensate flow rate through the beds since regeneration of resin is no longer performed at Unit 2. This approach to the operation of the demineralizers eliminates the potential for contamination from the products formed during acid and caustic regeneration of the resin.

Safety Evaluation Summary:

The condensate demineralizer resin will be replaced such that adequate remaining capacity will exist to handle the postulated main condenser leak event within the time permitted for an orderly shutdown. The effluent quality of the demineralizer system will satisfy the acceptable limit found in Table 2 of Regulatory Guide 1.56, Rev. 1, July 1978. The replacement of condensate demineralizer resins at these frequencies is in compliance with paragraph C of the regulatory guide.

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

Safety Evaluation No.: 94-070

Implementation Document No.: Simple Design Change SC2-0028-94

USAR Affected Pages: Figure 10.1-6b

System: Feedwater (FWS)

Title of Change: Replacement of Drain Valves 2FWS*V89A&B and 2FWS*V90A&B

Description of Change:

This change replaced drain valves 2FWS*V89A&B and V90A&B for containment isolation valves 2FWS*V12A&B. These valves are 3/4" NPS and are normally closed. The valves are not active components and their only safety function is pressure retention.

These valves are located in the primary containment and were replaced to enhance leak-tightness and prevent leakage in the drywell.

Safety Evaluation Summary:

Installation of the replacement drain valves does not impact the design of the FWS system. The system can still provide its intended flow and new drain valves 2FWS*V89A&B and V90A&B will assure system integrity.

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

Safety Evaluation No.: 94-071

Implementation Document No.: N/A

USAR Affected Pages: 2.2-1, 2.2-3, 2.2-8; Table 2.2-3; Figure 2.1-2

System: N/A

Title of Change: Gas Pipeline #63 to Sithe Energies USA Plant and #58 to Indeck Energy

Description of Change:

Sithe Energies USA has constructed a 1000-MW natural gas-fired electrical generating station known as Independence Generation Plant. It is a cogeneration plant located in Oswego County, New York.

Two natural gas pipelines lie within 8 km (5 miles) of the Nine Mile Point Station. One pipeline (#63) supplies gas to the Sithe plant and the other pipeline (#58) to Indeck Energy.

Safety Evaluation Summary:

The nearest point of the pipelines is over 2 miles from Unit 2. Due to the distance between the pipelines and Nine Mile Station, atmospheric dispersion would conservatively reduce the natural gas concentration below its lower explosive limit more than 1 mile from Nine Mile Point. The detonation of an unconfined natural gas dispersal in air is not a credible event.

Due to the distance from the pipelines to the Unit 2 Control Room air intake (>2 miles), the resultant atmospheric dispersion would conservatively reduce the natural gas concentration at the intakes to less than 9 g/m^3. This is well within the natural gas toxicity limit of 287 g/m^3.

Based on the evaluation performed, it is concluded that installation of the gas pipelines does not involve an unreviewed safety question.

Safety Evaluation No.: 94-073

Implementation Document No.: Simple Design Change SC2-0142-94

USAR Affected Pages: 9.4-71; Figures 9.4-22b through d

System: Heating and Glycol (HVG), Water Treatment (WTS)

Title of Change: Water Treating Makeup Water Supply to HVG

Description of Change:

This change isolated the hot water heating and glycol (HVG) system from makeup water supplied by the makeup water treatment system (WTS) because WTS has been repeatedly contaminated by glycol intrusion through leaky HVG valves. The needed makeup water for HVG was supplied manually by Operations under procedural controls. The isolation was accomplished by removing the in-line check valves and blanking off the lines.

Safety Evaluation Summary:

Neither the HVG nor the WTS systems have any safety-related functions. Failure or malfunction of the systems or components will not compromise any safety-related systems or components or prevent a safe reactor shutdown.

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

Safety Evaluation No.: 94-076

Implementation Document No.: DER 2-93-1935

USAR Affected Pages: Table 6.2-56 Sh 2, 4, 7, 10, 11, 12, 20, 21

System: Primary Containment

Title of Change: Appendix J Discrepancies, DER 2-93-1935

Description of Change:

DER 2-93-1935 addressed discrepancies involving editorial changes to the UFSAR clarifying the Unit 2 Appendix J Program.

This safety evaluation addresses changes to USAR Table 6.2-56 as follows:

1. Deleted reference to Type C testing for the following valves:

   2RHS*MOV1A, 2RHS*MOV1B, 2RHS*MOV1C, 2CSH*MOV118, 2CSL*MOV112 and 2ICS*MOV136

2. Revised Notes 23, 24 and 25 to delete reference to Type A testing of the following relief, safety, check and vacuum breaker valves:

   2RHS*SV34A/B, 2RHS*SV62A/B, 2RHS*RV56A/B, 2RHS*V20, 2RHS*V19, 2RHS*V117, 2RHS*V118, 2RHS*RVV35A/B, 2RHS*RVV36A/B, 2RHS*RV108, 2RHS*RV20A/B/C, 2RHS*RV61A/B/C, 2RHS*RV110, 2CSL*RV123, 2CSL*RV105, 2RHS*RV139, 2CSH*RV113 and 2CSH*RV114

3. Revised Table 6.2-56 to indicate a Type C test for valves 2CCP*RV170 and 2CCP*RV101.

Safety Evaluation Summary:

The valves in Item 1 are ECCS suction valves that take suction from the suppression pool at an elevation below minimum suppression pool water level of 199'-6" and, as such, are waterfilled post-LOCA. The valves in Item 2 are relief, safety, check and vacuum breaker valves that terminate in the suppression pool below the minimum water level of 199'-6". The suppression pool water effectively seals these containment isolation valves from the primary containment atmosphere, thereby preventing gaseous releases from the primary containment. Since these

Safety Evaluation No.: 94-076 (cont'd.)

Safety Evaluation Summary: (cont'd.)

valves do not see containment atmosphere post-LOCA, they do not represent potential containment atmospheric leakage paths and are not subject to leak testing as defined in Appendix J.

Relief valves 2CCP*RV170 and 2CCP*RV171 are located inside primary containment and their outlets terminate open ended inside the primary containment at an equipment drain and, as such, provide an atmospheric leak path from the primary containment. Therefore, these valves are considered containment isolation valves as defined in Appendix J and are Type C tested to satisfy the requirements of Appendix J.

This change revises USAR Table 6.2-56 to accurately denote the proper leak testing provisions for the valves in Items 1, 2 and 3. This is an editorial change and does not change leak testing requirements or leak testing methods of the Unit 2 Appendix J test program. The applicable containment isolation valves in Items 1, 2 and 3 will continue to be properly leak tested per existing procedures to ensure leak-tight integrity as required by 10CFR50 Appendix J and ASME XI.

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

Safety Evaluation No.: 94-077

Implementation Document No.: Procedure N2-OP-100B

USAR Affected Pages: 9.5-43, 9.5-69

System: Emergency Diesel Generator Lube Oil

Title of Change: Div. 3 EDG Lube Oil Temperature

Description of Change:

The lube oil temperature for the Div. 3 emergency diesel generator (2EGS*EG2) was stated in the USAR to be above 120°F during standby conditions. However, actual lube oil temperature was observed to be between 90°F and 110°F during standby conditions. With the vendor's concurrence, it has been established that the minimum standby lube oil temperature requirement for the Div. 3 diesel generator is 85°F.

Safety Evaluation Summary:

Establishing the requirement of 85°F as the minimum lube oil temperature during standby conditions is consistent with the acceptance criteria in NUREG-0800, that the temperature of the lubricating oil is maintained above a minimum value to enhance the "first-try" starting reliability of the engine in the standby condition. This change is also consistent with the vendor's recommendation for minimum lube oil temperature and will supply proper standby lubrication to the engine. Therefore, the actual lube oil temperature of between 90°F and 110°F is sufficient to verify that engine lube oil temperature requirements are met under standby conditions.

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

Safety Evaluation No.: 94-078

Implementation Document No.: Procedure N2-OPS-RPS-W001

USAR Affected Pages: N/A

System: MSS, HRS

Title of Change: Turbine Steam Valve Surveillance Test Interval Extension

Description of Change:

This safety evaluation evaluates changing the testing requirement for turbine control, stop, and intercept valves from weekly to monthly.

(NOTE: Subsequent changes have been evaluated under Safety Evaluation 95-032).

Safety Evaluation Summary:

The purpose of this testing is to discover any valve malfunctions that could contribute to a turbine overspeed event causing turbine components to become high-energy debris (missiles) capable of striking and damaging safety-related equipment.

The NRC Safety Evaluation Report for Unit 2 (NUREG-1047, Section 3.5.1.3.10), with regard to the turbine missile issue, concluded that the probability of unacceptable damage to safety-related structures, systems and components due to turbine missiles is acceptably low (i.e., 10^{-7} per year), provided that the total turbine missile generation probability is such that conformance with the NRC criteria (i.e., $P1 < 10^{-4}$ for favorably oriented turbines, $P1 < 10^{-5}$ for unfavorably oriented turbines) is maintained throughout the life of the plant by acceptable inspection and test programs. In reaching the conclusion, the NRC staff factored into consideration the favorable orientation of the Unit 2 turbine generator. Also in the Unit 2 Safety Evaluation, the NRC identified that the relevant General Electric (GE) missile probability analysis may be used in determining the inspection interval for the turbine discs at Unit 2.

The existing requirement for surveillance testing of turbine stop valves (TSVs), turbine control valves (TCVs) and turbine contained stop and intercept valves (CIVs) is to perform these tests on a weekly basis. In order to assure plant availability and decrease any potential of plant scrams, the surveillance testing

Safety Evaluation No.: 94-078 (cont'd.)

Safety Evaluation Summary: (cont'd.)

frequency of these valves is being temporarily extended (up to Refuel Outage 4 (RF04)) from a weekly to a monthly interval. Justification for this change in frequency is provided below. During RF04, the low-pressure turbine rotors are being replaced by monoblock rotors. Missile generation is not a concern for monoblock rotors. As part of the monoblock rotor modification, a separate safety evaluation will be prepared which will identify surveillance testing requirements for TSVs, TCVs and CIVs. It is anticipated that with the replacement of the existing "built-up" low-pressure rotors with monoblock rotors, surveillance testing frequency of TSVs, TCVs and CIVs can further be reduced (from monthly to quarterly).

In response to Niagara Mohawk's request, GE recalculated wheel missile probabilities for the Unit 2 low-pressure turbine rotors. These new calculations were based on the revised calculation procedure that 1) included updated failure rate data on the primary steam valves of GE nuclear units, and 2) included the capability of calculating wheel missile probabilities for extended time intervals between the GE normally recommended functional tests of the steam valves.

GE evaluation indicates that, based on NRC criteria (P1 $< 1\times10^{-4}$ for favorably oriented units), Q/Q/Q testing, and with no pre-warming, the inspection interval is reduced to 5.7 years for the A rotor. Considering pre-warming, M/Q/M testing, and based on GE recommendation (P1 $< 1\times10^{-5}$ for the unit), the A rotor inspection interval is reduced to 2.8 years. The existing inspection interval is 6 years. Based on existing frequency of testing (W/W/W) and test schedule of M/M/M (conservatively, Unit 2 shall utilize M/M/M schedule instead of M/Q/M analyzed by GE), which is anticipated to be utilized for a very short duration of the present operating cycle, the reduced inspection interval of 2.8 years would require A rotor inspection by R5.

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

Safety Evaluation No.: 94-079

Implementation Document No.: Procedures N2-FSP-FPM-A001-1 through 5

USAR Affected Pages: 9.5-2, 9A.3-43, 9A.3-44; Table 9.5-3, Sh 7 & 8

System: Fire Detection System

Title of Change: Performance Based Fire Detector Surveillance Testing

Description of Change:

This change added a clarifying paragraph that indicates that subsequent editions of NFPA codes and standards may be used for subsequent plant modifications and program revisions. Clarification was also added that a deviation from NFPA-72 specified fire detector test frequencies is utilized for fire zones that do not contain any equipment considered important to safety. Also, a deviation from NFPA-72 code specified testing requirements in fire zones containing safety-related equipment has been adopted. This deviation is based on obtaining equivalent reliability between test intervals as allowed by NFPA code equivalency provisions. The revised testing scheme uses a 10%, 20% expanded, total zone expanded, rotating test sample population for testing that is conducted on an annual basis.

Safety Evaluation Summary:

The clarifications are information additions that do not affect safety-related equipment and are not changes from present operating policies. The change to the testing scheme for fire detection in safety-related equipment areas is based on the plant-specific failure rate (failure to alarm under simulated fire conditions) of fire detection instruments. The scheme adopted provides an equivalent or reduced probability of a detector failure between test intervals than that generally assured by NFPA-72 annual test intervals using generic failure probability. The testing scheme change does not increase the probability of a postulated fire in the Fire Hazards Analysis, nor does it increase or decrease the severity of the fire.

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

Safety Evaluation No.: 94-080

Implementation Document No.: Mod. PN2Y94MX006

USAR Affected Pages: 3.7B-17, 3.9A-3, 3.9A-4, 3.9B-49; Table 3.9B-2m Sh 1 & 2

System: Hydrogen Recombiner (HCS), Reactor Water Recirculation (RCS), Residual Heat Removal (RHS), Service Water (SWP)

Title of Change: NMP2 Snubber Reduction

Description of Change:

This modification reduced the number of mechanical snubbers on Unit 2 safety-related piping systems by reanalyzing the piping systems for snubber removal or snubber replacement with struts.

Safety Evaluation Summary:

Due to failure rates associated with snubbers, snubber removal results in piping systems that are more reliable. Other benefits of the program include reduced long-term maintenance, inspection and test requirements.

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

Safety Evaluation No.: 94-082

Implementation Document No.: DER 2-93-2060

USAR Affected Pages: Figures 10.1-9e, 10.4-9 Sh 10

System: Condensate Demineralizer (CND)

Title of Change: CND Ultrasonic Resin Cleaner (URC) Level Element 2CND-LE225 Not Installed

Description of Change:

This safety evaluation documents the as-built plant condition for the CND system ultrasonic resin cleaner tank as not having a level element installed and the corresponding level alarms inoperative. Additionally, the associated sluicing water flow control valve has been maintained in the full open position since plant startup with approved holdout tag preventing misoperation.

Safety Evaluation Summary:

The ultrasonic resin cleaning equipment does not interface with or affect any equipment important to safety, and the CND system is not required to effect or support safe shutdown of the reactor or to perform in the operation of any reactor safety features.

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

Safety Evaluation No.:	94-083
Implementation Document No.:	Procedure N2-FSP-FPM-R001
USAR Affected Pages:	Table 9A.3-18 Sh 2
System:	Fire Detection (FPM)
Title of Change:	Elimination of Periodic Test Requirements for Thermal Fire Detectors in Fire Zone 252SW

Description of Change:

This change removed four thermal fire detectors from the scope of USAR-specified periodic testing. The thermal detectors are located in the SFP Phase Separator Tank Room on Reactor Building elevation 289'-0" and immediately outside the room. The area outside the room is also provided detection coverage by ionization detectors in another loop of the detection zone. The combustible loading within the room is insufficient to warrant fire detection. This change left the thermal detectors installed but will not require periodic testing in order for the zone to be considered operable. In addition, should one of the detectors go into alarm due to some future failure, this change allows the loop to be bypassed in the panel without declaring the zone of detection inoperable. This change was implemented since very high radiation levels normally present in the SFP Phase Separator Tank Room prevent testing in accordance with previous requirements.

Safety Evaluation Summary:

The reliability and margin of safety discussed in USAR Section 9A.3.6.1.7 will be maintained by this change since redundant ionization detection in the area will be maintained. The combustible loading within the SFP Phase Separator Tank Room is insufficient to warrant detection and has been documented in a fire protection engineering evaluation. The safe shutdown analysis is unaffected by this change as is the Fire Hazards Analysis.

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

Safety Evaluation No.: 94-087

Implementation Document No.: NUREG-0654, NUREG-0696, NUREG-0737

USAR Affected Pages: 13.3-2

System: N/A

Title of Change: Eliminate Corporate Emergency Operations Center

Description of Change:

NUREGS 0654, 0696, and 0737 contain criteria pertaining to emergency response facilities. There is no requirement specific to maintaining an offsite Corporate Emergency Operations Center (CEOC). There is, however, a requirement to maintain a backup Emergency Operations Facility (EOF) should the onsite EOF become uninhabitable. The offsite location previously used to obtain Engineering support was referred to as the CEOC in the Site Emergency Plan. The CEOC has historically provided a convenient location for obtaining Engineering support during emergency scenarios. The CEOC contained resources (drawings, calculations, personal references) typically used by Engineering personnel. The location was outside the ten-mile emergency zone and dedicated phone lines were used to ensure communication with the Technical Support Center (TSC) and EOF. The Engineering Department recently relocated to a building on site and could be affected by evacuation requirements. For emergency events which do not require evacuation, communication via phone lines ensures access to the same level of technical support previously available. However, if site evacuation is required, access to the resources contained in the Engineering Building could be lost. Under these scenarios, technical support would come solely from the TSC and EOF and would be dependent on the amount of technical information available in those facilities.

Safety Evaluation Summary:

Conformance with applicable criteria is assured since: 1) there is not a regulatory requirement for a CEOC, and 2) except under scenarios requiring evacuation, the proximity of Engineering resources to the plant will improve support. Therefore, there is no net negative impact from relocating the Engineering Department to the site and eliminating the CEOC.

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

Safety Evaluation No.: 94-088

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

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

System: Reactor Coolant System (RCS)

Title of Change: Recirculation Flow Control Valve Minimum Position Change

Description of Change:

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

Safety Evaluation Summary:

The peak neutron flux that will result from the increased flow when the recirculation pumps are upshifted is conservatively below the high neutron flux scram.

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

Safety Evaluation No.: 94-089

Implementation Document No.: Simple Design Change SC2-0019-94

USAR Affected Pages: Figures 10.4-1, 10.4-2a

System: Condensate Air Removal (ARC)

Title of Change: Design Change to 2ARC-AOV104

Description of Change:

Valve 2ARC-AOV104 failed to open when required. The root cause for this deviation was determined to be improperly sized spring for the design condition. The most cost-effective repair was to retrofit this actuator to open (and close) on air. This design change deleted the fail open requirement for this valve.

Safety Evaluation Summary:

Since the valve is nonsafety related, no safety concerns exist in allowing this valve to open on air. The air will provide the necessary force to break the valve away from its seat. Additionally, should the valve fail once open, it will remain open maintaining condenser vacuum. Allowing the valve to open and close on air will not adversely affect nuclear safety.

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

Safety Evaluation No.:	95-005
Implementation Document No.:	Procedure NSAS-POL-01, Procedure NEP-POL-0101
USAR Affected Pages:	13.1-4, 13.1-5, 13.1-7; Figures 13.1-3, 13.1-5
System:	N/A
Title of Change:	Reorganization of the Information Management Branch to the Nuclear Safety Assessment and Support Department

Description of Change:

The functions of the Information Management Branch have been relocated from the Nuclear Engineering organization to the Nuclear Safety Assessment and Support (NSAS) organization.

Safety Evaluation Summary:

Relocation of the administrative support functions provided by the Information Management Branch to the NSAS Department is consistent with the charter and responsibilities of that department and maintains clear management control and effective lines of authority and communication between the organizational units involved in the management, operation, and technical support for the operation of the facility.

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

Safety Evaluation No.:	95-030
Implementation Document No.:	Procedures S-RTP-165, S-RPIP-3.11
USAR Affected Pages:	Tables 1.8-2 Sh 6, 1.9-1 Sh 49 & 50, 12.5-3
System:	N/A
Title of Change:	Use of Audible Alarm Dosimeters and Personnel Air Samplers

Description of Change:

This change revised the USAR to agree with current Radiation Protection Program procedures for the use of audible alarm dosimeters and personnel air samplers.

Safety Evaluation Summary:

The proposed changes to the Unit 2 USAR will meet the intent of 10CFR20 and comply with applicable portions of regulatory guidelines.

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

Safety Evaluation No.:	95-032 Rev. 0 & 1
Implementation Document No.:	Mod. PN2Y93MX005
USAR Affected Pages:	3.5-8, 3.5-9, 3.5-16, 10.2-1, 10.2-8, 10.2-9, 10.2-10, 10.2-11, 10.2-12, 10.2-13; Tables 3.5-3, 3.5-6, 3.5-9, 3.5-12, 3.5-15, 3.5-16, 10.2-1 Sh 1 & 2
System:	TMS
Title of Change:	Low-Pressure Turbine Monoblock Rotor Replacement

Description of Change:

This modification replaced the three existing low-pressure turbine rotors (2TMS-T2A,B,C) with General Electric (GE) monoblock design rotors. The previous low-pressure turbine rotors were of a built-up design (shrunk-on wheels). The shrunk-on wheel design has a potential of developing cracking in the keyway, web and hub area due to stress corrosion cracking (SCC). The monoblock rotor design has been adopted as a corrective measure against SCC.

The benefit of replacement of the existing low-pressure turbine rotors with monoblock rotor is as follows:

a. Recovery of lost MWe due to wheel removal. GE guarantees a 28.3 MWe recovery.

b. Reduced low-pressure turbine rotor inspections. The recommended inspection frequency reduces from 6 to 10 years.

c. Reduced turbine valve testing.

d. Replacement monoblock rotors support reduced outage durations.

e. Cobalt reduction. The previous low-pressure turbine last-stage buckets utilized stellite erosion shields. The replacement buckets are flame hardened. The replacement last-stage buckets result in a reduction in radiation exposure to plant personnel.

Safety Evaluation No.: 95-032 Rev. 0 & 1 (cont'd.)

Safety Evaluation Summary:

The monoblock rotors were designed to meet the requirements of the previous design/operating conditions, including transient operating conditions. At the time of scheduled installation (1995), Unit 2 will be undergoing a power uprate. The monoblock rotors, therefore, conform to the design requirements established for power uprate. The requirements include the following:

Guaranteed Rating:	1,210,902 kW
Initial Steam Conditions:	1003 psia
Exhaust Pressure:	2.0" Hg Abs backpressure
Guaranteed Flow:	13,583,244 lbs/hr

The turbine rotors are designed with 5 percent flow margin above the flow required to meet the maximum guaranteed output for power uprate. The turbine generator is not required to effect or support safe shutdown of the reactor or to perform in the operation of reactor safety features. The turbine generator is, however, designed to minimize the possibility of turbine rotor failure that might produce a high-energy missile that could damage a safety-related component.

Replacement with monoblock rotors reduces the probability of missile generation by removing the potential for SCC at the interface between a rotor and wheel via the use of a monoblock construction. The overall probability of damage by turbine missiles (for monoblock rotors) will be within the acceptance value of 10^{-6}/yr, as outlined in SRP 2.2.3, and the acceptance value of 10^{-7}/yr, as specified in Regulatory Guide 1.115. The existing overspeed protection controls will prevent the rotor from exceeding the maximum transient speed of 120 percent (design overspeed) of rated turbine speed. This safety evaluation also re-evaluated testing requirements for various turbine devices based on monoblock rotor replacement.

Adequate procedural controls shall be maintained such that there will be no adverse effect to nuclear safety both during the transportation and installation of the monoblock rotors and during storage of the removed rotors.

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

Safety Evaluation No.: 95-033

Implementation Document No.: Mod. PN2Y94MX003

USAR Affected Pages: Figures 9.2-8b, 9.2-9b

System: Domestic Water (DWS), Sanitary Plumbing (PBS), Auxiliary Service Building HVAC (HVL), Paging System (COP)

Title of Change: Auxiliary Building Renovation, RFO4 Scope

Description of Change:

This change renovated the Auxiliary Service Building elevation 261'-0" to allow use as controlled personnel ingress and egress to/from the Turbine and Reactor Buildings via the linkway during Refueling Outage 4 (RFO4). This change involved making an opening in the 13 line wall at elevation 261' near the entrance to the south electrical tunnel stairwell, installation of an additional 1.5-hr. fire-rated door (ET262-6) for stairway isolation, removal of lockers, removal of the drinking fountain and wash basins, the capping of floor drain(s) in the temporary access passageway, and the removal of door AS261-7 for improved access. Also, door ET262-4 was removed while the area was being used only for access and egress during RFO4. The temporary access passageway was created by installing painted gypsum wallboard partitions. The ceiling tile grid and associated services were revised in the area of the passageway. These changes are partial scope for this modification. After the 1995 refuel outage (RFO4), the modification was resumed to provide a radiation protection calibration laboratory, storage room, separate male/female personnel decontamination facilities, removal of the 1,980-gallon hot water heater and replace it with a 120-gallon capacity, install a different design door for ET262-4, and install an equipment lift from Turbine Building elevation 250'-0" and elevation 261'-0" of the Auxiliary Service Building. The changes to be made after completion of RFO4 will be addressed in a subsequent safety evaluation.

Safety Evaluation Summary:

The changes being made to the south electrical tunnel stairwell were addressed for conformance with General Design Criterion (GDC) 2 and no adverse impact was created. Potential impact to adjacent safety-related areas and conformance to GDC 3 and 10CFR50 Appendix R were evaluated and conformance was maintained. Building services were revised and did not impact any operation of equipment important to safety. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.: 95-034

Implementation Document No.: Calculation ES-216-00B
Calculation H21C-038-01B

USAR Affected Pages: Tables 15.6-13 Sh 2 through 11, 15.6-16b

System: Reactor Containment Purge (CPS), Gas-Nitrogen (GSN)

Title of Change: Revised Bypass Leakage Design Basis 2GSN*V205

Description of Change:

This change revised the design basis analyses for determination of the bypass leakage through the CPS wetwell and drywell supply lines and resultant doses due to the increased leakage. The increased leakage was due to removal of leakage mitigation credit for check valve 2GSN*V205. Since this valve is not part of a leak test program, and will not be added to one, credit cannot be taken for leakage reduction following a design basis LOCA, as had been assumed in the original design basis analyses.

Safety Evaluation Summary:

This safety evaluation concludes that an unreviewed safety question does not exist as a result of removing leakage mitigation credit for check valve 2GSN*V205. This conclusion is based on the calculation of the additional leakage attributable to deleting credit for the check valve, and determination that the resultant doses will not cause the limits of 10CFR100 or 10CFR50 GDC 19 to be exceeded. The calculated doses at the exclusion area boundary, low population zone, and in the Control Room increase as a result of this change; however, they remain below the limits of 10CFR100 and GDC 19. Since Unit 2 is licensed to the limits of 10CFR100 and GDC 19, the consequences of a design basis accident are determined not to have increased.

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

Safety Evaluation No.: 95-035

Implementation Document No.: Simple Design Change SC2-0148-94

USAR Affected Pages: Figures 9.2-6a, 9.2-17b

System: Condensate Makeup and Drawoff (CNS), Makeup Water (MWS)

Title of Change: Elimination of Hose Connection Headers for MWS and CNS Systems

Description of Change:

This change eliminated hose connection headers and installed new permanent hose connections to allow movement of large equipment into the decontamination room of the Dirty Workshop on elevation 261'.

Safety Evaluation Summary:

An engineering review found that eliminating hose connection headers in the decontamination room and installing new permanent hose connections will improve movement of large equipment and still supply CNS and MWS to the decontamination room.

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

Safety Evaluation No.:	95-036
Implementation Document No.:	Simple Design Change SC2-0157-94
USAR Affected Pages:	9.5-17, 9.5-18, 9.5-21; Table 8.3-1 Sh 3 & 4; Figures 8.3-2, 8.3-3, 8.3-4, 8.3-6 Sh 5 & 27
System:	LAR, EJS, NJS
Title of Change:	Change Power Supply to 2LAR-PNL200

Description of Change:

This change separated the nondivisional Reactor Building normal lighting system from its present Divisional Class 1E power source 2EJS*US1 and connected it to its originally designed nondivisional source 2NJS-US2. This change also disconnected a loss-of-coolant accident signal circuitry to 2EJS*US1 which provided for tripping the breaker feeding the lighting panel and the associated computer points and annunciation circuitry to the plant process computer and to panel 2CEC*PNL852, respectively. Due to a NRC approved extension in the amount of time allowed for the Reactor Building drawdown, a new drawdown analysis has determined that the heat load generated from the Reactor Building normal lighting system would no longer prevent the drawdown from being achieved.

Safety Evaluation Summary:

The work scope is minimal and involves the disconnection and connection of existing cables and functional testing. No new cables or raceways will be installed and those cables to be spared will be abandoned in place. Separation criteria, Appendix R requirements, and electrical protection will be maintained.

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

Safety Evaluation No.: 95-037

Implementation Document No.: Simple Design Change SC2-0019-95

USAR Affected Pages: Figure 9.5-42

System: EGS

Title of Change: Reroute of the Governor Oil Cooling Line

Description of Change:

The cooling arrangement for the governor oil cooler was found to be inadequate to keep the governor's oil temperature to the vendor's recommended values. The cause of this condition was attributed to inadequate cooling water flow rate through the cooler. This simple design change improved this cooling arrangement by rerouting the return line from the governor oil cooler to a low pressure point of the jacket water system to increase the differential pressure and the flow rate.

Safety Evaluation Summary:

This change will maintain diesel generator reliability by providing proper cooling water flow to the governor oil cooler. In addition, a throttle valve will be added to the return line to obtain optimum governor cooler temperature. Operation of this valve will be controlled by procedure N2-OP-100A to maintain governor temperature between 120°F and 200°F.

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

Safety Evaluation No.: 95-038

Implementation Document No.: Calculation MS-4361

USAR Affected Pages: 9.1-12, 9.1-37, 9.1-38

System: FHS

Title of Change: Clarification of Design Basis for Spent Fuel Pool Rack External Loading

Description of Change:

This change corrected the discrepancy between the design basis calculation, fuel handling procedures, and Section 9.1 of the USAR. The design basis calculation was revised to include the case for the fuel bundle dropping onto the spent fuel pool racks while being moved with the polar crane 1/2-ton hoist. Additionally, the calculation revision addressed the case of a fuel bundle and grapple being dropped over the spent fuel pool racks from a maximum height of 30 inches above the racks. Use of the 25-ton auxiliary hoist for transfer of new fuel bundles and the position of the fuel bundle crate when opened now better describe the actual new fuel receipt activities. The required changes indicate the use of the 1/2-ton hoist for the transfer of new fuel bundles to either the new fuel storage vault or the new fuel inspection stand. Also, the new fuel bundle crate may be opened in the horizontal position provided that the crate still functions to support the fuel bundles.

Safety Evaluation Summary:

Clarification between the procedure and the design basis documents for the spent fuel storage racks' external loading was done in accordance with the design basis for the spent fuel pool racks as referenced in USAR Section 9.1. The correction of USAR Section 9.1 regarding new fuel receipt activities and associated procedures was done in accordance with the heavy load commitments per NUREG-0612, as referenced in USAR Appendix 9C.

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

Safety Evaluation No.:	95-043
Implementation Document No.:	LDCR 2-95-UFS-038
USAR Affected Pages:	15.7-8 through 15.7-14
System:	N/A
Title of Change:	Update SAR Description of the Bundle Drop Accident

Description of Change:

The fuel handling accident analysis for the bundle drop accident in Section 15.7.4 of the Unit 2 USAR has been revised to incorporate several changes in the assumptions of the analysis. The accident occurs during a refueling operation when a fuel assembly is moved over the top of the core. While the fuel grapple is in the overhoist condition (bottom of the assembly 32.95 feet above the top of the core), a main hoist cable fails allowing the assembly, the fuel grapple mast, and head to fall on top of the core impacting a group of four assemblies. The grapple head and mast are fixed vertically to the dropped assembly such that all the kinetic energy is transferred through the dropped assembly to the group of impacted assemblies. The dropped assembly impacts the core at a slight angle and the rods in this assembly are subjected to bending. After the assembly impacts the core, the assembly, grapple head and mast fall onto the core horizontally without contacting the side of the pressure vessel.

Safety Evaluation Summary:

The number of failed fuel rods for the bundle drop accident is determined by balancing the energy of the dropped assemblage against the energy required to fail a rod. The dropped assembly is considered to impact at a small angle, subjecting all the fuel rods in the dropped assembly to bending moments. The fuel rods are expected to absorb little energy prior to failure as a result of bending. For this reason it is assumed that all the rods in the dropped assembly fail. Therefore, the total number of failed rods on initial impact is 62+33=95. The assembly is assumed to tip over and impact horizontally on the top of the core. The energy from this second impact will result in 9 more failed rods. Consequently, the total number of failed rods from both impacts is determined to be 104. This compares with 124 failed rods from the analysis presented in the current USAR. Since the new analysis shows fewer failed rods, the radiological consequences are bounded by those of the previous analysis. The revised bundle drop accident methodology incorporates several conservative assumptions (i.e., including the weight of the fuel

Safety Evaluation No.: **95-043 (cont'd.)**

Safety Evaluation Summary: (cont'd.)

assembly and the mast and assuming a greater drop height) while maintaining the radiological consequences of this accident within the limits of the current analysis in the USAR. This methodology is the standard methodology used by General Electric (GE) for the licensing of all new fuel types and is included in GE's Standard Application for Reactor Fuel (GESTAR-II).

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

Safety Evaluation No.: 95-049 Rev. 0 & 1

Implementation Document No.: Procedure N2-OP-52

USAR Affected Pages: 6.2-66, 6.2-68, 6.2-69; Figure 6.2-77

System: Standby Gas Treatment (GTS)

Title of Change: 1-Hour Drawdown Analysis

Description of Change:

This safety evaluation addresses the revision to the drawdown analysis and procedural changes to reflect the analysis. The revised drawdown analysis removed existing conservatism (reduce the spent fuel pool heat load, and reduce allowable 2HVR*UC413A, B degradation from 40% to 20%) for the following six improvements:

1. Reduce ΔT requirement and eliminate periodic ΔT monitoring
2. Eliminate the normal lighting trip upon LOCA signal
3. Use only one of the two ECCS pump room unit coolers
4. Eliminate ΔT penalty curve for general area unit coolers
5. Restore use of electrical heaters in the ECCS cubicles and allow heating in secondary containment based on specific engineering evaluation
6. Increase the GTS and service water (SWP) systems initiation time following LOCA

These changes were made to improve plant flexibility and ease of maintenance work. The revised drawdown parameters are as follows:

- The longest calculated drawdown time is 57 minutes.

- Emergency unit coolers 2HVR*UC413A, B degradation can be as high as 20%.

- The GTS and SWP/unit coolers system initiation time following a loss-of-coolant accident (LOCA) can be as high as 60 and 90 seconds, respectively, from drawdown considerations only.

Safety Evaluation Summary:

The consequences of these changes have been evaluated against the current requirements. It is concluded that the 1-hour drawdown time requirement is not impacted.

Safety Evaluation No.: **95-049 Rev. 0 & 1 (cont'd.)**

Safety Evaluation Summary: (cont'd.)

These improvements do not impact GTS/HVR systems capacity to restore and maintain required vacuum in the secondary containment following a LOCA.

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

Safety Evaluation No.: 95-050

Implementation Document No.: N/A

USAR Affected Pages: N/A

System: High-Pressure Core Spray (CSH)

Title of Change: Temporary Enclosure for CSH Strainer

Description of Change:

The high-pressure core spray (HPCS) system takes water from the suppression pool through suction strainer 2CSH*STR1, penetration Z-12, and suction valve 2CSH*MOV118. Penetration Z-12 and suction valve 2CSH*MOV118 are at elevation 194'-11 15/16" and suction strainer 2CSH*STR1 is at elevation 189'-8". With minimum suppression pool water level at elevation 199'-6", maintenance/repair work on 2CSH*MOV118 cannot be performed without isolating the suppression pool.

This safety evaluation was issued to address the installation of a temporary enclosure on the suction strainer, 2CSH*STR1, in order to support subsequent repair and maintenance work to be performed on 2CSH*MOV118. The enclosure on the strainer will ensure sufficient isolation of the suppression pool from suction valve 2CSH*MOV118.

Safety Evaluation Summary:

The enclosure will be utilized only for repair and maintenance activities on 2CSH*MOV118. Administrative controls per the work order shall be in place as part of the maintenance work package, which will not allow for work to be done on 2CSH*MOV118 if the enclosure leaks.

The plant will be in Mode 5 with the reactor vessel head removed, the cavity flooded, the spent fuel pool gates removed, and the water level maintained within the limits of Technical Specifications 3.9.8 and 3.9.9. Therefore, HPCS will not be required to be operational per Technical Specification 3.5.2.

The suppression pool is not required to be operable during this activity per Technical Specification 3.5.3. However, suppression pool level will be maintained between elevation 199'-6" and 201'-0" to ensure adequate net positive suction head for emergency core cooling system pumps needed for Shutdown Safety Criteria N + 1.

Safety Evaluation No.: **95-050 (cont'd.)**

Safety Evaluation Summary: (cont'd.)

The valve pit wall elevation is equal to maximum suppression pool water level. Therefore, should the temporary enclosure fail, the valve pit will contain leakage from the suppression pool. Based on the size of the valve pit, the total leakage from the suppression pool would amount to approximately 5,723 gallons. This would result in lowering the suppression pool water level by approximately 1-1/2". Therefore, the availability of the suppression pool for water inventory control will not be affected.

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

Safety Evaluation No.: 95-051 Rev. 0 & 1

Implementation Document No.: Mod. PN2Y94MX013

USAR Affected Pages: 8.3-15; Tables 3.9A-12 Sh 8, 6.2-56 Sh 2, 8.3-1 Sh 17 & 20, 8.3-2 Sh 16, 17, 20, 8.3-4 Sh 8, 15, 8.3-5 Sh 1, 2, 3, 4, 8.3-6 Sh 2, 3, 4

System: Residual Heat Removal (RHS)

Title of Change: New Limitorque Actuators for 2RHS*MOV15A/B and MOV25A/B

Description of Change:

Based on revised sizing calculations due to changes made to the Unit 2 motor-operated valve (MOV) sizing calculation methodology, the motor output torque/thrust capability for containment spray isolation valves 2RHS*MOV15A/B and MOV25A/B under reduced voltage condition was not adequate to close the valves against the maximum expected differential pressure.

Containment spray injection valves 2RHS*MOV15A/B and MOV25A/B required replacement of their SMB-1-40 Limitorque motor/actuators with SMB-2-80 motor/actuators. The new motor/actuators are rated at 5.2 HP and 80 ft-lbs. The new motor/actuators meet valve operation test and evaluation system (VOTES) testing requirements per Generic Letter 89-10 and verify operation under design bases conditions.

Safety Evaluation Summary:

Replacement of the Limitorque motor/actuators for valves 2RHS*MOV15A/B and MOV25A/B with larger size motor/actuators will provide an acceptable torque switch setting thrust range to allow the valve to operate as intended during design basis conditions. This new range will also accommodate the use of the VOTES diagnostic test equipment and allow for actuator degradation.

Qualification for the new Limitorque motor/actuators has been performed to ensure continued structural integrity and operability of the modified valve assembly.

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

Safety Evaluation No.: 95-053

Implementation Document No.: Procedure N2-EMP-GEN-660

USAR Affected Pages: 8.3-74, 8.3-81; Table 1.8-1 Sh 63

System: BYS

Title of Change: Change of IEEE-Std 484 Year of Issue

Description of Change:

Unit 2 replaced the Div. I safety-related dc battery during Refuel Outage 3. The Div. II battery was replaced during Refuel Outage 4. Unit 2 is committed to comply with IEEE-Std 484-1975, "IEEE Recommended Practice for Installation Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations," for new battery installation. Since the time of the installation, the standard has been revised several times. The 1987 issue is now in effect. According to IEEE, the latest issue of the standard reflects the current state of the art and is recommended for use. The criteria provided in the 1987 issue of the standard generally encompass or exceed the criteria of the 1975 issue. The new criteria will increase safety during installation and testing and reduce the installation time. The 1987 issue provides a wider range of acceptance criterion for the intercell connectors resistance that may facilitate installation and testing. This change has no impact on battery characteristics or performance. Unit 2 dc system design criterion is to maintain 105 V dc minimum at the battery terminals regardless of the intercell connection resistance. This criterion is satisfied.

Safety Evaluation Summary:

The analysis performed revealed that the new resistance criteria for intercell connections does not compromise the ability of the battery to perform the safety-related function as designed and as described in the USAR. Engineering calculation performed for the most loaded battery determined that the impact of the new resistance criterion on the total battery voltage during discharge cycle is negligible. The battery's capacity, short circuit capability, and heat release are not affected either. Technical Specification operability criteria and surveillance requirements are also satisfied.

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

Safety Evaluation No.: 95-055 Rev. 0 & 1

Implementation Document No.: Simple Design Change SC2-0107-94

USAR Affected Pages: 9.2-14; Figure 9.2-3C;

System: Reactor Building Closed Loop Cooling Water (CCP)

Title of Change: Alternate Drywell Cooling

Description of Change:

In order to provide an alternate drywell cooling system to be used during outages, this simple design change added two permanent changes:

- Two piping penetrations through the southeast quadrant of the Reactor Building wall
- New 4" hose connection on the CCP supply and return headers

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

Safety Evaluation Summary:

The permanent changes are designed in accordance with design criteria for CCP. The Reactor Building penetrations are designed to ASME III NC-3600 requirements and include redundant spring-loaded check valves/blind flanges to assure that secondary containment integrity is maintained when alternate drywell cooling is operating/secured. The hoses will be routed so as to prevent physical interaction with safety-related items in the event of connector failure. All potentially affected essential equipment or systems are designed for flood or spray.

The implementation of this change will ensure that drywell temperature is controlled during an outage such that personnel stay times are maximized.

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

Safety Evaluation No.:	95-056
Implementation Document No.:	Simple Design Change SC2-0114-94
USAR Affected Pages:	Table 6.2-56, Sh 4
System:	High-Pressure Core Spray (CSH)
Title of Change:	Replace Valve Stem, Disc and Operator Gear Set for 2CSH*MOV105

Description of Change:

This simple design change changed the operator gear set and replaced the valve stem and disc for safety-related motor-operated valve (MOV) 2CSH*MOV105. The new gear set will increase the actuator output capacity under reduced voltage conditions and the new stem and disc will provide higher ASME allowable stresses. These changes, in turn, will increase the thrust window to accommodate diagnostic test equipment for torque switch setting as required by the Generic Letter 89-10 program. As a result of the actuator gear set change, the valve closure time will be increased.

Safety Evaluation Summary:

An engineering evaluation of the proposed change concluded that the replacement of the stem and disc with a higher ASME allowable and an increase in the stroke time due to the replacement gear set for the subject valve has no impact on the containment isolation requirements and the high-pressure core spray system operation as described in the USAR. The higher stroke time is still within the design basis of the system requirements. The MOV as well as the system will perform its intended safety function during and after an accident.

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

Safety Evaluation No.: 95-057

Implementation Document No.: Simple Design Change SC2-0095-94

USAR Affected Pages: 3B-2, 3B-3, 3B-5, 3B-6, 5.4-24; Table 6.2-56 Sh 5

System: Reactor Core Isolation Cooling (ICS)

Title of Change: Actuator Gear Set Changes for 2ICS*MOV121 and 2ICS*MOV128

Description of Change:

This change replaced the actuator gear sets for the subject valves in order to provide a sufficient thrust window for the valve operation test and evaluation system (VOTES) diagnostic equipment. As a result of the gear set change, the stroke time for these valves will increase from 15 seconds to 30 seconds.

Safety Evaluation Summary:

An engineering review of the requested change, which includes the effects of the change on the system's operability, reliability, maintainability, structural integrity and system interactions, has found that the implementation of this change will have no change on the safety or operability of the ICS system.

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

Safety Evaluation No.: 95-059 Rev. 0 & 1

Implementation Document No.: Procedures N2-TSP-CNT-@001, N2-TSP-CNT-@003, N2-TDP-IIT-0201, N2-TTP-CNT-001

USAR Affected Pages: 6.2-104; Figures 6.2-71a, 6.2-71b, 6.2-73a

System: N/A

Title of Change: Primary Containment Integrated Leak Rate Test

Description of Change:

The procedures for the integrated leak rate test were revised for the Type A test to be performed in Refueling Outage 4 (RFO4). The changes are as follows:

Item 1 (Rev. 0)

To allow the use of ANSI/ANS-56.8-1987, Containment System Leakage Testing Requirements, for the "Mass Point" method. NUREG-1047, Safety Evaluation Report related to the operation of Nine Mile Point Nuclear Station, Unit No. 2, Section 6.2.6, states the Type A test data will be analyzed using the "Mass Point" method in ANSI/ANS-56.8-1981. The "Mass Point" method will still be used to analyze the Type A test data, but a more current revision to ANSI/ANS-56.8 was used. The reason for the change is that 10CFR50 Appendix J was revised in 1988 to accept the "Mass Point" method described in ANSI/ANS-56.8-1987 but Nine Mile Point did not update the licensing base to reflect the change in 10CFR50 Appendix J.

Item 2 (Rev. 0)

To allow the installation and use of temporary instrumentation to monitor drywell parameters during the Type A test. USAR Section 6.2.6 states that two independent quartz digital-type absolute pressure manometers are connected to the leakage monitoring system (LMS) to monitor primary containment pressure during the Type A test. USAR Section 6.2.6 also states that 18 temperature elements and 6 humidity analyzers are provided in the containment atmosphere monitoring system (CMS) to monitor dry-bulb and dewpoint temperatures, respectively. The temporary instrumentation will be placed in the same locations as the permanent plant equipment.

Safety Evaluation No.: **95-059 Rev. 0 & 1 (cont'd.)**

Description of Change: (cont'd.)

The reason for the change is that the instrumentation provides the test data for the Type A test. Advances in electronic technology have resulted in more reliable and reduced installation times over conventional instrumentation. The result is reduced costs in man-hours and man-rem during the installation and removal phases.

<u>Item 3</u> (Rev. 1)

Procedure N2-TSP-CNT-@001 is being revised to allow the installation of temporary depressurization flanges on 2-CPS-014-9-4 and piping penetration 2PCB*Z74. These flanges will be used during the Type A test to reduce primary containment pressure. The reason for the change is to allow a safe and controlled depressurization of the primary containment.

Safety Evaluation Summary:

This safety evaluation has concluded that an unreviewed safety question does not exist as a result of evaluating a 24-hour Type A leakage rate in accordance with the "Mass Point" method described in ANSI/ANS-56.8-1987. ANSI/ANS-56.8-1987 provides recommendations for the Type A test instrumentation. These recommendations include calibration requirements, in-situ checks, and minimum quantities and loss criteria. N2-TTP-CNT-001 and N2-TSP-CNT-@001 were written to ensure that the recommendations of ANSI/ANS-56.8-1987 are met. Also, the temporary instrumentation will be placed in the corresponding locations of the permanent plant equipment. Therefore, an unreviewed safety question does not exist as a result of using temporary instrumentation to monitor primary containment parameters during the Type A test. The temporary flanges will be installed only in Operational Conditions 4 or 5 and are bounded by the USAR load combinations and stress limits for pipes and pipe penetrations. Therefore, this safety evaluation has concluded that an unreviewed safety question does not exist as a result of connecting temporary flanges to 2-CPS-014-9-4 and 2PCB*Z74.

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

Safety Evaluation No.: 95-060

Implementation Document No.: Simple Design Change SC2-0028-95

USAR Affected Pages: 10.4-30; Figure 8.3-1

System: Feedwater (FWS)

Title of Change: Feedwater Pump Motors HP Upgrade

Description of Change:

On January 17, 1995, feedwater pump motor 2FWS-M1B tripped while running at approximately full power. The trip occurred due to the action of the motor relay protection. The investigation of the event revealed that the insulation of the stator winding of the motor failed causing the action of the relay protection and motor trip. The motor was sent to Monarch Electric Service Co. for repair. Root cause evaluation performed by Monarch Co. identified that the motor insulation failure occurred due to corona erosion of ground wall insulation. The motor was rewound and returned to Unit 2. In the process of repair, a new type of insulation was used and the HP rating of the motor was increased from 12,000 to 14,100.

Since the motors 2FWS-M1A and 2FWS-M1C may also be susceptible to the same mode of failure, the decision was made to rewind these motors and upgrade the HP rating.

Safety Evaluation Summary:

The upgrading of HP of the feedwater pump motors satisfies functional requirements of the system. The performance of the pumps is not affected. The system and components will perform as designed and as described in the USAR. The upgraded HP of the motors is adequate for the power uprate of the plant.

The upgraded HP of the motors does not adversely affect the mechanical interface systems. According to NMPC Mechanical Engineering evaluation, the maximum HP requirement for the pump for power uprate condition is 13,190 HP. Therefore, the upgraded 14,100 HP is adequate.

The electrical equipment such as cables, circuit breakers, current transformers, and relays ratings were evaluated for the upgraded HP of the motors and were found to be adequate.

Safety Evaluation No.: **95-060 (cont'd.)**

Safety Evaluation Summary: (cont'd.)

The feedwater pumps are nonsafety-related components and are not required for safe shutdown of the plant. The upgraded HP of the motors has no impact on safety-related systems and components.

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

Safety Evaluation No.: 95-062

Implementation Document No.: Simple Design Change SC2-0029-95

USAR Affected Pages: Appendix 9C Table 3-2

System: MHR

Title of Change: Alternate Means for the SRV Removal (Replacing 2MHR-CRN200 With 2MHR-CRN200A and 2MHR-CRN200B)

Description of Change:

The purpose of this simple design change is to provide an interchangeable handling system for the removal and replacement of the safety relief valves (SRVs). Hoists 2MHR-CRN200A and 2MHR-CRN200B will replace crane 2MHR-CRN200 for the SRV removal activity. The SRVs were originally shipped to the site in the horizontal position and could be handled with a single 4-ton hoist. The SRVs now arrive in the vertical position, requiring a second hoist to safely remove the valves from their shipping container and reposition the valve into the horizontal position. Upon completion of the removal and replacement of the SRVs, hoists 2MHR-CRN200A and 2MHR-CRN200B can be removed from the monorail and crane 2MHR-CRN200 can be reinstalled.

Safety Evaluation Summary:

The improvement being made by this simple design change with the use of alternate hoists 2MHR-CRN200A and 2MHR-CRN200B in the place of crane 2MHR-CRN200 meets the requirements of the seismic evaluation of nonsafety-related components in safety-related areas and does not affect the safety and reliability of Unit 2. There is no safety-related equipment that would be affected by a load drop involving hoists 2MHR-CRN200A or 2MHR-CRN200B.

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

Safety Evaluation No.: 95-064

Implementation Document No.: Calculations EC-044 Rev. 11, EC-045, Rev. 7, EC-046 Rev. 5, EC-097 Rev. 2

USAR Affected Pages: 8.3-76; Tables 8.3-11 through 8.3-15

System: BYS, BWS

Title of Change: Nonsafety-Related Batteries Load Profile Update

Description of Change:

The battery sizing calculations were updated to reflect as-built dc loads of the nonsafety-related system, and to account for the plant modification which was implemented without revising these calculations.

Safety Evaluation Summary:

The revised battery sizing calculations conclude that the change of the dc loads is within the capabilities of the batteries and the chargers. The nonsafety-related dc system will continue to perform as designed and as described in the USAR, with the updated loads each battery is still capable of performing its duty cycle following the loss of charger while fully charged at 65°F, and with capacity deteriorated to 80 percent. Each battery can start and operate all required loads for the duration of the discharge cycle according to the battery load profile without battery terminal voltage falling below 105 V for 125 V dc system, and 21 V for 24 V dc system.

Each battery charger can still supply the continuous updated load on the battery while recharging the battery from the designed minimum charge state to the fully charged state in less than 24 hours.

The impact of this change on the plant response to station blackout event (SBO) has also been evaluated. Based on additional battery calculations performed for the new revised loads, the conclusion is made that nonsafety-related batteries still meet the 4-hour capability requirement as specified in NUMARC 87-00 and Regulatory Guide 1.155, and as demonstrated in the SBO study performed by General Electric.

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

Safety Evaluation No.: 95-066

Implementation Document No.: Temporary Mod. 95-011

USAR Affected Pages: N/A

System: Reactor Core Isolation Cooling (ICS)

Title of Change: Manual Operation of 2ICS*AOV130

Description of Change:

The ICS system is designed to assure sufficient reactor water inventory is maintained in the reactor vessel to permit adequate reactor core cooling. This temporary modification manually opened valve 2ICS*AOV130 and maintained it in the open position until the next system outage because the valve actuator is not capable of keeping the valve open due to diaphragm failure. This valve is one of the two normally open valves in series on the drain pot drain line of ICS en-route to the Reactor Building equipment drain system (DER).

Safety Evaluation Summary:

Based on a review of the system design bases and configuration, there is no specific reason in the USAR for the double isolation arrangement. A system design review by General Electric determined that double isolation arrangement was intended to provide redundancy of the drain valve closure during ICS operation. This redundancy was to minimize the spread of contamination and radiation release in the Reactor Building in case of high radiation levels in the steam supply line to the ICS turbine. Based on the reviews performed, it has been determined that this isolation capability can still be maintained via a single valve with no impact on nuclear safety.

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

Safety Evaluation No.: 95-068

Implementation Document No.: Procedure N2-MPM-GEN-R901

USAR Affected Pages: 9.1-39, 9.1-41

System: FHP

Title of Change: Revision to Fuel Pool Gate Removal Process in USAR Section 9.1

Description of Change:

This change revised the USAR to indicate the option to remove both the inner and the outer spent fuel pool gates after completion of flood-up activities prior to refueling, as described in procedure N2-MPM-GEN-R901, Rev. 1.

Safety Evaluation Summary:

The revision of the USAR to indicate the option to remove both the inner and the outer spent fuel pool gates after the completion of flood-up activities results in a more conservative plant configuration during reactor vessel disassembly activities. This was performed in accordance with the Guidelines for the Control of Heavy Loads (NUREG-0612) as described in USAR Appendix 9C.

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

Safety Evaluation No.: 95-069

Implementation Document No.: N/A

USAR Affected Pages: N/A

System: 345-kV Transmission Output, 115-kV Offsite Power Sources

Title of Change: Scriba Station, 345-kV B Bus Connection

Description of Change:

A sixth 345-kV transmission line was added to Scriba Station and connected to the 345-kV A bus in June 1994. Scriba Station is a 345-kV breaker and 1/2 station with an A and a B bus. The new transmission line was connected to the 345-kV B bus.

Safety Evaluation Summary:

The plant will be shut down for refuel during the period when the work will take place. All applicable Technical Specification requirements will be met. The work and the schedule have been reviewed for safe shutdown criteria.

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

Safety Evaluation No.: 95-071

Implementation Document No.: N/A

USAR Affected Pages: A.0-1, A.4.3-1, A.4.4-3, A.5.2-1, A.5.2-2, A.5.2-4, A.6-1, A.6-2, A.15.0-2, A.15.0-7, A.15.1-4, A.15.1-9, A.15.2-5, A.15.2-12, A.15.4-9, A.15B-1, A.15D-1; Tables A.5.2-1, A.5.2-2, A.6-2, A.15.0-4 Sh 1, 2, 3

System: Various

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

Description of Change:

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

Safety Evaluation Summary:

The reload analyses and evaluations are performed based on the General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A-10 and NEDE 24011-P-A-10-US (GESTAR II). This document describes the fuel licensing acceptance criteria; the fuel thermal-mechanical, nuclear, and thermal-hydraulic analyses bases; and the safety analysis methodology. For Reload 4, the evaluations included transients and accidents likely to limit operation because of minimum critical power ratio considerations; overpressurization events; loss-of-coolant accident; and stability analysis. Appropriate consideration of equipment out of service was included. Limits on plant operation were established to assure that applicable fuel and reactor coolant system safety limits are not exceeded.

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

Safety Evaluation No.:	95-077
Implementation Document No.:	Calculations H21C-038-01C, H21C-043-00B, A10.1-E-130
USAR Affected Pages:	15.6-13; Tables 15.6-13 Sh 10 & 11, 15.6-16b
System:	Residual Heat (RHS)
Title of Change:	Revise/Delete the Leak Rate Acceptance Criteria and Test Frequency for RHS Valves 2RHS*MOV142, MOV149, SOV35A/B and SOV36A/B

Description of Change:

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

Safety Evaluation Summary:

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

Safety Evaluation No.: 95-078

Implementation Document No.: Procedure N2-FHP-021

USAR Affected Pages: N/A

System: FNR, FNS, GTS

Title of Change: Revision to Control Blade Movement Procedure

Description of Change:

This safety evaluation was written to revise procedure N2-FHP-021, "Control Rod Uncoupling, Removal, and Installation." This revision allows control rod uncoupling, removal, and installation without secondary containment integrity and SGTS operability, provided seven days have elapsed since reactor shutdown and all movements are of objects totalling less than 617 pounds (the estimated weight of a fuel bundle).

Safety Evaluation Summary:

The revised control blade movement procedure provides the same level of safety to the Control Room and public as was previously available. The change does not alter Technical Specifications, or guidance provided by the vendor. Radiological analysis has shown that the proposal allows Unit 2 to meet 10CFR100 limits and remain in compliance with the plant safety analysis report.

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

Safety Evaluation No.: 95-079

Implementation Document No.: DER 2-95-1183

USAR Affected Pages: 8.3-11, 8.3-12

System: VBB

Title of Change: Revise UPS 2VBB-UPS1C/1D Acceptable Voltage Output Criteria

Description of Change:

Uninterruptible power supplies (UPSs) 2VBB-UPS1C and 2VBB-UPS1D feed selected lighting and communications loads. The original UPS units installed during construction were purchased from Exide. In 1991, these units were replaced with units purchased from HDR under Modification PN2Y89MX042. During the replacement, it was discovered that these units did not meet the output voltage acceptance criteria of Specification E-147, i.e., $\pm$2% of output voltage variation. Engineering evaluation of the deficiency was performed and units were accepted as supplied by the vendor with voltage output acceptable up to $\pm$3% based on the type of loads these units are feeding. The engineering specification E-147 was revised to allow the new acceptance criteria for UPSs 2VBB-UPS1C and 1D.

Safety Evaluation Summary:

The proposed change of accepting the $\pm$3% of output voltage variation does not affect the performance of the connected loads. The purpose of specifying precise output voltage regulation for the UPSs is to meet the requirements of the precision instrumentation and control equipment which they feed. Most of this equipment will require power supply voltage variation not to exceed 2%. The two UPSs involved in this change, 2VBB-UPS1C and 2VBB-UPS1D provide power supply only to the essential lighting, egress lighting, and page party/public address (PP/PA) communication system loads. The essential and egress lighting system equipment are designed for $\pm$10% supply voltage variation. The PP/PA communication system equipment are designed for 90-140 V as shown in Gaitronic specification. Therefore, an output voltage variation of $\pm$3% for 2VBB-UPS1C and 2VBB-UPS1D will not adversely affect operation of any of their connected equipment and is acceptable.

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

Safety Evaluation No.: 95-080 Rev. 0 & 1

Implementation Document No.: Procedure N2-PM-S012

USAR Affected Pages: 3C-25, 3C-28, 3C-29

System: N/A

Title of Change: Change the Visual Leak/Flood Detection Walkdowns to a Minimum of Once per Calendar Day

Description of Change:

As depicted in the USAR, area walkdowns by plant personnel for visual leak/flood detection were performed once every 8-hour shift. In the past, plant operations personnel were scheduled for three 8-hour shifts. Plant operations has revised their shift work schedules from three 8-hour shifts to two 12-hour shifts.

Calculations were revised to reflect the increase in water levels due to the change of the visual leak/flood detections from 12 hours to a minimum of once per calendar day, not to exceed a 24-hour time period (36-hour time period for the Control Building basement) between inspections.

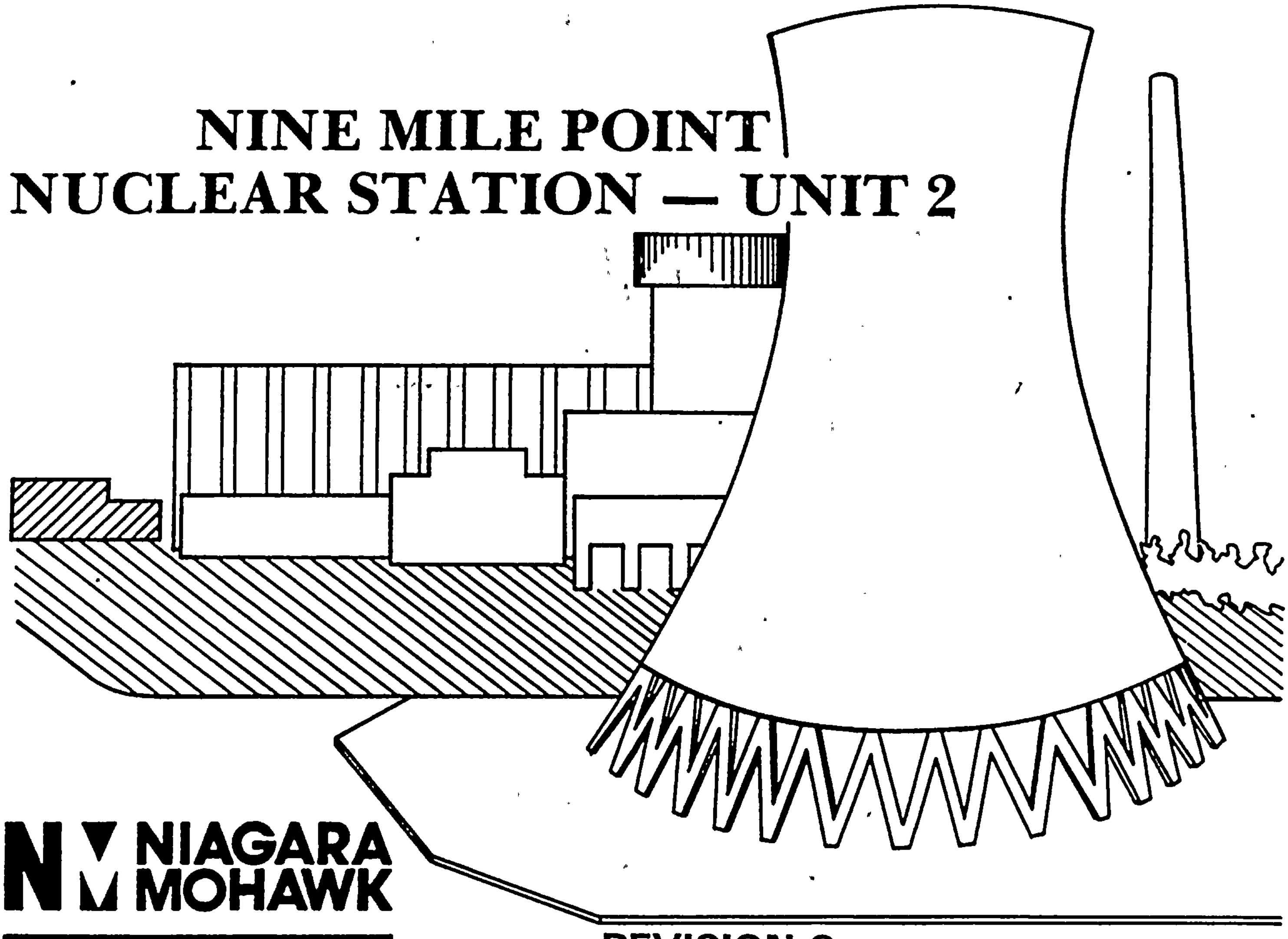
Safety Evaluation Summary:

The revision to the USAR to change the time intervals of the visual flood/leak detection walkdowns from every 12½ hours to a minimum of once per calendar day, not to exceed a 24-hour time period (36-hour time period for the Control Building basement) between inspections, does not affect the safety and reliability of Unit 2. The loss of safety-related equipment due to flooding has already been evaluated in the USAR Appendix 3C Spray/Flooding Evaluation and will not be changed.

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

UPDATED SAFETY ANALYSIS REPORT

NINE MILE POINT NUCLEAR STATION — UNIT 2



NIAGARA MOHAWK

REVISION 8

INSERTION INSTRUCTIONS

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

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

INSERTION INSTRUCTIONS

LISTS OF EFFECTIVE PAGES

REMOVE	INSERT
EP i	EP i
EP 1-1 through EP 1-4	EP 1-1 through EP 1-5
EP 2-1 through EP 2-21	EP 2-1 through EP 2-21
EP 3-1 through EP 3-12	EP 3-1 through EP 3-11
EP 4-1 through EP 4-2	EP 4-1 through EP 4-2
EP 5-1 through EP 5-2	EP 5-1 through EP 5-2
EP 6-1 through EP 6-6	EP 6-1 through EP 6-5
EP 7-1 through EP 7-5	EP 7-1 through EP 7-4
EP 8-1 through EP 8-3	EP 8-1 through EP 8-3
EP 9-1 through EP 9-11	EP 9-1 through EP 9-10
EP 10-1 through EP 10-2	EP 10-1 through EP 10-2
EP 11-1 through EP 11-2	EP 11-1 through EP 11-2
EP 12-1 through EP 12-2	EP 12-1 through EP 12-2
EP 13-1	EP 13-1
EP 14-1 through EP 14-4	EP 14-1 through EP 14-3
EP 15-1 through EP 15-6	EP 15-1 through EP 15-5
EP 16-1	EP 16-1
EP 17-1	EP 17-1
EP 18-1	EP 18-1
EP A-1	EP A-1
EP B-1	EP B-1
DAR-1 through DAR-4	DAR-1 through DAR-3
SSA-1 through SSA-2	SSA-1 through SSA-2

INSERTION INSTRUCTIONS

VOLUME 1

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-
1-i/ii	1-i/ii
1-v/vi	1-v/vi
1.1-1/2	1.1-1/2
F 1.1-1	F 1.1-1
1.2-5/6	1.2-5/6
1.2-13/14	1.2-13/14
1.2-21/22	1.2-21/22
F 1.2-1	F 1.2-1
F 1.2-7 (Sh 2 of 2)	F 1.2-7 (Sh 2 of 2)
F 1.2-11 (Sh 3 of 4)	F 1.2-11 (Sh 3 of 4)
F 1.2-40	F 1.2-40
T 1.3-1 (Sh 1 of 3)	T 1.3-1 (Sh 1 of 3)
T 1.3-2 (Sh 1 of 2)	T 1.3-2 (Sh 1 of 2)
T 1.3-3 (Sh 1 of 2)	T 1.3-3 (Sh 1 of 2)
T 1.3-4	T 1.3-4
T 1.3-5 (Sh 1 of 2)	T 1.3-5 (Sh 1 of 2)
T 1.3-6	T 1.3-6
T 1.3-7	T 1.3-7
T 1.3-8 (Sh 1 of 2)	T 1.3-8 (Sh 1 of 2)
T 1.3-9 (Sh 1 of 8)	T 1.3-9 (Sh 1 of 8)

INSERTION INSTRUCTIONS

VOLUME 2

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-
T 1.8-1 (Sh 1/2)	T 1.8-1 (Sh 1/1a)
---	T 1.8-1 (Sh 1b/2)
T 1.8-1 (Sh 51/52)	T 1.8-1 (Sh 51/52)
T 1.8-1 (Sh 53/54)	T 1.8-1 (Sh 53/54)
T 1.8-1 (Sh 63/64)	T 1.8-1 (Sh 63/64)
T 1.8-1 (Sh 75/76)	T 1.8-1 (Sh 75/76)
T 1.8-1 (Sh 77/78)	T 1.8-1 (Sh 77/78)
T 1.8-2 (Sh 5/6)	T 1.8-2 (Sh 5/6)
T 1.9-1 (Sh 35/36)	T 1.9-1 (Sh 35/36)
T 1.9-1 (Sh 47/48)	T 1.9-1 (Sh 47/48)
T 1.9-1 (Sh 49/50)	T 1.9-1 (Sh 49/50)
1.10-47/48	1.10-47/48
---	1.10-48a/48b
1.10-57/58	1.10-57/58
---	1.10-58a/58b
T II.B.3-1 (Sh 1 and 2 of 2)	T II.B.3-1 (Sh 1 and 2 of 2)
1.10-121/122	1.10-121/122
1.11-1/-	1.11-1/-
T 1.11-1 (Sh 1 thru 6 of 6)	T 1.11-1 (Sh 1 thru 6 of 6)

INSERTION INSTRUCTIONS

VOLUME 3

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-
2-xiii/xiv	2-xiii/xiv
F 2.1-2	F 2.1-2
2.2-1/2	2.2-1/2
2.2-3/4	2.2-3/4
2.2-7/8	2.2-7/8
T 2.2-1 thru T 2.2-3	T 2.2-1 thru T 2.2-3
T 2.2-5 (Sh 1 and 2 of 2)	T 2.2-5 (Sh 1 and 2 of 2)
T 2.2-6	T 2.2-6
T 2.3-4A	T 2.3-4A
T 2.3-6	T 2.3-6
2.4-1/2	2.4-1/1a
---	2.4-1b/2
T 2.4-6	T 2.4-6
T 2.4-7	T 2.4-7
T 2.4-9 thru T 2.4-11	T 2.4-9 thru T 2.4-11
T 2.4-13 thru T 2.4-15	T 2.4-13 thru T 2.4-15
F 2.4-1	F 2.4-1
T 2.5-3 (Sh 1 thru 3 of 3)	T 2.5-3 (Sh 1 thru 3 of 3)
T 2.5-4	T 2.5-4
T 2.5-6 thru T 2.5-8	T 2.5-6 thru T 2.5-8
T 2.5-10 thru T 2.5-13	T 2.5-10 thru T 2.5-13
T 2.5-15 thru T 2.5-19	T 2.5-15 thru T 2.5-19
T 2.5-21 thru T 2.5-23	T 2.5-21 thru T 2.5-23
T 2.5-25 thru T 2.5-27	T 2.5-25 thru T 2.5-27
T 2.5-28A	T 2.5-28A
T 2.5-30 thru T 2.5-32	T 2.5-30 thru T 2.5-32
T 2.5-45	T 2.5-45
T 2.5-48A thru T 2.5-48C	T 2.5-48A thru T 2.5-48C

INSERTION INSTRUCTIONS

VOLUME 4

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-

INSERTION INSTRUCTIONS

VOLUME 5

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-

INSERTION INSTRUCTIONS

VOLUME 6

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-

INSERTION INSTRUCTIONS

VOLUME 7

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-

INSERTION INSTRUCTIONS

VOLUME 8

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-
3-v/vi thru 3-xlvii/-	3-v/vi thru 3-xxxix/-
3.1-23/24	3.1-23/24
T 3.2-1 (Sh 18 of 34)	T 3.2-1 (Sh 18 of 34)
T 3.2-2 thru T 3.2-4	T 3.2-2 thru T 3.2-4
T 3.3-1	T 3.3-1
T 3.4-1 thru T 3.4-7	T 3.4-1 thru T 3.4-7

INSERTION INSTRUCTIONS

VOLUME 9

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-
3.5-7/8	3.5-7/8
3.5-9/10	3.5-9/9a
---	3.5-9b/10
3.5-15/16	3.5-15/16
T 3.5-3	T 3.5-3
T 3.5-6	T 3.5-6
T 3.5-9	T 3.5-9
T 3.5-12	T 3.5-12
T 3.5-15 thru T 3.5-18	T 3.5-15 thru T 3.5-18
T 3.6A-72 (Sh 1 thru 4 of 4)	T 3.6A-72 (Sh 1 thru 4 of 4)
T 3.6A-73 (Sh 1 and 2 of 2)	T 3.6A-73 (Sh 1 and 2 of 2)

INSERTION INSTRUCTIONS

VOLUME 10

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-
3.6B-1/2 thru 3.6B-21/-	3.6B-1/2 thru 3.6B-19/-
3.7A-1/2 thru 3.7A-33/34	3.7A-1/2 thru 3.7A-33/34
T 3.7A-1	T 3.7A-1
T 3.7A-11	T 3.7A-11/T 3.7A-12
T 3.7A-12	---
3.7B-1/2 thru 3.7B-27/-	3.7B-1/2 thru 3.7B-23/-
T 3.7B-1	T 3.7B-1/T 3.7B-2
T 3.7B-2	---
3.8-1/2 thru 3.8-77/78	3.8-1/2 thru 3.8-73/74
T 3.8-1 (Sh 1 thru 6 of 6)	T 3.8-1 (Sh 1 thru 6 of 6)
T 3.8-8 thru T 3.8-12	T 3.8-8 thru T 3.8-12
T 3.8-14	T 3.8-14

INSERTION INSTRUCTIONS

VOLUME 11

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-
3.9A-1/1a thru 3.9A-27b/28	3.9A-1/2 thru 3.9A-39/-
T 3.9A-1 (Sh 1 and 2 of 2)	T 3.9A-1 (Sh 1 and 2 of 2)
T 3.9A-3 thru T 3.9A-8	T 3.9A-3 thru T 3.9A-8
T 3.9A-11 thru T 3.9A-13	T 3.9A-11 thru T 3.9A-13
T 3.9A-15 (Sh 1 thru 2a of 2)	T 3.9A-15 (Sh 1 thru 3 of 3)
3.9B-1/2 thru 3.9B-73/-	3.9B-1/2 thru 3.9B-71/-
T 3.9B-1 (Sh 1 and 2 of 2)	T 3.9B-1 (Sh 1 and 2 of 2)
T 3.9B-2 (Sh 1 thru 5 of 5)	T 3.9B-2 (Sh 1 thru 4 of 4)
T 3.9B-2d thru T 3.9B-2f	T 3.9B-2d thru T 3.9B-2f
T 3.9B-2h (Sh 1 thru 4 of 4)	T 3.9B-2h (Sh 1 thru 4 of 4)
T 3.9B-2j (Sh 1 thru 6 of 6)	T 3.9B-2j (Sh 1 thru 5 of 5)
T 3.9B-2k (Sh 1 thru 8 of 8)	T 3.9B-2k (Sh 1 thru 8 of 8)
T 3.9B-2m (Sh 1 and 2 of 2)	T 3.9B-2m (Sh 1 and 2 of 2)
T 3.9B-2p	T 3.9B-2p
T 3.9B-4 thru T 3.9B-8	T 3.9B-4 thru T 3.9B-8

INSERTION INSTRUCTIONS

VOLUME 12

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-
3.10A-1/1a thru 3.10A-5/-	3.10A-1/2 thru 3.10A-5/-
3.10B-1/2 thru 3.10B-7/-	3.10B-1/2 thru 3.10B-5/6
T 3.10B-2	---
F 3.10B-1 thru F 3.10B-4	---
3.11-1/1a thru 3.11-7/-	3.11-1/2 thru 3.11-5/6
3A-v/vi	3A-v/vi
T 3A.1-1	T 3A.1-1
T 3A.7-1	---
F 3A.7-1	---
F 3A.7-2	---
T 3A.16-1 thru T 3A.16-5	---
T 3A.17-3 thru T 3A.17-5	T 3A.17-3 thru T 3A.17-5
T 3A.18-1	T 3A.18-1
T 3A.21-1/T 3A.21-2	T 3A.21-1/T 3A.21-2
T 3A.22-1/T 3A.22-2	T 3A.22-1/T 3A.22-2
F 3A.22-1	---
T 3A.23-1 (Sh 1/2)	T 3A.23-1 (Sh 1/2)
T 3A.26-1	T 3A.26-1
T 3A.27-1	T 3A.27-1
T 3A.33-1	T 3A.33-1/T 3A.33-2
T 3A.33-2	---
3B-i/ii thru 3B-v/vi	3B-i/ii thru 3B-v/vi
3B-1/2 thru 3B-11/12	3B-1/2 thru 3B-11/-
T 3B-1 thru T 3B-3	T 3B-1 thru T 3B-3
T 3B-7 thru T 3B-18	T 3B-7 thru T 3B-18
3C-i/ii	3C-i/ii
3C-1/2 thru 3C-33/34	3C-1/2 thru 3C-29/-

INSERTION INSTRUCTIONS

VOLUME 12 (Cont'd.)

REMOVE	INSERT
T 3C.4-6	T 3C.4-6
T 3C.5-1 thru T 3C.5-2	T 3C.5-1 thru T 3C.5-2
4-i/ii	4-i/ii
4-v/vi	4-v/vi
4.1-13/14	4.1-13/14
---	4.1-15/-
4.3-1/2	4.3-1/2
4.3-3/4	4.3-3/4
4.4-1/2	4.4-1/2
4.4-5/6	4.4-5/6
4.4-7/8	4.4-7/8
4.4-11/12	4.4-11/12
T 4.4-1 thru T 4.4-8	T 4.4-1 thru T 4.4-8
F 4.4-1	F 4.4-1
4.6-13/14	4.6-13/14
F 4.6-5b	F 4.6-5b
F 4.6-5c	F 4.6-5c
F 4.6-7 (Sh 1 and 2 of 3)	F 4.6-7 (Sh 1 and 2 of 3)
4A-1/2	4A-1/2

INSERTION INSTRUCTIONS

VOLUME 13

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-
5-i/ii	5-i/ii
5-iii/iv	5-iii/iv
5-ix/x	5-ix/x
F 5.1-1a	F 5.1-1a
5.2-3/4	5.2-3/4
5.2-5/6	5.2-5/6
5.2-15/16	5.2-15/16
5.2-21/22	5.2-21/21a
---	5.2-21b/22
5.2-47/48	5.2-47/48
T 5.2-2 thru T 5.2-4	T 5.2-2 thru T 5.2-4
T 5.2-6	T 5.2-6
F 5.2-1	F 5.2-1
F 5.2-4 (Sh 1 of 2)	F 5.2-4 (Sh 1 of 2)
5.3-5/6	5.3-5/6
---	5.3-6a/6b
T 5.3-2a	T 5.3-2a
5.4-5/6 thru 5.4-9/10	5.4-5/6 thru 5.4-9/10
5.4-17/18	5.4-17/18
5.4-23/24	5.4-23/24
5.4-25/26	5.4-25/26
5.4-29/30	5.4-29/30
5.4-43/44	5.4-43/44
5.4-45/46	5.4-45/46
5.4-51/52	5.4-51/52
T 5.4-1 thru T 5.4-3	T 5.4-1 thru T 5.4-3
F 5.4-2b	F 5.4-2b
F 5.4-2d	F 5.4-2d
F 5.4-10 (Sh 1 and 2 of 2)	F 5.4-10 (Sh 1 and 2 of 2)
F 5.4-12	F 5.4-12
F 5.4-13d	F 5.4-13d

INSERTION INSTRUCTIONS

VOLUME 13 (Cont'd.)

REMOVE	INSERT
F 5.4-13e	F 5.4-13e
F 5.4-13g	F 5.4-13g
F 5.4-14 (Sh 2 of 3)	F 5.4-14 (Sh 2 of 3)
F 5.4-16a	F 5.4-16a
F 5.4-16f	F 5.4-16f
F 5.4-17 (Sh 2 of 3)	F 5.4-17 (Sh 2 of 3)
5A-i/-	5A-i/-
5A-1/2 thru 5A-3/-	5A-1/2 thru 5A-3/-
T 5A-4	T 5A-4
5B-1/2 thru 5B-3/-	5B-1/2 thru 5B-3/-
6-i/ii thru 6-xxiii/-	6-i/ii thru 6-xxv/-

INSERTION INSTRUCTIONS

VOLUME 14

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-
6.2-1/2 thru 6.2-89/-	6.2-1/2 thru 6.2-107/-
T 6.2-1 thru T 6.2-43A	T 6.2-1 thru T 6.2-43A
T 6.2-44	T 6.2-44
T 6.2-45B	T 6.2-45B
T 6.2-47 thru T 6.2-52	T 6.2-47 thru T 6.2-52
T 6.2-54	T 6.2-54
T 6.2-55	T 6.2-55
T 6.2-55c thru T 6.2-55d	T 6.2-55c thru T 6.2-55d
T 6.2-56 (Sh 1 thru 5 of 24)	T 6.2-56 (Sh 1 thru 5 of 24)
T 6.2-56 (Sh 7 of 24)	T 6.2-56 (Sh 7 of 24)
T 6.2-56 (Sh 10 thru 12 of 24)	T 6.2-56 (Sh 10 thru 12 of 24)
T 6.2-56 (Sh 20 thru 24 of 24)	T 6.2-56 (Sh 20 thru 24 of 24)
T 6.2-57 thru T 6.2-62	T 6.2-57 thru T 6.2-62
T 6.2-64	T 6.2-64
F 6.2-2 thru F 6.2-12	F 6.2-2 thru F 6.2-12
F 6.2-15 thru F 6.2-27	F 6.2-15 thru F 6.2-27
F 6.2-29	F 6.2-29
F 6.2-34 thru F 6.2-37	F 6.2-34 thru F 6.2-37
F 6.2-44	F 6.2-44
F 6.2-71a	F 6.2-71a
F 6.2-71b	F 6.2-71b
F 6.2-72D thru F 6.2-72I	F 6.2-72D thru F 6.2-72I
F 6.2-73a	F 6.2-73a
F 6.2-77	F 6.2-77
---	F 6.2-95A thru F 6.2-95D

INSERTION INSTRUCTIONS

VOLUME 15

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-
6.3-19/20	6.3-19/20
6.3-37/38	6.3-37/38
T 6.3-1 (Sh 1/2)	T 6.3-1 (Sh 1/2)
T 6.3-1 (Sh 3/4)	T 6.3-1 (Sh 3/4)
T 6.3-4 (Sh 1/2)	T 6.3-4 (Sh 1/2)
T 6.3-5	T 6.3-5
F 6.3-1 (Sh 1 and 2 of 2)	F 6.3-1 (Sh 1 and 2 of 2)
6.4-1/2 thru 6.4-7/-	6.4-1/2 thru 6.4-5/6
6.5-5/6	6.5-5/6
T 6.5-1	T 6.5-1
6.6-1/1a thru 6.6-3/-	6.6-1/2 thru 6.6-3/-
6B-i/-	6B-i/-
6B-1/2 thru 6B-9/-	6B-1/2 thru 6B-7/8
6C-1/2 thru 6C-25/26	6C-1 thru 6C-19/-
---	T 6C-1/T 6C-2
F HC-1	F 6C-1
7-i/ii thru 7-xi/xii	7-i/ii thru 7-xiii/-
7.1-1/1a thru 7.1-7b/8	7.1-1/2 thru 7.1-11/-
T 7.1-2 (Sh 1 thru 3 of 3)	T 7.1-2 (Sh 1 thru 3 of 3)
7.2-1/2 thru 7.2-21/22	7.2-1/2 thru 7.2-21/22
7.3-1/2 thru 7.3-35/-	7.3-1/2 thru 7.3-33/34
T 7.3-1 thru T 7.3-17	T 7.3-1 thru T 7.3-17
F 7.3-10 (Sh 1 of 2)	F 7.3-10 (Sh 1 of 2)

INSERTION INSTRUCTIONS

VOLUME 16

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-
7.4-1/2 thru 7.4-23d/24	7.4-1/2 thru 7.4-25/-
T 7.4-1	T 7.4-1
F 7.4-1 (Sh 1 of 6)	F 7.4-1 (Sh 1 of 6)
7.5-1/1a thru 7.5-9/10	7.5-1/2 thru 7.5-9/-
T 7.5-2 (Sh 1 thru 13 of 13)	T 7.5-2 (Sh 1 thru 8 of 8)
7.6-1/2 thru 7.6-27/-	7.6-1/2 thru 7.6-35/36
T 7.6-1 thru T 7.6-6	T 7.6-1 thru T 7.6-6
7.7-1/1a thru 7.7-41/42	7.7-1/2 thru 7.7-37/-
T 7.7-1	T 7.7-1/T 7.7-2
T 7.7-2	---
F 7.7-2 (Sh 1 of 35)	F 7.7-2 (Sh 1 of 35)
F 7.7-2 (Sh 6 of 35)	F 7.7-2 (Sh 6 of 35)
F 7.7-2 (Sh 33 of 35)	F 7.7-2 (Sh 33 of 35)
F 7.7-6 (Sh 3 of 7)	F 7.7-6 (Sh 3 of 7)
7A.1-1/2 thru 7A.1-15/16	7A.1-1/2 thru 7A.1-15/-
7B-1/-	7B-1/-
8-i/ii thru 8-vb/vi	8-i/ii thru 8-vii/-
8.1-1/2 thru 8.1-7/8	8.1-1/2 thru 8.1-7/-
F 8.1-1	F 8.1-1
8.2-1/1a thru 8.2-25/26	8.2-1/2 thru 8.2-29/-
F 8.2-1	F 8.2-1
F 8.2-1a	F 8.2-1a
F 8.2-1b	F 8.2-1b
F 8.2-4a	---
---	F 8.2-6d thru F 8.2-6u
F 8.2-9	F 8.2-9
8.3-1/2 thru 8.3-71/72	8.3-1/2 thru 8.3-83/-

INSERTION INSTRUCTIONS

VOLUME 17

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-
T 8.3-1 (Sh 3 of 32)	T 8.3-1 (Sh 3 of 32)
T 8.3-1 (Sh 4 of 32)	T 8.3-1 (Sh 4 of 32)
T 8.3-1 (Sh 17 of 32)	T 8.3-1 (Sh 17 of 32)
T 8.3-1 (Sh 20 of 32)	T 8.3-1 (Sh 20 of 32)
T 8.3-2 (Sh 16 of 31)	T 8.3-2 (Sh 16 of 31)
T 8.3-2 (Sh 17 of 31)	T 8.3-2 (Sh 17 of 31)
T 8.3-2 (Sh 20 of 31)	T 8.3-2 (Sh 20 of 31)
T 8.3-3A	T 8.3-3A
T 8.3-4 (Sh 1 thru 26 of 26)	T 8.3-4 (Sh 1 thru 25 of 25)
T 8.3-5 (Sh 1 thru 6 of 6)	T 8.3-5 (Sh 1 thru 5 of 5)
T 8.3-6 (Sh 1 thru 6 of 6)	T 8.3-6 (Sh 1 thru 6 of 6)
T 8.3-10 thru T 8.3-15	T 8.3-10 thru T 8.3-15
F 8.3-1	F 8.3-1
F 8.3-2	F 8.3-2
F 8.3-3 (Sh 1 of 2)	F 8.3-3 (Sh 1 of 2)
F 8.3-4	F 8.3-4
F 8.3-6 (Sh 5 of 31)	F 8.3-6 (Sh 5 of 31)
F 8.3-6 (Sh 27 of 31)	F 8.3-6 (Sh 27 of 31)
F 8.3-8B (Sh 12 of 13)	F 8.3-8B (Sh 12 of 13)
F 8.3-10	F 8.3-10
9-i/ii thru 9-xxi/xxii	9-i/ii thru 9-xxi/-
9.1-1/2 thru 9.1-45/46	9.1-1/2 thru 9.1-47/48
T 9.1-1 thru T 9.1-3	T 9.1-1 thru T 9.1-3
T 9.1-5	T 9.1-5/T 9.1-6
T 9.1-6	---

INSERTION INSTRUCTIONS

VOLUME 18

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-
9.2-1/2 thru 9.2-51/-	9.2-1/2 thru 9.2-47/48
T 9.2-1B (Sh 1 thru 5 of 5)	T 9.2-1B (Sh 1 thru 3 of 3)
T 9.2-3 (Sh 1 and 2 of 2)	T 9.2-3
F 9.2-1c	F 9.2-1c
F 9.2-1e thru F 9.2-1g	F 9.2-1e thru F 9.2-1g
F 9.2-1j	F 9.2-1j
F 9.2-1L	F 9.2-1L
F 9.2-1m	F 9.2-1m
F 9.2-1p	F 9.2-1p
F 9.2-3c	F 9.2-3c
F 9.2-3e	F 9.2-3e
F 9.2-5e	F 9.2-5e
F 9.2-6a	F 9.2-6a
F 9.2-8b	F 9.2-8b
F 9.2-9b	F 9.2-9b
F 9.2-17b	F 9.2-17b
F 9.2-19d	F 9.2-19d

INSERTION INSTRUCTIONS

VOLUME 19

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-
9.3-1/1a thru 9.3-35b/36	9.3-1/2 thru 9.3-39/40
T 9.3-1 (Sh 1 thru 10 of 10)	T 9.3-1 (Sh 1 thru 9 of 9)
T 9.3-2 (Sh 1 and 2 of 2)	T 9.3-2 (Sh 1 and 2 of 2)
T 9.3-3	T 9.3-3
F 9.3-1f	F 9.3-1f
F 9.3-5b thru F 9.3-5d	F 9.3-5b thru F 9.3-5d
F 9.3-5g	F 9.3-5g
F 9.3-7 (Sh 5 of 8)	F 9.3-7 (Sh 5 of 8)
F 9.3-9a	F 9.3-9a
F 9.3-9b	F 9.3-9b
F 9.3-12h	F 9.3-12h
F 9.3-12j	F 9.3-12j
F 9.3-12k	F 9.3-12k
F 9.3-16 (Sh 7 of 9)	F 9.3-16 (Sh 7 of 9)
F 9.3-17a	F 9.3-17a
F 9.3-20b	F 9.3-20b

INSERTION INSTRUCTIONS

VOLUME 20

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-
9.4-3/4	9.4-3/4
---	9.4-4a/4b
9.4-7/8	9.4-7/8
9.4-23/24	9.4-23/24
9.4-47/48	9.4-47/48
9.4-53/54	9.4-53/54
9.4-63/64	9.4-63/64
9.4-69/70	9.4-69/70
9.4-71/72	9.4-71/72
T 9.4-1 (Sh 4 of 5)	T 9.4-1 (Sh 4 of 5)
T 9.4-2 (Sh 1 thru 14 of 14)	T 9.4-2 (Sh 1 thru 14 of 14)
T 9.4-5 thru T 9.4-9	T 9.4-5 thru T 9.4-9
F 9.4-3d	F 9.4-3d
F 9.4-6 (Sh 6 of 7)	F 9.4-6 (Sh 6 of 7)

INSERTION INSTRUCTIONS

VOLUME 21

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-
F 9.4-9 (Sh 1 thru 3 of 25)	F 9.4-9 (Sh 1 thru 3 of 25)
F 9.4-10a	F 9.4-10a
F 9.4-10e	F 9.4-10e
F 9.4-22a thru F 9.4-22e	F 9.4-22a thru F 9.4-22e
9.5-1/1a thru 9.5-67b/68	9.5-1/2 thru 9.5-87/88
T 9.5-3 (Sh 1 thru 11 of 11)	T 9.5-3 (Sh 1 thru 12 of 12)
T 9.5-3a	T 9.5-3a

INSERTION INSTRUCTIONS

VOLUME 22

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-
F 9.5-1b	F 9.5-1b
F 9.5-1g	F 9.5-1g
F 9.5-3c	F 9.5-3c
F 9.5-8 (Sh 1 and 2 of 2)	F 9.5-8 (Sh 1 and 2 of 2)
F 9.5-10 (Sh 1 of 2)	F 9.5-10 (Sh 1 of 2)
F 9.5-24	F 9.5-24
F 9.5-29	F 9.5-29
F 9.5-42	F 9.5-42
F 9.5-52a	F 9.5-52a
F 9.5-52c	F 9.5-52c

INSERTION INSTRUCTIONS

VOLUME 23

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-
9A-i/ii thru 9A-vii/viii	9A-i/ii thru 9A-vii/-
9A.1-1/-	9A.1-1/-
9A.2-1/1a thru 9A.2-5/6	9A.2-1/2 thru 9A.2-5/-
9A.3-1/1a thru 9A.3-59/60	9A.3-1/2 thru 9A.3-65/-
T 9A.3-7 (Sh 1 thru 4 of 4)	T 9A.3-7 (Sh 1 thru 3 of 3)
T 9A.3-15 (Sh 3/4)	T 9A.3-15 (Sh 3/4)
T 9A.3-15 (Sh 5/6)	T 9A.3-15 (Sh 5/6)
T 9A.3-16	T 9A.3-16
T 9A.3-18 (Sh 2 of 6)	T 9A.3-18 (Sh 2 of 6)
T 9A.3-19	T 9A.3-19
T 9A.3-20 (Sh 1 and 2 of 2)	T 9A.3-20 (Sh 1 and 2 of 2)
F 9A.3-1	F 9A.3-1
F 9A.3-5	F 9A.3-5
9C-i/ii thru 9C-iii/-	9C-i/ii thru 9C-iii/-
9C.1-1/-	9C.1-1/-
9C.2-1/-	9C.2-1/-
9C.3-1/2 thru 9C.3-9/10	9C.3-1/2 thru 9C.3-9/10
T 3-1	T 3-1
T 3-2 (Sh 3 of 3)	T 3-2 (Sh 3 of 3)
T 3-3 (Sh 1 of 3)	T 3-3 (Sh 1 of 3)
T 3-4	T 3-4
T 3-5	T 3-5
9C.4-1/-	9C.4-1/-
T 4-1 (Sh 2 of 4)	T 4-1 (Sh 2 of 4)
9C.5-1/-	9C.5-1/-
F 5-1	F 5-1
F 5-2	F 5-2
9C.6-1/-	9C.6-1/-
9C.7-1/-	9C.7-1/-
9C.8-1/2 thru 9C.8-5/-	9C.8-1/2 thru 9C.8-5/-
9C.9-1/-	9C.9-1/-
9C.10-1/2 thru 9C.10-5/-	9C.10-1/2

INSERTION INSTRUCTIONS

VOLUME 23 (Cont'd.)

REMOVE	INSERT
10-i/ii thru 10-v/vi	10-i/ii thru 10-v/vi
10.1-1/2	10.1-1/2
10.1-3/-	10.1-3/-
T 10.1-1	T 10.1-1
F 10.1-2	F 10.1-2
F 10.1-5b	F 10.1-5b
F 10.1-6b	F 10.1-6b
F 10.1-6c	F 10.1-6c
F 10.1-8b	F 10.1-8b
F 10.1-9a thru F 10.1-9c	F 10.1-9a thru F 10.1-9c
F 10.1-9e	F 10.1-9e
10.2-1/2 thru 10.2-13/14	10.2-1/2 thru 10.2-13/14
T 10.2-1 (Sh 1 thru 3 of 3)	T 10.2-1 (Sh 1 thru 3 of 3)
F 10.2-3 (Sh 1 of 3)	F 10.2-3 (Sh 1 of 3)

INSERTION INSTRUCTIONS

VOLUME 24

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-
10.3-1/2	10.3-1/2
10.4-1/2	10.4-1/2
10.4-11/12	10.4-11/12
10.4-13/14	10.4-13/14
10.4-21/22 thru 10.4-29/30	10.4-21/22 thru 10.4-29/30
10.4-33/34	10.4-33/34
F 10.4-1 (Sh 2 of 5)	F 10.4-1 (Sh 2 of 5)
F 10.4-2a	F 10.4-2a
F 10.4-7d	F 10.4-7d
F 10.4-9 (Sh 7 thru 10 of 10)	F 10.4-9 (Sh 7 thru 10 of 10)
11-i/ii	11-i/ii
11-v/vi	11-v/vi
11.1-1/2	11.1-1/2
11.1-13/-	11.1-13/-
T 11.1-1 thru T 11.1-8	T 11.1-1 thru T 11.1-8
11.2-15/11.2-16	11.2-15/11.2-16
T 11.2-2	T 11.2-2
T 11.2-5 (Sh 1/2)	T 11.2-5 (Sh 1/2)
T 11.2-6 (Sh 1 and 2 of 2)	T 11.2-6 (Sh 1/2)
F 11.2-1d	F 11.2-1d
F 11.2-1g	F 11.2-1g

INSERTION INSTRUCTIONS

VOLUME 25

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-
11.3-9/-	11.3-9/-
T 11.3-1 thru T 11.3-3	T 11.3-1 thru T 11.3-3
F 11.3-1a	F 11.3-1a
F 11.3-2	---
11.4-1/2 thru 11.4-9/-	11.4-1/2 thru 11.4-7/-
T 11.4-2	T 11.4-2
T 11.4-4 thru T 11.4-7	T 11.4-4 thru T 11.4-7
11.5-11/12	11.5-11/12
11.5-13/-	11.5-13/-
T 11.5-2 (Sh 1 thru 4 of 4)	T 11.5-2 (Sh 1 thru 4 of 4)
11A-i/ii	11A-i/ii
11A.1-1/2	11A.1-1/2
T 11A.1-1 thru T 11A.1-19	T 11A.1-1 thru T 11A.1-19
11A.2-1/2	11A.2-1/2
T 11A.2-1	T 11A.2-1/T 11A.2-2
T 11A.2-2	---
12-i/ii thru 12-xi/xii	12-i/ii thru 12-xi/xii
12.2-1/1a thru 12.2-15/16	12.2-1/2 thru 12.2-13/-
T 12.2-1 thru T 12.2-6	T 12.2-1 thru T 12.2-6
T 12.2-8 thru T 12.2-11	T 12.2-8 thru T 12.2-11
T 12.2-13 (Sh 1 thru 3 of 3)	T 12.2-13 (Sh 1 and 2 of 2)
T 12.2-14	T 12.2-14
T 12.2-16	T 12.2-16
12.3-29/30	12.3-29/30
T 12.3-3 (Sh 1 and 2 of 2)	T 12.3-3 (Sh 1 and 2 of 2)
T 12.3-4 (Sh 1 thru 4 of 4)	T 12.3-4 (Sh 1 and 2 of 2)
F 12.3-7	F 12.3-7
F 12.3-40	F 12.3-40

INSERTION INSTRUCTIONS

VOLUME 26

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-
12.4-1/1a thru 12.4-3/-	12.4-1/2 thru 12.4-5/6
T 12.4-1 thru T 12.4-6	T 12.4-1 thru T 12.4-6
T 12.4-12	T 12.4-12
T 12.4-13	T 12.4-13
T 12.5-3	T 12.5-3
T 12.5-4	T 12.5-4
13-i/ii	13-i/ii
13-vii/-	13-vii/-
13.1-3/4 thru 13.1-7/8	13.1-3/4 thru 13.1-7b/8
13.1-15/-	13.1-15/-
F 13.1-2 (Sh 1 and 2 of 2)	F 13.1-2
F 13.1-3	F 13.1-3
F 13.1-5	F 13.1-5
13.2-7/8	13.2-7/8
---	13.2-8a/8b
13.3-1/2	13.3-1/2
T 13.5-3 thru T 13.5-4	T 13.5-3 thru T 13.5-4
13.6-1/-	13.6-1/-
14-i/ii thru 14-xib/xii	14-i/ii thru 14-xi/xii
14.1-1/-	14.1-1/-
14.2-1/2 thru 14.2-37/-	14.2-1/2 thru 14.2-33/-
T 14.2-1 (Sh 1 thru 6 of 6)	T 14.2-1
T 14.2-2 thru T 14.2-24	T 14.2-2
T 14.2-25 thru T 14.2-28	T 14.2-25 thru T 14.2-28
T 14.2-31 thru T 14.2-35	T 14.2-31 thru T 14.2-35

INSERTION INSTRUCTIONS

VOLUME 26 (Cont'd.)

REMOVE	INSERT
T 14.2-37 (Sh 1/2)	T 14.2-37
T 14.2-40 thru T 14.2-42	T 14.2-40 thru T 14.2-42
T 14.2-43A (Sh 1/2)	T 14.2-43A (Sh 1/2)
T 14.2-44 thru T 14.2-48	T 14.2-44 thru T 14.2-48
T 14.2-53 thru T 14.2-55	T 14.2-53 thru T 14.2-55
T 14.2-59 (Sh 1/2)	T 14.2-59 (Sh 1/2)
T 14.2-60 (Sh 1/2)	T 14.2-60
T 14.2-64 thru T 14.2-66	T 14.2-64 thru T 14.2-66
T 14.2-69 (Sh 1/2)	T 14.2-69 (Sh 1/2)
T 14.2-70 (Sh 1/2)	T 14.2-70
T 14.2-73 thru T 14.2-76	T 14.2-73 thru T 14.2-76
T 14.2-78 thru T 14.2-81	T 14.2-78 thru T 14.2-81
T 14.2-83 thru T 14.2-94	T 14.2-83
---	T 14.2-86 (Sh 1/2)
---	T 14.2-87
T 14.2-96 thru T 14.2-99	T 14.2-96
T 14.2-102 thru T 14.2-106	T 14.2-102 thru T 14.2-106
T 14.2-111	T 14.2-111
T 14.2-113 (Sh 1/2)	T 14.2-113
T 14.2-115 (Sh 1/2)	T 14.2-115
F 14.2-236-1	---
T 14.2-241a	---
T 14.2-241b	---

INSERTION INSTRUCTIONS

VOLUME 27

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-
---	14.3-1/-
15-i/ii thru 15-xxvii/-	15-i/ii thru 15-xxiii/-
15.0-1/1a thru 15.0-11d/12	15.0-1/2 thru 15.0-17/-
T 15.0-1 thru T 15.0-6	T 15.0-1 thru T 15.0-6
F 15.0-1	F 15.0-1
F 15.0-2	F 15.0-2
15.1-1/2 thru 15.1-17/-	15.1-1/2 thru 15.1-15/16
T 15.1-1 thru T 15.1-6	T 15.1-1 thru T 15.1-6
F 15.1-1 thru F 15.1-4	F 15.1-1 thru F 15.1-4
15.2-1/2 thru 15.2-33/34	15.2-1/2 thru 15.2-31/32
T 15.2-1 thru T 15.2-13	T 15.2-1 thru T 15.2-13
F 15.2-1 thru F 15.2-9	F 15.2-1 thru F 15.2-9
Notes for F 15.2-11 (Sh 1 and 2 of 2)	Notes for F 15.2-11 (Sh 1 and 2 of 2)
---	F 15.2-18 thru F 15.2-20
15.3-1/2 thru 15.3-13/14	15.3-1/2 thru 15.3-11/12
T 15.3-1 thru T 15.3-5	T 15.3-1 thru T 15.3-5
F 15.3-1 thru F 15.3-5	F 15.3-1 thru F 15.3-5
15.4-1/2 thru 15.4-21/-	15.4-1/2 thru 15.4-21/-
T 15.4-1 thru T 15.4-16	T 15.4-1 thru T 15.4-16
F 15.4-1 thru F 15.4-8	F 15.4-1 thru F 15.4-8
15.5-1/2 thru 15.5-3/-	15.5-1/2 thru 15.5-3/-
T 15.5-1	T 15.5-1
F 15.5-1	F 15.5-1
15.6-1/2 thru 15.6-17/18	15.6-1/2 thru 15.6-21/-
T 15.6-1 thru T 15.6-21	T 15.6-1 thru T 15.6-21
15.7-1/2 thru 15.7-17/18	15.7-1/2 thru 15.7-19/-

INSERTION INSTRUCTIONS

VOLUME 27 (Cont'd.)

REMOVE	INSERT
T 15.7-1 thru T 15.7-17	T 15.7-1 thru T 15.7-17
15.8-1/2 thru 15.8-5/-	15.8-1/2 thru 15.8-5/-

INSERTION INSTRUCTIONS

VOLUME 28

REMOVE	INSERT
i/ii	i/ii
iii/iv	iii/iv
v/-	v/-
15A-i/ii thru 15A-vii/viii	15A-i/ii thru 15A-vii/viii
15A-1/2 thru 15A-59/-	15A-1/2 thru 15A-49/50
15B-i/ii thru 15B-iii/-	15B-i/ii thru 15B-iii/-
15B.1-1/2	15B.1-1/2
15B.2-1/2 thru 15B.2-5/6	15B.2-1/2 thru 15B.2-3/-
---	F 15B.2-1
15B.3-1/2 thru 15B.3-19/-	15B.3-1 thru 15B.3-5/6
---	T 15B.3-1 thru T 15B.3-5
---	F 15B.3-1 thru F 15B.3-6
15B.4-1/2 thru 15B.4-3/-	15B.4-1/2
15B.5-1/2 thru 15B.5-3/-	15B.5-1/2 thru 15B.5-3/-
15B.6-1/2 thru 15B.6-3/-	15B.6-1/-
---	T 15B.6-1
15B.7-1/2	15B.7-1/2
15B.8-1/2	15B.8-1/2
15C-1/2	15C-1/2
15D-i/ii thru 15D-iii/-	15D-i/ii thru 15D-iii/-
15D-1/1a thru 15D-7/8	15D-1/2 thru 15D-9/-
15E-i/ii thru 15E-iii/-	15E-i/ii thru 15E-iii/-
15E.1-1/2	15E.1-1/2
15E.3-1/2 thru 15E.3-17/18	15E.3-1/2 thru 15E.3-7/8
---	F 15E.3-1 thru F 15E.3-5
15E.4-1/2 thru 15E.4-5/6	15E.4-1/2 thru 15E.4-3/-
---	T 15E.4-1
---	F 15E.4-1
15E.5-1/-	15E.5-1/-

INSERTION INSTRUCTIONS

VOLUME 28 (Cont'd.)

REMOVE	INSERT
15G-i/ii thru 15G-iii/-	15G-i/ii thru 15G-iii/-
15G-1/2 thru 15G-11/12 T 15G-11 (Sh 1 and 2 of 2) T 15G-12	15G-1/2 thru 15G-11/- T 15G-11 (Sh 1/2) T 15G-12
17-i/ii thru 17-v/-	17-i/ii thru 17-v/
17.0-1/2 thru 17.0-3/-	17.0-1/2
17.1-1/1a thru 17.1-53/-	17.1-1/2 thru 17.1-55/56
17.2-1/-	17.2-1/-
18-i/ia thru 18-iii/iv	18-i/ii thru 18-iii/iv
18.1-1/1a thru 18.1-1b/2	18.1-1/2 thru 18.1-3/-
18.2-1/2 thru 18.2-17/18	18.2-1/2 thru 18.2-15/16
A-i/ii thru A-v/vi	A-i/ii thru A-v/vi
A.0-1/2 thru A.0-3/4 F A.0-1	A.0-1/2 thru A.0-3/4 F A.0-1
A.4.1-1/-	A.4.1-1/-
A.4.2-1/-	A.4.2-1/-
A.4.3-1/-	A.4.3-1/-
A.4.4-1/2 thru A.4.4-3/- F A.4-1	A.4.4-1/2 thru A.4.4-3/- ---
A.5.2-1/2 thru A.5.2-3/4 T A.5.2-1 thru T A.5.2-2 T A.5.2-3 F A.5.2-1	A.5.2-1/2 thru A.5.2-3/4 T A.5.2-1 thru T A.5.2-2 --- ---

INSERTION INSTRUCTIONS

VOLUME 28 (Cont'd.)

REMOVE	INSERT
A.6-1/2	A.6-1/2
T A.6-2	T A.6-2
F A.6-1 thru F A.6-4	---
A.15.0-1/2 thru A.15.0-7/-	A.15.0-1/2 thru A.15.0-7/-
T A.15.0-1 thru T A.15.0-3	T A.15.0-1 thru T A.15.0-3
T A.15.0-4 (Sh 1 thru 3 of 3)	T A.15.0-4 (Sh 1 thru 3 of 3)
F A.15.0-1	---
F A.15.0-2	---
A.15.1-1/2 thru A.15.1-9/-	A.15.1-1/2 thru A.15.1-7/8
T A.15.1-1	---
T A.15.1-2	---
F A.15.1-1	---
F A.15.1-2	---
A.15.2-1/2 thru A.15.2-11/12	A.15.2-1/2 thru A.15.2-9/-
T A.15.2-1	---
T A.15.2-2	---
F A.15.2-1	---
F A.15.2-2	---
A.15.4-1/2 thru A.15.4-9/-	A.15.4-1/2 thru A.15.4-7/-
T A.15.4-1 thru T A.15.4-3	---
F A.15.4-1	---
F A.15.4-2	---
A.15B-1/-	A.15B-1/-
A.15C-1/-	A.15C-1/-
A.15D-1/-	A.15D-1/-
B.1-3/4	B.1-3/4
T B-1 (Sh 1 and 2 of 2)	T B-1 (Sh 1 and 2 of 2)

INSERTION INSTRUCTIONS

APPENDIX 6A

REMOVE	INSERT
6A-i/ii thru 6A-xxiii/-	6A-i/ii thru 6A-xxi/xxii
6A.1-1/2 thru 6A.1-5/-	6A.1-1/2 thru 6A.1-3/4
6A.2-1/2 thru 6A.2-13/14	6A.2-1/2 thru 6A.2-11/12
F 6A.2-21	F 6A.2-21
F 6A.2-25	F 6A.2-25
F 6A.2-27	---
6A.3-1/2 thru 6A.3-13/-	6A.3-1/2 thru 6A.3-13/-
T 6A.3-1/T 6A.3-2	T 6A.3-1/T 6A.3-2
6A.4-1/2 thru 6A.4-41/-	6A.4-1/2 thru 6A.4-41/-
T 6A.4-1 thru T 6A.4-3	T 6A.4-1 thru T 6A.4-3
T 6A.4-3a	---
F 6A.4-2 thru F 6A.4-5	F 6A.4-2 thru F 6A.4-5
F 6A.4-7	F 6A.4-7
6A.5-1/2 thru 6A.5-5/6	6A.5-1/2 thru 6A.5-5/6
6A.6-1/2 thru 6A.6-3/4	6A.6-1/2 thru 6A.6-3/-
F 6A.6-1 thru F 6A.6-7	---
6A.7-1/2 thru 6A.7-7/-	6A.7-1/2 thru 6A.7-5/6
F 6A.7-1	---
6A.8-1/2	6A.8-1/2
6A.9-1/2 thru 6A.9-11f/12	6A.9-1/2 thru 6A.9-15/-
T 6A.9-4 thru T 6A.9-6	T 6A.9-4 thru T 6A.9-6
6A.10-1/2 thru 6A.10-9b/10	6A.10-1/2 thru 6A.10-9/-
T 6A.10-1	T 6A.10-1
T 6A.10-2 (Sh 1 and 2 of 2)	T 6A.10-2 (Sh 1/2)
T 6A.10-5 thru T 6A.10-6	T 6A.10-5 thru T 6A.10-6
T 6A.10-8	---
T 6A.10-9	---
---	F 6A.10-6a

INSERTION INSTRUCTIONS

APPENDIX 9B

REMOVE	INSERT
9B-i/ia thru 9B-iii/iv	9B-i/ii thru 9B-iii/iv
9B.1-1/-	9B.1-1/-
9B.2-1/2 thru 9B.2-3/-	9B.2-1/2
9B.3-1/-	9B.3-1/-
9B.4-1/2 thru 9B.4-7/8	9B.4-1/2 thru 9B.4-7/-
9B.5-1/2 thru 9B.5-5/-	9B.5-1/2 thru 9B.5-3/4
9B.6-1/2 thru 9B.6-3/4	9B.6-1/2 thru 9B.6-3/-
F 9B.6-2 thru F 9B.6-6	---
9B.7-1/-	9B.7-1/-
9B.8-1/2 thru 9B.8-7/-	9B.8-1/2 thru 9B.8-7/-
T 9B.8-1 (Sh 1 thru 40 of 40)	T 9B.8-1 (Sh 1 thru 41 of 41)
T 9B.8-2 (Sh 1 thru 38b of 38)	T 9B.8-2 (Sh 1 thru 15 of 15)
9B.9-1/-	9B.9-1/-
9B.10-1/-	9B.10-1/-
9B.11-1/-	9B.11-1/-
9B.12-1/2 thru 9B.12-11/-	9B.12-1/2 thru 9B.12-9/-

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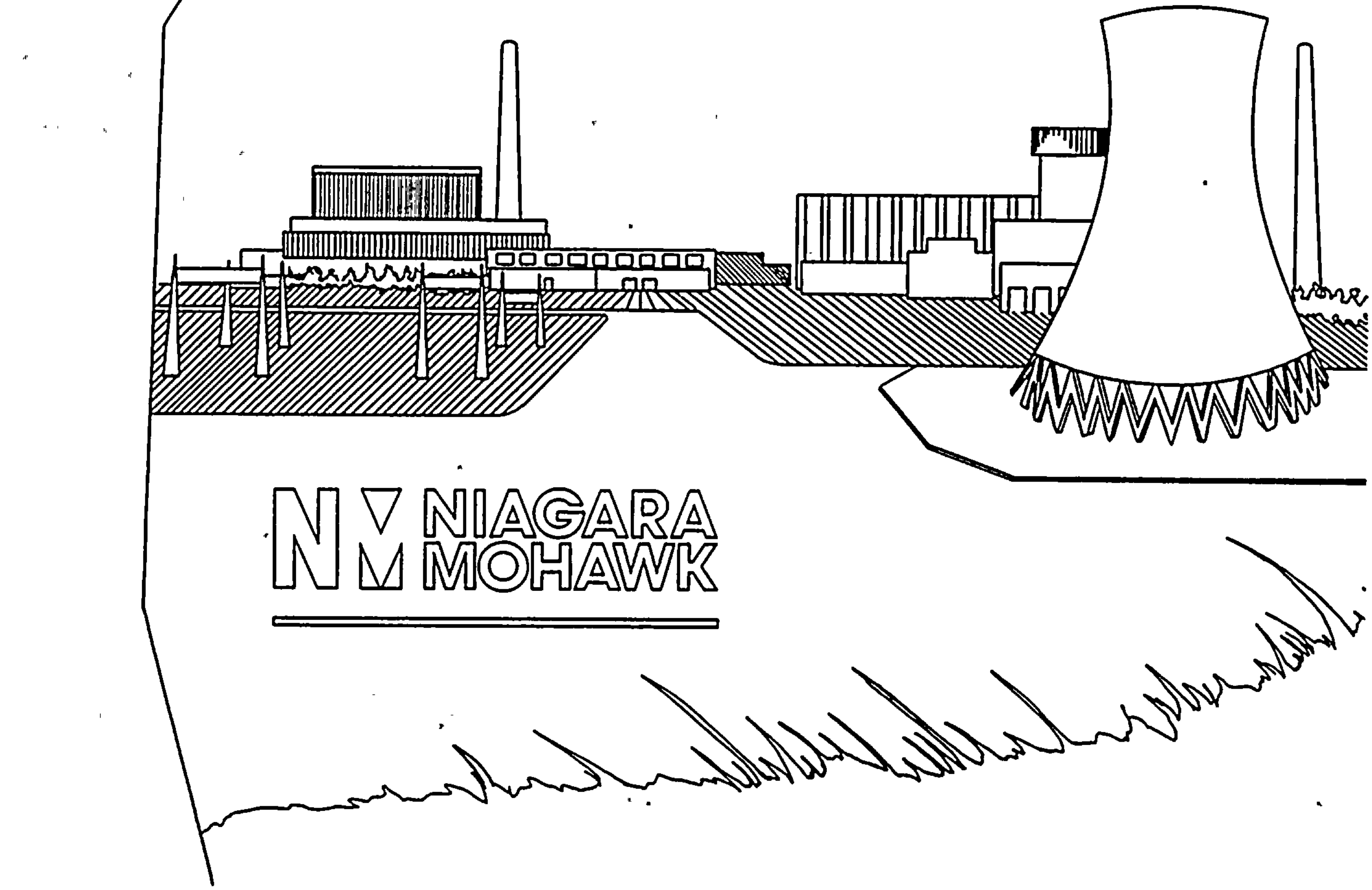

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F 1.7-4	R00	T 1.8-1 Sh 42	R06	T 1.8-1a Sh 10	R06
		T 1.8-1 Sh 43	R06	T 1.8-1a Sh 11	R06
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T 2.5-22 Sh 1	R08	T 2.5-49	A17	F 2.5-44	A00
T 2.5-22 Sh 2	R08	T 2.5-50	A22	F 2.5-45	A00
T 2.5-23	R08	F 2.5-1	A00	F 2.5-46	A00
T 2.5-24	R07	F 2.5-2	A00	F 2.5-47	A00
T 2.5-25	R08	F 2.5-3	A00	F 2.5-48	A00
T 2.5-26	R08	F 2.5-4	A00	F 2.5-49	A00
T 2.5-27	R08	F 2.5-5	A00	F 2.5-50	A00
T 2.5-28	R07	F 2.5-6	A00	F 2.5-51	A00

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F 2.5-54	A00	F 2.5-96D	A05	F 2.5-120 Sh 4	A00
F 2.5-55	A00	F 2.5-96E	A27	F 2.5-121 Sh 1	A00
F 2.5-56	A00	F 2.5-96F	A27	F 2.5-121 Sh 2	A00
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F 2.5-59	A00	F 2.5-99	A00	F 2.5-123	A00
F 2.5-60	A00	F 2.5-100	A00	F 2.5-124	A00
F 2.5-61	A00	F 2.5-101	A00	F 2.5-125	A00
F 2.5-62	A00	F 2.5-102	A00	F 2.5-126	A00
F 2.5-63	A00	F 2.5-102A	A05	F 2.5-127	A22
F 2.5-64	A00	F 2.5-102B	A05	F 2.5-127a	A09
F 2.5-65	A00	F 2.5-103	A00	F 2.5-127b	A09
F 2.5-66	A00	F 2.5-104	A00	F 2.5-128	A23
F 2.5-67	A00	F 2.5-105	A00	F 2.5-129	A05
F 2.5-68	A00	F 2.5-105A	A05	F 2.5-130	A05
F 2.5-69	A00	F 2.5-105B	A05	F 2.5-131	A05
F 2.5-70	A00	F 2.5-105C	A05	F 2.5-132	A05
F 2.5-71	A03	F 2.5-105D	A05	F 2.5-133	A05
F 2.5-72	A00	F 2.5-105E	A05	F 2.5-134	A05
F 2.5-73	A00	F 2.5-105F	A05	F 2.5-135	A11
F 2.5-74	A00	F 2.5-105G	A05	F 2.5-136	A12
F 2.5-75	A00	F 2.5-105H	A05	F 2.5-137	A12
F 2.5-76	A00	F 2.5-105I	A05	F 2.5-138	A12
F 2.5-77	A00	F 2.5-105J	A11	F 2.5-139	A12
F 2.5-78	A00	F 2.5-106	A00	F 2.5-140	A12
F 2.5-79	A00	F 2.5-107	A00	F 2.5-141	A12
F 2.5-80	A00	F 2.5-108	A00	F 2.5-142	A12
F 2.5-81	A00	F 2.5-109	A00	F 2.5-143	A12
F 2.5-82	A00	F 2.5-110	A05	F 2.5-144	A12
F 2.5-83	A00	F 2.5-111	A05	F 2.5-145	A12
F 2.5-84	A06	F 2.5-112	A00	F 2.5-146	A12
F 2.5-85	A00	F 2.5-113	A05	F 2.5-147	A12
F 2.5-86	A00	F 2.5-114	A00	F 2.5-148	A12
F 2.5-87	A00	F 2.5-115 Sh 1	A00	F 2.5-149	A12
F 2.5-88	A00	F 2.5-115 Sh 2	A00	F 2.5-150	A28
F 2.5-89	A00	F 2.5-116	A00	F 2.5-151	A13
F 2.5-90	A00	F 2.5-117 Sh 1	A00	F 2.5-152	A13
F 2.5-91	A00	F 2.5-117 Sh 2	A00	F 2.5-153	A13
F 2.5-92	A00	F 2.5-118	A00	F 2.5-154	A22
F 2.5-93	A00	F 2.5-119 Sh 1	A00	F 2.5-155	A22
F 2.5-94	A00	F 2.5-119 Sh 2	A00	F 2.5-156	A14
F 2.5-95	A00	F 2.5-119 Sh 3	A00	F 2.5-157	A14
F 2.5-96	A00	F 2.5-119 Sh 4	A00	F 2.5-158	A14
F 2.5-96A	A05	F 2.5-120 Sh 1	A00	F 2.5-159	A14

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F 2.5-172	A14	F 2.5-202	A14	T 2B-2A	A05
F 2.5-173	A14	F 2.5-203	A14	T 2B-2B	A05
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F 2.5-175	A14	F 2.5-205	A14	T 2B-3 Sh 2	A00
F 2.5-176	A14	F 2.5-206	A14	T 2B-3 Sh 3	A00
F 2.5-177	A14	F 2.5-207	A14	T 2B-3 Sh 4	A00
F 2.5-178	A14	F 2.5-208 Sh 1	A14	T 2B-3 Sh 5	A00
F 2.5-179	A14	F 2.5-208 Sh 2	A14	T 2B-3 Sh 6	A00
F 2.5-180	A14	F 2.5-208 Sh 3	A14	T 2B-3 Sh 7	A00
F 2.5-181	A14	F 2.5-208 Sh 4	A14	T 2B-3 Sh 8	A00
F 2.5-182	A14	F 2.5-208 Sh 5	A14	T 2B-3 Sh 9	A00
F 2.5-183	A14	F 2.5-208 Sh 6	A14	T 2B-4 Sh 1	A00
F 2.5-184	A14	F 2.5-208 Sh 7	A14	T 2B-4 Sh 2	A00
F 2.5-185	A14	F 2.5-208 Sh 8	A14	T 2B-4 Sh 3	A00
F 2.5-186	A14	F 2.5-208 Sh 9	A14	T 2B-4 Sh 4	A00
F 2.5-187	A14	F 2.5-208 Sh 10	A14	T 2B-4 Sh 5	A00
F 2.5-188	A27	F 2.5-209	A23	T 2B-4 Sh 6	A00
F 2.5-189	A14	F 2.5-210 Sh 1	A26	T 2B-4 Sh 7	A00
F 2.5-190	A14	F 2.5-210 Sh 2	A26	T 2B-4 Sh 8	A00
F 2.5-191	A14	F 2.5-210 Sh 3	A26	T 2B-4 Sh 9	A00
F 2.5-192 Sh 1	A14	F 2.5-210 Sh 4	A26	T 2B-5 Sh 1	A00
F 2.5-192 Sh 2	A14	F 2.5-211 Sh 1	A26	T 2B-5 Sh 2	A00
F 2.5-193 Sh 1	A14	F 2.5-211 Sh 2	A26	T 2B-5 Sh 3	A00
F 2.5-193 Sh 2	A14	F 2.5-212 Sh 1	A26	T 2B-5 Sh 4	A00
F 2.5-194	A14	F 2.5-212 Sh 2	A26	T 2B-5 Sh 5	A00
F 2.5-195	A14	F 2.5-212 Sh 3	A26	T 2B-5 Sh 6	A00
F 2.5-196	A14			T 2B-5 Sh 7	A00
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T 2B-38 Sh 12	A00	T 2B-38 Sh 58	A00	T 2B-38 Sh 104	A00
T 2B-38 Sh 13	A00	T 2B-38 Sh 59	A00	T 2B-38 Sh 105	A00
T 2B-38 Sh 14	A00	T 2B-38 Sh 60	A00	T 2B-38 Sh 106	A00
T 2B-38 Sh 15	A00	T 2B-38 Sh 61	A00	T 2B-38 Sh 107	A00
T 2B-38 Sh 16	A00	T 2B-38 Sh 62	A00	T 2B-38 Sh 108	A00
T 2B-38 Sh 17	A00	T 2B-38 Sh 63	A00	T 2B-39 Sh 1	A00
T 2B-38 Sh 18	A00	T 2B-38 Sh 64	A00	T 2B-39 Sh 2	A00
T 2B-38 Sh 19	A00	T 2B-38 Sh 65	A00	T 2B-39 Sh 3	A00
T 2B-38 Sh 20	A00	T 2B-38 Sh 66	A00	T 2B-39 Sh 4	A00
T 2B-38 Sh 21	A00	T 2B-38 Sh 67	A00	T 2B-39 Sh 5	A00
T 2B-38 Sh 22	A00	T 2B-38 Sh 68	A00	T 2B-39 Sh 6	A00
T 2B-38 Sh 23	A00	T 2B-38 Sh 69	A00	T 2B-39 Sh 7	A00
T 2B-38 Sh 24	A00	T 2B-38 Sh 70	A00	T 2B-39 Sh 8	A00
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T 2B-38 Sh 28	A00	T 2B-38 Sh 74	A00	T 2B-39 Sh 12	A00
T 2B-38 Sh 29	A00	T 2B-38 Sh 75	A00	T 2B-39 Sh 13	A00
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T 2B-38 Sh 33	A00	T 2B-38 Sh 79	A00	T 2B-39 Sh 17	A00
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T 2B-38 Sh 38	A00	T 2B-38 Sh 84	A00	T 2B-39 Sh 22	A00
T 2B-38 Sh 39	A00	T 2B-38 Sh 85	A00	T 2B-39 Sh 23	A00
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T 2B-38 Sh 43	A00	T 2B-38 Sh 89	A00	T 2B-39 Sh 27	A00
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T 2B-38 Sh 45	A00	T 2B-38 Sh 91	A00	T 2B-39 Sh 29	A00
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T 2B-38 Sh 47	A00	T 2B-38 Sh 93	A00	T 2B-39 Sh 31	A00
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T 2B-38 Sh 52	A00	T 2B-38 Sh 98	A00	T 2B-39 Sh 36	A00
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T 2B-39 Sh 43	A00	T 2B-39 Sh 89	A00	2C-4	A00
T 2B-39 Sh 44	A00	T 2B-39 Sh 90	A00	T 2C-1	A00
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T 2B-39 Sh 46	A00	T 2B-39 Sh 92	A00	F 2C-2	A00
T 2B-39 Sh 47	A00	T 2B-39 Sh 93	A00	F 2C-3	A00
T 2B-39 Sh 48	A00	T 2B-39 Sh 94	A00		
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T 2B-39 Sh 50	A00	T 2B-39 Sh 96	A00	2D-i	A00
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T 2B-39 Sh 52	A00	T 2B-39 Sh 98	A00	2D-2	A00
T 2B-39 Sh 53	A00	T 2B-39 Sh 99	A00	2D-3	A00
T 2B-39 Sh 54	A00	T 2B-39 Sh 100	A00	2D-4	A00
T 2B-39 Sh 55	A00	T 2B-39 Sh 101	A00	2D-5	A00
T 2B-39 Sh 56	A00	T 2B-39 Sh 102	A00	2D-6	A00
T 2B-39 Sh 57	A00	T 2B-39 Sh 103	A00	2D-7	A00
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T 2B-39 Sh 59	A00	T 2B-39 Sh 105	A00	2D-9	A00
T 2B-39 Sh 60	A00	T 2B-39 Sh 106	A00	F 2D-1	A00
T 2B-39 Sh 61	A00	T 2B-39 Sh 107	A00	F 2D-2	A00
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T 2B-39 Sh 63	A00	T 2B-40	A05		
T 2B-39 Sh 64	A00	T 2B-41	A00	App 2E	A00
T 2B-39 Sh 65	A00	T 2B-42	A00	2E-1	A00
T 2B-39 Sh 66	A00	T 2B-42A	A05	2E-2	A00
T 2B-39 Sh 67	A00	T 2B-43	A00	2E-3	A00
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T 2B-39 Sh 69	A00	T 2B-44 Sh 2	A00	App 2F	A00
T 2B-39 Sh 70	A00	T 2B-45	A00	2F-i	A23
T 2B-39 Sh 71	A00	T 2B-46	A00	2F-ia	A23
T 2B-39 Sh 72	A00	T 2B-47	A00	2F-ib	A23
T 2B-39 Sh 73	A00	T 2B-48 Sh 1	A00	2F-ii	A00
T 2B-39 Sh 74	A00	T 2B-48 Sh 2	A00	T 2F-1 Sh 1	A23
T 2B-39 Sh 75	A00	T 2B-49	A00	T 2F-1 Sh 2	A23
T 2B-39 Sh 76	A00	T 2B-50	A00	T 2F-1 Sh 3	A23
T 2B-39 Sh 77	A00	T 2B-51 Sh 1	A00	T 2F-2 Sh 1	A13
T 2B-39 Sh 78	A00	T 2B-51 Sh 2	A00	T 2F-2 Sh 2	A13
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T 2B-39 Sh 83	A00	T 2B-54 Sh 2	A03	T 2F-2c Sh 2	A23
T 2B-39 Sh 84	A00			T 2F-3	A23
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T 2F-6 Sh 3	A00	2H-ia	A09	F 2H-33	A00
T 2F-6 Sh 4	A23	2H-ib	A09	F 2H-34	A00
T 2F-6 Sh 5	A00	2H-ii	A00	F 2H-35	A00
T 2F-6 Sh 6	A23	2H-iii	A00	F 2H-36	A00
T 2F-6 Sh 7	A23	2H-iv	A00	F 2H-37	A00
T 2F-6 Sh 8	A23	2H-v	A00	F 2H-38	A00
T 2F-6 Sh 9	A23	2H-vi	A00	F 2H-39	A00
T 2F-6 Sh 10	A23	2H-vii	A09	F 2H-40	A00
T 2F-6 Sh 11	A23	2H-1	A00	F 2H-41	A00
T 2F-6 Sh 12	A23	2H Notes Sh 1	A00	F 2H-42	A00
T 2F-7 Sh 1	A24	2H Notes Sh 2	A00	F 2H-43	A00
T 2F-7 Sh 2	A00	2H Notes Sh 3	A00	F 2H-44	A00
T 2F-7 Sh 3	A00	F 2H-1	A00	F 2H-45	A00
T 2F-7 Sh 4	A00	F 2H-1A	A09	F 2H-46	A00
T 2F-7 Sh 5	A00	F 2H-2	A00	F 2H-47	A00
T 2F-7 Sh 6	A00	F 2H-3	A00	F 2H-48	A00
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T 2F-7 Sh 8	A00	F 2H-5	A00	F 2H-50	A00
T 2F-7 Sh 9	A00	F 2H-6	A00	F 2H-51	A00
T 2F-8	A23	F 2H-7	A00	F 2H-52	A00
T 2F-9	A00	F 2H-8	A00	F 2H-53	A00
T 2F-10	A00	F 2H-9	A00	F 2H-54 Sh 1	A00
T 2F-11 Sh 1	A00	F 2H-10	A00	F 2H-54 Sh 2	A00
T 2F-11 Sh 2	A00	F 2H-11	A00	F 2H-55	A00
T 2F-11 Sh 3	A23	F 2H-12	A00	F 2H-56	A00
T 2F-11 Sh 4	A23	F 2H-13	A00	F 2H-57	A00
T 2F-11 Sh 5	A23	F 2H-14	A00	F 2H-58	A00
T 2F-11 Sh 6	A23	F 2H-15	A00	F 2H-59	A00
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T 2F-11 Sh 8	A23	F 2H-17	A00	F 2H-61	A00
T 2F-11 Sh 9	A23	F 2H-18	A00	F 2H-62	A00
		F 2H-19	A00	F 2H-63	A00
App 2G	A00	F 2H-20	A00	F 2H-64	A00
2G-i	A23	F 2H-21	A00	F 2H-65	A00
T 2G-1	A00	F 2H-22	A00	F 2H-66	A00
T 2G-2	A00	F 2H-23	A00	F 2H-67	A00
T 2G-3	A00	F 2H-24	A00	F 2H-68	A00
T 2G-4	A00	F 2H-25	A00	F 2H-69	A00
T 2G-5	A00	F 2H-26	A00	F 2H-70	A00
T 2G-6	A23	F 2H-27	A00	F 2H-71	A00
T 2G-7	A23	F 2H-28	A00	F 2H-72	A00
T 2G-7a	A23	F 2H-29	A00	F 2H-73	A00
T 2G-8	A23	F 2H-30	A00	F 2H-74	A00
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F 2H-77	A00	2I-13	A00	2I-59	A00
F 2H-78	A00	2I-14	A00	2I-60	A00
F 2H-79	A00	2I-15	A00	2I-61	A19
F 2H-80	A00	2I-16	A00	2I-62	A00
F 2H-81	A00	2I-17	A00	2I-63	A00
F 2H-82	A00	2I-18	A00	2I-64	A00
F 2H-83	A00	2I-19	A00	2I-65	A00
F 2H-84	A00	2I-20	A00	2I-66	A00
F 2H-85	A00	2I-21	A00	2I-67	A00
F 2H-86	A00	2I-22	A00	2I-68	A00
F 2H-87 Sh 1	A00	2I-23	A00	2I-69	A00
F 2H-87 Sh 2	A00	2I-24	A00	2I-70	A00
F 2H-88	A00	2I-25	A00	2I-71	A00
F 2H-89	A00	2I-26	A00	2I-72	A00
F 2H-90	A00	2I-27	A00	2I-73	A00
F 2H-91	A00	2I-28	A00	2I-74	A00
F 2H-92	A00	2I-29	A00	2I-75	A00
F 2H-93	A00	2I-30	A00	2I-76	A00
F 2H-94	A00	2I-31	A00	2I-77	A00
F 2H-95	A00	2I-32	A00	2I-78	A00
F 2H-96	A09	2I-33	A00	2I-79	A00
F 2H-97	A09	2I-34	A00	2I-80	A00
F 2H-98	A09	2I-35	A00	2I-81	A00
F 2H-99	A09	2I-36	A00	2I-82	A00
F 2H-100	A09	2I-37	A00	2I-83	A00
F 2H-101	A09	2I-38	A00	2I-84	A00
		2I-39	A00	2I-85	A00
App 2I	A00	2I-40	A00	2I-86	A00
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2I-i	A00	2I-42	A00	2I-88	A00
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2I-iii	A00	2I-44	A00	2I-90	A00
2I-iv	A00	2I-45	A00	2I-91	A00
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2I-6	A00	2I-52	A00	2I-98	A00
2I-7	A00	2I-53	A00	2I-99	A00
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2I-106	A00	2J-5	A00	F 2K-12F	A00
2I-107	A00	2J-6	A00	F 2K-13A	A00
2I-108	A00	2J-7	A00	F 2K-13B	A00
2I-109	A00	2J-8	A00	F 2K-13C	A00
2I-110	A00	2J-9	A03	F 2K-13D	A00
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2I-114	A00			F 2K-14B	A00
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2I-117	A00	2K-ii	A00	F 2K-16A	A00
2I-118	A00	2K-iii	A00	F 2K-16B	A00
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2I-136	A00	F 2K-7C	A00	F 2K-22C	A00
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2I-139	A00	F 2K-8C	A00	F 2K-23C	A00
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2I-142	A00	F 2K-9C	A00	F 2K-24C	A00
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		F 2K-11A	A00	F 2K-24F	A00
App 2J	A00	F 2K-11B	A00	F 2K-25A	A00
2J-i	A00	F 2K-12A	A00	F 2K-25B	A00
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F 2K-26B	A00	F 2K-34C	A00	F 2K-47C	A00
F 2K-26C	A00	F 2K-35A	A00	F 2K-47D	A00
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F 2K-27A	A00	F 2K-36B	A00	2L-ii	A07
F 2K-27B	A00	F 2K-36C	A00	2L-iii	A00
F 2K-27C	A00	F 2K-37A	A00	2L-1	A07
F 2K-27D	A00	F 2K-37B	A00	2L-2	A07
F 2K-27E	A00	F 2K-37C	A00	T 2L-1 Sh 1	A07
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F 2K-28C	A00	F 2K-38A	A00	T 2L-3 Sh 1	A07
F 2K-28D	A00	F 2K-38B	A00	T 2L-3 Sh 2	A07
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F 2K-29C	A00	F 2K-39A	A00	T 2L-6 Sh 1	A07
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F 2K-30B	A00	F 2K-39F	A00	T 2L-8 Sh 2	A07
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F 2K-30F	A00	F 2K-41A	A00	T 2L-11	A07
F 2K-31A	A00	F 2K-41B	A00	T 2L-12	A07
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A.15.0-2	R08
A.15.0-3	R08
A.15.0-4	R08
A.15.0-5	R08
A.15.0-6	R08
A.15.0-7	R08
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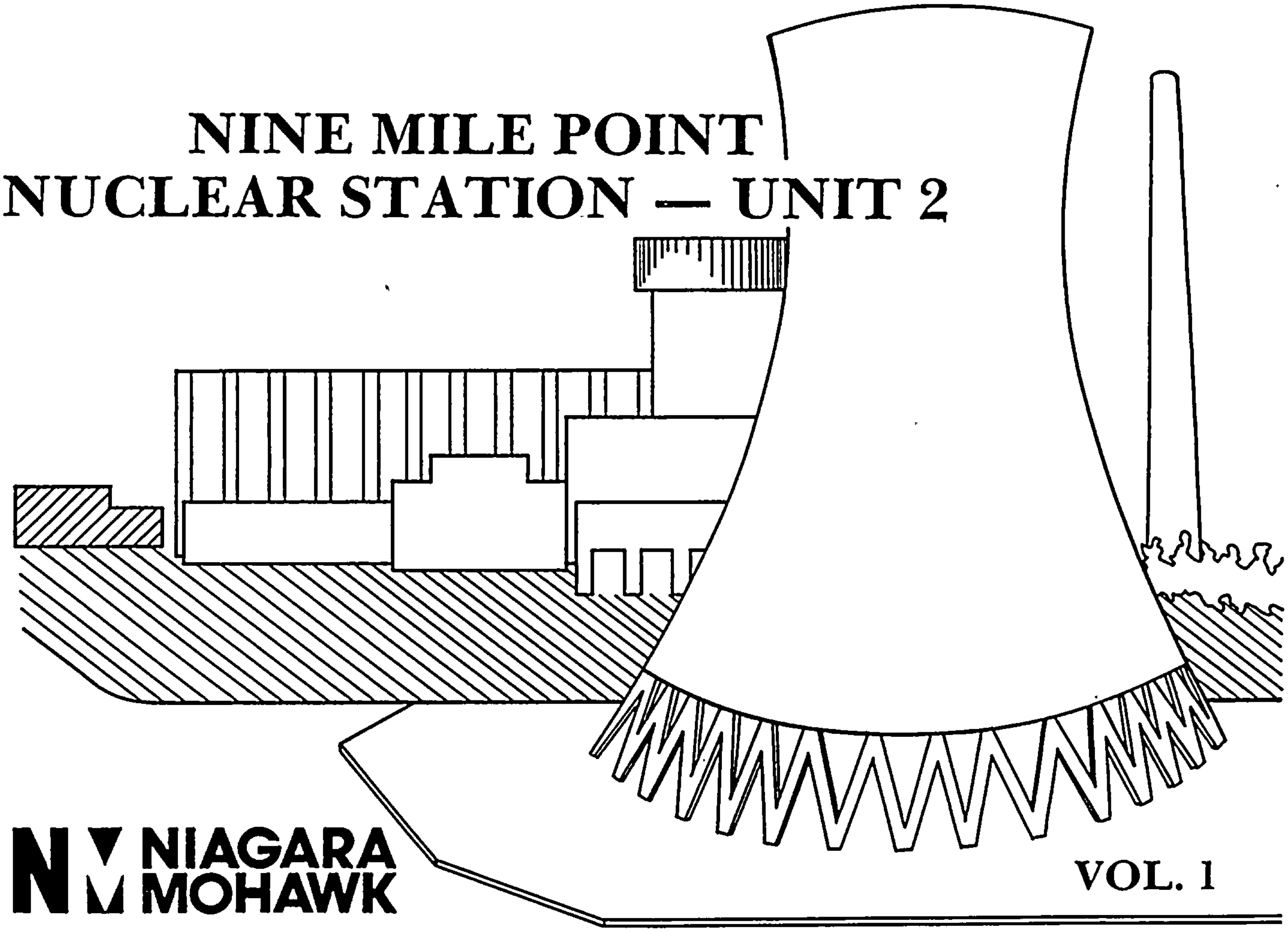

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INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

1.1 INTRODUCTION

This Final Safety Analysis Report (FSAR) is submitted by the Niagara Mohawk Power Corporation (Applicant) and its co-owners (Central Hudson Gas and Electric Corporation, Long Island Lighting Company, New York State Electric and Gas Corporation, and Rochester Gas and Electric Corporation) in support of the application for a Class 103 operating license for the nuclear power station designated Nine Mile Point Nuclear Station - Unit 2 (Unit 2).

Unit 2 is located on a 364-ha (900-acre) site owned by Niagara Mohawk Power Corporation (NMPC), and is situated on the southeast shore of Lake Ontario, Oswego County, NY, approximately 10 km (6.2 mi) northeast of the city of Oswego. Unit 2 and support facilities occupy about 18.2 ha (45 acres), and share the site with the existing Nine Mile Point Nuclear Station - Unit 1 (Unit 1) (Docket No. 50-220) which has been in commercial operation since 1969. The Nine Mile Point site is adjacent to the James A. FitzPatrick Nuclear Power Plant owned by the New York Power Authority (NYPA); Unit 2 is located 274 m (900 ft) east of Unit 1 and about 716 m (2,350 ft) west of the James A. FitzPatrick plant.

Unit 2 employs a nuclear steam supply system (NSSS) consisting of a single-cycle, forced circulating boiling water reactor (BWR). The plant-rated core thermal power level (Figure 1.1-1) is 3,467 MWt corresponding to a net electrical output of 1,144 MWe, and design thermal power of 3,536 MWt corresponding to a gross electrical output of 1,207 MWe. The thermal power used for the plant transient and loss-of-coolant accident (LOCA) analyses is 3,536 MWt. All safety systems have been designed for a thermal power of 3,536 MWt. The NSSS supplier is General Electric Company-Nuclear Energy Operations (GE-NEO). The balance of the plant is designed and constructed by Stone & Webster Engineering Corporation (SWEC). Other plants designed by SWEC that are similar in concept are currently under review by the Nuclear Regulatory Commission (NRC). These are the Shoreham Nuclear Power Station, Brookhaven, Long Island, NY, and the River Bend Station, St. Francisville, LA.

The containment design employs the BWR Mark II concept of over-under pressure suppression with multiple downcomers connecting the reactor drywell to the water-filled pressure suppression chamber. The primary containment is a steel-lined, reinforced concrete enclosure housing the reactor and the suppression pool.

The reactor building completely encloses the primary containment. The structure provides secondary containment when the primary containment is closed and in service, and provides primary containment when the primary containment is open, as during refueling. The reactor building houses the refueling and reactor servicing equipment, new and spent fuel storage facilities, and other reactor auxiliary and service equipment. The primary purpose of the reactor building is to minimize ground-level release of airborne radioactive material.

The outer wall of the reactor building is reinforced concrete up to the crane rail level above the refueling floor. Above the crane rail level, the superstructure is a steel frame using metal wall panels with sealed joints. Access to the building is through airlocks.

The power generation complex includes several contiguous buildings: the reactor building with two auxiliary bays, the control building, the turbine building, and the radwaste building. Other buildings, such as the security facility, are also located in the general plant area. A screenwell for the circulating and service water systems is located approximately 107 m (350 ft) northwest of the centerline of the reactor building.

Condenser cooling for Unit 2 is provided from a counterflow, natural-draft, hyperbolic, concrete cooling tower located approximately 330 m (1,000 ft) south of the centerline of the reactor building. The ultimate heat sink for emergency core cooling is Lake Ontario. Below grade and north of the screenwell building, there are two concrete tunnels that convey the service water intake, service water discharge, and cooling tower blowdown. A safety-related intake pipe is enclosed in each tunnel. The intake pipes extend from the intake shaft approximately 396 m (1,300 ft) northward under Lake Ontario to the submerged intake structures. One tunnel also contains the discharge pipe which extends approximately 550 m (1,800 ft) to the discharge diffuser.

Radionuclides are emitted to the atmosphere from two locations at Unit 2. These are the stack and the combined vent for the radwaste and reactor buildings. Liquid radwaste is stored for decay or concentrated to a solid waste for controlled disposal at regulated storage sites.

The shielding design and plant layout are based on extensive experience of NMPC and SWEC in controlling radiological exposures to as low as reasonably achievable (ALARA) levels. Estimated radiological doses for normal operations and postulated accidents are all fractional parts of the doses listed in federal radiological guidelines for siting and operation of nuclear power plants. Environmental impacts are described in the separate Environmental Report-Operating License Stage (ER-OLS) being submitted for Unit 2.

1.11 ABBREVIATIONS AND ACRONYMS

Table 1.11-1 is a list of abbreviations used in this FSAR.

TABLE 1.11-1

ABBREVIATIONS AND ACRONYMS USED IN FSAR

ADS	Automatic depressurization system
ALARA	As low as reasonably achievable
AOV	Air-operated valve
AP	Annulus pressurization
APRM	Average power range monitor
ARI	Alternate rod insertion
ARMS	Area radiation monitoring system
ATWS	Anticipated transient without scram
BCP	Bottom center pressure
BOC	Beginning of cycle
BSW	Biological shield wall
BTP	Branch technical position
BWR	Boiling water reactor
CAD	Containment atmosphere dilution (device)
CAM	Continuous air monitor
CCW	Closed cooling system
CGCS	Combustible gas control system
CHF	Critical heat flux
CIV	Combined intermediate valve
CIVM	Collision-imported-velocity method
CMFA	Common mode failure analysis
CND	Condensate demineralizer
CO	Condensation oscillation
CPR	Critical power ratio
CRD	Control rod drive
CRDA	Control rod drop accident
CRPI	Control rod position indication
CRVICS	Containment and reactor vessel isolation control system
CUF	Cumulative usage factor
CWS	Circulating water system
DAR	Design Assessment Report for Hydrodynamic Loads
DB	Design basis
DBA	Design basis accident
DBE	Design basis earthquake
DBFL	Design basis flood level
DCDT	Direct current differential transducer
DER	Double-end rupture
DG	Diesel generator
DRMS	Digital radiation monitoring system
EAB	Exclusion area boundary
ECA	Engineering change authorization
ECCS	Emergency core cooling system
ECN	Engineering change notice

TABLE 1.11-1 (Cont'd.)

EFCV	Excess flow check valve
EHC	Electrohydraulic control
EIC	Energy Information Center
EOC	End of cycle
EOF	Equivalent occurrence factor
EPA	Electric protective assembly
EPZ	Emergency planning zone
EQD	Environmental qualification document
ERF	Emergency response facility
ESF	Engineered safety feature
ETS	Emergency trip system
FA	Full arc (mode of TCV operation)
FAS	Fluid actuator system
FATT	Fracture appearance transition temperature
FCD	Functional control diagram
FDDR	Field deviation disposition request
FLECHT	Full-length emergency cooling heat transfer
FMEA	Failure modes and effects analysis
FMH	Fixture mounting height
FPCC	Fuel pool cooling and cleanup
FPS	Fire protection system
FSAR	Final safety analysis report
GDC	General design criterion
GE	General Electric Company
GETAB	GE thermal analysis basis
HAZ	Heat affected zone
HCU	Hydraulic control unit
HDFM	Heavy density fill material
HELB	High energy line break
HEM	Homogeneous equilibrium model
HEPA	High-efficiency particulate air/absolute (filter)
HEPCO	Hydro-Electric Power Commission of Ontario
HPCI	High pressure coolant injection
HPCS	High pressure core spray
HPU	Hydraulic power unit
HX	Heat exchanger
HVAC	Heating, ventilating, and air conditioning
HVRS	Reactor building ventilation system
IAC	Interim acceptance criteria (NCR)
IAS	Instrument air service
IBA	Intermediate break accident
ICC	Inadequate core cooling
IDC	Incident detection circuitry
IDS	Instrument data sheet

TABLE 1.11-1 (Cont'd.)

IED	Instrument and electrical drawing
IGSCC	Intergranular stress corrosion cracking
ILRT	Integrated leakage rate test
IPCEA	Insulated Power Cables Engineers Association
IRM	Intermediate range monitor
LCO	Limiting condition of operation
LCS	Leakage control system
LDS	Leak detection system
LFMG	Low frequency motor generator
LHGR	Linear heat generation rate
LOCA	Loss-of-coolant accident
LOFW	Loss of feedwater
LOOP (LOP)	Loss of offsite power
LPAP	Low power alarm point
LPCI	Low pressure coolant injection
LPCS	Low pressure core spray
LPDS	Loose parts detection system
LPRM	Local power range monitor
LPSP	Low power set point
LPZ	Low population zone
LSA	Low specific activity
LSD	Lake survey datum (of 1935)
LSSS	Limiting safety system setting
LTC	Load tap changing (mechanism)
LWS	Liquid radwaste system
MAPLHGR	Maximum average planar linear heat generation rate
MBA	Misplaced bundle accident
MCC	Motor control center
M/CC	Maintenance and calibration communication (system)
MCPR	Minimum critical power ratio
MG	Motor generator set
MLD	Mean low water datum
MLHGR	Maximum linear heat generation rate
MMI	Modified Mercalli intensity
MOI	Method of images
MOV	Motor operated valve
MPC	Maximum permissible concentration
MSIV	Main steam isolation valve
MSIV-LCS	Main steam isolation valve leakage control system
msl	Mean sea level
MSL	Main steam line
MSLB	Main steam line break
MTV	Mechanical trip valve

TABLE 1.11-1 (Cont'd.)

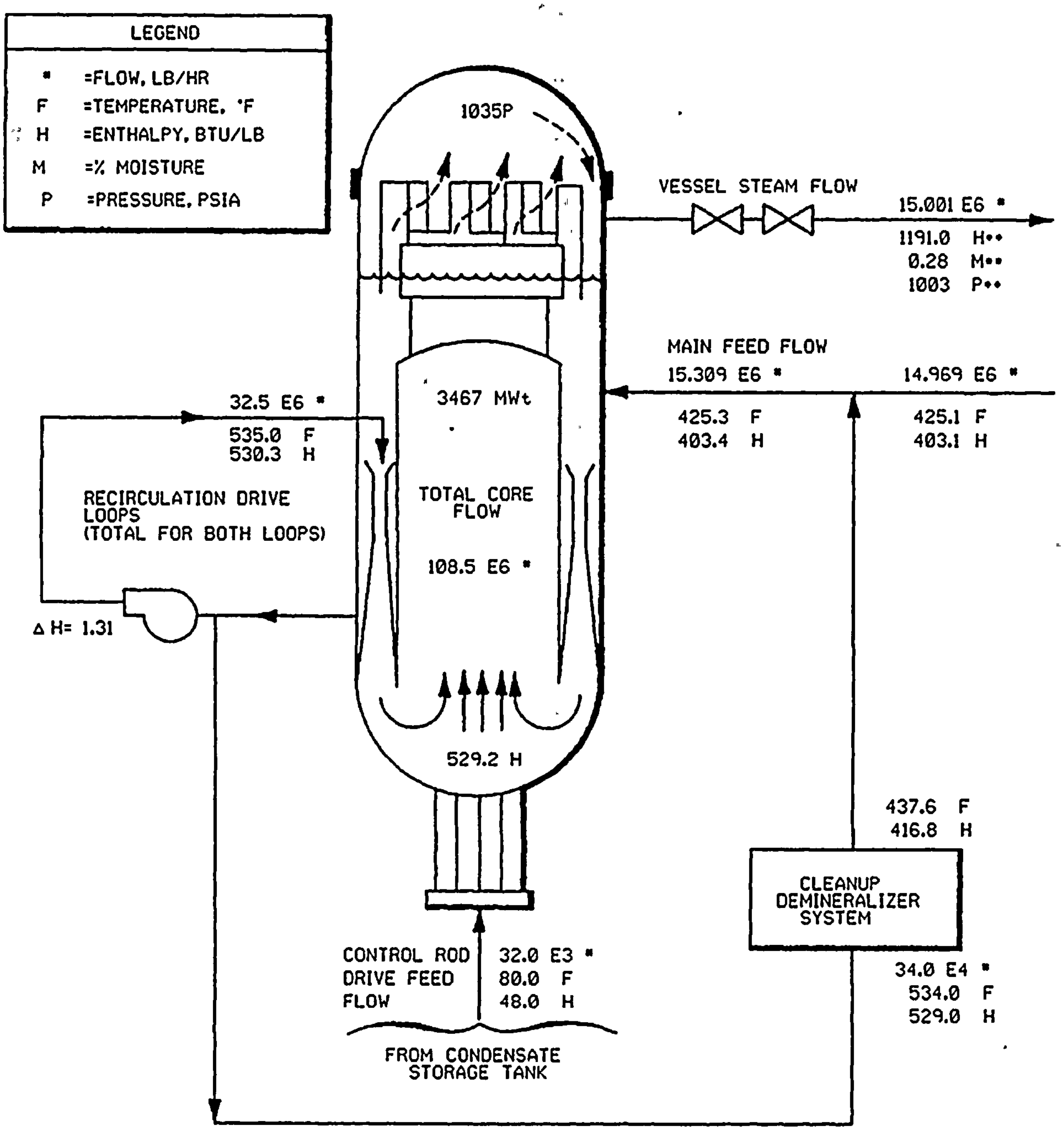
NB	Nuclear boiler
NBR	Nuclear boiler rated (power)
NBS	National Bureau of Standards
NDL	Nuclear data link
NDT	Nil ductility transition
NDTT	Nil ductility transition temperature
NED	Nuclear energy division (GE)
NIOSH	National Institute for Occupational Safety and Health
NMS	Neutron monitoring system
NPRDS	Nuclear plant reliability data system
NPSH	Net positive suction head
NRV	Nonreturn valve
NSS	Nonnuclear safety
NSOA	Nuclear safety operational analysis
NSSS	Nuclear steam supply system
NUMAC RWM	Nuclear measurement analysis and control rod worth minimizer
OBE	Operating basis earthquake
OFS	Orificed fuel support
ORE	Occupational radiation exposures
OT	Operational transient
PA	Public address (system)
PAM	Postaccident monitoring
PASNY	Power Authority of the State of New York
PCI	Pellet-cladding interaction
PCIOMR	Preconditioning cladding interim operating management recommendation
PCRVICS	Primary containment and reactor vessel isolation control system
PCS	Process computer system
PCT	Peak cladding temperature
p.f.	Power factor
PGCC	Power generating control center
P&ID	Piping and instrumentation diagram
PLU	Power load unbalance
PMF	Probable maximum flood
PMS	Probable maximum surge
PMWS	Probable maximum windstorm
PP/PA	Page party/public address (system)
PQL	Product quality checklist
PRM	Power range monitor
PSAR	Preliminary safety analysis report
PSD	Power spectrum density
PTPD	Project test program objectives
PVS	Plant vent stack
PWR	Pressurized water reactor

TABLE 1.11-1 (Cont'd.)

QA	Quality assurance
QC	Quality control
RAB	Restricted area boundary
RBCLCW	Reactor building closed loop cooling water (system)
RBM	Rod block monitor
RBPC	Reactor building polar crane
RCIC	Reactor core isolation cooling
RCPB	Reactor coolant pressure boundary
RCS	Reactor coolant system
RCSCM	RHR containment spray cooling mode
RDCS	Rod drive control system
RH	Relative humidity
RHR	Residual heat removal
RMCS	Reactor manual control system
RMS	Radiation monitoring system
RMS	Root mean square
RPC	Rod pattern controller
RPIS	Rod position information system
RPS	Reactor protection (trip) system
RPT	Recirculation pump trip
RPV	Reactor pressure vessel
RRCS	Redundant reactivity control system
RSCM	RHR reactor shutdown cooling mode
RSCS	Rod sequence control system
RSO	Reactor system outline
RSPCM	RHR suppression pool cooling mode
RSS	Remote shutdown system
RWCU	Reactor water cleanup
RWP	Radiation work permit
SACF	Single active component failure
SAR	Safety analysis report
SBA	Small break accident
SCA	Single-channel analyzer
SCBA	Self-contained breathing apparatus
SDIV	Scram discharge instrument volume
SDV	Scram discharge volume
SEF	Single equipment failure
SFC	Spent fuel pool cooling and cleanup system
SGTS	Standby gas treatment system
SLC	Standby liquid control
SMSA	Standard metropolitan statistical area
SOE	Single operator error
SOF	Single operator failure
SORC	Station Operations Review Committee
SPC	Sound-powered communication (system)
SPDS	Safety parameter display system

TABLE 1.11-1 (Cont'd.)

SPG	Substitute position generator
SRAB	Safety Review and Audit Board
SRDI	Safety-related display instrumentation
SRM	Source range monitor
SRM	Security-related materials
SRP	Standard Review Plan
SRSS	Square root of the sum of the squares
SRV	Safety/relief valve
SRVDL	Safety/relief valve discharge line
SS	Safe shutdown
SSE	Safe shutdown earthquake
SWP	Service water system
TBCLCW	Turbine building closed loop cooling water
TCV	Turbine control valve
TG	Turbine generator
TIP	Traversing in-core probe
TLD	Thermoluminescent dosimeter
TSS	Temperature sensor/switch
TSVC	Turbine stop valve closure
UHS	Ultimate heat sink
UPS	Uninterruptible power supply
ZPA	Zero period asymptote



• NOMINAL POWER UPRATE PLANT CONDITIONS

•• THESE REPRESENT CONDITIONS BEFORE THE TURBINE STOP VALVES.

SOURCE: NEDC-31994P

THIS DRAWING CREATED ELECTRONICALLY

FIGURE 1.1-1

HEAT BALANCE AT RATED POWER

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accidents that release radioactive material into the primary containment volume.

16. It is possible to test primary containment integrity and leak-tightness at periodic intervals.

17. A reactor building is provided that completely encloses both the primary containment and the fuel storage areas. The secondary containment includes a method for controlling release of radioactive materials from the barrier and includes a capability for filtering radioactive materials within the barrier.

18. The reactor building is designed to act as a radioactive material barrier, if required, when the primary containment is open for expected operational purposes.

19. The primary containment and reactor building, in conjunction with other engineered safeguards, limits radiological effects of accidents resulting in the release of radioactive material to the primary containment volume to significantly less than the requirements of 10CFR100.

20. Provisions are made for removing energy from within the primary containment to maintain the integrity of the primary containment system following accidents that release energy to the primary containment.

21. Piping that penetrates the primary containment structure and serves as a path for the uncontrolled release of radioactive material to the environs is automatically isolated whenever such potential for radioactive material release exists. Such isolation is effected in time to limit radiological effects to significantly less than the requirements of 10CFR100.

22. The emergency core cooling system (ECCS) is provided to limit fuel cladding temperature to 2,200°F as a result of a LOCA.

23. The ECCS provides for continuity of core cooling over the complete range of postulated break sizes in the RCPB.

24. The ECCS is diverse, reliable, and redundant.

25. Operation of the ECCS is initiated automatically when required, regardless of the availability of offsite power.

26. The main control room is shielded against radiation to permit continued occupancy under accident conditions.

27. In the event that the main control room becomes uninhabitable, it is possible to bring the reactor from power range operation to a cold shutdown condition by manipulating local controls and equipment available outside the main control room.

28. Backup reactor shutdown capability is provided independent of normal reactivity control provisions. This backup system shuts down the reactor from any normal operating condition and maintains the shutdown condition.

1.2.1.3 System-by-System Approach

The principal architectural and engineering criteria for design are summarized below on a system-by-system or system group basis. The system-by-system presentation facilitates understanding of the actual design of any one system. Only the most restrictive of any related criteria are stated for a system. Where the most restrictive criterion is classified as a power generation consideration, less restrictive safety criteria may not be stated in the system-by-system presentation. However, the actual design of a system must reflect all criteria that pertain to it.

1.2.1.3.1 Nuclear System Criteria

Principal design criteria for the reactor, ECCS, RCPB, and reactivity control systems are as follows:

1. The nuclear system is designed to support a GE BWR rated at 3,467 MWt.

2. Fuel cladding is designed to retain integrity as a radioactive material barrier throughout the design power range. Fuel cladding is designed to accommodate, without loss of integrity, the pressures generated by the fission gases released from fuel material throughout the design life of the fuel.

3. Fuel cladding, in conjunction with other unit systems, is designed to retain integrity throughout any abnormal operational transient.

4. Those portions of the nuclear system that form part of the RCPB are designed to retain integrity as a radioactive material barrier following abnormal operational transients and accidents.

5. Heat removal systems including the ECCS and makeup water supplies are provided in sufficient capacity, redundancy, and operational adequacy to remove heat generated in the reactor core for the full range of normal operational conditions from unit shutdown to design power and for any abnormal operational transient

The auxiliary boiler building (Figure 1.2-34), located north of the screenwell building, houses the electric boilers and accessories to supply steam to the plant during shutdown.

The standby gas treatment building and railroad access area (Figures 1.2-35 and 1.2-36) house the standby gas treatment filters and associated equipment and allow access for spent fuel shipping.

The condensate storage tank building (Figure 1.2-37) houses the condensate storage tanks and associated equipment.

The natural-draft cooling tower (Figures 1.2-38 and 1.2-39) provides the normal heat sink for heat transferred to the circulating water system from the main condensers.

The auxiliary service building (Figures 1.2-7 and 1.2-8), adjacent to the reactor building, houses the heating, ventilating and air conditioning (HVAC) room and decontamination and shower facilities for personnel.

The decontamination area (Figures 1.2-19 through 1.2-21, 1.2-23, and 1.2-24), immediately south of the radwaste building, provides the facility for decontamination of large tools and equipment, and a sample room. It also houses clean steam reboilers and related equipment.

The hydrogen storage area (for hydrogen cooling of the turbine generator, Figure 1.2-40) is located west of the offgas area. The hydrogen storage bottles are mounted on concrete pads and are in a fenced area.

1.2.4 Nuclear Steam Supply System

The nuclear system includes a direct-cycle, forced circulation, GE BWR that produces steam for direct use in the steam turbine. A heat balance showing the major parameters of the nuclear system for the warranted power condition is shown on Figure 1.1-1.

The NSSS is further discussed in Chapters 4 and 5.

1.2.4.1 Reactor Core and Control Rods

The reactor fuel and core design are described in Section 2 of Reference 5 and Section 1 of Reference 6.

Experience has shown that the control rods are not susceptible to distortion and have an average life expectancy many times the residence time of a fuel loading.

1.2.4.2 Reactor Vessel and Internals

The reactor vessel contains the core and supporting structures; the steam separators and dryers; the jet pumps; the control rod guide tubes; the distribution lines for the feedwater, core sprays, and standby liquid control; the in-core instrumentation; and other components. The main connections to the vessel include the steam lines, coolant recirculation lines, feedwater lines, CRD and in-core nuclear instrument housings, core spray lines, residual heat removal (RHR) lines, standby liquid control line, core differential pressure line, jet pump pressure-sensing lines, and water level instrumentation.

The reactor vessel is designed and fabricated in accordance with applicable codes for a pressure of 1,250 psig. The nominal operating pressure in the steam space above the separators is 1,035 psia. The vessel is fabricated of low-alloy steel and is clad internally with stainless steel (except for the top head nozzles and nozzle weld zones which are unclad).

The reactor core is cooled by demineralized water that enters the lower portion of the core and boils as it flows upward around the fuel rods. The steam leaving the core is dried by steam separators and dryers located in the upper portion of the RPV. The steam is then directed to the turbine through the main steam lines. Each steam line has two isolation valves in series, one on either side of the primary containment barrier.

1.2.4.3 Reactor Recirculation System

The reactor recirculation system consists of two recirculation pump loops external to the RPV. These loops provide the piping path for the driving flow of water to the RPV jet pumps. Each external loop contains one high-capacity motor-driven recirculation pump, two motor-operated maintenance valves, and one hydraulically-operated flow control valve. The variable position hydraulic flow control valve operates in conjunction with a low-frequency motor generator (MG) set to control reactor power level through the effects of coolant flow rate on moderator void content.

The jet pumps are RPV internals. They provide a continuous internal circulation path for the major portion of the core coolant flow. The jet pumps are located in the annular region between the core shroud and the vessel inner wall. Any recirculation line break still allows core flooding to approximately two-thirds of the core height, the level of the inlet of the jet pumps.

1.2.4.4 Residual Heat Removal System

The RHR system is a system of pumps, heat exchangers, and piping that fulfills the following functions:

1. Removes decay and sensible heat during and after plant shutdown.

water temperature, purity, clarity, and level. This process prevents the spent fuel from overheating and the buildup of excessive radioactive materials in the cooling water, thereby minimizing radiation levels.

The system includes two heat exchangers, each of which is capable of removing the full decay heat from a normal refueling offload of spent fuel. A cross-connection to the RHR system provides additional emergency backup cooling and cooling during a full core offload.

Chapter 9 gives further details of the fuel handling and storage system.

1.2.8 Power Conversion System

Chapter 10 provides a detailed discussion of the following equipment systems.

1.2.8.1 Turbine Generator

The turbine is a 1,800-rpm tandem-compound, six-flow, single-stage reheat unit with an electrohydraulic governor control. The turbine generator has an emergency trip system for turbine overspeed. The output of the turbine generator is 1,210.9 MWe at turbine guarantee conditions with 2.0 in Hg abs backpressure and 0 percent makeup.

The generator is a direct-driven, three-phase, 60-Hz, 25,000-V, 1,800-rpm hydrogen inner-cooled, synchronous generator rated at 1,348,400 kVA at 0.90 power factor, 0.58 short-circuit ratio at maximum hydrogen pressure of 75 psig.

1.2.8.2 Main Steam System

The main steam system delivers steam from the nuclear boiler system through four 26-/28-in OD steam lines to the turbine generator, turbine bypass valves, SJAEs, offgas preheaters, steam seal evaporator, and radwaste steam reboiler.

1.2.8.3 Main Condenser

The main condenser maintains 2.0 in Hg abs when operating at reactor warranty conditions with 66.0°F circulating water inlet temperature. The condenser includes provisions for accepting steam bypassed around the turbine generator. Deaeration of condensate is accomplished in the condenser.

1.2.8.4 Main Condenser Air Removal System

The main condenser air removal system, using air ejectors for normal operation and vacuum hogging pumps for startup, evacuates gases from the main turbine and condenser during plant startup and maintains the condenser essentially free of gases during operation. This system handles all in-leakage of noncondensable gases through the turbine seals, condensate, feedwater, and steam systems, and noncondensables that are generated in the reactor by disassociation of water.

1.2.8.5 Turbine Gland Sealing System

The turbine gland sealing system provides mildly radioactive steam to the seals of the turbine throttle valve stem glands and the turbine shaft glands. The sealing steam is supplied by a clean steam reboiler using condensate. The unit auxiliary boiler provides an auxiliary steam supply for startup and when reactor heating steam is not available. The steam packing exhauster collects and condenses the air and steam mixture and discharges the air and other noncondensables to the plant exhaust duct to the atmosphere, using a motor-driven exhauster.

1.2.8.6 Steam Bypass System and Pressure Control System

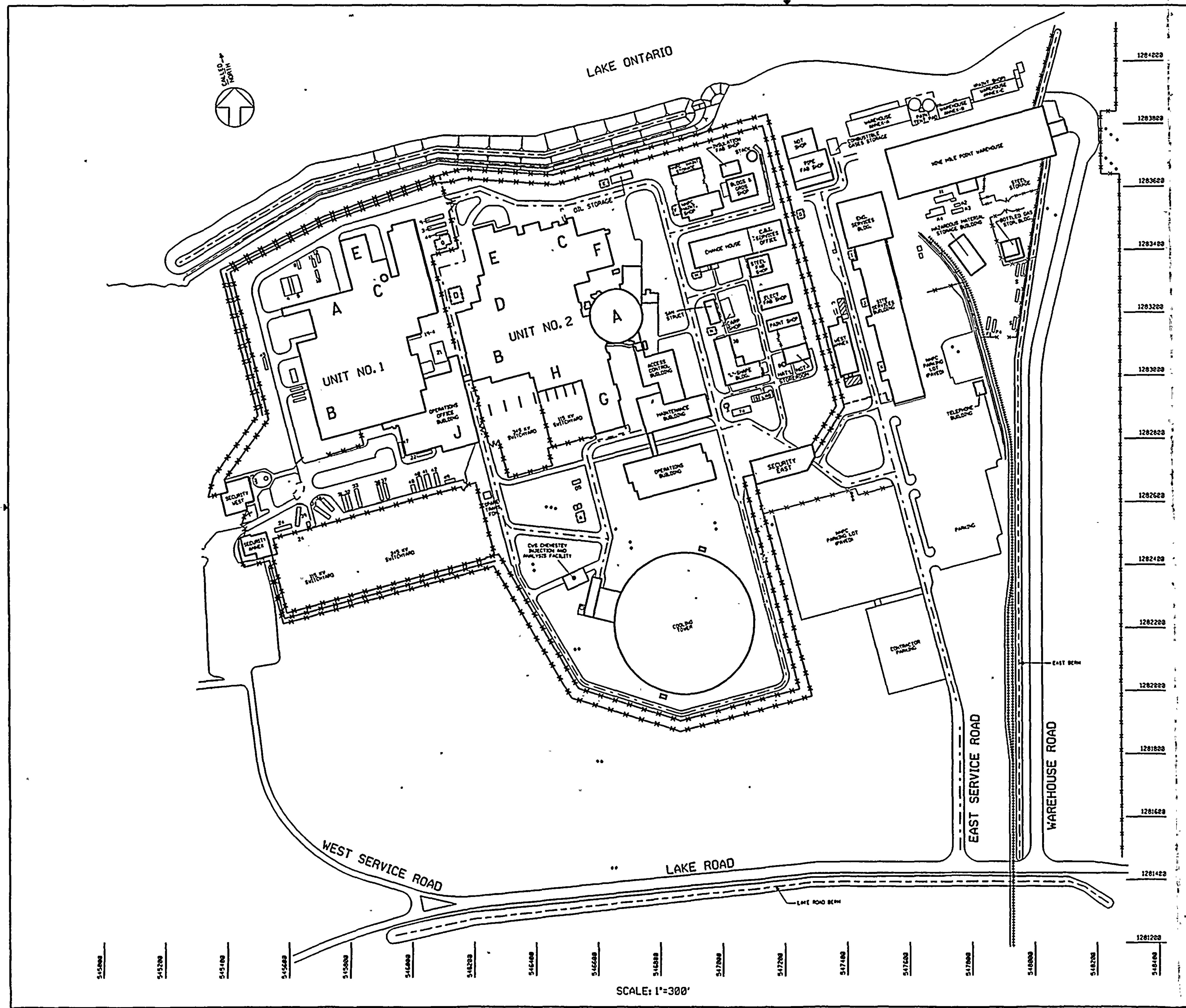
A turbine bypass system is provided which passes steam directly to the main condenser under control of the pressure regulator. Steam is bypassed to the condenser whenever the reactor steaming rate exceeds the load passed to the turbine generator. The capacity of the turbine bypass system is 25 percent of the turbine rated steam flow. The pressure regulation system provides main turbine control valve and bypass valve flow demands to maintain a nearly constant reactor pressure during normal plant operation. It also provides demands to the recirculation system to adjust power levels by changing reactor recirculation flow rates.

1.2.8.7 Circulating Water System

The circulating water system (CWS) provides the condenser with a continuous supply of cooling water. The CWS is a pumped closed loop system utilizing an air-cooled natural-draft cooling tower as a heat sink. Six one-sixth capacity circulating water pumps are provided to pump cooling water from the cooling tower basin through the main condenser and back to the top of the cooling tower. Makeup water is provided from Lake Ontario by the service water system.

1.2.8.8 Condensate and Feedwater Systems

The condensate and feedwater systems supply condensate from the condenser hotwell to the RPV. The condensate is pumped by two of the three condensate pumps through the full flow condensate demineralizer system, the intercooler of the air ejectors, and the steam packing exhauster to the condensate booster pumps. The condensate booster pumps pump the flow through three strings consisting of two drain coolers and five stages of low-pressure heaters each. In addition, three heater drain pumps provide approximately one-third of the feedwater flow requirements. The last low-pressure heaters discharge to the suction of three



IDENTIFICATION LEGEND

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

KEY

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

SOURCE: EY-8S

FIGURE 1.2-1

PLOT PLAN

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NINE MILE POINT-UNIT 2

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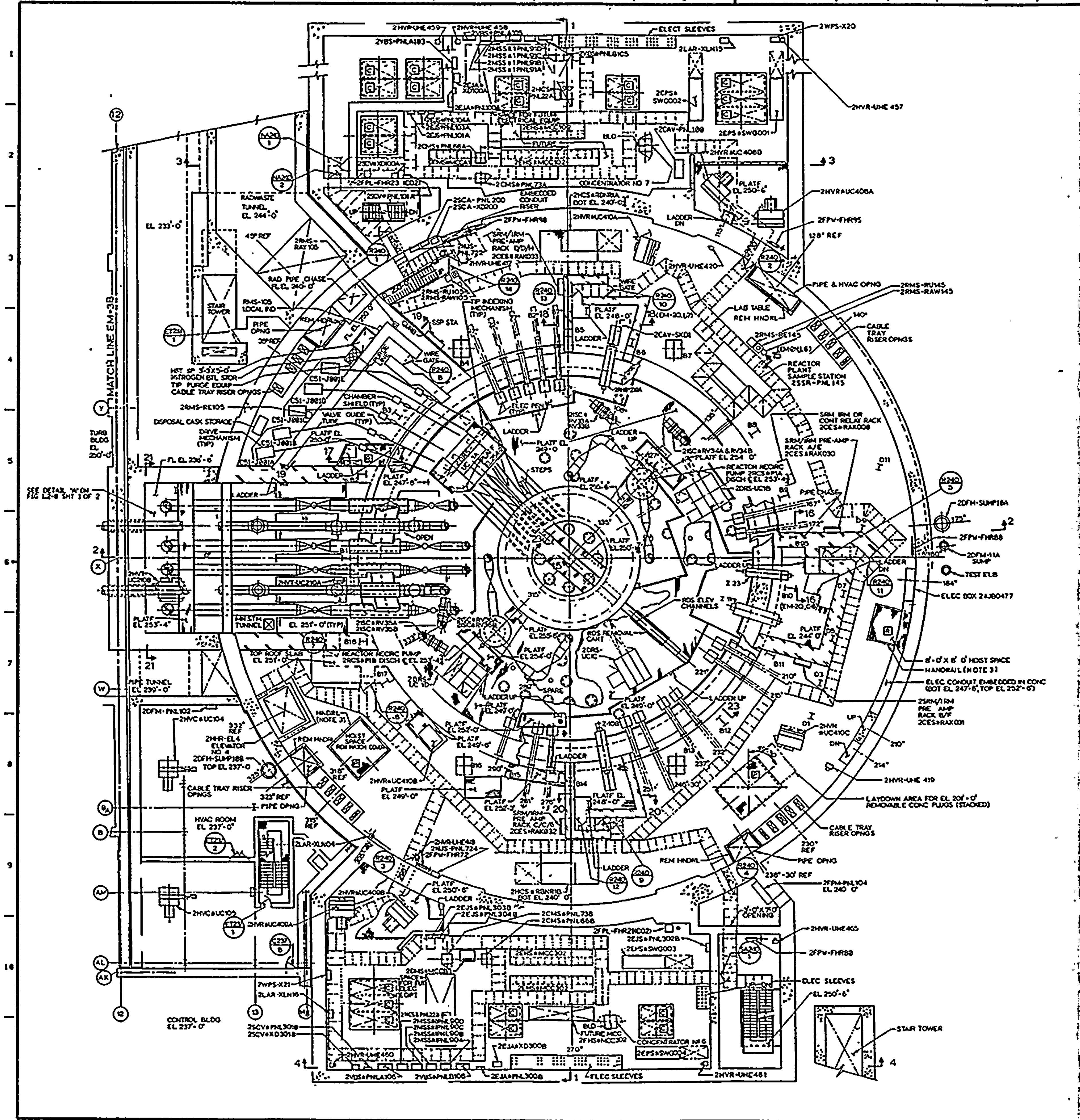

NOTES:
1. SCALE: NONE
2. GENERAL NOTES & REFERENCES SEE EM-2A

FIGURE 1.2-7

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

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NINE MILE POINT-UNIT 2
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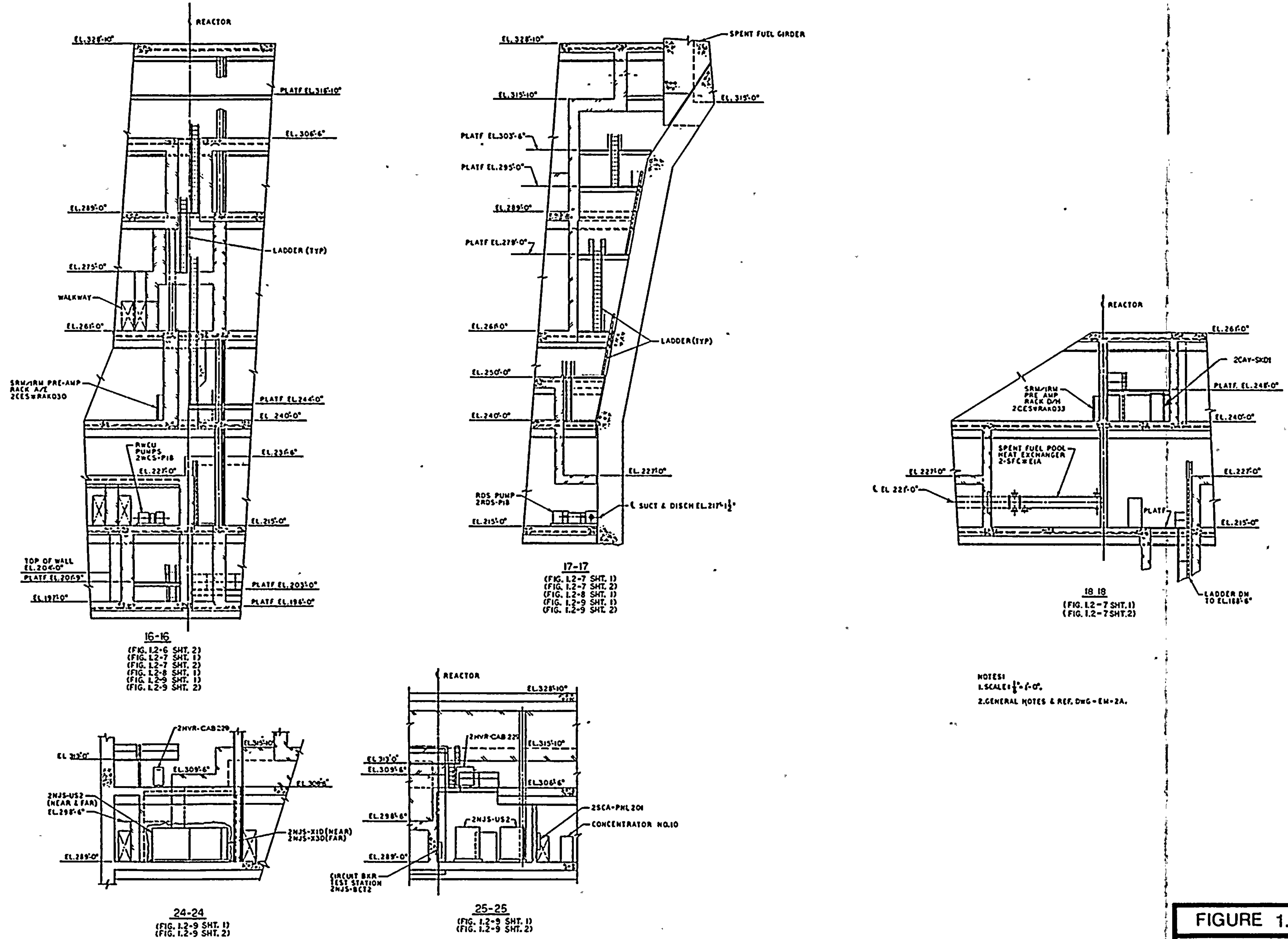

FIGURE 1.2-11

GENERAL ARRANGEMENT REACTOR BUILDING SECTIONS
SHEET 3 OF 4

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NINE MILE POINT-UNIT 2
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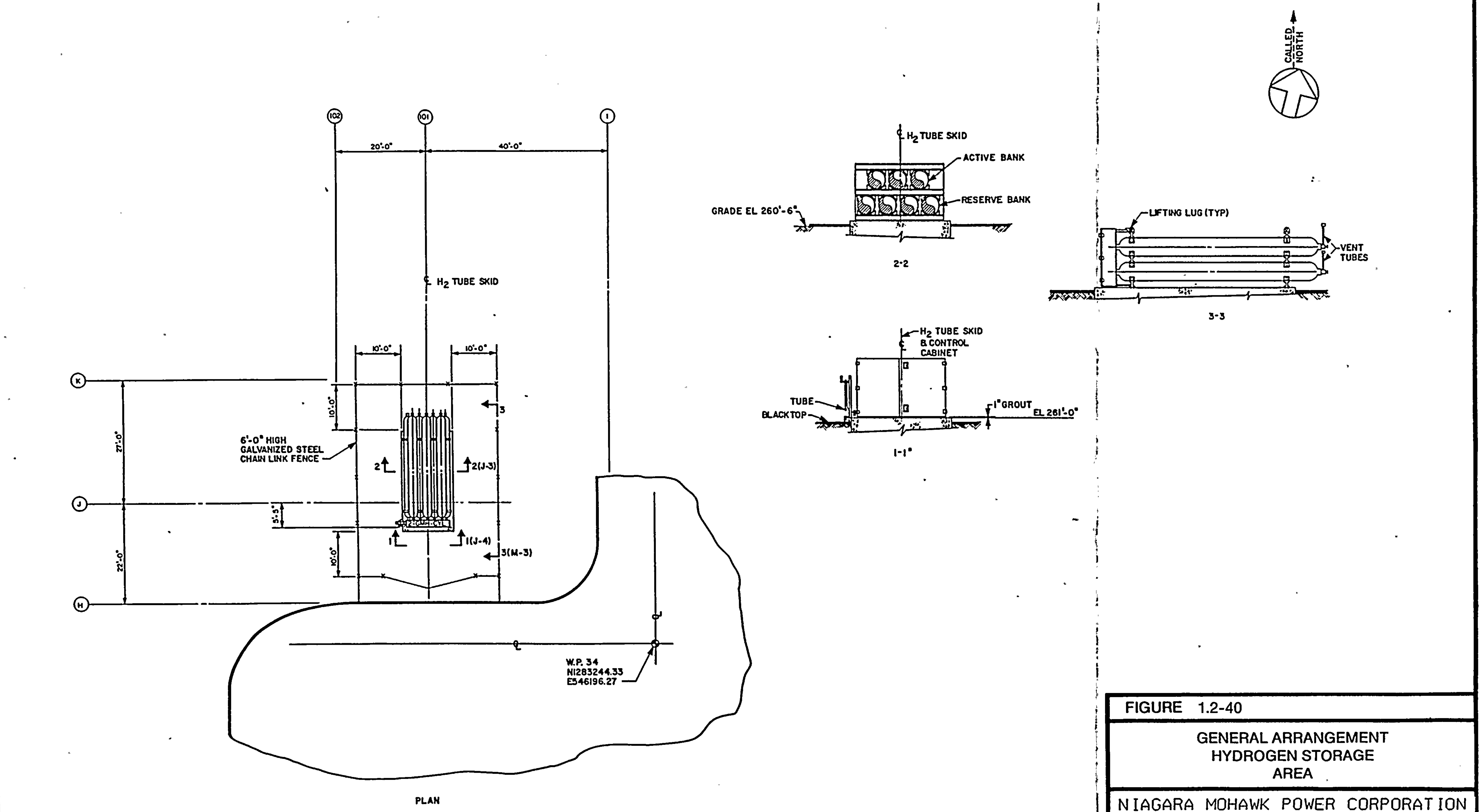

FIGURE 1.2-40

GENERAL ARRANGEMENT HYDROGEN STORAGE AREA

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TABLE 1.3-1

COMPARISON OF NUCLEAR STEAM SUPPLY SYSTEM DESIGN CHARACTERISTICS (HISTORICAL)

	Nine Mile Point Unit 2	WPPSS Unit 2	Zimmer Unit 1	La Salle Units 1, 2
THERMAL AND HYDRAULIC DESIGN (Section 4.4)				
Rated power, MWt	3,323	3,323	2,436	3,293
Design power, MWt (ECCS design basis)	3,463	3,468	2,550	3,434
Steam flow rate, millions lb/hr	14.263	14.295	10.477	14.166
Core coolant flow rate, millions lb/hr	108.5	108.5	78.5	106.5
Feedwater flow rate, millions lb/hr	14.564	14.256	10.477	14.127
System pressure, nominal in steam dome, psia	1,020	1,020	1,020	1,020
Average power density, kW/l	49.15	49.15	50.51	50.0
Minimum critical power flux ratio (MCPR)	1.24	1.24	1.24	1.28
Coolant enthalpy at core inlet, Btu/lb	527.5	527.6	527.4	527.1
Core max exit voids within assemblies	76.2	79	75	76
Core average exit quality, % steam	13.10	13.5	13.2	13.2
Feedwater temperature, °F	420	420	420	420
Design Power Peaking Factor				
Maximum relative assembly power	1.40	1.40	1.40	1.40
Axial peaking factor	1.40	1.4	1.4	1.40
Nuclear Design (First Core)				
See [4].				
CORE MECHANICAL DESIGN (Sections 4.2 and 7.6)				
Fuel Assembly				
See [4].				
Fuel Rods				
See [4].				
Fuel Pellets				
See [4].				

TABLE 1.3-2

COMPARISON OF ENGINEERED SAFETY FEATURES
DESIGN CHARACTERISTICS (HISTORICAL)

	Nine Mile Point Unit 2	WPPSS Unit 2	Zimmer Unit 1	La Salle Units 1, 2
EMERGENCY CORE COOLING SYSTEMS (Systems sized on design power) (Section 6.3)				
Low-Pressure Core Spray System				
No. loops	1	1	1	1
Flow rate, gpm	6,350 @ 128 psid	6,250 @ 122 psid	4,625 @ 119 psid	6,250 @ 122 psid
High-Pressure Core Spray System				
No. loops	1	1	1	1
Flow rate, gpm	1,550 @ 1,130 psid 6,350 @ 200 psid	1,650 @ 1,110 psid 6,250 @ 200 psid	1,330 @ 1,110 psid 4,725 @ 200 psid	1,650 @ 1,110 psid 6,250 @ 200 psid
Automatic Depressurization System				
No. systems	1	1	1	1
No. relief valves	7	7	6	7
Low Pressure Coolant Injection(1)				
No. LPCI systems	1	1	1	1
No. pumps	3	3	3	3
Flow rate, gpm/pump	7,450 @ 26 psid	7,450 @ 20 psid	5,050 @ 20 psid	7,067 @ 20 psid
AUXILIARY SYSTEMS				
Residual Heat Removal System (Section 5.4.7)				
No. loops	2	2	2	3
No. pumps	2	2	2	3
Flow rate, gpm/pump(2)	7,450	7,450	5,050	7,450
Duty, millions Btu/hr/heat exchanger(3)	41.6	41.6	30.8	46.6
No. heat exchangers	2	2	2	2
Primary containment cooling mode flow rate, gpm(4)	7,450	7,450	5,050	8,400

TABLE 1.3-3

COMPARISON OF CONTAINMENT DESIGN CHARACTERISTICS (HISTORICAL)

	Nine Mile Point Unit 2	WPPSS Unit 2	Zimmer Unit 1	La Salle Units 1, 2
Primary Containment[1] (Section 3.8)				
Type	Over & underpressure suppression Mark II	Over & underpressure suppression Mark II	Over & underpressure suppression Mark II	Over & underpressure suppression Mark II
Construction	Reinforced concrete steel liner	Steel freestanding	Concrete prestressed steel liner	Concrete posttensioned steel liner
Drywell	Frustum of cone, upper portion	Frustum of cone, upper portion	Frustum of cone, upper portion	Frustum of cone, upper portion
Pressure suppression chamber	Cylindrical lower portion	Cylindrical lower portion with elliptical bottom	Cylindrical lower portion	Cylindrical lower portion
Pressure suppression chamber - internal design pressure, psig	45	45	45	45
Pressure suppression chamber - external design pressure, psig	4.7	2	2	5
Drywell-internal design pressure, psig	45	45	45	45
Drywell-external design pressure, psig	4.7	2	2	5
Drywell free volume, ft^3	303,418	200,540[2]	180,000[3]	221,518
Pressure suppression chamber free volume (min), ft^3	192,028	144,184[4]	93,000	166,400
Pressure suppression pool water volume, ft^3	154,794[5]	112,197	102,120	109,096
Submergence of vent pipe below suppression pool surface, ft	9.5 min 11.0 max	11.67 min 12.00 max	10	12
Design environmental temperature of drywell, °F	340	340	340	340
Design environmental temperature of pressure suppression chamber, °F	270	275	275	275

TABLE 1.3-4

COMPARISON OF ELECTRICAL POWER SYSTEM DESIGN CHARACTERISTICS (HISTORICAL)

	Nine Mile Point Unit 2	WPPSS Unit 2	Zimmer Unit 1	La Salle Units 1, 2
Offsite Power System (Section 8.2)				
Outgoing lines (No.-rating)	1-345-kV	1-500-kV	3-345-kV	2-345-kV (per unit)
Incoming lines (No.-rating)	2-115-kV	1-230-kV 1-115-kV	1-69-kV 1-345-kV	2-345-kV (per unit)
Onsite ac Power System (Section 8.3.1)				
Normal station service transformers	1	2	1 (unit auxiliary)	1 per unit
Reserve station service transformers	3[1]	2	2	1 (system aux)
Standby diesel generators	3[2]	3[2]	3	3[3]
4,160-V ESF buses	3[2]	3[2]	3	3
ESF buses	3-600-V[2]	3-480-V[2]	5-480-V	4-480-V
dc Power Supply (Section 8.3.2)				
Batteries (No.-volts)	6-125-V[4] 4±24-V	4-24-V 5-125-V[4] 1-250-V	3-125-V 1-250-V	3-125-V 1-250-V
Buses (No.-volts)	6-125-V[4] 2±24-V	2-24-V 5-125-V[4] 1-250-V	3-125-V 1-250-V	3-125-V 1-250-V

(1) Includes one auxiliary boiler transformer.
(2) Includes a HPCS diesel generator.
(3) Five total for 2 units. One serves either unit.
(4) HPCS battery and bus included.

TABLE 1.3-5

COMPARISON OF RADIOACTIVE WASTE MANAGEMENT
DESIGN CHARACTERISTICS (HISTORICAL)

	Nine Mile Point Unit 2	WPPSS Unit 2	Zimmer Unit 1	La Salle Units 1, 2
Gaseous Radwaste (Section 11.3)				
Design basis, noble gases, uci/sec	100,000 after 30 min decay	100,000 after 30 min decay	100,000 after 30 min decay	100,000 after 30 min decay
Process treatment	Recombiner ambient charcoal	Low temperature charcoal	Chilled charcoal	Recombiner ambient charcoal
No. beds	8	8	5	8
Design condenser in-leakage, cfm	30	30	12.5	21
Release point, height above ground, ft	430 (stack) 187 (vent)	230	172	370
Liquid Radwaste* (Section 11.2)				
Treatment of:				
Floor drains	F or E, F, D returned to condensate storage, concentrates to radwaste solidification	F, D returned to condensate storage	F, E returned to condensate storage	E, D returned to condensate storage
Equipment drains	F, D returned to condensate storage	F, D returned to condensate storage	F, D returned to condensate storage	F, D returned to condensate storage
Chemical waste	E, F, D returned to condensate storage, concentrates to radwaste solidification	N, E, D returned to condensate storage	E, D concentrates to solid radwaste, distillate recycled	E, D concentrates to solid radwaste, distillate recycled

TABLE 1.3-6

COMPARISON OF POWER CONVERSION SYSTEM DESIGN CHARACTERISTICS (HISTORICAL)

	Nine Mile Point Unit 2	WPPSS Unit 2	Zimmer* Unit 1	La Salle Units 1, 2
Design power, MWt	3,463	3,468	2,550	3,434
Design power, MWe, gross	1,202	1,205	883	1,122
Generator speed, RPM	1,800	1,800	1,800	1,800
Design steam flow, lb/hr	14.3×10^6	15.0×10^6	11.0×10^6	14.2×10^6
Turbine inlet pressure, psia	965	970	965	965
Turbine Bypass System (Section 10.4.4)				
Capacity, percent of turbine design steam flow	25	25	25	25
Main Condenser (Section 10.4.1)				
Heat removal capacity, Btu/hr	$7,830 \times 10^6$	$7,702 \times 10^6$	$7,053 \times 10^6$	$7,609 \times 10^6$
Circulating Water System (Section 10.4.5)				
No. Pumps	6	8	3	3
Flow rate, gpm/pump	105,000	82,000	150,000	210,000
Condensate and Feedwater Systems (Section 10.4.7)				
Design flow rate, lb/hr	14.917×10^6	14.260×10^6	10.971×10^6	14.127×10^6
No. condensate pumps	3 running	3 running	3	3 plus 1 spare
No. condensate booster pumps	3 running	3 running	3	3 plus 1 spare
No. feedwater pumps	2 running 1 standby	2 running	2	3
Condensate pump drive	ac power	ac power	ac power	ac power
Condensate booster pump drive	ac power	ac power	ac power	ac power
Feedwater pump drive	ac power	Turbine	Turbine	Turbine 2 Motor 1

* Indicates parameters at rated power.

TABLE 1.3-7

COMPARISON OF STRUCTURAL DESIGN CHARACTERISTICS (HISTORICAL)

	Nine Mile Point Unit 2	WPPSS Unit 2	Zimmer Unit 1	La Salle Units 1, 2
Elevated Release Point (Section 11.3.3)				
Type	Stack, vent	Vent	Vent	Vent
Construction	Stack - reinforced concrete Vent - steel	Steel	Steel	Steel
Height (above ground), ft	430 (stack) 187 (vent)	200	172	370
Seismic Design (Section 3.7)				
Operating basis earthquake				
Horizontal, g	0.075	0.125	0.10	0.10
Vertical, g	0.075		0.07	0.07
Safe shutdown earthquake				
Horizontal, g	0.15	0.250	0.20	0.20
Vertical, g	0.15		0.14	0.14
Wind Design (Section 3.3)				
Maximum sustained, mph	90	100	90	90
Tornado				
Rotational, mph	290	300	300	300
Translational, mph	70	60	60	60
Total, mph	360	360	360	360

TABLE 1.3-8

COMPARISON OF FINAL AND PRELIMINARY DESIGN INFORMATION
FOR THE NSSS SCOPE OF SUPPLY (HISTORICAL)

Item	Change	Reason for Change	FSAR Reference
Control rod drive position	Changed to 11 wire probe and solid state.	Improved reliability and increased frequency of checking actual rod position.	7.7.1
Recirculation pump and motor	The flow rate and horsepower required have been reduced; voltage has changed from 4,160 V to 13,200 V.	Detailed system design.	5.4
Recirculation flow measurement	The recirculation flow measurement design was changed from a flow element to an elbow-tap type.	To improve flow measurement accuracy.	5.4
Recirculation system	The pressure interlock for RHR injection was changed.	IEEE-279 requirements.	7.3.1, 7.1
Feedwater and recirculation nozzle safe ends and thermal sleeves	Material/design change. Piping changed to type 316K from type 304.	Mitigate IGSCC.	5.3
Nuclear fuel	The number of fuel pins in each fuel bundle has been changed from 7x7 to 8x8.	Improved fuel performance by increasing safety margins.	4.2
Nuclear boiler	a. A turbine building high temperature trip for MSIVs was added.	Improve leak detection capability.	7.3
	b. Delete REVAB system.	GE Mark II suppression pool dynamics test program showed REVAB undesirable.	5.2, 5.4, 8.3.1
Main steam line isolation	A main condenser low vacuum initiation of the main steam line isolation was added.	NRC requirement.	7.3.1
Main steam line drain system	A main steam line drain system was improved.	Prevent accumulation of condensate in an idle line outboard of MSIV.	5.1
Feedwater sparger	The thermal sleeve was changed to provide welded design of sparger to nozzle.	To eliminate vibration and cracking.	5.3
RCIC steam supply	A warmup bypass line and valve were added.	Permits pressurizing and prewarming of the steam supply line downstream to the turbine during reactor vessel heatup.	5.4
Control rod drive system	Alternate rod injection and scram discharge volume modifications were implemented.	To reduce potential for failure to scram.	4.6.1

TABLE 1.3-9

COMPARISON OF FINAL AND PRELIMINARY DESIGN INFORMATION
FOR THE BALANCE OF PLANT (HISTORICAL)

Item	Change	Reason for Change	FSAR Reference
Reactor building	Addition of auxiliary bays.	Provide room to allow segregation of ECCS.	6.2.3, 3.2.1, 3.8.4
Summary description of structures	Additional buildings included in the unit.	Auxiliary bays, railroad access lock, condensate storage tank building, and demineralized water and waste neutralizer tank storage building added.	1.2, 3.2.1
Primary containment cooling	a. Power electric motor components through normal 4-kV switchgear.	Containment cooling is not a nuclear safety-related system.	8.3
	b. Revised arrangement and number of unit coolers.	Improve air distribution based on operating experience.	9.4.9
Standby gas treatment system (SGTS)	Add ASME Class 2 isolation valve between containment purge and SGTS.	To isolate containment purge (Class 4 outside containment) from Class 2 SGTS.	6.5.1
Reactor building ventilation system	a. Locate supply fans in SGT building.	Provides a more efficient isolation of the reactor building.	9.4.2
	b. Normal exhaust system to consist of two sets of two fans.	Improve system design for better air movement and optimum fan performance.	9.4.2
	c. Change from valves to zero-leakage dampers.	Reduce seismic load and closure time and compact valve design.	9.4.2
	d. Eliminate mixing box.	Credit taken for turbulent mixing during emergency operation.	9.4.2
Primary containment ventilation	All containment purge air is passed through the SGTS and vented out the stack.	To enhance safety of plant ventilation design.	6.5.1
High-density spent fuel storage	Changed spent fuel rack configuration to high-density storage design.	To increase onsite storage capacity of spent fuel.	9.1
Primary shield wall	Additional provisions added to supplement the commitment to use AISC Steel Construction Manual welding requirements.	AISC standards do not account for certain weld configurations necessary to achieve proper erection, hence additional standards used for those weld configurations.	3.8.3

UPDATED SAFETY ANALYSIS REPORT

NINE MILE POINT NUCLEAR STATION — UNIT 2

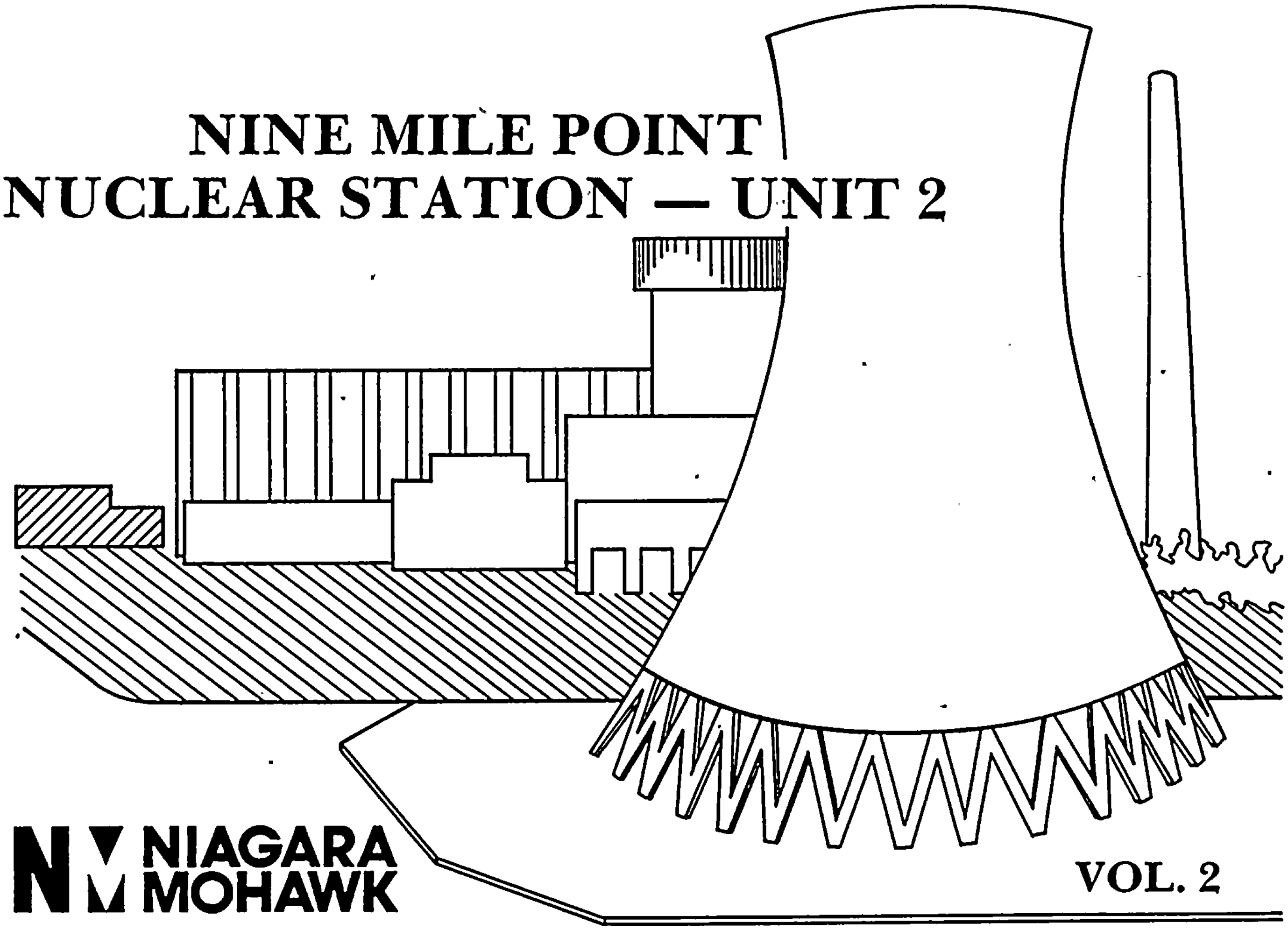

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TABLE 1.8-1

CONFORMANCE WITH DIVISION 1 NRC REGULATORY GUIDES

<u>Regulatory Guide 1.1, Revision 0 (November 1970)</u>

Net Positive Suction Head for Emergency Cooling and Containment Heat Removal System Pumps

<u>FSAR Section</u> 6.3.2.2

<u>Position</u> The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

The physical location of the RHR, LPCS, and HPCS pumps in relation to the minimum suppression pool water level is such that the required NPSH is maintained on these pumps under the conditions of zero psig containment pressure and 212°F suppression pool water temperature for all operating modes. Adequate NPSH is verified by system calculations.

<u>Regulatory Guide 1.2, Revision 0 (November 1970)</u>

Thermal Shock to Reactor Pressure Vessels

<u>FSAR Section</u> 5.3.3

<u>Position</u> The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

<u>Regulatory Guide 1.3, Revision 2 (June 1974)</u>

Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors

<u>FSAR Sections</u> 3.11.5, 15.6.5

<u>Position</u> The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

RG 1.3, together with TID 14844 models, has been used to arrive at the estimated site boundary 2-hr dose, and low population zone (LPZ) 30-day dose for a LOCA. Regulatory Position C.1.f of RG 1.3 is replaced by Standard Review Plan (NUREG-0800) Section 6.5.5 for the fission product scrubbing and iodine retention in the suppression pool, which is included in the containment leakage and TIP leakage release calculations during the secondary containment drawdown period.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.3, Revision 2 (June 1974) (cont'd.)
(See Section 15.6.5 for details). All other releases are evaluated using the RG 1.3 criteria.
However, the meteorological assumptions were based upon Murphy and Campe and RG 1.145, as discussed in Section 2.3.4.3.

TABLE 1.8-1 (Cont'd.)

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Regulatory Guide 1.4, Revision 2 (June 1974)

Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors

Position RG 1.4 applies to PWR plants and therefore is not applicable to Unit 2.

Regulatory Guide 1.5, Revision 0 (March 1971)

Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors

FSAR Section 15.6.3

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

The assumptions in RG 1.5 have been used to arrive at the site boundary and LPZ doses for the steam line break accident. The coolant activity level given by the NRC in the Standard Technical Specification for GE BWRs, NUREG-0123, is used as a reference point for the accident analysis and is given as the final coolant activity level for the Unit 2 Technical Specifications.

Regulatory Guide 1.6, Revision 0 (March 1971)

Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems

FSAR Sections 8.3.1.2, 8.3.2.2

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide for Division I, II, and III diesels.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.94, Revision 1 (April 1976) (cont'd.)

(a) after cutting, the edges of the cut will be ground or reamed back a minimum of 1/32 in, and (b) the final bolt hole dimensions will not exceed those given in the Specification for Structural Joints Using ASTM A325 or A490 bolts.

2. ANSI N45.2.5-1974 Section 5.4 For the Unit 2 project, the criterion established for correct bolt length is one thread extending beyond the face of the nut.

3. ANSI N45.2.5-1974 Section 5.5 All reinforcing bar splices made by arc welding, except those splices welded to metal embedments, will be selected on a random basis for radiography as specified in the Unit 2 PSAR Section 12.6.3, and inspected in accordance with AWS D12.1. Splices welded to metal embedments will be inspected in accordance with AWS 12.1. Additionally, sister splice testing will be done in accordance with Specification No. NMP2-S203C with the same frequency as specified for B-series sister splices when required by the engineers.

4. ANSI N45.2.5-1974 Section 6.2.2 Exceptions regarding mechanical splicing of QA Category I reinforcing bars can be found in Unit 2 Project Position 1.10.

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* This commitment is modified at the time of the QA Topical Report implementation. At that time, the QATR (see Appendix B) supersedes this commitment.

Regulatory Guide 1.95, Revision 1 (January 1977)

Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release

FSAR Section 6.4

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.96, Revision 1 (June 1976)

Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants

FSAR Sections 1.2.9.11, 5.4.5, 6.2.3.2.3, 15.6.5

Position MSIV leakage, at the maximum rate allowed by the Technical Specifications, has been included in the secondary containment bypass leakage analysis (Section 6.2.3.2.3) and in the LOCA radiological consequence analysis (Section 15.6.5). These design-basis analyses demonstrate that the calculated exposures are within the guidelines of 10CFR100 and 10CFR50 Appendix A, General Design Criteria 19.

In addition, a qualitative comparison has been made between Unit 2 and the plant used as the basis for analyses presented in NUREG-1169, "Technical Findings Related to Generic Issue C-8; Boiling Water Reactor Main Steam Isolation Valve Leakage and Leakage Treatment Methods." This comparison demonstrated that the design features of Unit 2 are sufficiently similar to the NUREG-1169 base plant, such that the conclusions of NUREG-1169 are considered directly applicable to Unit 2. NUREG-1169 concluded that the overall risks from the accident sequences in which MSIV leakage could be a significant factor are low without a leakage control system, and alternate fission product handling techniques, which make use of the holdup volume of the main steam lines and condenser, produce significant reductions in offsite dose consequences. It is therefore concluded that a MSIV leakage control system is not required for Unit 2.

Regulatory Guide 1.97, Revision 3 (May 1983)

Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

FSAR Section 1.10, 7.1.2, 7.5.2.1

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described in Section 7.5.2.1.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.98, Revision 0 (March 1976)
(For Comment)

Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor

FSAR Section 15.7.1

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

TABLE 1.8-1 (Cont'd.)

<u>Regulatory Guide 1.99, Revision 2 (May 1988)</u> Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials <u>FSAR Sections</u> 5.3.1, 5.3.2 <u>Position</u> The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described in Sections 5.3.1 and 5.3.2.
<u>Regulatory Guide 1.100, Revision 1 (August 1977)</u> Seismic Qualification of Electric Equipment for Nuclear Power Plants <u>FSAR Sections</u> 3.10A, 3.10B, 7.1, 8.3 <u>Position</u> The Unit 2 project addresses the Regulatory Position (Paragraph C) of this guide through the alternate approach described in Sections 3.10A and 3.10B.
<u>Regulatory Guide 1.101, Revision 2 (October 1981)</u> Emergency Planning for Nuclear Power Plants <u>FSAR Section</u> Site Emergency Plan <u>Position</u> The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below. The Emergency Plan has been written to comply with NUREG-0654.
<u>Regulatory Guide 1.102, Revision 1 (September 1976)</u> Flood Protection for Nuclear Power Plants <u>FSAR Sections</u> 2.4.2, 2.4.10, 3.4 <u>Position</u> The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

TABLE 1.8-1 (Cont'd.)

<u>Regulatory Guide 1.125, Revision 1 (October 1978)</u>

Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants

<u>FSAR Section</u> 2.4

<u>Position</u> The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

<u>Regulatory Guide 1.126, Revision 1 (March 1978)</u>

An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification

<u>FSAR Section</u> Chapter 5

<u>Position</u> The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A (latest approved revision), and General Electric Standard Application for Reactor Fuel - United States Supplement, NEDE-24011-P-A-US (latest approved revision), are used to comply with the requirement of this regulatory guide.

<u>Regulatory Guide 1.127, Revision 1 (March 1978)</u>

Inspection of Water-Control Structures Associated with Nuclear Power Plants

<u>Position</u> Not applicable to Unit 2.

<u>Regulatory Guide 1.128, Revision 1 (October 1978)</u>

Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants

<u>FSAR Sections</u> 8.1, 8.3.2

<u>Position</u> The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach. Instead of IEEE-484-1975, the latest issue of IEEE-484-1987 is used. According to IEEE, the latest issue of a standard represents the current state of the art and is recommended for use.

TABLE 1.8-1 (Cont'd.)

<u>Regulatory Guide 1.129, Revision 1 (February 1978)</u>

Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants

<u>FSAR Sections</u> 8.1, 8.3.2

<u>Position</u> The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

<u>Regulatory Guide 1.130, Revision 1 (October 1978)</u>

Design Limits and Loading Combinations for Class 1 Plate-and-Shell Type Component Supports

<u>FSAR Section</u> 3.9.3.B

RG 1.130 Revision 0 (July 1977) was issued after the docketing date for Unit 2 and work was in progress. However, Unit 2 complies with the indicated items of the Regulatory Position of this guide through the alternative approach, described as follows. The remaining design analysis criteria of this regulatory guide are adequately addressed by conservatisms in the existing ASME III Code.

1. <u>Paragraph C.2</u> Ultimate strength temperature correlation of this guide was used in regions adjacent to pipe having high temperatures.

2. <u>Paragraph C.3</u> Regulatory Position C.4, with alternate conservative collapse criteria developed by the NSSS supplier for plates and shells, was used in lieu of Regulatory Position C.3.

<u>Regulatory Guide 1.131, Revision 0 (August 1977)</u>
<u>(For Comment)</u>

Qualification Tests of Electrical Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants

<u>FSAR Section</u> 3.11

<u>Position</u> The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.148 (March 1981) (cont'd.)

BOP

Other safety-related valve assemblies classified as Quality Group A, B, or C in RG 1.26 comply with the regulatory guides as described below.

a. Section C.2.a, Valve Application Characteristics

The frequency of use for each safety-related valve assembly is not specified. The normal (open/closed) position is not specified, except in the case of safety-related butterfly and solenoid valve assemblies.

b. Section C.2.b, Structural Requirements

The dynamic loading and the piping frequency response spectra are not specified. Potential water hammer is not considered when establishing the maximum differential pressure across a valve.

c. Section C.2.c, Operational Requirements

The safety-related function (open/close, remain-as-is) is not specified, except in the cases of ball, butterfly, and solenoid valve assemblies. Motor power requirements for valve assemblies are not specified.

NSSS

Fast-closing isolation valve assemblies classified as Quality Group D in RG 1.26 meet the requirements of ANSI B31.1.0, 1977. They also comply with RG 1.148, dated March 1981, with the following clarification:

a. Section C.2.a, Valve Application Characteristics

The frequency of use for each safety-related valve assembly is not specified. The normal (open/closed) position is not specified, except in the case of safety-related butterfly and solenoid valve assemblies.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.148 (March 1981) (cont'd.)

b. Section C.2.b, Structural Requirements

The dynamic loading and the piping frequency response spectra are not specified. Potential water hammer is not considered when establishing the maximum differential pressure across a valve.

Main steam isolation valve assemblies comply with RG 1.148, dated March 1981, with the following clarification:

a. Section C.1.c.2, Applicability and Relationship with other Standards

The functional specification does not reference the design specification.

Regulatory Guide 1.149 (April 1981)

Nuclear Power Plant Simulators for Use in Operator Training

FSAR Section Chapter 13.2

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide for design and construction.

Regulatory Guide 1.149, Revision 1 (April 1987)

Nuclear Power Plant Simulation Facilities for Use in Operator License Examinations

FSAR Section Chapter 13.2

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide for simulator certification.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.150, Revision 1 (February 1983)

Ultrasonic Testing of Reactor Vessel Welds During Preservice and In-service Examination

FSAR Section PSI/ISI Plan

Position The Preservice Inspection Plan for Unit 2 consists of three separate documents as follows:

1. Preservice Inspection Plan for Nuclear Piping Systems and the Reactor Pressure Vessel.

2. In-service Testing Plan for Pumps and Valves.

3. Preservice Inspection Plan for Nuclear Piping System and Component Supports.

Part 1 was submitted for the Staff's review by letter dated October 15, 1985, and included all NDE for items required by ASME Section XI for nuclear piping systems, and the RPV, its internals and safe ends. Also included in that submittal is a listing of PSI ultrasonic procedures and the PSI isometric drawings. Part 2 was transmitted for the Staff's review by letter dated November 27, 1985, and Part 3 was submitted on December 17, 1985.

As discussed in Section 1.7 of the PSI Plan (Part 1), the required examinations for PSI shall be performed prior to initial startup of the plant. The applicable code for Unit 2 PSI is ASME Section XI, 1980 Edition through the Winter 1980 Addenda, as discussed in Section 1.1.1 of the PSI Plan (Part 1). Section 1.2.1 of the PSI Plan (Part 1) discusses the application of the examination selection criteria. Selected for volumetric examination were 7.5 percent of the welds in the RHR system, HPCS system, and LPCS system, normally excluded from preservice volumetric examination.

The degree of compliance with RG 1.150 Revision 1 is provided in Table 1.8-1a. This document presents the alternate method of compliance with the regulatory guide, how the compliance is achieved (under the column "Response"), and what documents or procedures reference the compliance implementation (under the column "Procedures and References"). The Degree of Compliance position is incorporated in the RPV examination procedures. A technique qualification shall be performed using the ultrasonic examination systems that will be employed during the automated PSI examinations of the RPV. Actual RPV and

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.150, Revision 1 (February 1983) (cont'd.)

nozzle segments containing known size reflectors located in the ID surface shall be used. This qualification shall be witnessed by NMPC, its Agents, and the NMPC ANII, as a minimum. The qualification shall demonstrate that flaws of the maximum allowable limits are detectable.

Regulatory Guide 1.155 (August 1988)

Station Blackout

FSAR Section 8.3.1.5

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Unit 2 is evaluated against the requirements of the Station Blackout Rule, 10CFR50.63, using the guidance contained in NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," NUMARC 87-00 Supplemental Questions/Answers, dated December 27, 1989, and NUMARC 87-00 Major Assumptions, dated December 27, 1989, except where RG 1.155 takes precedence. Table 1 of RG 1.155 provides a cross-reference between the regulatory guide and NUMARC 87-00. Any exceptions to the NUMARC guidance taken by NMPC are identified in the SBO documentation maintained by NMPC (see Letter No. NMP2L 1230, dated April 3, 1990, to NRC, TAC No. 68571).

TABLE 1.8-2 (Cont'd.)

Regulatory Guide 8.14, Revision 1 (August 1977)

Personnel Neutron Dosimetry

FSAR Section None

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Regulatory Guide 8.15, Revision 0 (October 1976)

Acceptable Programs for Respiratory Protection

FSAR Section None

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Paragraph C.3.n Exception is taken to the recommendations of Section 13.2 of NUREG-0041 relative to prohibiting the use of contact lenses with full face-piece respirators.

Regulatory Guide 8.19, Revision 1 (June 1977) (For Comment)

Occupational Radiation Dose Assessment in Light-Water Reactor Nuclear Power Plants - Design Stage Man-Rem Estimates

FSAR Section None

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Regulatory Guide 8.20, Revision 1 (September 1979)

Applications of Bioassay for I-125 and I-131

FSAR Section

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

TABLE 1.8-2 (Cont'd.)

<u>Regulatory Guide 8.26, Revision 0 (September 1980)</u> Applications of Bioassay for Fission and Activation Products <u>FSAR Section</u> None <u>Position</u> See Sections 12.5.3 and Exhibit 12.1-2 for an assessment of this Regulatory Guide.
<u>Regulatory Guide 8.27, Revision 0 (March 1981)</u> Radiation Protection Training for Personnel at Light-Water Cooled Nuclear Power Plants <u>FSAR Section</u> None <u>Position</u> See Section 12.5.3 for an assessment of this Regulatory Guide.
<u>Regulatory Guide 8.28, Revision 0 (August 1981)</u> Audible Alarm Dosimeters <u>FSAR Section</u> <u>Position</u> The Unit 2 project complies with this guide with the following clarification: Audible alarm dosimeters may be used in areas of high noise provided that the frequency of observation of accrued dose is increased.
<u>Regulatory Guide 8.29, Revision 0 (July 1981)</u> Instruction Concerning Risks From Occupational Radiation Exposure <u>FSAR Section</u> None <u>Position</u> See Section 12.5.3 for an assessment of this Regulatory Guide.

37. STANDARD REVIEW PLAN 5.4.6, REVISION 2 - REACTOR CORE ISOLATION COOLING SYSTEM (BWR)

Difference TMI action items are not discussed in this section.

Discussion See FSAR Section 1.10 for NUREG-0737.

38. STANDARD REVIEW PLAN 5.4.8, REVISION 2 - REACTOR WATER CLEANUP SYSTEM

Difference 1 All RWCU system components are not drained and vented through closed systems.

Discussion Vents and drains associated with the pumps and the regenerative and nonregenerative heat exchangers are routed to the reactor building equipment drain system through open drains which are vented to secondary containment atmosphere. The pumps and heat exchanger vents are used to vent the equipment when filling the system. The drains are used to empty the components prior to maintenance.

The temperature of the water will be low enough during these draining and venting operations that the possibility of airborne contamination will be minimal. Therefore, the routing of these lines to an open drain connection is acceptable.

Difference 2 Evaluation of compliance with the Technical Specifications for water chemistry parameter limits is not provided.

Discussion Reactor water purity will be maintained by the system to yield effluent water in accordance with the requirements of RG 1.56 (FSAR Section 5.4.8.1.2) and the Technical Specifications for Water Chemistry within limits described in Technical Specifications.

39. STANDARD REVIEW PLAN 6.1.2, REVISION 2 - PROTECTIVE COATING SYSTEMS (PAINTS) - ORGANIC MATERIALS

Difference For a small fraction of the exposed surfaces in the drywell, the recommendation of RG 1.54 is not met.

Discussion See FSAR Sections 6.1.2.1 and 6.1.2.2. Protective coatings are generally not used in the suppression pool. The majority of the exposed surfaces within the drywell (i.e., primary containment lines,

TABLE 1.9-1 (Cont'd.)

drywell head, biological shield wall, structural steel, cranes, pipe rupture restraints, pipe supports, piping, and concrete) are coated with materials qualified in accordance with ANSI N101.2 and applied in accordance with RG 1.54. The balance of the exposed surfaces within the drywell (i.e., valve bodies, hand wheels, electrical and control panels, loudspeakers, and emergency light cases), constituting a small fraction of the total exposed surfaces, do not satisfy RG 1.54 conditions.

40. STANDARD REVIEW PLAN 6.2.1.1.c, REVISION 4, JULY 1981, APPENDIX I TO STANDARD REVIEW PLAN 6.2.1.1.c, REVISION 1, JULY 1981 - PRESSURE-SUPPRESSION TYPE BWR CONTAINMENTS

Difference 1 Peak calculated temperature for the wetwell airspace exceeds the design temperature of the suppression pool.

Discussion Peak calculated containment pressure and deck differential pressure are within design limits. Drywell calculated environment temperature is below its design value. However, following the steam bypass transient, the atmospheric temperature in the suppression chamber is greater than 212°F (superheated). For a small-break LOCA with steam bypass, the temperature is determined to be approximately 250°F. Any Category 1 equipment in the suppression chamber will be qualified to the maximum envelope value of 270°F, which has been specified in environmental qualification documents. However, the structure temperature, i.e., steel liner, remains below the saturation temperature of the suppression chamber atmosphere for the duration of the transient. Since the liner temperature is below 212°F, the design temperature of the suppression chamber structure is not exceeded.

Difference 2 Suppression chamber spray is not autoactuated following a LOCA.

Discussion One of the SRP requirements concerns the automatic suppression chamber spray limiting containment pressure to 45 psig considering steam bypass. Analysis for Unit 2 shows that containment spray is not necessary for the first 30 min following a LOCA; therefore, manual spray is justified. This will eliminate the potential for inadvertent spray due to the malfunction of an automatic control.

TABLE 1.9-1 (Cont'd.)

63. STANDARD REVIEW PLAN 11.3, REVISION 0, BRANCH TECHNICAL POSITION ETSB 11-5 - POSTULATED RADIOACTIVE RELEASES DUE TO A WASTE GAS SYSTEM LEAK OR FAILURE

Difference A comparison of the main parameters of the waste gas system event analysis, as presented in this SRP and those actually used in FSAR Section 15.7.1, is provided below.

Parameter	NUREG-0800 BTP ETSB 11-5	FSAR Section 15.7.1
Accident/event	Bypass of charcoal delay units, release of undelayed offgas activities	Failure of charcoal delay beds, release of total bed activity
Source term	7 x normal operation source term 7x50,000 uCi/s = 350,000 uCi/s	100 uCi/s/MWt (100x3,536 MWt) = 353,600 uCi/s
Source term decay time	30 min	30 min
Isotopes considered	Xe, Kr, Ar	Xe, Kr
Holdup time on charcoal beds	Not applicable	Xe - 178 days Kr - 278 hr
Release point	Ground level	Ground level
Duration of release	2 hr	2 hr
Value of X/Q	5% overall site short term	.5% maximum sector short term
Duration of exposure	2 hr	2 hr
Dose calculations	Semi-infinite cloud	Semi-infinite cloud
Exposure limit	<0.5 Rem total body	<5 Rem whole body (calculated .24 Rem); <30 Rem Beta (calculated .22 Rem)

TABLE 1.9-1 (Cont'd.)

Discussion The analysis of the failure of the offgas system, provided in Section 15.7.1, is more conservative than the analysis proposed in this SRP, in terms of duration, X/Q, and transit time. Therefore, the existing analysis envelops that proposed by BTP ETSB 11-5.

64. STANDARD REVIEW PLAN 12.2, REVISION 2 - RADIATION SOURCES

Difference 1 Shielding and ventilation design fission product source terms were not developed using these bases:

1. An offgas rate of 100,000 uCi/sec after 30 min delay for BWRs.

2. 0.25 percent fuel cladding defects for PWRs.

Discussion The general basis for the shielding design is stated in Section 12.2.1.1. Sections 12.2.1.2 through 12.2.1.5 provide source data that were used in shielding designs. Sources of airborne radiation to be considered in ventilation design are discussed in Section 12.2.2. Criterion (1) is discussed in Section 11.1, and criterion (2) does not apply.

65. STANDARD REVIEW PLANS 12.3 AND 12.4, REVISION 2 - RADIATION PROTECTION DESIGN FEATURES

Difference 1 The following items required by NUREG-0800, Section II.1, are not presented in the FSAR.

1. Access control to spent fuel transfer canal should be more stringent than that required by 10CFR20.203.

2. All accessible portions of the spent fuel transfer canal that are capable of having radiation levels greater than 100 rads/hr shall be shielded during fuel transfer.

3. Removable shielding may be used (for Item b) but must be explicitly marked. Local audible and visible alarming radiation monitors must be installed to alert personnel if the temporary shielding is removed during fuel transfer operations.

4. All accessible portions of the spent fuel transfer tube shall be clearly marked with a

TABLE 1.9-1 (Cont'd.)

sign stating that potentially lethal radiation fields are possible during fuel transfer.

5. Similar precautions to those described in Items a through d shall also apply to any other radiation source having radiation levels higher than 100 Rem/hr.

Discussion

1. Because of the procedures and shield design described below, access control in accordance with 10CFR20 is considered to be adequate.

2. A portable shield or access control will be used to limit dose rates in areas of the drywell accessible during fuel transfer to <20 mRem/hr.

3. Refueling procedures will either mandate the placement of the radiation shield or implement access controls before fuel transfer operations. Portable monitors will be used to alarm audibly and visibly in the drywell if the portable shield is not installed or is removed during fuel transfer.

4. Not applicable to Unit 2 design.

5. Precautions similar to those described above may also be taken for other radiation sources having radiation levels in excess of 100 Rem/hr.

Difference 2 Area radiation monitors are required by NUREG-0800, paragraph II.4.A.3, to remain on-scale when measuring dose rates during accidents and anticipated operational occurrences. A description of vital area monitoring has not been provided in Section 12.3.4.

Discussion Postaccident vital area monitors meet the criterion of NUREG-0800, paragraph II.4.A.3, and will be addressed in an amendment to FSAR Section 12.3.

66. STANDARD REVIEW PLAN 12.5, REVISION 2 - OPERATIONAL RADIATION PROTECTION PROGRAM

Difference 1 No personnel count rate meters are provided.

TABLE 1.9-1 (Cont'd.)

Discussion The use of count rate meters on protective clothing will provide little, if any, additional radiation protection in view of the extensive personnel monitoring that will be implemented.

Difference 2 TLDs are processed quarterly.

Discussion Although Unit 2 does conform to Regulatory Guide 8.3, in 1987 10CFR20 was amended to require all licensees to have personnel dosimetry devices that are utilized to comply with NRC regulations processed by processors that have been accredited by the National Voluntary Laboratory Accreditation Program (NVLAP) of the National Bureau of Standards. In the statement of consideration for this amendment, the NRC specified that dosimetry processors would demonstrate compliance with ANSI N13.11-1983 through testing. Nine Mile Point Dosimetry Facility is accredited by NVLAP to process TLDs by virtue of actual demonstration of compliance with ANSI N13.11-1983 through testing. Based on "fade" studies, processing TLDs quarterly instead of monthly does not affect the dosimetry facility's NVLAP accreditation and complies with 10CFR20.

67. STANDARD REVIEW PLAN 13.4, REVISION 2 - OPERATIONAL REVIEW

Difference Independent review is not performed by an ISEG.

Discussion Independent review is performed by the SRAB and the Onsite Technical Services Group, as described in Section 1.10 and Chapter 13. The approach given meets the intent of the requirements stated.

68. STANDARD REVIEW PLAN 14.2, REVISION 2 - INITIAL PLANT TEST PROGRAM

Difference The test abstracts contain significant parameters but do not include plant performance characteristics.

The following table summarizes these considerations:

Containment	LOCA Source Term (Noble Gas/Iodine/ Particulate)	Non-LOCA High Energy Line Break Source Term (Noble Gas/Iodine/ Particulate)
Outside	% (100/50/1) in RCS	% (10/10/0) in RCS
Inside	Larger of (100/50/1) in containment or (100/50/1) in RCS	(10/10/0) in RCS

Nine Mile Point Unit 2 Position

Analyses have been performed to quantify the postaccident radiation levels throughout the Unit 2 plant based upon the source terms presented below.

These radiation conditions are being used in conjunction with other environmental conditions (pressure, temperature, and humidity) for the equipment qualification program. Safety-related equipment is being qualified in accordance with NUREG-0588.

A description of the Unit 2 postaccident shield design review is given in Section 12.3.1.3. Areas where access is vital after an accident are analyzed for personnel occupancy to ensure that doses to personnel performing vital postaccident functions are less than GDC 19 limits. This information is provided in Table 12.3-3. A dose rate map for potentially-occupied areas is provided on Figure 12.3-69 and corresponding Table 12.3-4.

Source Term

Radioactive source release and distribution assumptions for Unit 2 are as follows.

Radioactive Source Release

1. The percentages of core inventory radioactive fission products assumed to be released from the fuel rods are:

Noble gases (Kr, Xe)	100%
Halogens	50%
Others	1%
Cesium	50%

2. This entire release is assumed to occur instantaneously at the start of the accident.

Radioactive Source Distribution

To envelop the full spectrum of break sizes and depressurization rates, two bounding events and source distributions were considered.

1. LOCA The following fission products are considered to be uniformly mixed in the following volumes:

   a. Suppression Pool and Reactor Coolant System

Noble gases	0%
Halogens	50%
Others	1%
Cesium	50%

   This distribution assumes short-term reactor depressurization and is consistent with a scenario which leads to gross fission product release from the fuel.

   b. Combined Drywell/Wetwell Air Space

Noble gases	100%
Halogens	50%*

   Using this distribution, time history radiation zones are established throughout the Unit 2 plant as follows:

   a. The above sources will be distributed in the following system piping to establish time history radiation zones for the primary containment and reactor building. These systems were conservatively assumed to operate concurrently.

      (1) Main steam system (primary only).
      (2) RCIC.
      (3) RHR.
      (4) LPCS and HPCS.
      (5) SGTS.
      (6) RCS/RWCU (primary only).
      (7) Containment atmosphere monitoring.
      (8) Hydrogen recombiner system.

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* The fraction of airborne halogens available for release to the environment is 25 percent of the core inventory in accordance with RG 1.3. This percentage is further reduced due to suppression pool scrubbing credit in accordance with SRP Section 6.5.5.

b. In addition to radiation shine from system piping and components, the primary containment is assumed

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Criterion 5

The time for a chloride analysis to be performed depends upon two factors: (a) if the plant's coolant water is seawater or brackish water, and (b) if there is only a single barrier between primary containment systems and the cooling water. Under both of the above conditions the licensee shall provide for a chloride analysis within 24 hr of the sample being taken. For all other cases, the licensee shall provide for the analysis to be completed within 4 days. The chloride analysis does not have to be done onsite.

Clarification 5

BWRs on sea or brackish water sites, and plants which use sea or brackish water in essential heat exchangers (e.g., shutdown cooling) that have only single barrier protection between the reactor coolant are required to analyze chloride within 24 hr. All other plants have 96 hr to perform a chloride analysis. Samples diluted by up to a factor of 1,000 are acceptable as initial scoping analysis for chloride, provided (1) the results are reported as ± 2 ppm Cl (the licensee should establish this value; the number in the blank should be no greater than 10.0 ppm Cl) in the reactor coolant system, and (2) that dissolved oxygen can be verified at <0.1 ppm, consistent with the guidelines above in Clarification No. 4. Additionally, if chloride analysis is performed on a diluted sample, an undiluted sample need also be taken and retained for analysis within 30 days, consistent with ALARA.

Position 5

Chloride in the reactor coolant can be determined within 96 hr by using a specific ion electrode. Unit 2 does not use brackish water for plant coolant. An undiluted reactor coolant sample is treated with a sodium bromate-nitric acid solution to eliminate halogen interferences. The sample is handled in a fume hood with long tongs and lead brick shielding to reduce radiation exposure.

Additionally, NMPC participates in the Pooled Inventory Management Program and should have a postaccident sampling cask from Nuclear Packaging, Inc., available for sample transport to an offsite facility for further analysis.

Criterion 6

The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiation exposures to any individual exceeding the criteria of GDC 19 (Appendix A, 10CFR50), i.e., 5 rem whole body, 75 rem extremities. (Note that the design and operational review criterion was changed from the operational limits of 10CFR20

(NUREG-0578) to the GDC 19 criterion (October 30, 1979, letter from H. R. Denton to all licensees).

Clarification 6

Consistent with Regulatory Guide 1.3 or 1.4 source terms, provide information on the predicted personnel exposures based on person-motion for sampling, transport and analysis of all required parameters.

Position 6

As shown in Table II.B.3-1, whole body exposure and extremity exposure* would be less than 0.98 R and 16 R, respectively. Individual exposure would be at even lower levels because more than one person would be performing the required tasks.

Criterion 7

The analysis of primary coolant samples for boron is required for PWRs. (Note that Regulatory Guide 1.97, Rev. 2, specifies the need for primary coolant boron analysis capability at BWR plants).

Clarification 7

PWRs need to perform boron analysis. The guidelines for BWRs are to have the capability to perform boron analysis, but they do have to do so unless boron was injected.

Position 7

Boron concentration in the primary coolant can be determined by the carminic acid method of analysis on the diluted reactor coolant samples. Reactor coolant flows into the PASS through a ball valve bored out to 0.10 ml. The valve is rotated 90 degrees, and a syringe is used to flush the sample plus 10 ml of demineralized water into a sample bottle. The bottle is transported to the laboratory in a lead-shielded cask. The sample is handled in the laboratory with tongs and lead brick shielding to reduce radiation exposure.

Criterion 8

If in-line monitoring is used for any sampling and analytical capability specified herein, the licensee shall provide backup sampling through grab samples, and shall demonstrate the capability of analyzing the samples. Established planning for analysis at offsite facilities is acceptable. Equipment provided

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* The referenced exposures are based on a power level of 3,323 MWt. Due to power uprate to 3,467 MWt, the exposure values shown must be multiplied by a factor of 1.0136.

for backup sampling shall be capable of providing at least one sample per day for 7 days following onset of the accident, and at least one sample per week until the accident condition no longer exists.

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TABLE II.B.3-1

TIME AND DOSE PROJECTIONS FOR PASS SAMPLING, TRANSPORT, AND ANALYSIS

Task	Time (min)		Exposure[2][3] (mR)			Notes
	Start	Stop	Persons[1]	Whole Body	Extremities	
Decision to take sample	0	0	N/A	N/A	N/A	Assumes TSC and OSC activated and sample room habitated
Read containment atmosphere H_2 levels in control room	0	5	1	NEG	N/A	
Operate control panel for dilute reactor coolant	0	20	4	9.5	9.5	6" lead shielding
Transport dilute reactor coolant to laboratory	20	42	2	3.6+1	2.5+2	6" lead shielding (Max) 3" lead shielding (Min)
Prepare coolant for isotopic	42	44.5	1	5.0-1	6.3+1	4" lead glass for W.B. (Max) 1/2" lead shielding (Min)
Perform isotopic analysis of coolant	44.5	49.5	1	2.2-4	2.0-1	
Analyze coolant for boron	49.5	54.2	1	2.5	8.6+1	4" lead glass + 2" lead for W.B. 1/2" lead shielding
Prepare sample panel for containment atmosphere	20	20	2	0	0	6" lead shielding
Operate control panel for containment atmosphere	20	35	2	4.8+0	4.8+0	2" lead shielding
Transfer containment atmosphere to small cask	35	39.8	1	1.8+1	2.4+2	2" lead shielding
Transport containment atmosphere to laboratory	39.8	58.5	2	5.8+2	2.4+3	3" lead shielding
Prepare containment atmosphere for isotopic	58.5	63.9	1	3.3	5.2+2	4" lead glass & 2" lead for W.B. (Max) 1/2" lead shielding (Min)
Perform isotopic analysis of containment atmosphere	63.9	68.9	1	2.7-3	2.0-0	
Operate control panel for dissolved gas	39.8	109.8	3	2.5+1	2.5+1	6" lead shielding

TABLE II.B.3-1 (Cont'd.)

Task	Time (min)		Exposure[2][3] (mR)			Notes
	Start	Stop	Persons[1]	Whole Body	Extremities	
Operate control panel for 10-ml reactor coolant	109.8	119.8	3	3.6+0	3.6+0	6" lead shielding
Transport 10-ml reactor coolant to laboratory	119.8	179.1	3	6.0+1	3.8+3	6" lead shielding (Max) 2" lead shielding (Min)
Analyze 10-ml reactor coolant for chloride	179.1	183.6	1	2.4+2	8.1+3	4" glass lead & 2" lead for W.B. (Max) 1/2" lead shielding (Min)

(1) Number of persons performing particular task.
(2) Doses are based on the assumption that the decision to take a sample is made 1 hr after reactor scram.
(3) The exposure values shown in Table II.B.3-1 are based on a power level of 3,323 MWt. Due to a power increase to 3,467 MWt, the exposure values shown must be multiplied by a factor of 1.0136.

is the manual equivalent of the LLS relief logic. The plant EOPs incorporate this particular instruction of the EPGs.

Closure of the MSIVs is initiated when reactor water level reaches Level 1.

The SER specifies a recommended simmer margin of 120 psi. The plant vessel operating pressure is less than 1,020 psig, whereas the lowest spring set pressure is 1,165 psig. Consequently, for the limiting SRV under limiting operating pressure, a simmer margin of at least 120 psi is maintained.

II.K.3.17 REPORT ON OUTAGES OF EMERGENCY CORE COOLING SYSTEMS LICENSEE REPORT AND PROPOSED TECHNICAL SPECIFICATION CHANGES

FSAR Cross-Reference

Section 6.3 and the Technical Specifications

NUREG-0737 Position

Several components of the ECCSs are permitted by Technical Specifications to have substantial outage times (e.g., 72 hr for one diesel generator; 14 days for the HPCI system). In addition, there are no cumulative outage time limitations for ECCSs. Licensees should submit a report detailing outage dates and lengths of outages for all ECCSs for the last 5 yr of operation. The report should also include the causes of the outages (i.e., controller failure, spurious isolation).

The present Technical Specifications contain limits on allowable outage times for ECCSs and components. However, there are no cumulative outage time limitations on these same systems. It is possible that emergency core cooling equipment could meet present Technical Specification requirements but have a high unavailability because of frequent outages within the allowable Technical Specifications.

The licensees should submit a report detailing outage dates and length of outages for all ECCSs for the last 5 yr of operation, including causes of the outages. This report will provide the staff with a quantification of historical unreliability due to test and maintenance outages, which will be used to determine if a need exists for cumulative outage requirements in the Technical Specifications.

Based on the above guidance and clarification, a detailed report should be submitted. The report should contain (1) outage dates and duration of outages; (2) cause of the outage; (3) ECCSs or components involved in the outage; and (4) corrective action taken. Test and maintenance outages should be included in the above listings which are to cover the last 5 yr of operation.

The licensee should propose changes to improve the availability of emergency core cooling equipment, if needed.

Applicant for an operating license shall establish a plan to meet these requirements.

Nine Mile Point Unit 2 Position

NMPC will report ECCS outages via LERs and Annual Summary Reports as required by Technical Specifications.

II.K.3.18 ADS ACTUATION LOGIC

FSAR Cross-Reference

Sections 6.3, 7.3

NUREG-0737 Position

The ADS actuation logic should be modified to eliminate the need for manual actuation to assure adequate core cooling. A feasibility and risk assessment study is required to determine the optimum approach. One possible scheme that should be considered is ADS actuation on low reactor vessel water level provided no HPCI or HPCS system flow exists and a low-pressure ECCS is running. This logic would complement, not replace, the existing ADS actuation logic.

Nine Mile Point Unit 2 Position

NMPC has participated in the BWROG evaluation of logic modifications to simplify ADS actuation.

Based on the BWROG design modifications found to be acceptable by the NRC staff, NMPC has removed the high drywell pressure trip in conjunction with the addition of a manual switch to inhibit ADS actuation (Option 2, NEDE-30045).

II.K.3.21 CORE SPRAY AND LPCI AUTO RESTART

FSAR Cross-Reference

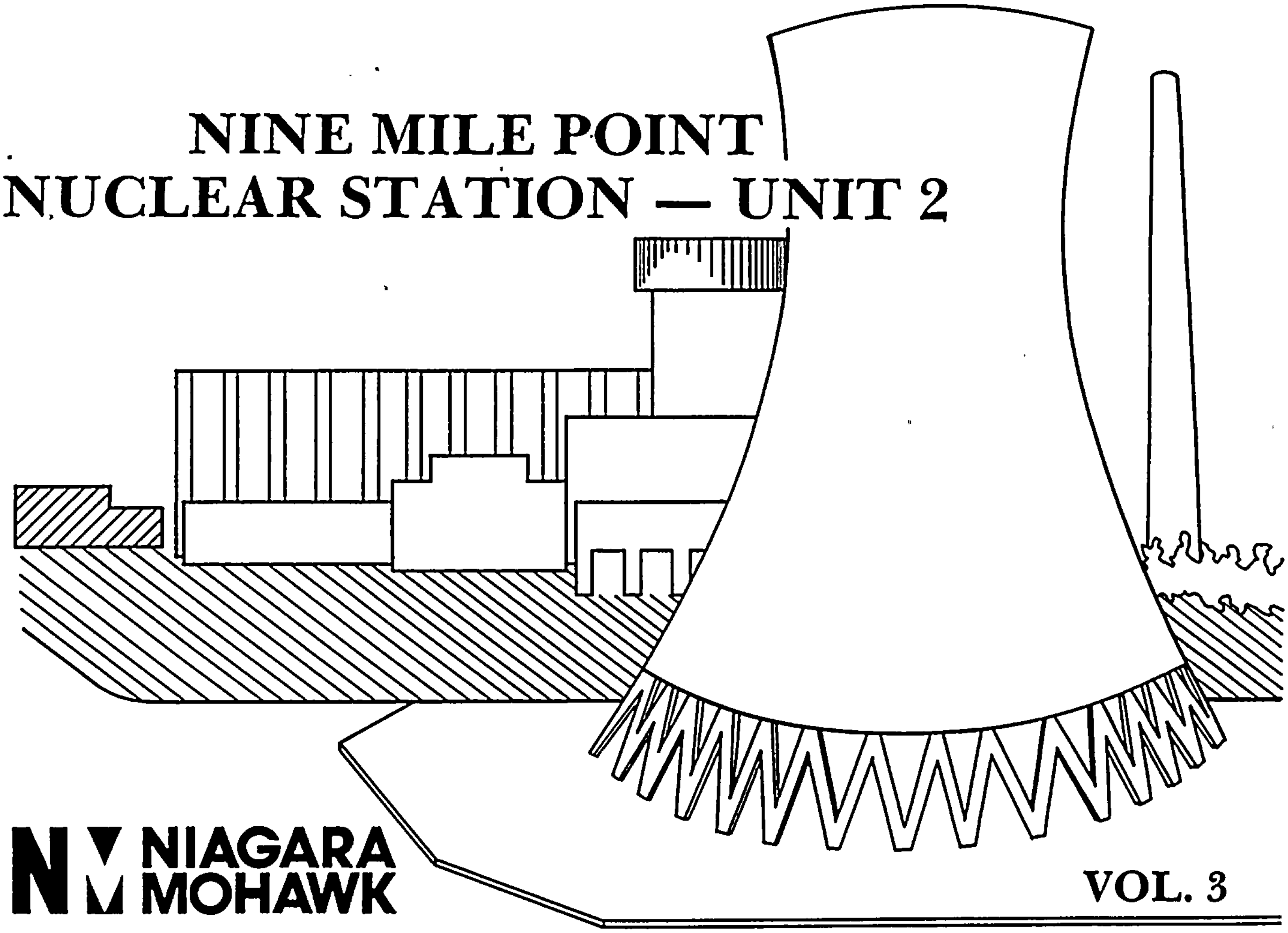
Sections 6.3, 7.3

NUREG-0737 Position

The core spray and 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

UPDATED SAFETY ANALYSIS REPORT

NINE MILE POINT NUCLEAR STATION — UNIT 2



NIAGARA MOHAWK

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TABLE 2.2-1

DAILY TRAFFIC VOLUME OF COUNTY HIGHWAYS IN THE VICINITY OF UNIT 2

Highway	Daily Traffic Volume	Date of Survey	Distance and Direction from Unit 2		
			km	Direction	mi
County Rte 29					
Between Lake Rd and Rte 1	1,729	April 1978	2	ESE	1.2
Between Rte 1 and 104	2,856	May 1978	3	SE	1.9
Between Rte 51A and 4	1,229	April 1979	7	SSE	4.3
County Rte 63 at Miner Rd	671	June 1978	4	SE	2.5
County Rte 1					
Between Lake Rd and Cremery Rd	1,341	April 1978	5	SW	3.1
Between Cremery Rd and Lakeview Rd	1,305	April 1978	4	SSW	2.5
Between Lakeview Rd and Rte 29	972	April 1978	4	S	2.5
Between Rte 29 and 44	1,312	July 1978	5	SE	3.1
Between Rte 44 and Hickory Grove Dr	1,312	July 1978	7	SSE	4.3
Between Hickory Grove Dr and Rte 104B	964	September 1978	8	SSE	5.0
Middle Rd between Rte 1 and Cremery Rd	885	April 1979	6	SW	3.7
Cremery Rd					
Between Rte 1 and Middle Rd	1,011	April 1979	4	SW	2.5
Between Middle Rd and Rte 104	1,558	October 1979	6	SW	3.7
Kocher Rd between Rte 1 and 104	3,063	April 1978	7	SSW	4.3
County Rte 53 between Rte 104 and 4	702	May 1980	7	SSW	4.3
Klocks Corner Rd between Rte 104 and 4	826	April 1979	7	SSW	4.3
County Rte 51A					
Between Rte 104 and 29	792	October 1979	6	S	3.7
Between Rte 29 and 51	595	October 1979	7	SSE	4.3
County Rte 51					
Between Rte 104 and 51A	205	October 1979	6	SW	3.7
Between Rte 51A and Mud Lake Rd	402	October 1979	9	SSE	5.6
County Rte 6					
Between Rte 1 and 104B	602	April 1979	8	ESE	5.0
Between Rte 104 and 64	702	April 1979	8	SE	5.0

SOURCE: Reference 1

TABLE 2.2-2

1978 FREIGHT TRAFFIC FOR LAKE ONTARIO AROUND OSWEGO HARBOR
(short tons)

Commodity	Total	Foreign		Domestic	
		Overseas Imports	Canadian Imports	Lakewise Receipts	Local
Barley and rye	5,880			5,880	
Rice	5	5			
Wheat	22,515			22,515	
Cocoa beans	559	559			
Fresh and frozen vegetables	6	6			
Animals and animal products	11	11			
Fresh fish, except shellfish	86		52		34
Crude petroleum	155,745	155,745			
Nonmetallic minerals	40,189		40,189		
Alcoholic beverages	131	131			
Miscellaneous food products	93	93			
Basic textile products	1,700	1,700			
Printed matter	14	2	12		
Radioactive materials, wastes	60	60			
Basic chemicals and products	37	37			
Paints	18	18			
Phosphatic chemical fertilizers	19,953		19,953		
Residual fuel oil	857,677		857,677		
Building cement	72,335	912	71,423		
Nonmetallic mineral products	2	2			
Iron and steel pipe and tube	62	62			
Unworked aluminum and alloys	37,949	15,958	21,991		
Fabricated metal products	40	40			
Machinery, except electrical	188	181	7		
Electrical machinery and equipment	3		3		
Motor vehicles, parts, and equipment	20	20			
Iron and steel scrap	581		581		
Nonferrous metal scrap	120	120			
TOTAL	1,215,979	175,662	1,011,888	28,395	34

SOURCE: Reference 2

TABLE 2.2-5

HAZARDOUS MATERIALS STORED/USED BY INDUSTRIES WITHIN 8 KM (5 MI)

Material	Industrial User	Storage on Premises (Max. at One Time)	Shipment				
			Mode	Average Size	Maximum Quantity Shipped	Frequency	Average Quantity Shipped/yr
Carbon dioxide	Alcan	65 tons	Truck	6 tons	10 tons	Weekly	250 tons
	FitzPatrick	26,000 lb	Truck	6,900 gal	6,900 gal	Infrequently, as needed system	Small quantities used to recharge
	NMP Unit 1	20,000 lb	Truck	5,000 lb	6,300 gal	Monthly	56,000 gal
Chlorine	Alcan	30 tons	Truck	1 (1 ton) cylinder	12 tons	Biweekly	350 cylinders
Helium	Alcan	1,917 ft^3	Truck	213 ft^3	9 cylinders (213 ft^3 each)	As needed	94 cylinders (20,022 ft^3)
Hydrochloric acid	Alcan	500 gal	Truck	55 gal	385 gal	Weekly	16,000 gal
	Oswego Wire	4 gal	Truck	4 gal	4 gal	Biannually	8 gal
Hydrogen	FitzPatrick	308,000 ft^3	Truck	128,000 ft^3	128,000 ft^3	Biweekly	5,881,928 ft^3
	NMP Unit 1	12,000 ft^3	Truck	24,000 ft^3	24,000 ft^3	Bimonthly	72,000 ft^3
	Oswego Wire	6,450 ft^3	Truck	215 ft^3	5,375 ft^3	10 Weeks	27,305 ft^3
Nitrogen	Alcan	50,000 lb	Truck	6,000 lb	6,000 lb	Triannually	18,000 lb
	FitzPatrick	10,000 gal	Truck	6,900 gal	6,900 gal	Monthly	129,000 gal
	NMP Unit 1	15,300 gal	Truck	6,300 gal	6,300 gal	Monthly	56,000 gal
	Oswego Wire	160,000 ft^3	Truck	145,000 ft^3	160,000 ft^3	Weekly	8,000,000 ft^3
Propane	FitzPatrick	1,000 gal	Truck	22,500 lb	22,500 lb	Monthly	3,000 gal
	Alcan	80,000 lb	Truck	7,000 lb	7,000 lb	Biweekly	200,000 gal
	Oswego Wire	500 gal	Truck	275 gal	450 gal	Biweekly	7,000 gal

TABLE 2.2-5 (Cont'd.)

Material	Industrial User	Storage on Premises (Max. at One Time)	Shipment				
			Mode	Average Size	Maximum Quantity Shipped	Frequency	Average Quantity Shipped/yr
Sodium hydroxide	FitzPatrick	5,000 gal	Truck	3,000 lb	3,000 lb	Monthly	246,000 lb
	Alcan	3,000 gal	Truck	55 gal	1,000 gal	Monthly	10,000 gal
	NMP Unit 1	165 gal	Truck	55 gal	55 gal	As needed	100 gal
	Oswego Wire	385 gal	Truck	55 gal	55 gal	Monthly	660 gal
Sulfuric Acid	Alcan	375 gal	Truck	375 gal	375 gal	As needed	4,875 gal
	FitzPatrick	5,000 gal	Truck	3,000 lb	3,000 lb	Monthly	396,000 lb
	NMP Unit 1	165 gal	Truck	55 gal	55 gal	As needed	100 gal
	Oswego Wire	110 gal	Truck	55 gal	55 gal	Bimonthly	1,320 gal
Isopropyl alcohol	Oswego Wire	110 gal	Truck	55 gal	55 gal	Biweekly	1,430 gal

TABLE 2.2-6

1978 LAKE ONTARIO CARGO TRANSPORT FOR OSWEGO HARBOR
(short tons)

Hazardous Commodity Designation	Commodity Description	Total	Foreign Imports	Domestic Receipts
1311	Crude petroleum	155,745	155,745	
2810	Sodium hydroxide			
2811	Crude products from coal tar, petroleum, and natural gas			
2812	Dyes, organic pigment, tanning materials			
2813	Alcohols			
2817	Benzene and toluene			
2818	Sulfuric acid			
2819	Basic chemicals and products	37	37	
2841	Soap, detergents, and cleaning preparations			
2861	Gum and wood chemicals			
2871	Nitrogenous chemical fertilizers			
2872	Potassic chemical fertilizers			
2873	Phosphatic chemical fertilizers	19,953	19,953	
2879	Fertilizers not elsewhere classified			
2891	Miscellaneous chemical products			
2911	Gasoline			
2912	Jet fuel			
2913	Kerosene			
2914	Distillate fuel oil			
2915	Residual fuel oil	857,677	857,677	
2916	Lubricating oils and greases			
2917	Naphtha, solvents			
2918	Asphalt, tar, pitches			
2920	Coke			
2921	Liquified petroleum gases, natural gases			
2991	Petroleum and coal products not elsewhere classified			
Hazardous commodities		1,033,412	1,033,412	0
All commodities		1,215,979	1,187,550	28,429
All commodities for entire Lake Ontario		49,887,155	49,501,197	385,958

TABLE 2.3-6

FASTEST MILE WIND SPEEDS AT OSWEGO, SYRACUSE, AND ROCHESTER

Station	Year	True Speed (Fastest Mile)		Anemometer Height		Fastest Mile 10 m (33 ft) or 30 m (98 ft)			
		m/sec	mph	m	ft	m/sec	mph	m/sec	mph
Oswego Weather Bureau Office[1]	1893	28	62	26	84			38	84
	1926	28	62	26	85			37	83
Oswego NWS Station[2]	1964	38	85	20	65	34	77		
Rochester Weather Bureau Office[3]	1922	27	60	31	102			34	76
Rochester NWS Rochester Airport[4]	1950	33	73	21	69	29	65		
	1979	27	60	6	20	30	66		
Syracuse Weather Bureau Office[5]	1921	34	75	34	113			41	92
Syracuse NWS Hancock Airport[6]	1954	28	63	22	72	25	56		
	1967	28	62	6	21	30	67		

[1] Period of record 1887 through 1952
[2] Period of record 1953 through 1967, generally April 10 through December 15
[3] Period of record 1887 through 1940
[4] Period of record 1941 through 1979
[5] Period of record 1903 through 1940
[6] Period of record 1941 through 1979

Source: Reference 57

TABLE 2.4-6

PREDICTED HOURLY VALUES OF PRESSURE, WIND SPEED, AND WIND DIRECTION
FOR PMWS ON LAKE ONTARIO

Sequential Time During Storm (hr)	Time (EST)	P1 (mb)	S1		D1 (deg)	P2 (mb)	S2		D2 (deg)	P3 (mb)	S3		D3 (deg)	P4 (mb)
			km/hr	mph			km/hr	mph			km/hr	mph		
0	0800	989	21	13	170	996	11	7	170	1005	11	7	170	1011
1	0900	983	32	20	170	992	19	12	170	1000	11	7	170	1006
2	1000	2980	55	34	170	987	22	14	170	995	10	6	170	1003
3	1100	977	77	48	170	982	37	23	170	986	18	11	170	992
4	1200	978	64	40	160	979	71	44	170	986	31	19	160	993
5	1300	970	53	33	160	975	72	45	170	981	64	40	170	990
6	1400	966	53	33	170	972	42	26	170	978	87	54	170	984
7	1500	964	64	40	200	968	53	33	170	974	58	36	160	983
8	1600	962	69	43	210	966	61	38	190	972	55	34	170	978
9	1700	961	79	49	220	964	64	40	210	968	63	39	170	974
10	1800	961	113	70	230	963	74	46	210	964	63	39	200	970
11	1900	964	124	77	260	964	108	67	220	963	74	46	210	966
12	2000	968	143	89	270	961	117	73	260	961	79	49	250	961
13	2100	971	148	92	270	964	130	81	280	961	108	67	260	959
14	2200	972	158	98	270	968	148	92	280	962	127	79	270	958
15	2300	976	161	100	280	972	151	94	280	968	140	87	280	957
16	2400	982	161	100	280	972	161	100	290	970	146	91	290	958
17	0100	986	148	92	290	979	161	100	280	973	161	100	270	962
18	0200	990	143	89	290	983	148	92	280	975	161	100	290	965
19	0300	993	143	89	300	988	148	92	290	982	156	97	290	968
20	0400	997	132	82	300	991	137	85	290	986	148	92	290	972
21	0500	999	119	74	290	993	135	84	290	989	145	90	280	979
22	0600	1000	113	70	300	999	130	81	290	993	142	88	290	984
23	0700	1002	111	69	310	1000	122	76	310	997	132	82	300	988
24	0800	1004	74	46	330	1001	100	62	310	999	116	72	300	990
25	0900	1006	60	37	320	1003	72	45	330	1000	93	58	330	995
26	1000	1009	63	39	340	1006	60	37	320	1002	69	43	320	999
27	1100	1011	48	30	320	1008	56	35	330	1005	64	40	320	1000

KEY: P = Pressure
S = Wind speed
D = Wind direction

NOTE: Numbers shown in pressure, speed, and distance columns indicate region of Lake Ontario shown on Figure 2.4-6.

TABLE 2.4-7

BRETSCHNEIDER'S JOINT DISTRIBUTION OF $\overline{H}$ AND $\overline{T}$ FOR ZERO CORRELATION

Number of Waves per 1,000 Consecutive Waves for Various Ranges in Height and Period

Range in Relative Height (H/H)	Range in Relative Period T/$\overline{T}$											
	0-0.2	0.2-0.4	0.4-0.6	0.6-0.8	0.8-1.0	1.0-1.2	1.2-1.4	1.4-1.6	1.6-1.8	1.8-2.0	0-2.0	Accumulative
0-0.2	0.03	0.50	2.05	4.86	7.68	8.09	5.31	1.92	0.34	0.03	30.81	30.8
0.2-0.4	0.10	1.41	5.81	13.78	21.76	23.92	15.05	5.44	0.98	0.07	88.32	119.13
0.4-0.6	0.14	2.06	8.54	20.23	31.95	33.65	22.10	7.99	1.44	0.11	128.21	247.34
0.6-0.8	0.16	2.40	9.91	23.48	37.08	39.06	25.65	9.27	1.67	0.12	148.80	396.14
0.8-1.0	0.16	2.40	9.92	23.51	37.13	39.11	25.69	9.28	1.67	0.12	148.99	545.13
1.0-1.2	0.15	2.14	8.87	21.02	33.19	34.97	22.96	8.30	1.49	0.11	133.20	678.33
1.2-1.4	0.12	1.74	7.21	17.07	26.96	28.40	18.65	6.74	1.21	0.09	108.19	786.52
1.4-1.6	0.09	1.30	5.37	12.72	20.09	21.16	13.90	5.02	0.90	0.07	80.62	867.14
1.6-1.8	0.06	0.90	3.72	8.82	13.93	14.67	9.64	3.48	0.63	0.05	55.90	923.04
1.8-2.0	0.03	0.48	1.99	4.72	7.45	7.85	5.15	1.86	0.33	0.03	29.89	952.93
2.0-2.2	0.03	0.42	1.72	4.09	6.45	6.80	4.47	1.61	0.29	0.02	25.90	978.83
2.2-2.4	0.01	0.18	0.76	1.80	2.84	2.99	1.97	0.71	0.13	0.01	11.40	990.23
2.4-2.6	0.01	0.09	0.39	0.93	1.47	1.55	1.02	0.37	0.07		5.90	996.13
2.6-2.8		0.04	0.18	0.43	0.67	0.71	0.47	0.17	0.03		2.70	998.83
2.8-3.0												
0-3.0	1.09	16.06	66.44	157.46	248.65	262.93	172.03	62.16	11.18	0.83		
Accumulative	1.09	17.15	83.59	241.05	489.70	752.63	924.66	986.82	998.00	998.83		

KEY: $\underline{T}$ = Wave period
$\overline{T}$ = Mean wave period
$\underline{H}$ = Wave height
$\overline{H}$ = Mean wave period

SOURCE: Reference 8

TABLE 2.4-9

PUBLIC WATER SUPPLY DATA

Distance From Site (mi)	Number[1]	Town	Estimated Population Served (1980)	Average Output (mgd)	Source of Water Supply
0 to 10	1	Onondaga County Water Authority	40,000	22-24[2]	Lake Ontario (intake at Oswego)
	2	Oswego	30,270	14[2]	Lake Ontario (intake at Oswego)
10 to 20	3	Village of Mexico	1,725	0.24[3]	3 wells: 2 40-ft deep, 1 38-ft deep; average yield 275 gpm; probably in alluvium
	4	Village of Pulaski	2,700	0.025[3]	Springs
	5	City of Fulton	15,000	2.0[3]	12 wells: 30- to 70-ft deep; in alluvium
	6	Village of Sandy Creek	1.435	0.33[3]	2 wells: 21-ft deep, average yield 275 gpm; probably in alluvium
20 to 30	7	Village of Central Square	1,427	0.96[3]	2 wells: 1 24-ft deep, 1 10-ft deep; in alluvium
	8	Town of Orwell	250	0.015[3]	Spring
	9	Village of Phoenix	2,600	1.0[3]	2 wells: 1 25-ft deep, 1 45-ft deep; average yield 400 gpm; probably in alluvium
	10	Baldwinsville	10,000	1.0	4 wells: 1 93-ft deep, yield 1,500 gpm; 3 shallow wells, in alluvium
	11	Fairhaven	765	0.15	Spring; 1 well 46-ft deep, yield 300 gpm
	12	Cato	500	0.033	3 wells: 2 55-ft deep, 1 70-ft deep; average yield 350 gpm
	13	Wolcott	1,640	0.220	Lake Ontario
	14	Adams	1,735	0.3	Spring, infiltration gallery
	15[4]	Red Creek[4]		0.03	Wells and springs
	16	Constantia	3,060	0.20	Spring-fed reservoir

(1) Refer to Figure 2.4-9.
(2) Average output for 1982; current design capacity is 36 mgd for both Onondaga County Water Authority and City of Oswego.
(3) Average output based upon currently available data, Oswego County Public Health Department.
(4) Red Creek is an industrial, not public, supply source.

TABLE 2.4-10

DOMESTIC WELLS WITHIN 2-MI RADIUS OF PLANT*

Well No.	Well Depth (ft)	Approx. Land Surface El (ft)	Depth to Water Level (ft)	Approx. El of Water Level (ft)	Type of Well	Estimated Pumpage Rate (gpd)	Name of Owner
1	18	275	10	265	Dug, 3' diam.	Not in use	Jack Timon
2	43	275			Drilled	150	Jack Timon
3	43	275	7	268	Drilled	300	E. Roy
4	25	280	8	272	Drilled	300	J. Roy
5	28	280	4	278	Drilled	100	Mason
6	30	275			Drilled	225	Barns
7	45	275	8	267	Drilled, 6"	150	Malone
8	40	270	8	262	Drilled, 6"	(for lawn only)	Malone
9	30	270	8	262	Drilled, 6"	(for lawn only)	Malone
10	35	270	8	262	Drilled, 6"	975	Malone
11	40	270	8	262	Drilled, 6"	975	Malone
12	60	285			Drilled	Not in use	Hudson
13	60	275			Drilled	375	Upcraft
14	25	275	Near to Surface		Drilled	Not in use	R. Fauata
15	80	280			Drilled	150	R. Dickenson-Brown
16	20	285			Dug	225	R. Dickenson-Brown
17	38	275	11	264	Drilled	1,000	J. E. Reardon
18	60	285	25	260	Drilled	375	J. Murray
19		280				500	Donahue
20		275				Not in use	Ketchem
21	12	260	5	255	Dug	400	R. Palmateer
22	25	255	8	247	Drilled, 6"	Up to 1,500	Malone (campground)
23	70	310	10	300	Drilled	Not in use	D. Stevens
24	70	310			Drilled	Not in use	D. Stevens
25	30	290			Dug	Not in use	Simineau
26	12	290	5	285	Dug	100	Simineau
27	80	285			Drilled	Not in use	Simineau
28	15	285	5	280	Dug	Not in use	Simineau
29		285	0	285	Dug	Not in use	Simineau
30	20	285	0	265	Dug	100	Whiting
31	40	285	0	285	Drilled	Not in use	Whiting
32		290	0	290	Dug	Not in use	J. McLean
33	42	330	30	300	Dug	375	Adkins
34		330			Dug	Not in use	C. Upcraft
35		330			Dug	Not in use	C. Upcraft
36		340				Not in use	Pryor
37	60	350	7	343	Dug	Not in use	R. W. Rasmussen
38	18	310			Dug	225	J. O'Conner
39	22	320	18	302	Dug	100	E. LaBouef
40		330			Drilled	150	F. Peck
41	42	330	12	318	Drilled	150	Randall

Table 2.4-10 (Cont'd.)

Well No.	Well Depth (ft)	Approx. Land Surface El (ft)	Depth to Water Level (ft)	Approx. El of Water Level (ft)	Type of Well	Estimated Pumpage Rate (gpd)	Name of Owner
42	100	330			Drilled	300	Pitcher
43	45	330	17	312	Drilled	700	Hopkins and Kersey
44		325			Drilled	Not in use	Unknown
45	15	325	10	315	Dug	100	E. Whaley
46		325	5	320	Drilled	50	L. Whaley
47	12	325			Dug	300	L. Whaley
48	25	325	12	313	Drilled	375	Dickenson
49	6	325	3	322	Dug	300	R. LaBouef
50	38	340	21	319	Drilled	375	Carpenter
51	28	330	11	319	Drilled	375	Nelson
52		330				Not in use	Unknown
53	22	335	9	327	Dug	450	M. Coe
54	60	340	25	315	Drilled	375	Upcraft
55	27	335			Dug	100	F. A. Newstead
56	30	340	12	328	Drilled	150	L. F. Dillenbeck
57	30	340	15	325	Drilled, 6"	150	Lawton
58	30	340	15	325	Drilled, 6"	300	Woods
59		340			Dug	150	Unknown
60	30	340	15	325	Drilled	600	Goodness
61	30	345	27	318	Drilled	50	Vandish
62	39	340	15	325	Drilled, 4"	500	Richardson
63	30	335	24	311	Drilled	300	Albright
64	15	325			Drilled	Not in use	Unknown
65	25	340			Dug	Not in use	Prosser (temp. vacant)
66	65	325			Drilled	500	Read and Ocheebein
67		325	3	320	Drilled, 6"	500	LaBouef
68		325			Dug	375	Wills
69	58	330	8	322	Drilled, 6"	200	G. Drake
70	25	335	0	335	Dug	300	C. Drake
71	30	325	3	322	Dug	Not in use	Brandon (temp. vacant)
72	15	330	3	327	Dug	Not in use	Klesinger (temp. vacant)
73	65	335			Drilled	400	Conroy
74	32	340	15	325	Dug	400	S. McLean
75	44	330	14	316	Drilled	50	E. Patrick
76	22	335	10	325	Dug	800	France
77	30	315	25	290	Drilled, 6"	300	Whaley
78		290			Drilled, 3"	150	F. O'Conner
79	9	290	4	286	Dug	Not in use	F. O'Conner
80	42	290			Drilled	225	L. Whaley
81		75				Not in use	Unknown
82	54	275	12	263	Drilled	375	J. O'Conner
83	45	270	8	262	Drilled	100	J. T. O'Conner
84	18	280	8	272	Dug	100	E. Henry

Table 2.4-10 (Cont'd.)

Well No.	Well Depth (ft)	Approx. Land Surface Elevation (ft)	Depth to Water Level (ft)	Approx. El of Water Level (ft)	Type of Well	Estimated Pumpage Rate (gpd)	Name of Owner
85	6	275	4	271	Dug	Not in use	E. Hutchins
86		280			Dug	10,500+	C. Parkhurst
87	26	300	15	285	Dug	8,500+	K. Parkhurst
88		315	20	295	Dug	100	M. Goewey
89	6	320	0	320	Dug	850+	J. Parkhurst
90	10	325	3	322	Dug	4,200+	Woolson
91	90	325			Drilled	150	W. Woolson
92		290			Dug	300+	King
93	90	295			Drilled	375	King
94	31	300	25	275	Drilled	400	Barton
95	12	280	2	278	Dug	225	Parkhurst
96		280				Not in use	Unknown
97	8	270	0	270	Dug	225	Bellemo
98	24	275	3	270	Dug	150	R. Fox
99	30	280	15	265	Dug	375	Fox
100		265				Not in use	Jansen
101	10	300	3	297		Not in use	Unknown (summer home)
102		285				Not in use	Unknown (summer home)

* For location of wells, see Figure 2.4-10.

TABLE 2.4-11

PUBLIC AND PRIVATE WATER SUPPLY SYSTEMS IN THE UNITED STATES DRAWING FROM LAKE ONTARIO WITHIN 80 KM (50 MI) OF UNIT 2

Map No.*	Name of System (Intake County)	Distance (km/mi) and Direction from Unit 2	Distance (km/mi) by Water from Unit 2	Average Withdrawal Rate 1980-81		Type of Use	Population Served	Production Capacity		Comments
				cu m/day	mgd			cu m/day	mgd	
1	Rochester Gas & Electric - Robert E. Ginna Nuclear Power Plant (Wayne County)	78/49 WSW	78/49	2,180,160	576.00	Industrial cooling	-	2,180,160	576.00	-
2	Ontario Town Water District (Wayne County)	74/46 WSW	74/46	11,355	3.00	Domestic, industrial	5,000	11,355	3.00	Expanded system startup summer 1981
3	Williamson Water District (Wayne County)	66/41 WSW	66/41	6,813	1.80	Domestic, industrial	4,700	14,762	3.90	Apr-Jun (avg) 4,921 cu m/day (1.3 mgd); Sep-Dec can reach 9,463 cu m/day (2.5 mgd)
4	Sodus Village (Wayne County)	58/36 WSW	58/36	984	0.26	Domestic, industrial	1,800	3,785	1.00	Jan-Jun lows of 265 to 492 cu m/day (0.07 to 0.13 mgd); Aug-Nov 1 highs of 3,747 cu m/day (0.99 mgd)
5	Sodus Point (Wayne County)	53/33 SWS	53/33	757	0.20	Domestic	4,500	2,839	0.75	Winter min. 454 cu m/day (0.12 mgd); peak in dry summer weather 1,703 cu m/day (0.45 mgd)

TABLE 2.4-11 (Cont'd.)

Map No.*	Name of System (Intake County)	Distance (km/mi) and Direction from Unit 2	Distance (km/mi) by Water from Unit 2	Average Withdrawal Rate 1980-81		Type of Use	Population Served	Production Capacity		Comments
				cu m/day	mgd			cu m/day	mgd	
6	Wolcott Village (Wayne County)	41/25 WSW	41/25	908	0.24	Domestic, Industrial	2,500	3,785	1.00	Avg. winter usage (Jan-Mar) approx. 681 cu m/day (0.18 mgd); avg. peak usage Jun-Nov 1,363 cu m/day (0.36 mgd)
7	NMPC Oswego Steam Station - Unit 5 (Oswego County)	15/10 WSW	15/10	1,558,814	411.84	Industrial cooling	-	1,558,814	411.84	-
8	NMPC Oswego Steam Station - Units 1-4 (Oswego County)	15/10 WSW	15/10	452,383	119.52	Industrial cooling	-	452,383	119.52	-
9	NMPC Oswego Steam Station - Unit 6 (Oswego County)	15/10 WSW	15/10	1,771,380	468.00	Industrial cooling	-	1,771,380	468.00	-
10	City of Oswego (Oswego County)	13/8 WSW	13/8	37,850	10.00	Domestic, industrial	32,000	60,560	16.00	Winter, 30,280 cu m/day (8 mgd); summer, 37,850 cu m/day (10 mgd)
11	Metropolitan Water Board of Onondaga County, Syracuse, NY (Oswego County)	13/8 WSW	13/8	90,840	24.00	Domestic, Industrial	120,000	136,260	36.00	Winter 75,700 cu m/day (20.0 mgd); summer, 98,410-105,980 cu m/day (26.0-28.0 mgd); to Onondaga County Water Authority; remainder to city of Syracuse

TABLE 2.4-11 (Cont'd.)

Map No.*	Name of System (Intake County)	Distance (km/mi) and Direction from Unit 2	Distance (km/mi) by Water from Unit 2	Average Withdrawal Rate 1980-81		Type of Use	Population Served	Production Capacity		Comments
				cu m/day	mgd			cu m/day	mgd	
12	NMPC Scriba, NY, Unit 1 (Oswego County)	-	750 ft (Unit 2 discharge to Unit 1 intake)	1,444,356	381.60	Industrial cooling	-	1,444,356	381.60	-
13	Power Authority of the State of New York, Scriba, NY (Oswego County)	-	3,500 ft (Unit 2 discharge to FitzPatrick intake)	2,158,358	570.24	Industrial cooling	-	2,158,358	570.24	-
14	Sacketts Harbor Village (Jefferson County)	49/31 NNE	51/32	568	0.15	Domestic	1,200	1,893	0.50	Withdrawals fluctuate in summer from 492 cu m/day (0.13 mgd) in Jun to 681 cu m/day (0.18 mgd) in Aug and Sep
15	Chaumont Village (Jefferson County)	60/37 NNE	61/38	265	0.07	Domestic	550	908	0.24	Winter (Dec-Mar) usage is approx. 189 cu m/day (0.05 mgd); summer usage (Jun-Sep) avg. 341 cu m/day (0.09 mgd)
16	Cape Vincent Village (Jefferson County)	65/41 N	65/41	757	0.20	Domestic	750	908	0.24	Withdrawals fluctuate between Jun and Sep from 473 to 1,136 cu m/day (0.125 to 0.3 mgd)

* Locations corresponding to map numbers are shown on Figure 2.3-17

TABLE 2.4-11 (Cont'd.)

SOURCES:

New York State Department of Health. Selected Public Water Supply Inventory. Albany, NY, July 22, 1981.

Personal communication between C. Gaye, Metropolitan Water Board of Onondaga County, Syracuse, NY, and C. S. Ellis, Stone & Webster Engineering Corporation, Boston, MA, August 11, 1981; February 2, 1982; and June 1, 1982.

Personal communication between Mrs. Frantz, Ontario Town Water District, Ontario, NY, and C. S. Ellis, Stone & Webster Engineering Corporation, Boston, MA, August 11, 1981.

Personal communication R. Walvoord, Williamson Water District, Williamson, NY, and C. S. Ellis, Stone & Webster Engineering Corporation, Boston, MA, August 11, 1981.

Personal communication between Mr. Wilkinson, City of Oswego Water Supply, Oswego, NY, and C. S. Ellis, Stone & Webster Engineering Corporation, Boston, MA, August 11, 1981.

TABLE 2.4-13

UNITED STATES IRRIGATION INTAKES ON LAKE ONTARIO WITHIN 80 KM (50 MI) OF UNIT 2

Farmer	Location of Intake (County)	Distance in km (mi) by Water from Discharge	Area in ha (acres)	Average Water Use		Total Water Use/ Application cu m (mgd)	Frequency of Application
				cm/ha (in/acre)	l/ha (gal/acre)		
J. Simplaar Mexico, NY	On Lake Ontario, between Demster Beach Road and Hickory Grove Road (Oswego County)	8.2 (5.1)	24.3 (60)	7.6 (3)	762,000 (81,463)	18,510 (4.89)	Once per year, 1 year in 4
L. Hurlbutt Mexico, NY	South side of Butterfly Swamp (Oswego County)	9.9 (6.2)	8.1 (20)	7.6 (3)	762,000 (81,463)	6,170 (1.63)	Once per year, dry weather only
D. Ouellette Sterling, NY	East Branch of Sterling Creek (Cayuga County)	38.6 (24.1)	28.3 (70)	5.1 (2)	508,000 (54,308)	14,380 (3.80)	Once per year, 1 year in 5

NOTE: Irrigated crop at each location, apples.

SOURCES: Personal communication between J. Simplaar, Mexico, NY, and C. S. Ellis, Stone & Webster Engineering Corporation, Boston, MA, June 10, 1981.

Personal communication between L. Hurlbutt, Mexico, NY, and C. S. Ellis, Stone & Webster Engineering Corporation, Boston, MA, June 9, 1981.

Personal communication between D. Ouellette, Sterling, NY, and C. S. Ellis, Stone & Webster Engineering Corporation, Boston, MA, June 15, 1981.

TABLE 2.4-14

CANADIAN IRRIGATION INTAKES ON LAKE ONTARIO WITHIN 80 KM (50 MI) OF UNIT 2

Name	Location	Rate Not to Be Exceeded		Amount Not to Be Exceeded	
		lpm	gpm	cu m/day	mgd
Picton Golf and Country Club	Hallowell Township	454	120	189	0.05
G. Vader	Athol Township	2,044	540	1,136	0.30
K. Perry	Athol Township	1,249	330	1,173	0.31
R. K. Hicks	North Marysburgh Township	1,423	376	2,044	0.54
Windy Acres Farms	Hallowell Township	1,635	432	1,363	0.36
B. McArthur (West Lake Farms Ltd.)	Hallowell Township	908	240	984	0.26
C. Foster	Hallowell Township	1,703	450	1,438	0.38
G. Bosma	South Marysburgh Township	568	150	530	0.14
Point Pleasant Farms, Ltd.	North Marysburgh Township	1,703	450	2,460	0.65
Waupoos Canning Co., Ltd.	North Marysburgh Township	1,703	450	2,044	0.54
J. Carter	North Marysburgh Township	2,502	661	2,233	0.59
R. & K. Carson	North Marysburgh Township	1,703	450	1,438	0.38
E. Vowinckel	South Marysburgh Township	2,275	601	3,255	0.86
R. R. Dodokin	South Marysburgh Township	454	120	681	0.18
W. Hicks	South Marysburgh Township	908	240	227	0.06
C. A. McCormack	South Marysburgh Township	840	222	757	0.20
Cataraqui Golf and Country Club	Kingston	1,590	420	2,271	0.60

SOURCE: Ontario Ministry of the Environment. Data on Public and Private Water Supply Systems Drawing From Lake Ontario. Kingston, Ontario, July 24 and August 20, 1981.

TABLE 2.4-15

SUMMARY OF RESULTS ANALYSIS OF BUILDING FLOODING DUE TO PMP BASED ON HYDROMET 51 AND 52

Building	Door	Length (ft)	Total Flow (ft^3)	Bldg. Depth (in)	Distribution of Flow/Flux
Diesel generator	Stop logs	11 15	404.2 551.1	9	Evenly over floor; provided caulking material to make stop logs watertight.
Control building	C261-29 C261-24 C261-31	7 3 3	257.2 110.2 110.2	1	C261-29 and C261-31 flows will combine and spread through El 261 corridors and into some rooms. C261-24 will flow down stairwell and spread through corridors on lower floors. No fix is required.
Auxiliary bay south	SA262-3	3	63.6	1	Flow will be confined to auxiliary bay stairwell area. No fix is required.
Electrical tunnel - south area	ET262-4 ET262-3	3 3	63.6 63.6	<1 <1	Flow will be spread evenly through each tunnel. Flow will be spread evenly through each tunnel.
Electrical tunnel - north area	ET261-1 ET261-2	3 3	102.6 102.6	<1 <1	Flow will be spread evenly through each tunnel. Flow will be spread evenly through each tunnel.
Auxiliary bay north	NA262-1	3	63.6	1	Flow will be confined to auxiliary bay stairwell area. No fix is required.
RB railroad track bay	RR-261-1	17	581.5	4	Doors are equipped with weatherstrip and 1/16" neoprene loop. The doors are airtight; water leakage will be negligible. No safety-related equipment is in this area.
Standby gas treatment	SG261-2 SG261-1 SG261-6	8 8 3	273.6 273.6 102.6	4	Flow is spread evenly throughout the building (see Note 5).
Service water pump room (north) from auxiliary boiler room	AB261-3	3	408.1	4.4	Flow into auxiliary boiler building is distributed into the pumphouse if AB261-3 is open.

NOTES:

1. Use hydrographs from Calculation No. 12177-WH(B)-062 for water surface elevations.
2. Door sill elevation is in the door identification number, e.g., SG261-1, where "261" is the door sill elevation.

TABLE 2.4-15 (Cont'd.)

3. Calculation method for in-flow through doorways from PMF submerged orifice discharge equation:
   $Q = CA\sqrt{2gh}$
   Q = Flow (cfs)
   C = Discharge coefficient = 0.6
   A = Cross-sectional area of flow = length of door (L) x crack width of door opening (1/16 in due to flow "necking")
   g = Gravity
   h = Headwater surface elevation on exterior of door
4. No credit has been taken in the analysis for sump water retainage or sump pump operation.
5. Since equipment structural pad height is 6 in, resultant building water depths less than this are acceptable.

TABLE 2.5-3

IN SITU STRESS MEASUREMENTS BY OVERCORING

No.	Reference	Location	Horizontal Stress Maximum (psi)	Horizontal Stress Minimum (psi)	Average Orientation Maximum Horizontal Stress	Depth of Measurement (ft)
212	Hooker and Johnson	Proctor, VT				
		Borehole No. 1	1,427	470	N5E	0.5-1.8
		Borehole No. 2	1,132	517	N24W	0.5-1.8
213	Kaprowski	Sterling, NY				
		Borehole No.1	1,200	850	N20W	33.2
			1,500	950	N50W	42.8
			1,450	750	N70W	46.7
			1,300	650	N50W	59.3
		(Average)	1,360	800	N50W	
214	Franklin and Hungr	Scarborough, Ont	245	230	N90E	230
215	Dames & Moore	Somerset, NY				
		Borehole No. 1	640	460	N15W	27.7
			450	40	N45W	31.2
			295	60	N60W	32.6
			340	170	N15W	71.3
		(Average)	430	180	N34W	
		Borehole No. 2	450	240	N15W	
			265	-40	N10W	
			500	140	N15E	
		(Average)	400	110	N3W	
216	Palmer and Lo	Thorold, Ont.				
		Borehole No.1	1,880	1,750	N27W	41.7
			1,310	756	N88W	51.0
			1,180	956	N62E	53.1
			1,300	966	N76E	56.6
			2,130	1,600	N60E	60.0
			2,130	1,620	N58E	61.0
			2,000	961	N56E	65.0
			1,520	990	N60E	81.0
		(Average)	1,680	1,200	N60E	
217	Lo	Mississauga, Ont.	NA	1,100	N-NE[1]	<85 ft[1]
218	Morton et al					
217	Lo	Duffin Creek, Ont.	NA	1,000	N-NE[1]	<85 ft[1]

TABLE 2.5-3 (Cont'd.)

No.	Reference	Location	Horizontal Stress Maximum (psi)	Horizontal Stress Minimum (psi)	Average Orientation Maximum Horizontal Stress	Depth of Measurement (ft)
218	Morton et al					
217	Lo	Wesleyville, ONT.	1,600	1,400	N13W	<85 ft[1]
218	Morton et al					
120	Sbar and Sykes	Niagara, NY Borehole No. 1[2]	1,000	-10	N34-55E	6-70
219	American Falls International Board	Borehole No. 2[2]	870	-330	N34-55E	6-70
214	Franklin and Hungr	Niagara, ONT. Borehole No. 1	1,495	NA	NA	115
220	Goldberg et al	Rochester, NY Borehole No. 1	800	500	N76E	24.5
			1,600	1,000	N80W	35.9
			2,000	600	N80W	40.8
			4,000	1,400	N78W	45.1
			4,300	1,500	N86E	46.4
		(Average)	2,540	1,000	N88W	
		Borehole No. 2[3]	900	-700	N10E	32.2
			1,000	-208	N62E	37.3
			1,900	-900	N18E	46.1
			2,400	-400	N39E	50.6
			1,700	300	N11E	55.4
		(Average)	1,580	-381	N28E	
		Borehole No. 3	3,000	2,000	N76E	19.2
			3,000	1,400	N32W	40.5
			-700	-1,400	N38W	48.2
			1,400	900	N22W	51.2
		(Average)	1,675	725	N49W	
81	NYSE&G	New Haven, NY Borehole G-84	250	157	N78E	21.7
			754	440	N55W	24.5
			275	-71(T)	N21E	30.3
			708	510	N84W	38.0
			1,500	516	N40W	42.3
			767	438	N13W	50.6
		(Average)	709	332	N16W	

TABLE 2.5-3 (Cont'd.)

No.	Reference	Location	Horizontal Stress Maximum (psi)	Horizontal Stress Minimum (psi)	Average Orientation Maximum Horizontal Stress	Depth of Measurement (ft)
		Borehole G-85	356	94	N30E	21.2
			156	13	N04W	22.8
			223	38	N14E	25.5
			263	-84(T)	N29W	29.1
			693	259	N49W	35.6
			546	248	N48W	37.5
		(Average)	373	95	N14W	
		Borehole G-86	1,301	666	N38W	21.8
			443	263	N42W	31.2
			481	460	Hydrostatic	35.4
			900	772	N04W	38.7
			731	452	N20W	41.9
		(Average)	771	523	N26W	

(1) Orientation and depth reported by Morton et al[218].
(2) Sites close to the Niagara Gorge; orientation and magnitude of stress may be influenced by this fact. Note Borehole No.2 is closer to gorge.
(3) Data from Borehole No. 2 may be influenced by effects associated with a nearby open vertical fracture.

NOTE: Positive numbers indicate compressive stresses; negative numbers indicate tensile stresses.

TABLE 2.5-4

OC-100 OVERCORE RESULTS

Test	Depth[1]	Rock Type	P (psi)	Q (psi)	Θp	x 10^6 psi E_{AVG}	E_1	E_2	E_3	Remarks
1	11' 2"	Massive sandstone	-162	-318	N43E	4.7	4.7	4.4	5.1	
5	43' 6"	Shale/silty sandstone/silty sandstone with shaly laminae	+419	+210	N20E	2.9[2]	-	-	-	Modulus assumed from Test No. 6; core fractured at 43' 7" when recovered.
6	45' 0"	Graywacke	+395	+38	N37E	2.9	2.9	2.9	2.8	
9	49' 3"	Silty sandstone, interbedded shale, and shale clasts	+129	-331	N35E	3.1	3.2	3.2	2.8	Biaxial core fractured 1,000 psi. Buttons at silty sandstone/silty sandstone with shale clasts contact.
10[3]	52' 2"	Silty sandstone	+35(?)	-263(?)	N82W	1.8	2.4	1.7	1.2	Used 1/2" as zero; biaxial core fractured 1,600 psi of first load cycle.
10[4]	63' 1"	Silty sandstone with shaly laminae at 63' 2 3/4"	+157(?)	+87(?)	N43E	3.5	3.5	3.5	3.6	Biaxial test depth at 62' 10".

(1) Below casing collar.
(2) Assumed; negative values denote tensile stresses.
(3) Data termed uncertain due to drift of all axes at the end of test.
(4) Data termed uncertain due to drift of all axes at the beginning and end of test.

TABLE 2.5-6

IN SITU ELASTIC PROPERTIES FROM CROSSHOLE SURVEYS
BORING GP-1 TO BORING GP-2

(Time Correction Applied)

Depth (ft)	Distance (ft)			Compressional Time (msec)	Shear Time (msec)	Compressional Velocity (ft/sec)			Shear Velocity (ft/sec)			Specific Gravity (g/cc)	Shear Modulus (psi x 10^6)			Poisson's Ratio
	Min	Max	Avg			Min	Max	Avg	Min	Max	Avg		Min	Max	Avg	
0	30.95	30.95	30.95		0.00											
21	31.00	31.10	31.05	2.24	6.28	13478	14266	13862	5805	6051	5926	2.625	1.193	1.295	1.242	0.388
25	31.05	31.10	31.07	2.24	5.92	13500	14266	13873	6235	6506	6368	2.671	1.400	1.524	1.460	0.367
35	31.10	31.15	31.12	2.24	5.78	13522	14289	13895	6426	6713	6566	2.655	1.478	1.613	1.543	0.356
40	31.15	31.25	31.20	2.12	5.78	14289	15170	14717	6436	6735	6582	2.698	1.506	1.650	1.576	0.375
55	31.20	31.35	31.27	2.20	5.08	13805	14650	14216	7536	7957	7741	2.684	2.055	2.291	2.168	0.289
66	31.25	31.50	31.37	2.20	5.04	13827	14720	14261	7622	8077	7844	2.668	2.089	2.346	2.213	0.283
70	31.30	31.50	31.40	2.20	4.90	13850	14720	14273	7904	8378	8135	2.686	2.262	2.541	2.396	0.259
85	31.25	31.70	31.48	1.96	4.86	15470	16684	16059	7972	8522	8240	2.711	2.322	2.654	2.481	0.321
100	31.45	31.85	31.65	2.16	5.18	14167	15167	14653	7417	7884	7645	2.668	1.979	2.235	2.102	0.313
115	31.60	32.10	31.85	2.12	4.96	14495	15583	15024	7861	8403	8125	2.652	2.209	2.524	2.360	0.293
125	31.70	32.20	31.95	2.00	4.48	15388	16598	15975	8955	9641	9288	2.692	2.910	3.373	3.130	0.245
130	31.70	32.25	31.98	2.00	4.84	15388	16624	15987	8128	8716	8414	2.681	2.388	2.746	2.559	0.308
137	31.80	32.35	32.07	2.04	4.76	15143	16338	15723	8325	8936	8622	2.667	2.491	2.871	2.673	0.285
145	31.85	32.45	32.15	2.20	4.92	14093	15164	14614	8003	8585	8286	2.721	2.349	2.703	2.518	0.263
160	31.85	32.55	32.20	2.04	4.88	15167	16439	15784	8084	8703	8385	2.708	2.385	2.765	2.567	0.303
175	31.95	32.65	32.30	2.20	4.84	14137	15257	14682	8192	8824	8500	2.651	2.398	2.783	2.582	0.248
190	32.10	32.80	32.45	2.08	4.84	15000	16238	15601	8231	8865	8539	2.661	2.430	2.819	2.616	0.286
205	32.25	33.00	32.62	2.16	4.84	14527	15714	15104	8269	8919	8586	2.670	2.461	2.863	2.653	0.261
220	32.40	33.10	32.75	2.12	4.92	14862	16068	15448	8141	8757	8441	2.683	2.397	2.773	2.577	0.287
232	32.50	33.25	32.87	2.20	4.88	14381	15537	14943	8249	8890	8561	2.736	2.509	2.915	2.703	0.256
235	32.55	33.25	32.90	1.98	4.88	15956	17318	16616	8261	8890	8568	2.742	2.523	2.921	2.713	0.319
250	32.70	33.45	33.08		4.80				8472	9139	8797	2.714	2.626	3.056	2.831	
265	32.90	33.55	33.22	2.14	4.88	14955	16130	15526	8350	8971	8652	2.721	2.558	2.952	2.746	0.275
280	33.05	33.70	33.37	2.00	4.68	16044	17371	16687	8837	9520	9169	2.735	2.879	3.341	3.099	0.284
295	33.25	33.90	33.58	2.08	4.72	15537	16782	16142	8796	9469	9124	2.712	2.829	3.278	3.043	0.265
300	33.30	33.95	33.62	2.12	4.84	15275	16481	15861	8538	9176	8849	2.736	2.689	3.105	2.888	0.274
310	33.50	34.05	33.77	2.16	4.92	15090	16214	15637	8417	9008	8705	2.711	2.589	2.965	2.769	0.275
316	33.55	34.10	33.82	2.08	4.72	15678	16881	16262	8876	9525	9192	2.717	2.885	3.323	3.094	0.265
321	33.65	34.20	33.92	2.18	4.96	15022	16132	15562	8371	8953	8654	2.727	2.576	2.946	2.753	0.276
340	33.90	34.35	34.12	2.16	5.20	15270	16357	15799	7958	8461	8203	2.710	2.313	2.615	2.458	0.315
353	34.10	34.65	34.38	2.08	5.24	15935	17153	16526	7930	8451	8185	2.721	2.307	2.620	2.457	0.337

TABLE 2.5-7

IN SITU ELASTIC PROPERTIES FROM CROSSHOLE SURVEYS
BORING GP-1 TO BORING RT-1

(Time Correction Applied)

Depth (ft)	Distance (ft)			Compressional Time (msec)	Shear Time (msec)	Compressional Velocity (ft/sec)			Shear Velocity (ft/sec)			Specific Gravity (g/cc)	Shear Modulus (psi x 10^6)			Poisson's Ratio
	Min	Max	Avg			Min	Max	Avg	Min	Max	Avg		Min	Max	Avg	
0	15.50	15.50	15.50													
21	15.30	15.35	15.32	1.14	3.50	12750	14213	13443	5977	6504	6230	2.625	1.264	1.497	1.373	0.363
25	15.25	15.35	15.30	1.16	3.56	12500	13955	13190	5821	6343	6071	2.671	1.220	1.449	1.327	0.366
35	15.05	15.10	15.07	1.14	4.06	12542	13981	13224	4824	5171	4992	2.655	.833	.957	.892	0.417
40	14.95	15.05	15.00	1.02	3.16	13843	15677	14706	6734	7450	7075	2.698	1.649	2.019	1.821	0.349
55	14.70	14.80	14.75	1.12	2.92	12458	13962	13170	7424	8315	7846	2.684	1.994	2.501	2.227	0.225
66	14.50	14.60	14.55	1.14	2.92	12083	13619	12763	7323	8202	7739	2.668	1.929	2.420	2.154	0.209
70	14.45	14.60	14.52	1.10		12457	14038	13205				2.686				
85	14.30	14.45	14.38	0.96	2.78	14020	16056	14974	7772	8811	8261	2.711	2.207	2.837	2.494	0.281
100	14.20	14.35	14.27	1.04	3.02	12909	14643	13726	6827	7633	7210	2.668	1.676	2.095	1.869	0.310
115	14.20	14.35	14.27	1.04	2.68	12909	14643	13726	7320	8247	7758	2.652	1.915	2.431	2.152	0.265
125	14.20	14.40	14.30	1.00	2.76	13396	15319	14300	7802	8889	8314	2.692	2.209	2.867	2.508	0.245
130	14.25	14.45	14.35	0.92	2.76	14541	16802	15598	7830	6920	8343	2.681	2.216	2.875	2.516	0.300
137	14.30	14.55	14.43	0.96	2.68	14020	16167	15026	8218	9448	8796	2.667	2.428	3.209	2.781	0.239
145	14.40	14.60	14.50	1.04	2.80	13091	14898	13942	7742	8795	8239	2.721	2.198	2.837	2.490	0.232
160	14.65	14.80	14.72	0.92	2.94	14949	17209	16005	7325	8222	7750	2.708	1.959	2.466	2.193	0.347
175	15.05	15.10	15.07	1.10	3.04	12974	14519	13705	7167	7947	7537	2.651	1.835	2.257	2.030	0.283
190	15.50	15.65	15.57	1.06	2.96	13839	15650	14693	7673	6599	8112	2.661	2.112	2.652	2.360	0.281
205	15.90	16.25	16.07	1.12	2.92	13475	15330	14353	8030	9129	8551	2.670	2.321	3.000	2.631	0.225
220	16.50	16.80	16.65	1.18	3.12	13306	15000	14110	7569	8485	8005	2.683	2.072	2.604	2.317	0.263
232	16.95	17.35	17.15	1.16	3.28	13893	15773	14784	7244	8107	7656	2.736	1.935	2.424	2.162	0.317
235	17.10	17.50	17.30	1.12	3.16	14492	16509	15446	7703	6663	6160	2.742	2.193	2.774	2.461	0.306
250	17.70	18.25	17.98	1.12	3.12	15000	17217	16049	8119	9217	8642	2.714	2.412	3.108	2.732	0.296
265	16.20	19.00	18.60	1.24	3.38	14000	16102	15000	7459	6482	7949	2.721	2.041	2.639	2.317	0.305
280	19.05	19.75	19.40	1.14	3.38	15675	18267	17018	8072	9144	8584	2.735	2.402	3.082	2.717	0.329
285	19.75	20.50	20.12	1.30	3.42	14522	16532	15481	7964	6991	8456	2.712	2.319	2.955	2.614	0.287
300	20.00	20.75	20.37	1.24	3.44	15365	17585	16431	6000	9022	8490	2.736	2.360	3.002	2.658	0.318
310	20.50	21.30	20.90	1.32	3.48	14855	16905	15833	8071	9103	6566	2.711	2.380	3.028	2.681	0.293
316	20.65	21.60	21.23	1.34	3.54	14893	16875	15840	7898	8852	8366	2.717	2.284	2.870	2.557	0.307
321	21.10	21.85	21.48	1.36	3.76	14659	16808	15790	7482	6340	7895	2.727	2.058	2.557	2.291	0.333

TABLE 2.5-8

IN SITU ELASTIC PROPERTIES FROM CROSSHOLE SURVEYS
BORING RT-1 TO BORING GP-2

(Time Correction Applied)

Depth (ft)	Distance (ft)			Compressional Time (msec)	Shear Time (msec)	Compressional Velocity (ft/sec)			Shear Velocity (ft/sec)			Specific Gravity (g/cc)	Shear Modulus (psi x 10^6)			Poisson's Ratio
	Min	Max	Avg			Min	Max	Avg	Min	Max	Avg		Min	Max	Avg	
0	27.27	27.27	27.27													
21	26.85	26.90	26.98	2.00		13034	13866	13437				2.625				
25	26.65	26.70	26.57	1.88	0.00	13737	14670	14189				2.671				
35	26.30	26.40	26.35	1.92	4.32	13283	14194	13724	7781	8302	8034	2.655	2.167	2.467	2.310	0.239
40	25.10	26.20	26.15	1.84	4.16	13737	14719	14212	8106	8675	8381	2.698	2.389	2.737	2.555	0.234
55	25.45	25.65	25.55	1.72	4.32	14298	15452	14855	7530	8066	7790	2.684	2.051	2.354	2.195	0.310
66	24.95	25.20	25.08	1.90	4.28	13414	14483	13931	7470	8025	7789	2.668	2.007	2.316	2.154	0.277
70	24.75	25.00	24.87	1.76	4.22	13599	14706	14134	7546	8117	7822	2.686	2.062	2.385	2.215	0.279
85	23.95	24.40	24.17	1.50	4.00	14428	15844	15109	7827	8531	8167	2.711	2.239	2.660	2.438	0.294
100	23.35	23.75	23.55	1.54	3.98	13735	15032	14360	7681	8363	8010	2.668	2.122	2.515	2.808	0.274
115	22.70	23.15	22.92	1.56	3.68	14012	15433	14696	8285	9114	8684	2.652	2.454	2.970	2.696	0.232
125	22.25	22.70	22.48	1.44	3.80	14833	16449	15608	7780	8534	8143	2.692	2.196	2.643	2.406	0.313
130	22.00	22.45	22.23	1.40	3.68	15068	16754	15875	8029	8839	8419	2.681	2.330	2.823	2.561	0.304
137	21.70	22.15	21.92	1.40	3.72	14863	16580	15661	7806	8585	8181	2.667	2.190	2.650	2.406	0.312
145	21.20	21.75	21.48	1.56	3.94	13086	14900	13766	7067	7768	7405	2.721	1.832	2.213	2.011	0.296
160	20.40	21.10	20.75	1.22	3.78	15938	18190	17008	7183	7992	7573	2.708	1.883	2.332	2.094	0.376
175	19.65	20.40	20.02	1.40	3.72	13459	15224	14304	7068	7907	7472	2.651	1.785	2.234	1.995	0.312
190	18.80	19.70	19.25	1.32	3.36	13623	15635	14583	7769	8874	8297	2.661	2.165	2.825	2.470	0.261
205	18.00	18.85	18.42	1.28	3.30	13433	15451	14395	7627	8727	8153	2.670	2.094	2.741	2.392	0.261
220	17.20	18.20	17.70	1.18	3.12	13871	16250	15000	7890	9192	8510	2.683	2.251	3.056	2.619	0.263
232	16.65	17.65	17.15	1.16	3.04	13648	16045	14784	7929	9289	8575	2.736	2.318	3.183	2.712	0.246
235	16.45	17.45	16.95	1.08	3.08	14430	17108	15694	7687	8995	8309	2.742	2.184	2.991	2.552	0.305
250	15.70	16.80	16.25	1.00	2.82	14811	17872	16250	8351	10000	9129	2.714	2.551	3.658	3.049	0.269
265	15.00	16.05	15.52	1.04	2.86	13636	16378	14928	7812	9331	8530	2.721	2.239	3.194	2.669	0.258
280	14.30	15.40	14.45	0.96	2.80	14020	17111	15469	7688	9277	8437	2.735	2.179	3.173	2.625	0.288
295	13.70	14.70	14.20	0.88	2.64	14574	17927	16136	8059	9800	8875	2.712	2.374	3.511	2.879	0.283
300	13.50	14.50	14.00	0.92	2.56	13776	16860	15217	8333	10211	9211	2.736	2.561	3.846	3.129	0.211
310	13.10	14.00	13.55	0.88	2.64	13936	17073	15398	7706	9333	8469	2.711	2.170	3.183	2.621	0.283
316	12.80	13.75	13.27	0.84	2.52	14222	17628	15804	8101	9964	8970	2.717	2.404	3.636	2.947	0.262
321	12.70	13.55	13.13	0.82	2.50	14432	17829	16006	8141	9963	8990	2.727	2.436	3.649	2.971	0.269

TABLE 2.5-10

MINERALOGICAL COMPOSITION
ESTIMATED MODAL ANALYSIS BY PETROGRAPHIC EXAMINATION OF THIN SECTIONS

				Component (%)				
Sample*	Depth	Rock Type (by Thin Section)	Geologic Unit	Quartz	Chert	Feldspar	Rock Fragments	Matrix and Accessory Minerals
SR-1	21'11" (6.68 m)	Submature lithic arenite	Oswego Sandstone	60	5	7	20	8
SR-2	23'0" (7.01 m)	Submature lithic arenite	Oswego Sandstone	53	7	8	25	7
SR-3	26'10" (8.18 m)	Lithic graywacke	Transition Zone	45	3	10	25	17
SR-4	27'11" (8.51 m)	Lithic graywacke	Transition Zone	46	2	5	30	17
SR-5	91'0" (27.74 m)	Shale	Pulaski Unit B	25	-	1	-	74
SR-6	92'3" (28.12 m)	Shale	Pulaski Unit B	25	-	1	-	74

* All samples are from Boring TD-1.

TABLE 2.5-11

MINERALOGICAL COMPOSITION
MODAL ANALYSIS BY X-RAY DIFFRACTION

Sample	Rock Type (by thin section)	Mineral (%)					
		Silica, Quartz, and Chert	Sodic Feldspar	Chlorite Group	Illite Group	Kaolinite	Other Undifferentiated or as Noted
SR-1	Submature lithic arenite	59	10	8	6	15	2
SR-2	Submature lithic arenite	58	10	10	6	15	1
SR-3	Lithic graywacke	52	12	9	7	17	3
SR-4	Lithic graywacke	50	9	11	9	18	3
SR-5	Shale	34	2	8	19	27	10*
SR-6	Shale	32	2	13	22	23	8*

* Includes a major percentage of halloysite, and probably organic material and rock fragments.

TABLE 2.5-12

MODULI FROM SONIC TESTING

Sample No.	Depth	Rock Type	Compression Wave Velocity (V_p) (ft/sec)	Shear Wave Velocity (V_s) (ft/sec)	Uniaxial Stress (psi)	Poisson's Ratio (υ)	Shear Modulus G (x 10^6 psi)	Young's Modulus E (x 10^6 psi)
GP-1-1	12'10" - 13'15"	Sandstone	13,560	8,782	10.0	0.139	2.676	6.096
GP-1-2	25' - 25'6"	Sandstone	15,171	9,435	10.0	0.185	3.150	7.464
GP-1-4	25'11" - 29'8"	Sandstone	14,429	9,829	20.0	0.201	2.787	6.692
GP-1-5	30'6" - 30'11"	Siltstone	11,488	5,028	20.0	0.382	0.912	2.520
GP-1-12	60'2" - 61'2"	Sandstone	15,215	9,819	40.0	0.143	3.400	7.773
GP-1-14	82'1" - 82'9"	Graywacke	18,397	11,238	60.0	0.202	4.744	11.409
GP-1-21	101' - 109'6"	Sandstone	17,372	10,993	75.0	0.166	4,385	10.226
GP-1-30	158'2" - 158'11"	Sandstone	18,728	11,529	110.0	0.195	5.002	11.952
GP-1-44	280'4" - 281'	Sandstone	16,891	10,941	800.0	0.139	4.387	9.991
GP-1-49	322'6" - 323'	Silicious	16,354	10,457	225.0	0.154	4.004	9.242
GP-1-7	40'7" - 41'5"	Graywacke	13,956	8,767	30.0	0.174	2.765	6.491
GP-1-9	53'5" - 53'11"	Graywacke	12,574	7,718	40.0	0.198	2.123	5.087
GP-1-13	70'11" - 71'10"	Graywacke	13,671	8,271	25.0	0.211	2.465	5.971
GP-1-22	110'11" - 111'10"	Sandy siltstone	12,444	5,923	70.0	0.354	1.263	3.419
GP-1-53	339'7" - 340'2"	Sandy siltstone	14,389	9,486	240.0	0.116	3.265	7.286
GP-1-42	251'2" - 252'0"	Shale	11,585 11,585	6,936 6,936	151.3 175.0	0.221 0.221	1.746 1.746	4.263 4.263
GP-1-29	155' - 155'4"	Shale	10,485 10,042	6,347 6,182	151.6 110.0	0.211 0.195	1.448 0.374	3.506 3.283
GP-1-6	39'8" - 40'0"	Shale	11,495 11,495	6,745 7,056	151.5 30.0	0.237 0.198	1.636 1.790	4.049 4.289
GP-1-47	303'5" - 304'0"	Silty shale	11,351 11,351	6,946 6,946	152.8 210.0	0.201 0.201	1.762 1.762	4.232 4.232
GP-1-24	121'9" - 122'4"	Shale	10,854	6,702	151.7	0.192	1.624	3.871

TABLE 2.5-12 (Cont'd.)

Sample No.	Depth	Rock Type	Compression Wave Velocity (V_p) (ft/sec)	Shear Wave Velocity (V_s) (ft/sec)	Uniaxial Stress (psi)	Poisson's Ratio (υ)	Shear Modulus G (x 10^6 psi)	Young's Modulus E (x 10^6 psi)
GP-1-55	350'1" - 350'6"	Silty shale	11,847	6,977	245.0	0.235	1.768	4.365
GP-1-36		Siltstone	10,881	6,367	150.9	0.240	1.475	3.658
GP-1-27		Shale	11,388 9,622	5,958 5,864	302.1 100.0	0.312 0.205	1.279 1.239	3.355 2.985
GP-1-15	88'11" - 89'5"	Interbedded sandstone and shale	12,003	7,266	151.0	0.211	1.902	4.605
GP-1-28	152'6" - 153'8"	Sandy siltstone	13,394	7,952	151.0	0.228	2.285	5.612
GP-1-35	195'0" - 195'8"	Shale	11,481	7,010	150.8	0.203	1.775	4.270
GP-1-39	239'3" - 239'8"	Silty sandy shale	12,140	7,461	301.8	0.196	2.023	4.841

TABLE 2.5-13

TRIAXIAL TEST RESULTS (T-3-3 SERIES)

Results from Triaxial Compression Testing of Rock Samples from Boring T-3-3

Rock Type	Formation	Confining Pressure		Sample Pore Pressure		Young's Modulus		Maximum Stress	
		psi	kg/cm^2	psi	kg/cm^2	psi x 10^6	x $10^5 kg/cm^2$	psi	kg/cm^2
Siltstone/shale	Pulaski Unit A	200	14	0		4.56	3.21	11,659	820
Siltstone/shale	Pulaski Unit A	200	14	0		4.62	3.25	15,697	1,103
Sandstone	Pulaski Unit A	200	14	0		6.70	4.71	37,435	2,632
Sandstone	Pulaski Unit A	200	14	0		6.71	4.72	40,906	2,877
Silty shale	Pulaski Unit B	200	14	0		3.64	2.56	10,410	2,876
Silty shale	Pulaski Unit B	200	14	0		4.10	2.88	10,532	740
Silty shale	Pulaski Unit B	400	28	0		3.60	2.53	13,536	952
Silty shale	Pulaski Unit B	400	28	200	14	3.98	2.80	7,459	524
Silty shale	Pulaski Unit B	400	28	200	14	3.90	2.74	8,128	571

NOTE: Long axis of cylindrical samples is perpendicular to bedding.

TABLE 2.5-15

TRIAXIAL TEST RESULTS (RT SERIES)

Test No.	Sample No.	Rock Type	Relation of Test Specification Axis to Bedding (deg)*	Confining Pressure (psi)	Maximum Point Differential Stress (psi)	Maximum Point Strain (%)	Young's Modulus (psi x 10^6)	Maximum Strain Under Confining Pressure (%)
1	RT 1-1	Sandstone	90	50	31,745	0.81	5.63	5.44
2	RT 1-1		90	750	37,012	0.97	5.62	5.75
3	RT 1-1		90	1,500	39,055	1.06	5.06	6.41
4	RT 2-7	Siltstone	90	50	9,588	0.78	1.61	3.03
5	RT 2-7		90	750	11,586	0.92	1.70	2.79
6	RT 2-7		90	1,500	12,408	1.05	1.75	3.12
7	RT 2-7		45	50	7,073	0.52	1.60	2.53
8	RT 2-7		45	750	7,864	0.67	1.58	3.01
9	RT 2-7		45	1,500	7,571	0.69	1.62	2.68
10	RT 2-7		0	50	9,861	0.47	2.42	2.85
11	RT 2-7		0	750	12,016	0.62	2.52	3.06
12	RT 2-7		0	1,500	12,499	0.74	2.42	3.11
13	RT 1-22	Graywacke	90	50	8,583	1.03	1.28	3.27
14	RT 1-22		90	750	9,485	1.15	1.44	3.25
15	RT 1-22		90	1,500	9,848	1.16	1.48	3.22
16	RT 1-22		45	50	4,114	0.53	0.81	2.06
17	RT 1-22		45	750	3,990	0.59	0.87	2.29
18	RT 1-22		45	1,500	5,913	0.94	1.00	3.43
19	RT 1-22		0	50	4,857	0.38	1.60	2.26
20	RT 1-22		0	750	8,286	0.58	1.98	3.87
21	RT 1-22		0	1,500	9,406	0.71	2.09	3.96
22	RT 1-51		90	50	12,054	0.90	1.96	3.43
23	RT 1-51		90	750	10,393	1.02	1.70	3.43
24	RT 1-51		90	1,500	12,250	1.22	1.58	3.49
25	RT 1-51		45	50	5,668	0.50	1.32	3.26
26	RT 1-51		45	750	7,116	0.69	1.46	2.71
27	RT 1-51		45	1,500	11,208	0.86	1.77	3.39
28	RT 1-51		0	50	9,488	0.40	2.96	2.97
29	RT 1-51		0	750	10,628	0.57	2.37	2.96
30	RT 1-51		0	1,500	11,155	0.67	2.37	3.16

* Bedding is taken to be perpendicular to the bulk sample core axis.

TABLE 2.5-16

UNIAXIAL MODULUS DETERMINATIONS

Specimen No.	Depth (ft)	Rock and Formation	i°*	10^6 psi E_z	u_{zx}	10^6 psi E_y	u_{yx}	u_{yz}	10^6 psi E_i	Stress Range (psi)
RT1-3A	18.9-19.1	Sandstone Oswego Sandstone	90	4.08 4.12 4.42	0.168 0.168 0.167					2nd loading 218-1865 2nd unloading 4662-311 3rd loading 311-3108
RT1-4A	20.5-20.65	Sandstone Oswego Sandstone	0			5.20 3.15 5.22	0.292 0.445 0.320	0.096 0.125 0.114		2nd loading 377-1840 2nd unloading 1950-220 3rd loading 220-3145
RT1-6A	23.9-24.25	Sandstone Oswego Sandstone	45						2.37 1.79 2.60	2nd loading 109-1876 2nd unloading 1076-125 3rd loading 125-3127
RT1-12A	31.3-31.5	Siltstone (sandy) Transition Zone	0			4.97 4.41 5.45	0.270 0.297 0.270	0.403 0.446 0.393		2nd loading 156-1907 2nd unloading 1907-156 3rd loading 156-3127
RT2-2A	25.55-25.95	Siltstone (shaly) Transition Zone	90	1.12 1.12 1.14	0.330 0.333 0.321					2nd loading 125-1870 2nd unloading 1870-125 3rd loading 125-1870
RT2-6A	26.9-27.3	Siltstone (sandy) Transition Zone	45						3.29 3.56 3.42	2nd loading 140-2341 2nd unloading 2341-94 3rd loading 94-1873
RT1-56A	96.7-97.1	Siltstone (shaly) Pulaski B	0			1.93	0.177	0.428		1st loading 188-1407 (specimen failed)
RT1-56B	96.7-97.1	Siltstone (shaly) Pulaski B	0			4.70 4.65 3.91	0.331 0.416 0.360	0.437 0.489 0.485		2nd loading 63-1408 2nd unloading 1408-125 3rd loading 125-1940
RT1-17A	41.6-42.0	Graywacke Pulaski A	90	6.27 4.82 6.28	0.331 0.350 0.330					2nd loading 94-1907 2nd unloading 1970-94 3rd loading 94-3127
RT1-20A	46.0-46.4	Graywacke (silty) Pulaski A	0			5.93 4.48 7.15	0.379 0.654 0.284	0.632 0.726 0.619		2nd loading 188-1878 2nd unloading 1878-156 3rd loading 156-1565
RT1-26A	51.2-51.35	Graywacke (shaly) Pulaski	45						3.20 3.18 3.57	2nd loading 1876-31 2nd unloading 1876-31 3rd loading 31-3127
RT1-36A	1.55-61.95	Silty sandstone and graywacke (silty) Pulaski Unit 1	90	7.26 6.76 7.94	0.175 0.242 0.211					2nd loading 157-1942 2nd unloading 1911-125 3rd loading 125-1927

TABLE 2.5-16 (Cont'd.)

Specimen No.	Depth (ft)	Rock and Formation	i°*	10^6 psi E_x	u_{zx}	10^6 psi E_y	u_{yx}	u_{yz}	10^6 psi E_i	Stress Range (psi)
RT1-61A	105.7-106.1	Shale (silty) Pulaski C	90	2.68 2.85 2.81	0.682 0.656 0.655					2nd loading 111-2222 2nd unloading 2222-167 3rd loading 167-2222
RT1-61B	105.7-106.1	Shale (silty) Pulaski C	0			4.61 4.07 4.71	0.437 0.423 0.401	0.517 0.537 0.524		2nd loading 56-1669 2nd unloading 1669-167 3rd loading 167-1669
RT1-62D	106.4-106.9	Shale (silty) Pulaski C	45						1.39 1.46 1.43	2nd loading 56-1680 2nd unloading 1680-112 3rd loading 112-1680

*i° = Inclination of core to bedding.

TABLE 2.5-17

YOUNG'S MODULUS DETERMINED BY BIAXIAL TEST

(modulus x 10^6 psi)

Lithologic Unit	Rock Type Tested	No. Tests	Minimum	Mean	Maximum	Standard Deviation (x 10^6 psi)
Oswego Sandstone	Sandstone	37	1.6	4.3	8.5	1.53
Transition Zone of Oswego Sandstone	Sandstone, shale, and graywacke	17	2.7	5.1	7.9	1.25
Pulaski Formation Unit A	Graywacke and sandstone	52	1.8	4.6	8.3	1.28
Pulaski Formation Unit B	Sandstone	5	3.2	4.1	4.9	0.70
Pulaski Formation Unit C	Shale, siltstone, and sandstone	31	1.9	3.4	7.1	0.99
Whetstone Gulf Formation Unit A	Sandstone	2	4.3	4.3	4.4	0.1
Whetstone Gulf Formation Unit B	Sandstone	4	4.1	4.7	5.8	0.8
Total		148	1.6	4.3	8.5	1.34

TABLE 2.5-18

SWELL TEST RESULTS

Sample No.	Test No.	Rock Type	Rock Unit	Test Type	Vertical Stress σ_v(psi)	Horizontal Stress σR(psi)	First Stress Invariant $\sigma_v+2\sigma_h$ (psi)	Vertical Swell Strain Rate (10^{-2}%/ $\log_{10}$ day)	Horizontal Swell Strain Rate (10^{-2}%/ $\log_{10}$ day)
RT2-1	5	Sandstone	Oswego	Swell Strain	25	75-140	175-305	0	0
TD-1	1	Sandstone	Oswego	Free Swell	0	0	0	0	0
TD-2	1	Sandstone	Oswego	Free Swell	0	0	0	0	0
RT1-14	4	Shale	Transition Zone	Swell Strain	37	44-75	125-187	12	1.7
RT1-14	4	Shale	Transition Zone	Swell Strain	19	88-107	195-233	12	0.9
RT1-15	8	Shale with siltstone lenses	Transition Zone	Swell Strain	37	60-112	157-261	11	1.2
RT1-15	8	Shale with siltstone lenses	Transition Zone	Swell Strain	19	110-140	254-299	11	0.55
RT1-9	18	Siltstone	Transition Zone	Swell Strain	72	56-72	180-220	0	0.7
RT1-9	18	Siltstone	Transition Zone	Swell Stress	85	75	238	0	0
RT1-10	25	Siltstone	Transition Zone	Swell Stress	70	75-111	220-292	0	0.75
RT1-10	25	Siltstone	Transition Zone	Swell Stress	120	104	328	0	0
RT2-4	21	Sandy siltstone	Transition Zone	Swell Stress	115	90-150	295-415	0	1.65
RT2-4	21	Sandy siltstone	Transition Zone	Swell Stress	115	155-178	425-471	0	0.4
TD-1	3	Graywacke	Transition Zone	Free Swell	0	0	0	12.4	4.2
TD-1	4	Shale siltstone	Transition Zone	Free Swell	0	0	0	10.4	3.2
TD-1	5	Sandstone	Transition Zone	Free Swell	0	0	0	0	0.35
RT1-23	6	Graywacke	Pulaski A	Swell Strain	47	110-130	267-307	4.0	0.9
RT1-23	6	Graywacke	Pulaski A	Swell Strain	23	130-148	283-319	3.0	0.8
RT1-24	10	Graywacke	Pulaski A	Swell Strain	23	133-161	289-345	0.25	0.5
RT1-29	2	Graywacke	Pulaski A	Swell Strain	23-47	52	127-151	0	0
RT1-37	14	Graywacke	Pulaski A	Swell Strain	63	93-148	249-359	5.0	2.1
RT1-17	14	Graywacke	Pulaski A	Swell Strain	63	150-170	360-403	5.0	0.9

TABLE 2.5-18 (Cont'd.)

Sample No.	Test No.	Rock Type	Rock Unit	Test Type	Vertical Stress σ_v(psi)	Horizontal Stress σR(psi)	First Stress Invariant $\sigma_v+2\sigma_h$ (psi)	Vertical Swell Strain Rate (10^{-2}%/ $\log_{10}$ day)	Horizontal Swell Strain Rate (10^{-2}%/ $\log_{10}$ day)
RT1-44	11	Graywacke	Pulaski A	Swell Strain	66	130-165	324-396	5.5	1.7
RT1-44	11	Graywacke	Pulaski A	Swell Strain	66	165-190	396-446	4.0	0.77
RT1-30	20	Graywacke	Pulaski A	Swell Stress	55-220	0-60	55-340	0	3.4
RT1-30	20	Graywacke	Pulaski A	Swell Stress	220-600	60-70	340-740	0	0.25
RT1-33	72	Graywacke	Pulaski A	Swell Stress	20-45	120-160	260-365	0	1.4
RT1-33	22	Graywacke	Pulaski A	Swell Stress	45-70	160-170	365-410	0	0.25
RT1-48	17	Graywacke	Pulaski A	Swell Stress	60	75-110	210-280	0	1.55
RT1-40	17	Graywacke	Pulaski A	Swell Stress	60-75	110-120	280-315	0	0.5
TD-1	6	Graywacke	Pulaski A	Free Swell	0	0	0	5.6	2.2
TD-1	7	Sandstone	Pulaski A	Free Swell	0	0	0	0.18	0.4
TD-1	8	Graywacke	Pulaski A	Free Swell	0	0	0	4.6	5.0
TD-1	9	Graywacke	Pulaski A	Free Swell	0	0	0	12.0	5.3
RT1-52	19	Shale	Pulaski B	Swell Stress	200-460	50-100	300-660	0	3.4
RT1-52	19	Shale	Pulaski B	Swell Stress	460-550	100	600-750	0	0.3
RT1-55	21	Shale	Pulaski B	Swell Stress	75-225	130-140	335-505	0	0
RT1-60	7	Silty shale	Pulaski C	Swell Stress	01	127-150	335-381	0.35	0.4
TD-1	10	Graywacke	Pulaski B	Free Swell	0	0	0	13.1	4.4
TD-1	11	Sandstone	Pulaski B	Free Swell	0	0	0	0.2	0.45
TD-1	12	Shale	Pulaski B	Free Swell	0	0	0	-	7.3
TD-1	13	Shale with sandstone lenses	Pulaski B	Free Swell	0	0	0	26.0	7.0
TD-1	14	Shale with sandstone interbeds	Pulaski C	Free Swell	0	0	0	17.0	6.4
TD-1	16	Interbedded shale, sandstone and siltstone	Pulaski C	Free Swell	0	0	0	16.4	6.7

TABLE 2.5-18 (Cont'd.)

Sample No.	Test No.	Rock Type	Rock Unit	Test Type	Vertical Stress σ_v(psi)	Horizontal Stress σR(psi)	First Stress Invariant $\sigma_v+2\sigma_h$ (psi)	Vertical Swell Strain Rate (10^{-2}%/ $\log_{10}$ day)	Horizontal Swell Strain Rate (10^{-2}%/ $\log_{10}$ day)
TD-1	17	Siltstone	Pulaski C	Free Swell	0	0	0	15.2	5.4
RT1-63	9	Silty shale	Whetstone Gulf A	Swell Strain	82	120-105	322-452	12.5	1.5
RT1-66	1	Silty shale	Whetstone Gulf A	Swell Strain	131	111-180	353-491	8.0	1.35
RT1-77	3	Silty shale	Whetstone Gulf A	Swell Strain	130	75-110	280-350	0.1	0.95
RT1-77	3	Silty shale	Whetstone Gulf A	Swell Strain	65	125-145	315-355	1.25	0.75

TABLE 2.5-19

TRIAXIAL SWELL AND CREEP TEST RESULTS

Rock Type	Sample No.	Cell Pressure (psi)	Strain Rates (10^{-9}/min)[1]					
			Axial			Circumferential		
			Cycle 1	Cycle 2	Cycle 3	Cycle 1	Cycle 2	Cycle 3
Silty shale	RT1-35C	900 406 29	1,300 -89 1,800	740 28 -160	13 -64 32	190 89 740	100 -37 -64	10 -74 5
Black shale	RT2-12B	900 406 29	7,200 -2,800 -2,800	1,000 -56 28	120 -32 250	1,500 0 -278	-42 0 56	40 7 54
Graywacke with silt lenses	RT1-47A	900 406 29	-280 -1,100 -1,800	-860 110 -320	210 110 -100	-5,000 560 -3,500	-250 140 -630	110 83 -390
Graywacke with silt lenses	RT1-28A	900 406 29	560 280 560	-610 -56 -28	160 -12 8	-1,700 -370 -1,000	-280 9 200	21 7 9
Graywacke with silt lenses	RT1-28B	406 203 29	-1,000 -89 0	-190 47 0	250 11 -12	-5,000 190 -190	-330 28 -9	120 7 -5
Silty shale	RT1-35A	1,393	12,000	-1,100	-200	-3,200	-240	-50
Silty shale[2]	RT1-35B	406 203 44	6,900 -740 -370	-260 -56 -110	-60 -32 -24	-1,100 0 280	-250 0 -19	-26 -5 -8

[1] Swell strains are positive; shrinkage strains are negative.
[2] Denotes tested specimen was sealed to prevent water loss or gain. Other tests were executed with water supplied to specimen.

TABLE 2.5-21

EXPLORATION AT THE UNIT 2 SITE
AFTER GENERAL SITE EXPLORATION (POST-PSAR)

Investigation	Date	Scope of Work	FSAR Section	Reference
General excavation mapping	5/75 through 8/76	Mapping and/or photographing all Category I excavations and anomalous features.	2.5.4.3 Appendix 2H 2.5.1.2.3	
Thrust structures within power block complex	8/76 through 9/76	Detailed mapping and photographs North and south radwaste trenches North wall heater bay Circulating water intake Southern circulating water piping encasement Normal switchgear building Northwest reactor notch		Reference 94 Vol. I, Section 3.4
		Radiometric dating (K/Ar) of clay		Dames & Moore, 1976
		Palynologic analysis of unconsolidated clay	2.5.1.2.3	Reference 94, Vol. I, Section 3.4
		Fracture and fold geometric analysis	2.5.1.2.3	Reference 94, Vol. I, Sections 3.2 and 3.4
Cooling Tower fault (geologic investigation)	9/76 through 4/78	Detailed mapping	2.5.1.2.3	
		Exploratory excavations		
		Pit 1		Reference 94, Vol. I, Section 4.0
		Trench 3		Reference 94, Vol. I, Section 4.0
		Trench 4		Reference 94, Vol. I, Section 4.0
		Trench 5		Reference 94, Vol. I, Section 4.0
		Existing excavations	Appendix 2H	Reference 94, Vol. I, Section 4.0
		Circulating water piping trench		Reference 94, Vol. I, Section 4.0
		Cooling tower excavation		Reference 94, Vol. I, Section 3.3
		Drainage ditch		Reference 94, Vol. I, Section 3.3
		Intake shaft		Reference 94, Vol. I, Section 3.4
		Geometric analysis of deformational features All exposures listed above	2.5.1.2.3	
Cooling Tower fault (geologic investigation)	9/76 through 4/78	Boring program	2.5.1.2.3	Reference 94, Vol. I, Section 5.0
		Subsurface stratigraphy and structure at two locations along Cooling Tower fault (total of 24 borings) T-3 and T-4 series		
		Standard boring and geophysical logs		Reference 94, Vol. I, Appendix I-A
		Downhole impression packer survey		Reference 94, Vol. I, Appendix I-B
		Mineralization studies	2.5.1.2.6	Reference 94, Vol. I, Section 6.0
		Fluid inclusion analysis of mineralization from surficial exposures and rock cores		Reference 94, Vol. I, Section 6.3 and Appendix I-E

TABLE 2.5-21 (Cont'd.)

Investigation	Date	Scope of Work	FSAR Section	Reference
		X-ray diffraction analysis of bedrock samples Uranium series age dating of calcite		Reference 94, Vol. I, Section 6.4 and Appendix I-F Reference 94, Vol. I, Appendix I-G
Cooling Tower fault (geomorphology investigation)	9/76 through 4/76	Stratigraphy and age of unconsolidated sediments C-14 age dating Palynologic Analyses Grain size analyses Mineralogic and X-ray diffraction analyses	2.5.1.2.2	Reference 94, Vol. II, Section 1.4 Reference 94, Vol. II, Section 1.4 and Appendix II-A Reference 94, Vol. I, Appendix I-A and Vol. II, Section 1.4 and Appendix II-B Reference 94, Vol. II, Section 1.4 and Appendix II-C Reference 94, Vol. II, Section 1.4 and Appendix II-D
		Aerial photograph interpretation	2.5.1.2	Reference 94, Vol. II, Section 1.4.3
Cooling Tower fault (rock mechanics investigation)	1/77 through 4/78	Stress measurements by overcoring Numerous measurements in borings OC-1 through OC-9 Logs of borings OC-1 through OC-9 Stressmeters installed in borings OC-1, OC-2, OC-5, and OC-7	2.5.4.1.4	Reference 94, Vol. III, Section 3.0 and Appendix III-C Reference 94, Vol. III, Appendix III-D
		Analysis of residual strain relief by undercoring Measurements from five cores	2.5.4.1.4	Reference 94, Vol. III, Section 4.0 and Appendix III-G
		Measurements of unconfined swelling strain Measurements of seventeen cores under variety of relative humidity conditions Samples from boring TD-1	2.5.4.1.4	Reference 94, Vol. III, Section 6.0 and Appendix III-K Reference 94, Vol. III, Appendix III-D
		Measurement of groundwater levels in vicinity of Cooling Tower fault Standpipes installed in borings T-4-7, T-4-9, T-4-10, and T-4-11 Water levels monitored 9/30/77 to 1/19/78	2.4.13	Reference 94, Vol. III, Section 5.0 and Appendix III-J
Additional rock mechanics investigation and initial rock monitoring program	3/77 through 6/82	Inclinometer installation and monitoring I-1 through I-4 around circulating water piping trench and intake shaft Boring logs I-1 through I-4	2.5.4.13 Appendix 2K	

TABLE 2.5-21 (Cont'd.)

Investigation	Date	Scope of Work	FSAR Section	Reference
	4/77 through 11/78	Test excavation and monitoring of trench 5 along Cooling Tower fault Inclinometers It-1 through IT-5 Tape extensometer measurement array Precise leveling survey	2.5.4.13 Appendix 2K	
	1985	Displacement monitoring of 12-line wall	2.5.4.13	
	6/77 through 6/82	Extensometer installation and monitoring MPX-1 and MPX-2 (multipoint wire extensometers) adjacent to intake shaft and reactor containment excavations Boring logs MPX-1 and MPX-2	2.5.4.13 Appendix 2K	
	11/77	Geophysical surveys Borings GP-1, GP-2, and RT-1 Standard logs of borings Geophysical logs (caliper, gamma, density, and 3-D velocity) Uphole compressional wave survey Crosshole compressional wave survey Crosshole shear wave velocity survey	2.5.4.4 Appendix 2K 2.5.4.4	
	11/77 through 2/78	Testing of rock samples from borings RT-1 and RT-2 Boring logs RT-1 and RT-2 Direct shear tests on bedding Ring tests; swelling strain, swelling stress, creep strain Triaxial swell and creep tests Uniaxial compression tests Triaxial strength tests Determination of elastic parameters: Young's modulus and Poisson's ratio parallel and perpendicular to bedding, and shear modulus	2.5.4.2 Appendix 2K	
	11/77 through 2/78	Stress measurements by overcoring Stress determinations in Boring OC-100 Log of Boring OC-100	2.5.4.1.4 Appendix 2K	

TABLE 2.5-21 (Cont'd.)

Investigation	Date	Scope of Work	FSAR Section	Reference
Radwaste thrust structure (geologic investigation)	4/79 through 10/80	Detailed mapping of various foundation excavations Circulating water piping excavation Relocated cooling tower excavation North radwaste trench East lake water intake tunnel	2.5.1.2.3 Appendix 2H	Reference 154, Section 2.1
		Analysis of structural elements from mapped areas	2.5.1.2.3	Reference 154, Section 2.1
	4/79 through 10/80	Boring program Subsurface stratigraphy and structure west of surficial exposure of radwaste fault (total of 9 borings) Standard boring and geophysical logs, Borings 801 through 810 Downhole television survey	2.5.1.2.3	Reference 154, Section 2.2 and Appendix A Reference 154, Appendix A
		Mineralization studies Fluid inclusion analysis of mineralization from exposures of thrust structures and rock cores C-13 and O-18 analysis of calcite	2.5.1.2.6	Reference 154, Section 2.3 and Appendix B.1 Reference 154, Appendix B.2
		Age dating C-14 analysis of calcite unconsolidated sediment from within thrust structures Uranium series dating of calcite Palynologic analysis of sediments	2.5.1.2.3 2.5.1.2.6	Reference 154, Section 2.3 Reference 154, Appendix B.2 Reference 154, Appendix B.3 Reference 154, Appendix C.1 and C.2
		Mineralogic analysis of unconsolidated sediments within thrust structures	2.5.1.2.3	Reference 154, Section 2.3, Appendix C.3
Rock mechanics (intake shaft)	11/79 through 1985	Installation of instrumentation in vicinity of intake shaft Inclinometers SI-1 through SI-7 Extensometers EX-1 through EX-4 (multipoint rod type) Piezometers PI-1 through PI-4 (2 transducers each) Boring logs of SI, EX, and PI borings Tape extensometer measurement arrays at four levels in shaft	2.5.4.13 2.4.13 Appendix 2K 2.5.4.2	

TABLE 2.5-21 (Cont'd.)

Investigation	Date	Scope of Work	FSAR Section	Reference
Radwaste thrust structure (rock mechanics investigation	2/80 through 10/80	Stress measurements by overcoring numerous measurements in borings RS-1 through RS-4 Logs of Borings RS-1 through RS-4 Installation of inclinometers in thrust block Inclinometers installed in Borings 803, 805, 806, 810, and RS-2	2.5.4.1.4 2.5.4.2	Reference 154, Section 3.0 and Appendices E.1 and E.3 Reference 154, Appendix E.2
Geophysical survey	7/80	Ground magnetometer survey (across trace of high-angle faults)	2.5.1.2.3	Reference 154, Response to NRC Question Q361.26
Rock mass displacement monitoring program	8/81 through 1985	Installation and monitoring of instruments in and around site excavations Inclinometers Intake shaft, SI-8, SI-9, SI-10 Reactor containment SI-20 through SI-23 Radwaste thrust block 820, 821 Extensometers (multipoint rod type) Intake shaft EX-5 and EX-6 Reactor containment EX-20 and IIEX-1 Piezometers Intake shaft PI-5 Reactor containment PI-20 and PI-21 Boring logs of above boreholes Various displacement monitoring gauges within site engineering structures	2.5.4.13 Appendix 2K	

TABLE 2.5-22

PARTICIPANTS OF POST-PSAR INVESTIGATIONS

Investigation	Date	Participant	Contribution
General excavation mapping	5/75 to 1983	Stone & Webster Engineering Corporation	Architect-engineer, all foundation excavation mapping
Thrust structures within the power block complex	8/76 through 9/76	Dames & Moore Dr. F. A. Donath, CGS, Inc. Krueger Enterprises, Inc., Cambridge, MA Dr. I. A. Sirkin - Adelphi University, NY	Principal investigator Assessment of geologic structures K/Ar age dating Palynologic analysis
Cooling tower fault	9/76 through 4/78	Dames & Moore Dr. F. A. Donath Dr. D. Coates, SUNY, Binghamton, NY Dr. S. Alexander, Pennsylvania State University Dr. H. L. Barnes, Pennsylvania State University Dr. A. H. Vassiliou, Rutgers, the State University Dr. Teh-Lung Ku - University of Southern California Dr. L. A. Sirkin Brock University, Toronto, Canada Teledyne Isotopes, Westwood, NJ Krueger Enterprises, Inc. Dr. N. J. Price, Imperial College, London, England	Principal investigator NMPC review panel member NMPC review panel member NMPC review panel member Fluid inclusion analysis, ore microscopy, X-ray diffraction Mineralogic analysis, X-ray diffraction, X-ray, fluorescence Uranium series age dating Palynologic analysis C-14 age dating C-14 age dating C-14 age dating Assessment of evolution of bedrock stresses
Additional rock mechanics investigation and initial monitoring program	11/77 through 2/78	Dames & Moore Birdwell Geophysical Company Franklin Trow Associates, Ltd., Toronto, Canada CGS, Inc., Urbana, IL Woodward Clyde Associates	Principal investigator Downhole geophysical survey Rock testing Rock testing Rock testing
Radwaste thrust structure	4/79 through 10/80	Dames & Moore Dr. F. A. Donath Dr. D. Coates Dr. S. Alexander Dr. S. S. Philbrick Dr. R. H. Johns Mr. W. W. Moore Dr. T. I. Pewe, Tempe, AZ Dr. C. Fairhurst, University of Minnesota, Minneapolis, MN	Principal investigator NMPC review panel member NMPC review panel member NMPC review panel member Dames & Moore review panel member Dames & Moore review panel member Dames & Moore review panel member Assessment of origin of clay deformation Assessment of potential for differential movement on radwaste structure

TABLE 2.5-22 (Cont'd.)

Investigation	Date	Participant	Contribution
		Dr. N. J. Price	Rock mechanics assessment of radwaste structure
		Dr. I. A. Sirkin	Assessment of origin of clay deformation and palynologic analyses
		Dr. J. Terasmae, Brock University	Palynologic analyses
		Dr. H. Krueger	Isotopic and radiometric analyses
		Dr. T. L. Ku	Uranium series age dating
		Dr. H. L. Barnes	Paragenetic and fluid inclusion analyses
		Dr. A. H. Vassiliou	Mineralogical analyses
Rock mechanics investigation of intake shaft	11/79 through 1985	Dames & Moore	Principal investigator
		Dr. C. Fairhurst	Reviewer
Rock mass displacement monitoring program	8/81 through 1985	Dames & Moore	Overall responsibility for installation and monitoring program
		Dr. C. Fairhurst	Principal investigator
		Dr. E. D. McKay, CGS, Inc.	Staff investigator providing coordination between Dames & Moore and principal investigator
		Dr. B. Sellers	Instrumentation consultant
		Mr. J. Dunnicliff, P.E.	Instrumentation consultant
		Dr. F. A. Donath	NMPC review panel member
		Dr. S. Alexander	NMPC review panel member
		Dr. D. Coates	NMPC review panel member

TABLE 2.5-23

SUMMARY OF BORINGS COMPLETED ON THE NINE MILE POINT SITE

Boring Series	Year(s) Completed	Total No. Borings	Range of Depth (ft)	Purpose	Core Sample (Yes/No)
100	1963	21	10-100	Foundation investigation Unit 1	Yes
200	1968	14	12-150	Foundation investigation proposed facility	Yes
300	1968	15	75-198	Foundation investigation JAF plant	Yes
400	1971-72	31	52-205	Foundation investigation Unit 2	Yes
T-3	1977	10	176-222	Cooling Tower fault subsurface investigation	Yes
T-4	1977	13	90-342	Cooling Tower fault subsurface investigation	Yes
A	1977	2	160-161	Cooling Tower fault subsurface investigation	Yes
OC	1977-78	10	22-116	Stress measurements	Yes
TD	1977	1	106	Free swell testing	Yes
GP	1978	2	360-361	Geophysical testing	Yes
RT	1977-78	2	40-350	Rock testing	Yes
L	1977	8	115-132	Foundation investigation lake water tunnels	Yes
RS	1980	4	94-125	Stress measurements	Yes
800	1980	11	90-300	Radwaste fault investigation and rock monitoring	Yes
MPX	1977-79	2	115-165	Rock monitoring Unit 2	Yes
EX	1979, 1981	7	151-162	Rock monitoring Unit 2	Yes
HEX	1981	1	155	Rock monitoring Unit 2	Yes
SI	1979, 1981	14	151-205	Rock monitoring Unit 2	Yes
PI	1979, 1981	7	88-127	Rock monitoring Unit 2	Yes
I	1977	4	50-150	Rock monitoring Unit 2	No
IT	1977	5	20	Rock monitoring Unit 2	No
P	1979	6	43-95	Piezometer installation Unit 2	No

TABLE 2.5-25

TRIAXIAL SHEAR TEST RESULTS

Source: Meany Airport Road Pit
Confining Pressure: σ_3 = 69kn/m^2 (10 psi)

Axial Deviator			Induced Principal Pore			Normalized Data		Stress Path			
Strain (%)	Stress		Pore Pressure (kn/m^2)	Stress Ratio	Pressure Parameter	Deviator Stress	Pore Pressure	τ45		Σ457	
	psi	kn/m^2						kn/m^2	psi	psi	kn/m^2
0.0	0.0	0.0	0.0	1.000	0.0	0.0	0.0	0.0	0.0	10.0	69.0
0.04	4.5	31.2	0.0	1.452	0.0	0.452	0.0	15.6	2.2	12.2	84.6
0.10	10.1	70.1	0.0	2.016	0.0	1.016	0.0	35.0	5.0	15.0	104.0
0.18	18.6	128.9	0.0	2.869	0.0	1.869	0.0	64.5	9.3	19.3	133.5
0.29	28.7	198.0	0.0	3.870	0.0	2.870	0.0	99.0	14.3	24.3	168.0
0.44	38.1	263.2	0.0	4.815	0.0	3.815	0.0	131.6	19.0	29.0	200.6
0.78	45.3	313.2	0.0	5.539	0.0	4.539	0.0	156.6	22.7	32.7	225.6
1.17	43.8	302.6	0.0	5.385	0.0	4.385	0.0	151.3	21.9	31.9	220.2
1.62	39.7	274.6	0.0	4.979	0.0	3.979	0.0	137.2	19.8	29.8	206.2
2.35	38.1	263.2	0.0	4.814	0.0	3.814	0.0	131.6	19.0	29.0	200.6
3.17	36.1	249.4	0.0	4.614	0.0	3.614	0.0	124.7	18.0	28.0	193.7
4.20	35.1	242.4	0.0	4.513	0.0	3.513	0.0	121.2	17.5	27.5	190.2
5.56	36.2	250.0	0.0	4.624	0.0	3.624	0.0	125.0	18.1	28.1	194.0
8.44	34.7	239.4	0.0	4.470	0.0	3.470	0.0	119.7	17.3	27.3	188.7
11.77	35.0	241.5	0.0	4.501	0.0	3.501	0.0	120.8	(17.5)	(27.5)	189.8
14.65	35.5	245.2	0.0	4.553	0.0	3.553	0.0	122.6	(17.7)	(27.7)	191.6

TABLE 2.5-26

TRIAXIAL SHEAR TEST RESULTS

Source: Meany Airport Road Pit
Confining Pressure: σ_3 = 138 kn/m^2 (20 psi)

Axial Deviator			Induced Principal Pore			Normalized Data		Stress Path			
Strain (%)	Stress		Pore Pressure (kn/m^2)	Stress Ratio	Pressure Parameter	Deviator Stress	Pore Pressure	τ45		Σ45	
	psi	kn/m^2						kn/m^2	psi	psi	kn/m^2
0.0	0.0	0.0	0.0	1.000	0.0	0.0	0.0	0.0	0.0	20.0	138.0
0.05	6.6	45.7	0.0	1.331	0.0	0.331	0.0	22.8	3.3	23.3	160.8
0.10	12.6	87.2	0.0	1.632	0.0	1.632	0.0	43.6	6.3	26.3	181.6
0.16	18.8	129.8	0.0	1.940	0.0	0.940	0.0	64.9	9.4	29.5	202.9
0.31	33.9	233.9	0.0	2.695	0.0	1.695	0.0	117.0	16.9	36.9	255.0
0.56	49.3	340.4	0.0	3.466	0.0	2.466	0.0	170.2	24.6	44.6	308.2
0.56	49.3	340.4	0.0	3.466	0.0	2.466	0.0	170.2	24.6	44.6	308.2
0.80	58.0	400.7	0.0	3.904	0.0	2.904	0.0	200.4	29.0	49.0	338.4
1.10	61.8	426.6	0.0	4.091	0.0	3.091	0.0	213.3	30.9	50.9	351.3
1.59	61.6	425.0	0.0	4.080	0.0	3.080	0.0	212.5	30.8	50.8	350.5
2.13	61.3	423.1	0.0	4.065	0.0	3.065	0.0	211.5	30.6	50.6	349.5
3.19	62.0	428.1	0.0	4.102	0.0	3.102	0.0	214.1	31.0	51.0	352.1
4.34	61.7	425.7	0.0	4.085	0.0	3.085	0.0	212.9	30.8	50.8	350.9
5.22	61.7	425.7	0.0	4.087	0.0	3.087	0.0	212.9	30.8	50.8	350.9
6.98	62.3	430.5	0.0	4.119	0.0	3.119	0.0	215.2	31.1	51.1	353.2
9.40	63.6	439.4	0.0	4.184	0.0	3.184	0.0	219.7	31.8	51.8	357.7
11.82	65.0	448.5	0.0	4.250	0.0	3.250	0.0	224.3	32.5	52.5	362.3
14.24	65.7	453.8	0.0	4.289	0.0	3.289	0.0	226.9	32.8	52.8	364.9

TABLE 2.5-27

TRIAXIAL SHEAR TEST RESULTS

Source: Meany Airport Road Pit
Confining Pressure: σ_3 = 276.0 kn/m^2 (40 psi)

Axial Deviator			Induced Principal Pore			Normalized Data		Stress Path			
	Stress							τ45		Σ45	
Strain (%)	psi	kn/m^2	Pore Pressure (kn/m^2)	Stress Ratio	Pressure Parameter	Deviator Stress	Pore Pressure	kn/m^2	psi	psi	kn/m^2
0.0	0.0	0	0.0	1.000	0.0	0.0	0.0		0.0	40.0	276.0
0.04	8.6	59.8	0.0	1.216	0.0	0.216	0.0	29.9	4.3	44.3	305.9
0.10	19.0	131.1	0.0	1.475	0.0	0.475	0.0	65.6	9.5	49.5	341.6
0.21	31.4	216.9	0.0	1.786	0.0	0.786	0.0	108.5	15.7	55.7	384.5
0.36	47.2	325.9	0.0	2.181	0.0	1.181	0.0	162.9	23.6	63.6	438.9
0.63	64.9	448.4	0.0	2.624	0.0	1.624	0.0	224.2	32.4	72.4	500.2
1.03	79.9	551.8	0.0	2.999	0.0	1.999	0.0	275.9	39.9	79.9	551.9
1.54	89.3	616.3	0.0	3.233	0.0	2.233	0.0	308.2	44.6	84.6	584.2
2.36	97.8	675.1	0.0	3.446	0.0	2.446	0.0	337.6	48.9	88.9	613.6
3.30	104.3	719.9	0.0	3.608	0.0	2.608	0.0	359.9	52.1	92.1	635.9
4.49	110.5	762.8	0.0	3.764	0.0	2.764	0.0	381.4	55.2	95.2	657.4
6.01	116.8	806.1	0.0	3.921	0.0	2.921	0.0	403.1	58.4	98.4	679.1
7.78	121.5	838.9	0.0	4.040	0.0	3.040	0.0	419.5	60.7	100.7	695.5
10.81	128.8	889.3	0.0	4.222	0.0	3.222	0.0	444.7	64.4	104.4	720.7
13.39	133.2	919.6	0.0	4.332	0.0	3.332	0.0	459.8	66.6	106.6	735.8
15.06	134.3	926.9	0.0	4.358	0.0	3.358	0.0	463.4	67.1	107.1	739.4

TABLE 2.5-30

PIEZOMETER READINGS, PI-SERIES

Date	Shaft Water Level Elev. (ft)	Water Level Elevation (ft)							
		PI-1		PI-2		PI-3		PI-4	
		A Sensor	B Sensor	A Sensor	B Sensor	A Sensor	B Sensor	A Sensor	B Sensor
11/12/79	NR	184.1	190.2	188.6	196.9	NR	NR	204.3	202.8
12/09/79	186.0	184.5	189.7	184.9	197.3	184.9	184.2	197.9	--
12/12/79	177.0	178.3	185.6	182.1	197.8	180.8	185.6	207.8	198.6
12/17/79	143.5	151.3	184.2	164.6	196.9	155.8	184.2	189.1	--
12/19/79	142.0	150.4	184.2	162.2	196.4	155.4	184.7	189.1	--
12/21/79	140.0	149.0	184.2	164.2	196.9	154.5	184.7	187.7	--
12/28/79	140.0	149.5	183.8	164.2	195.5	156.5	184.7	188.2	--
01/03/80	140.0	148.5	183.3	163.7	195.0	156.3	184.2	185.4	--
01/17/80	136.0	138.8	--	--	195.5	141.5	184.7	185.4	--
01/29/80	135.0	--	--	--	206.1	136.9	--	188.2	--
02/11/80	130.0	--	--	--	209.6	139.6	--	NR	NR
03/17/80	130.0	--	--	--	198.7	137.8	--	184.5	--
04/14/80	130.0	--	--	--	198.3	136.0	--	184.0	--
05/21/80	130.0	--	--	--	198.3	135.5	--	182.6	--
06/18/80	130.0	--	--	--	201.0	136.0	--	183.1	--
07/21/80	130.0	--	--	--	202.3	136.0	--	NR	NR
09/04/80	130.0	--	--	--	202.0	135.5	--	183.1	--
10/17/80	123.0	--	--	--	202.9	134.2	--	182.6	--
12/01/80	123.0	--	--	--	202.0	132.3	--	181.3	195.8

KEY: -- = Water level below sensor level
NR = No reading

NOTE: Sensor elevations are:

PI-1A	131.0	PI-1B	181.0
PI-2A	160.0	PI-2B	189.5
PI-3A	130.0	PI-3B	181.0
PI-4A	160.5	PI-4B	194.0

TABLE 2.5-31

SUMMARY OF CYCLIC TRIAXIAL TESTS
BORROW SOURCE: CHAUVIN PIT

Initial			After Consolidation			Effective Consolidation Pressure		B at End of Consolidation	Cyclic Deviator Stress		Number of Cycles to:			
Dry Unit Weight		Relative Density (%)	Dry Unit Weight		Relative Density (%)						Double Amplitude Strain			
kN/m³	pcf		kN/m³	pcf		kNm²	psi		kN/m⁻²	psi	2.5%	5%	10%	Failure in Extension
17.9	(113.7)	68	18.0	(113.8)	69	49.0	(7.11)	0.84	34.3	(4.98)	46	48	50	-
18.0	(113.8)	69	18.0	(113.9)	70	49.0	(7.11)	0.97	36.3	(5.26)	27	28	29	-
18.0	(114.3)	73	18.0	(114.4)	73	49.0	(7.11)	0.97	47.1	(6.83)	10	11	-	12
18.0	(114.3)	73	18.0	(114.5)	74	49.0	(7.11)	0.97	43.2	(6.26)	10	11	12	14
18.0	(113.9)	70	18.0	(114.4)	73	147.1	(21.33)	0.96	85.3	(12.38)	18	19	20	21
18.0	(114.4)	73	18.1	(114.9)	76	147.1	(21.33)	0.92	114.7	(16.64)	6	6	-	7
18.0	(113.9)	70	18.1	(114.8)	76	294.2	(42.66)	0.93	161.8	(23.47)	7	7	-	8
17.9	(113.7)	69	18.1	(114.6)	74	294.2	(42.66)	1.00	135.3	(19.63)	28	29	30	31
18.0	(114.1)	71	18.1	(115.0)	77	294.2	(42.66)	0.97	119.7	(17.26)	53	53	54	55
18.2	(116.1)	84	18.2	(116.2)	84	49.0	(7.11)	0.97	52.0	(7.54)	13	16	20	-
18.1	(115.1)	77	18.1	(115.4)	79	49.0	(7.11)	0.95	45.1	(6.54)	29	32	36	-
18.3	(116.5)	86	18.3	(116.6)	87	49.0	(7.11)	0.93	42.2	(6.12)	53	56	66	-
18.2	(115.9)	82	18.3	(116.4)	85	147.1	(21.33)	0.95	85.3	(12.38)	22	23	25	27
18.2	(116.2)	84	18.3	(116.7)	87	147.1	(21.33)	0.93	114.7	(28.03)	7	8	8	9
18.1	(115.7)	81	18.3	(116.6)	87	294.2	(42.66)	0.99	116.7	(16.93)	57	58	59	-
18.2	(115.8)	82	18.3	(116.6)	87	294.2	(42.66)	0.94	143.2	(20.77)	28	29	30	-
18.2	(115.9)	82	18.3	(116.7)	87	294.2	(42.66)	0.94	173.6	(25.18)	11	-	-	-
18.4	(117.1)	90	18.4	(117.2)	90	49.0	(7.11)	0.98	39.2	(5.69)	146	169	304	-
18.5	(118.0)	95	18.5	(118.1)	95	49.0	(7.11)	0.90	42.2	(6.12)	66	73	82	-
18.3	(117.0)	89	18.4	(117.1)	90	49.0	(7.11)	0.96	52.0	(7.54)	26	30	35	-
18.3	(117.0)	89	18.4	(117.1)	90	49.0	(7.11)	0.95	68.7	(9.96)	9	11	13	-
18.4	(117.1)	90	18.4	(117.5)	92	147.1	(21.33)	0.92	100.0	(14.51)	33	36	39	41
18.4	(117.5)	92	18.5	(118.0)	95	147.1	(21.33)	0.88	85.3	(12.38)	172	175	180	-
18.4	(117.2)	90	18.5	(117.7)	93	147.1	(21.33)	0.96	120.6	(17.50)	11	13	15	-
18.4	(117.3)	91	18.5	(118.1)	95	294.2	(42.66)	0.94	146.1	(21.20)	43	44	-	-
18.4	(117.1)	90	18.5	(117.8)	94	294.2	(42.66)	0.93	173.6	(25.18)	27	29	30	-
18.4	(117.3)	91	18.5	(118.1)	95	294.2	(42.66)	0.96	201.0	(29.16)	7	-	-	-

SOURCE: Stone & Webster Geotechnical Laboratory, Reference 208.

TABLE 2.5-32

SUMMARY OF CYCLIC TRIAXIAL TESTS
BORROW SOURCE: MEANY ENGLES ROAD PIT

Initial			After Consolidation			Effective Consolidation Pressure		B at End of Consolidation	Cyclic Deviator Stress		Number of Cycles to:			
Dry Unit Weight		Relative Density (%)	Dry Unit Weight		Relative Density (%)						Double Amplitude Strain			Failure in Extension
kN/m³	pcf		kN/m³	pcf		kNm²	psi		kN/m²	psi	2.5%	5%	10%	
18.1	(115.2)	50	18.1	(115.2)	50	147.1	(21.33)	0.96	82.4	(11.95)	5	6	6	7
18.1	(115.3)	50	18.1	(115.3)	50	147.1	(21.33)	0.98	60.8	(8.82)	19	20	21	23
18.1	(115.3)	50	18.1	(115.3)	50	147.1	(21.33)	0.96	47.1	(6.82)	90	92	93	97
18.1	(115.2)	50	18.1	(115.2)	50	294.2	(42.66)	0.95	163.8	(23.75)	6	6	7	9
18.2	(115.5)	51	18.2	(115.5)	51	294.2	(42.66)	0.95	138.3	(20.05)	10	11	12	14
18.1	(115.4)	51	18.1	(115.4)	51	294.2	(42.66)	0.95	111.8	(16.21)	15	15	16	18
18.1	(115.2)	50	18.2	(116.0)	52	440.3	(63.99)	0.95	242.2	(35.13)	5	5	6	7
18.2	(115.5)	51	18.2	(115.5)	51	440.3	(63.99)	0.95	182.4	(26.46)	10	11	11	11
18.2	(115.5)	51	18.2	(115.5)	51	440.3	(63.99)	0.95	124.6	(18.06)	71	73	73	73
18.6	(118.5)	60	18.6	(118.5)	60	147.1	(21.33)	0.95	61.8	(8.96)	16	18	20	24
18.7	(118.7)	61	18.7	(118.7)	61	147.1	(21.33)	0.95	88.3	(12.80)	8	10	11	12
18.6	(118.4)	60	18.6	(118.4)	60	147.1	(21.33)	0.95	51.0	(7.40)	55	58	62	82
18.6	(118.5)	60	18.6	(118.5)	60	294.2	(42.66)	0.93	166.7	(24.18)	8	9	11	12
18.6	(118.5)	60	18.6	(118.5)	60	294.2	(42.66)	0.95	127.5	(18.49)	22	23	25	26
18.6	(118.5)	60	18.6	(118.5)	60	294.2	(42.66)	0.94	98.1	(14.22)	82	85	89	96
18.6	(118.5)	60	18.6	(118.5)	60	440.3	(63.99)	0.90	257.9	(37.40)	6	7	9	9
18.7	(118.8)	61	18.7	(118.8)	61	440.3	(63.99)	0.94	197.1	(28.59)	18	20	24	25
18.6	(118.6)	60	18.7	(119.4)	62	440.3	(63.99)	0.95	163.8	(23.75)	39	41	42	42
19.2	(122.1)	70	19.2	(122.1)	70	147.1	(21.33)	0.95	78.5	(11.37)	26	32	45	48
19.2	(122.0)	70	19.2	(122.0)	70	147.1	(21.33)	0.95	62.8	(9.10)	84	93	151	175
19.3	(122.6)	72	19.3	(122.6)	72	147.1	(21.33)	0.98	108.9	(15.79)	9	13	16	18
19.2	(122.1)	70	19.2	(122.1)	70	147.1	(21.33)	0.95	175.6	(25.46)	8	12	17	20
19.2	(122.1)	70	19.2	(122.1)	70	147.1	(21.33)	0.95	369.7	(63.62)	2	4	5	8
19.2	(122.1)	70	19.2	(122.1)	70	294.2	(42.66)	0.96	128.5	(18.63)	27	31	36	37
19.2	(122.2)	71	19.2	(122.2)	71	294.2	(42.66)	0.95	112.8	(16.36)	86	92	109	141
19.2	(122.0)	70	19.2	(122.0)	70	294.2	(42.66)	0.93	136.3	(19.77)	26	29	35	36
19.2	(121.9)	70	19.2	(121.9)	70	294.2	(42.66)	0.96	272.6	(39.54)	8	12	16	17
19.2	(122.1)	70	19.2	(122.1)	70	294.2	(42.66)	0.95	303.0	(43.95)	5	7	9	9
19.2	(122.2)	71	19.2	(122.2)	71	294.2	(42.66)	0.96	363.8	(52.76)	4	7	9	9
19.2	(122.2)	71	19.2	(122.2)	71	440.3	(63.99)	0.91	317.8	(46.08)	9	13	18	19
19.2	(122.3)	71	19.2	(122.3)	71	440.3	(63.99)	0.91	386.4	(56.03)	7	11	15	15
19.2	(122.0)	70	19.2	(122.0)	70	440.3	(63.99)	0.91	257.9	(37.48)	16	21	29	30
19.2	(122.0)	70	19.2	(122.0)	70	440.3	(63.99)	0.95	447.2	(64.85)	5	8	10	11

SOURCE: Goldberg-Zoino-Dunnicliff & Associates, Inc., Reference 209.

TABLE 2.5-45

MAXIMUM RESPONSES OF INPUT EARTHQUAKES TO
DUCTLINES 907 AND 922 AND MANHOLE NO. 1, CL-1E

	Maximum Acceleration						Avg. Max. Acceleration of Three Earthquakes
	Helene		Santa Anita		Lake Hughes		
	Range	Avg.	Range	Avg.	Range	Avg.	
Ductlines 907 and 922, depth at 10'-14'	0.17 - 0.184 (g)	0.18 g	0.247 - 0.292 (g)	0.27 g	0.247 - 0.3 (g)	0.27 g	0.24 g
Manhole No. 1, CL-1E, depth at 0'-17'	0.16 - 0.205 (g)	0.18 g	0.208 - 0.328	0.27 g	0.208 - 0.338	0.27 g	0.24 g

NOTES:

1. Based on Figure 2.5-151.
2. The corresponding frequency for the buried ductlines and manhole is at least 20 Hz for the maximum acceleration of 0.24 g.

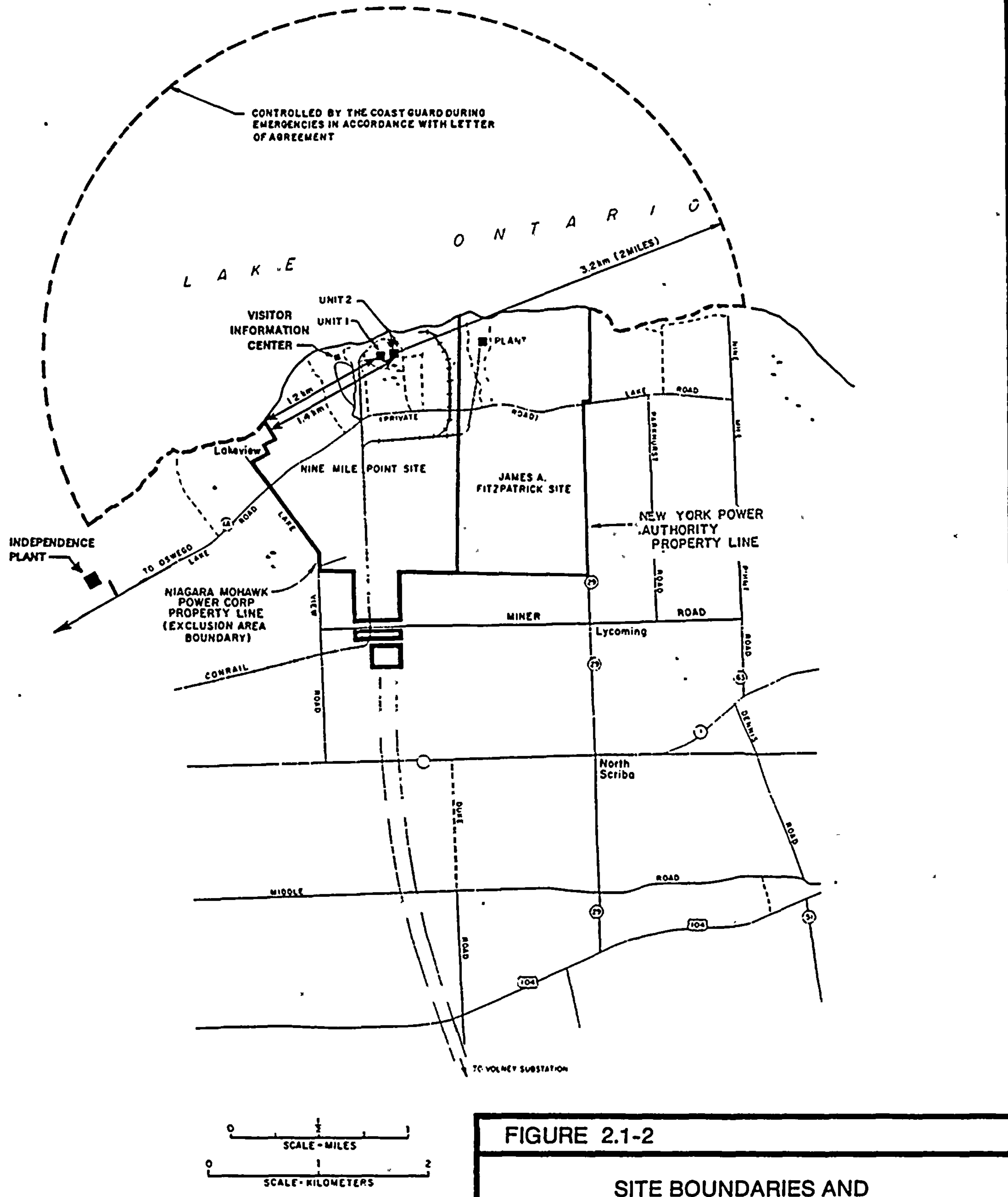

FIGURE 2.1-2

SITE BOUNDARIES AND TRANSPORTATION ROUTES

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

2.2 NEARBY INDUSTRIAL, TRANSPORTATION, AND MILITARY FACILITIES

2.2.1 Locations and Routes

Only one manufacturing or industrial plant, Alcan Aluminum Corporation's Alcan Sheet and Plate Division, is located within 8 km (5 mi) of Unit 2. There are also three electrical power generation facilities, the J. A. FitzPatrick Nuclear Power Plant operated by NYPA, Unit 1 operated by NMPC, and Independence Generation Plant operated by Sithe Energies USA, located within 8 km (5 mi) of Unit 2. Figure 2.1-2 shows the location of these three facilities relative to Unit 2.

The principal products of the Alcan Aluminum Corporation plant are aluminum sheet and plate. There are no chemical plants, refineries, military bases, or underground gas storage facilities within 8 km of the plant. In addition, no fuel storage facilities lie within the 8-km radius except those storage facilities associated with the Alcan plant, the FitzPatrick plant, Sithe plant, and Units 1 and 2. Two natural gas pipelines lie within 8 km of the plant; one pipeline supplies the Sithe plant and the other supplies INDECK Energy. Both are located on the north-south and east-west transmission line corridors. Finally, there are no hazardous waste storage or disposal sites permitted by the state in the 8-km radius from the plant.

Major transportation facilities are shown on Figure 2.2-1. The principal roadway within 8 km of Unit 2 is U.S. Route 104 which passes 6.2 km (3.9 mi) south of the plant and connects the city of Oswego and Mexico Village. Daily traffic volume for U.S. Route 104 was 5,841 vehicles in 1979. Highway access to the site is via two county routes, Route 1A to the southwest and Route 29 to the east. A private east-west road crosses the site and connects these two county routes. Other local roads in the vicinity generally had average daily traffic counts of fewer than 2,000 vehicles in 1978-1979, the most current survey dates[1]. Table 2.2-1 presents daily traffic volume counts for county highways within 8 km of the plant.

One railroad company, Conrail, transports freight in the vicinity of the plant. The rail lines and spurs serving Unit 2, as well as the J. A. FitzPatrick plant and Unit 1, are shown on Figure 2.1-2. The closest rail line to Unit 2 is the Oswego-Mexico branch of Conrail located approximately 2.5 km (1.5 mi) from the Nine Mile Point site. This branch line has daily service on demand and averages one train daily, 5 days a week. A rail spur was constructed to serve Unit 2 during construction and operation of the plant. Possible sources of traffic or hazardous materials utilizing ground transportation routes within 8 km of the plant are identified and detailed in Section 2.2.3.

The Oswego River passes within 11 to 12 km (6.6 to 7.2 mi) of Unit 2 at its nearest point and serves as a major route for waterborne commerce on Lake Ontario. Freight traffic statistics are maintained by the U.S. Army Corps of Engineers. Totals for the river section from New York State Barge Canal Lock No. 8 to the port of the City of Oswego are the only statistics applicable for the nearest reach of river to the Station. Table 2.2-2 details the 1978 freight traffic for this reach of river. The Port of Oswego, the easternmost port on Lake Ontario, is located approximately 11 km (6.6 mi) southwest of Unit 2 and provides a link with all ports on the Great Lakes and St. Lawrence River. Ships in normal commercial lanes bound to and from the Port of Oswego pass no closer than 11.3 km (7 mi) to the intake structures of Unit 2.

Regular commercial air service is provided at the Clarence E. Hancock Airport, located 49.8 km (31 mi) southeast of Unit 2 near Syracuse, NY. The nearest flight corridor associated with this airport is 22.2 km (13.8 mi) from the Nine Mile Point Station. Light plane traffic is handled at the Oswego County Airport in the town of Volney, approximately 19.3 km (12 mi) south of the Nine Mile Point site. Lakeside Airstrip, a private facility which operates primarily as a maintenance facility with very little air traffic, is located along Route 176 approximately 10 km (6.2 mi) south of the Nine Mile Point site. Helicopter service is provided for local transportation between Hancock Airport and the site. The service involves approximately 45 to 60 flights per year to three landing locations on the site approximately 1000 to 2000 feet west and south of the reactor building.

It is not anticipated that there will be any significant increase in the number of industrial, transportation, and military facilities located within an 8-km radius of Unit 2 over the plant lifetime. There are also no significant changes expected in the nature of existing facilities or the extent of their activities within the designated area over the projected lifetime of Unit 2.

2.2.2 Description

2.2.2.1 Description of Facilities

Major industrial facilities within 8 km of Unit 2 are listed in Table 2.2-3. Alcan Rolled Products, located approximately 4.5 km (2.7 mi) southwest of Nine Mile Point Station, is the largest employer with approximately 1,000 workers manufacturing aluminum sheet and plating. No hazardous materials are manufactured within the 8-km radius. Hazardous materials stored or used are discussed in Section 2.2.2.2. The New York State Department of Environmental Conservation records the type, amount, and route of hazardous materials carried in the state.

One rail line passes through the 8-km radius. Conrail has a branch line serving Alcan Aluminum, Units 1 and 2, and the James A. FitzPatrick Nuclear Power Plant. The main line is located approximately 3.5 km (2.1 mi) at its nearest point from Unit 2.

Waterborne commerce statistics prepared by the U.S. Army Corps of Engineers only partially identify specific hazardous materials on Lake Ontario. Commerce is recorded by general commodity[2]. A listing of hazardous materials identified and their commodity designations is provided in Table 2.2-4.

Hazardous materials transported by air have not been identified because of the distance of airways and facilities from Unit 2.

2.2.2.2 Description of Products and Materials

To identify hazardous materials regularly stored or used within 8 km of Unit 2, surveys were conducted of industrial firms, pipeline companies, and distributors that might be expected to handle toxic chemicals or explosives. Appendix 2A describes the method used to collect data regarding hazardous materials used by various industries near the site. Hazardous materials considered are included in Table 2.2-4. Toxic chemicals and explosives stored or used by industries or distributors in the vicinity of the Station are summarized in Table 2.2-5. Two natural gas pipelines and a small propane distribution company are located within 8 km of Unit 2.

Waterborne commerce for 1978 Lake Ontario traffic is described in Tables 2.2-2 and 2.2-6. Approximately 1.2 million tons of cargo were transported on Lake Ontario. Since more specific commodity categories are not used in data collection, there are no means of identifying types, frequency, and amounts of hazardous material shipments past the site.

The nearest passage of commercial vessels to Unit 2 occurs when navigating to and from the City of Oswego harbor, located approximately 10 km (6.2 mi) from Nine Mile Point Station. The Port of Oswego Authority indicates that none of the hazardous materials listed in Table 2.2-5 have been transported on Lake Ontario, either originating at or destined to the Port of Oswego. All industries reported receiving hazardous material shipments via U.S. Highway 104 and County Route 1 by truck.

2.2.2.3 Projections of Industrial Growth

There are no plans for major expansion in transportation, storage, or industrial facilities in the vicinity of Unit 2.

2.2.3 Evaluation of Potential Accidents

The consideration of a variety of potential accidents, and their effects on the plant or plant operation, is included in this section. Types of accidents considered include explosions, flammable vapor clouds, toxic chemicals, fires, collisions with intake structures, and liquid spills.

2.2.3.1 Determination of Design Basis Events

2.2.3.1.1 Explosions

Based on a comprehensive survey of industries within a 10-km (6.2-mi) radius of Unit 2, the nearest highway on which explosive materials can be transported is Route 104, which is a distance of about 6.2 km (3.9 mi) from safety-related structures. This separation distance far exceeds the safe distance for truck traffic (approximately 548.6 m, 1,800 ft) given on Figure 1 of RG 1.91.

In discussions with Conrail, it was determined that no explosive or flammable materials are transported to the Oswego terminal of the rail line between Oswego and Mexico, NY. In any event, the distance from this rail line to Unit 2 is much greater than the safe distance for rail traffic given in RG 1.91.

Since the nearest commercial shipping lanes on Lake Ontario are more than 10 km (6.2 mi) from Unit 2 (according to the U.S. Coast Guard), potential explosions on a ship or barge are not considered a design basis event[3]. This distance is well beyond the radius of the peak incident pressure of 1 psi as given in RG 1.91. Therefore, according to guidance contained in RG 1.91, explosions on nearby transportation routes are not considered design basis events due to the separation distances of potential sources of explosions from Unit 2.

2.2.3.1.2 Flammable Vapor Clouds (Delayed Ignition)

Propane stored at the James A. FitzPatrick plant is the only potential source of a flammable vapor cloud that might affect the Unit 2 site[4]. Approximately 3,785 l (1,000 gal) of propane at the James A. FitzPatrick plant is stored about 700 m (2,297 ft) from the Unit 2 containment building. An analysis has been performed to assess the potential of a 1-psi overpressure occurring at the Unit 2 containment building as a result of the delayed ignition of a flammable vapor cloud of propane. A 1-psi overpressure is that pressure below which no significant damage to critical plant structures is expected, as determined by the U.S. NRC in RG 1.91, "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants."

The delayed ignition analysis was performed utilizing a computer program (GASBLAST) that calculates the extent and volume of the mixture of air and gaseous explosive for a given set of meteorological conditions and rich and lean concentration detonability limits. The TNT equivalent and incident blast pressure isobars, represented by concentric circles emanating from the center of the plume, are then determined. The program assumes that the gas is released at ground level at a constant release rate. The analytical method that forms the basis of the program follows the computational model established by Burgess and Zabetakes[12]. The blast effect is determined by using the

Only those chemicals that have the potential to form a toxic vapor cloud or plume after release to the environment need to be evaluated. This criterion is met by all chemicals listed in Table 2.2-7.

Control Room Habitability Determination

The effect of an accidental release of each of the chemicals described in the previous section on control room habitability is evaluated by calculating vapor concentrations inside the control room as a function of time following the accident. This calculation is performed using the conservative methodology outlined in NUREG-0570 and utilizing the assumptions described in RG 1.78.

In a postulated accident, the entire content of the largest single storage container is released, resulting in a toxic vapor cloud and/or plume that is conservatively assumed to be transported by the wind directly toward the control room intake. The formation of the toxic cloud and/or plume is dependent on the characteristics of the chemical and the environment. The entire amount of a chemical stored as a gas is treated as a puff or cloud that has a finite volume determined from the quantity and density of the stored chemical. A substance stored as a liquid with a boiling point below the ambient temperature forms an instantaneous puff due to flashing (rapid gas formation) of some fraction of the stored quantity. The remaining liquid forms a puddle that quickly spreads into a thin layer on the ground, subsequently vaporizing and forming a ground-level vapor plume. A high boiling point liquid (above ambient temperature) forms a puddle that evaporates by forced convection with no flashing involved.

The calculations are done by a computer program (VAPOR) based on NUREG-0570 methodology that requires the following input information: chemical physical properties, control room parameters, meteorology, distance from the spill to the control room intake, quantity of chemical released, and toxicity limits. The following Unit 2 control room parameters are used: ventilation rate of 0.708 m^3/sec (1,500 ft^3/min), and net free volume of 5,935 m^3 (209,600 ft^3). The most conservative meteorological condition is assumed for the calculation, consisting of Pasquill Class G stability, a wind speed of 0.5 m/sec (1.6 ft/sec), and an ambient temperature of 33°C (91°F).

The criteria for determining chemical toxicity and setting limits for habitability determinations are taken from regulatory guidance documents. According to RG 1.78, the toxicity limit of a chemical is the maximum concentration that can be tolerated by an average human for 2 min without physical incapacitation (severe coughing, eye burn, severe skin irritation). Standard Review Plan (SRP) Section 6.4 states that acute effects should be reversible within a short period of time (several minutes) without the benefit of medication other than the use of self-contained breathing apparatus (SCBA). The acute toxicity limits listed in RG 1.78 are used in this study except that, where more appropriate, documented sources are available[5-9].

Nonguideline toxicity limits are based on concentrations that produce no effects or minor irritation affecting mental alertness and physical coordination, assuming a 15-min exposure time. In cases where appropriate human data are not available, data are used by applying a conservative factor of 10 to lower the acute exposure limit.

Results and Conclusions

The results of the analysis are summarized in Table 2.2-8, which indicates that none of the toxic chemicals evaluated have the potential to incapacitate the Control Room Operators.

2.2.3.1.4 Fires

The production of high heat fluxes and smoke from fires at industrial or storage facilities, oil and gas pipelines, transportation routes, or homes in the site vicinity does not present a hazard to the safe operation of the plant due to the large separation distances of these potential fires from the site. The nearest storage facilities of flammable materials in large quantities and the nearest oil pipeline are over 10 km (6.2 mi) from the Nine Mile Point site. The nearest gas pipeline is over 3.2 km (2 mi) from Nine Mile Point site. The nearest truck route (Route 104) passes the site at a distance of about 6.2 km (3.9 mi) from the plant. There are no known regular shipments of flammable materials on Route 104 with the exception of possible local gasoline deliveries. The nearest residence is approximately 1.6 km (1 mi) from the site.

The site is sufficiently cleared in areas adjacent to the plant that forest or brush fires pose no safety hazards. Onsite fuel storage fires do not jeopardize plant safety since these facilities are designed in accordance with applicable fire codes. A detailed description of the plant fire protection system is presented in Section 9.5.1.

2.2.3.1.5 Collisions with Intake and Discharge Structures

Oswego Harbor is located approximately 12 km (7.5 mi) southwest of the intake structures. The intake structures are located approximately 305 m (1,000 ft) offshore in a water depth of 6 m (20 ft) at the minimum controlled lake level.

In accordance with Coast Guard recommendations, the intake structures are constructed with the tops 3 m (10 ft) below the minimum controlled lake level during the navigational season. Even greater protection is afforded the discharge structure since it is located 1,500 ft offshore and covered by approximately an additional 12 ft of water.

TABLE 2.2-3

INDUSTRIAL FIRMS WITHIN 8 KM (5 MI) OF UNIT 2

Firm	Distance/ Direction from Site (km)	Products	Employment
Alcan Aluminum Corporation	4.5/SW	Aluminum sheet and plate	1,000
James A. FitzPatrick Nuclear Power Plant	<1/E	Electrical generation	515
Nine Mile Point Unit 1	Adjacent to Unit 2	Electrical generation	450
Sithe Energies USA Independence Generation Plant	3.5/SW	Electrical generation	75

SOURCE: References 4, 10, 11, 17, 18

TABLE 2.3-4A

METEOROLOGICAL INSTRUMENTATION SPECIFICATIONS

Operational Measurements Program

Parameter	Instrument	Specification	Value
Wind direction	Teledyne Geotech Sensor - 50.2C/50.2D Vane - 53.2 Processor - 21.21	Accuracy Damping ratio Distance constant Range Threshold	±2 deg 0.4 1.1 m (3.7 ft) 0-360/540 deg 0.30 m/sec (0.7 mph)
Wind speed	Teledyne Geotech Sensor - 50.1B Cups - 52.1 Processor - 40.12c & 21.11	Accuracy Distance constant Starting threshold Range	±0.67 m/sec (±0.15 mph) or 1% 1.5 m (5.0 ft) 0.27 m/sec (0.6 mph) 0-45 m/sec (0-100 mph)
Temperature	Teledyne Geotech Sensor - Platinum RTD Processor - 21.32 Aspirated Thermal Radiation Shield 327	Ambient temperature range Temperature difference Range Linearity Error	-40 to 43°C (-40 to 110°F) -4 to 11°C (-8 to 20°F) ±0.2°C (±0.4°F) 0.1°C (0.2°F) under radiation of 1.6 cal/cm^2/min (353.8 Btu/ft^2/hr)
Dew point	General Eastern 1200EPS or E1	Range	-40 to 43°C (-40 to 110°F)
Precipitation	Belfort Instrument Company Tipping Bucket	Calibration increment	0.25 mm (0.01 in)
Barometric Pressure	Yellow Springs Instrument Company Sensor - 2014-28/32-HA-3WH Teledyne Geotech Processor - 40.61 & 21.61	Range	948 to 1084 mb (28.00 to 32.00 in Hg)

2.4 HYDROLOGIC ENGINEERING

2.4.1 Hydrologic Description

2.4.1.1 Site and Facilities

Unit 2 is located on the western portion of the Nine Mile Point promontory on the southeastern shore of Lake Ontario in Oswego County, NY. All elevations in this report refer to the USLS 1935 Data.

1. To convert elevations from USLS 1935 to 1955 International Great Lakes Data, subtract 0.375 m (1.23 ft).

2. To convert elevations from USLS 1935 to 1985 International Great Lakes Data, subtract 0.217 m (0.71 ft).

The natural grade elevation of the Nine Mile Point site varies between el 78.03 m (256 ft) and el 80.77 m (265 ft). There are no perennial streams located on the site. Precipitation at the site is carried to Lake Ontario via drainage ditches, storm sewers, and groundwater flow.

A revetment ditch system is constructed along the lakeshore in front of Unit 2. The top of the revetment is at el 80.16 m (263 ft) and prevents possible plant flooding due to lake wave action (Section 2.4.5). A ditch located immediately south of the revetment collects rainfall runoff flowing north toward the lake and conveys the flow to both ends of the revetment.

All personnel entrances to Category I structures are at el 79.55 m (261 ft) or higher. A detailed description of the water level (flood) design is found in Section 3.4.

2.4.1.2 Hydrosphere

Lake Ontario, the easternmost of the Great Lakes, is an international body of water forming part of the border between the United States and Canada. The lake is 310.6 km (193 mi) long and 85.3 km (53 mi) wide at its largest points, and has a surface area of 19,010.6 sq km (7,340 sq mi) or 1.901 million ha (4.7 million acres). It has a maximum depth of 244.4 m (802 ft), an average depth of approximately 86.3 m (283 ft), and a volume of 1,638 cu km (393 cu mi) or 0.164 billion ha-m (1.34 billion acre-ft).

Inflow into the western end of Lake Ontario averages approximately 5,806 cu m/sec (205,000 cu ft/sec [cfs]). Runoff directly into Lake Ontario from 70,707 sq km (27,300 sq mi) of watershed in New York State and the province of Ontario amounts to an additional 1,020 cu m/sec (36,000 cfs). The main feeder is the Niagara River; other large rivers draining into the lake are the Genesee and the Oswego from the south shore, the Black River from the east shore, and the Trent River from the north shore. The outflow from the lake into the St. Lawrence River averages about 6,824 cu m/sec (241,000 cfs).

During the winter, ice cover forms in the slack water bays, but the lake itself is seldom more than 25 percent ice-covered. Lake

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Ontario's outflow river, the St. Lawrence, is ice-covered from late December until the end of March, all the way from the lake to the international boundary at Massena, NY.

Prior to the beginning of flow regulation, the elevation of the lake surface was controlled by a natural rock weir located about 6.4 km (4 mi) downstream from Ogdensburg, NY, in the Galop Rapids reach of the St. Lawrence River. The 111-yr record of the USLS (1860 to 1970) indicates a mean lake surface elevation of 75 m (246 ft). Over this period, the maximum monthly lake surface elevation was 75.98 m (249.29 ft) and the minimum was 73.97 m (242.68 ft), a range of 2.01 m (6.61 ft). The annual range of elevations varies between 1.09 and 0.21 m (3.58 and 0.69 ft).

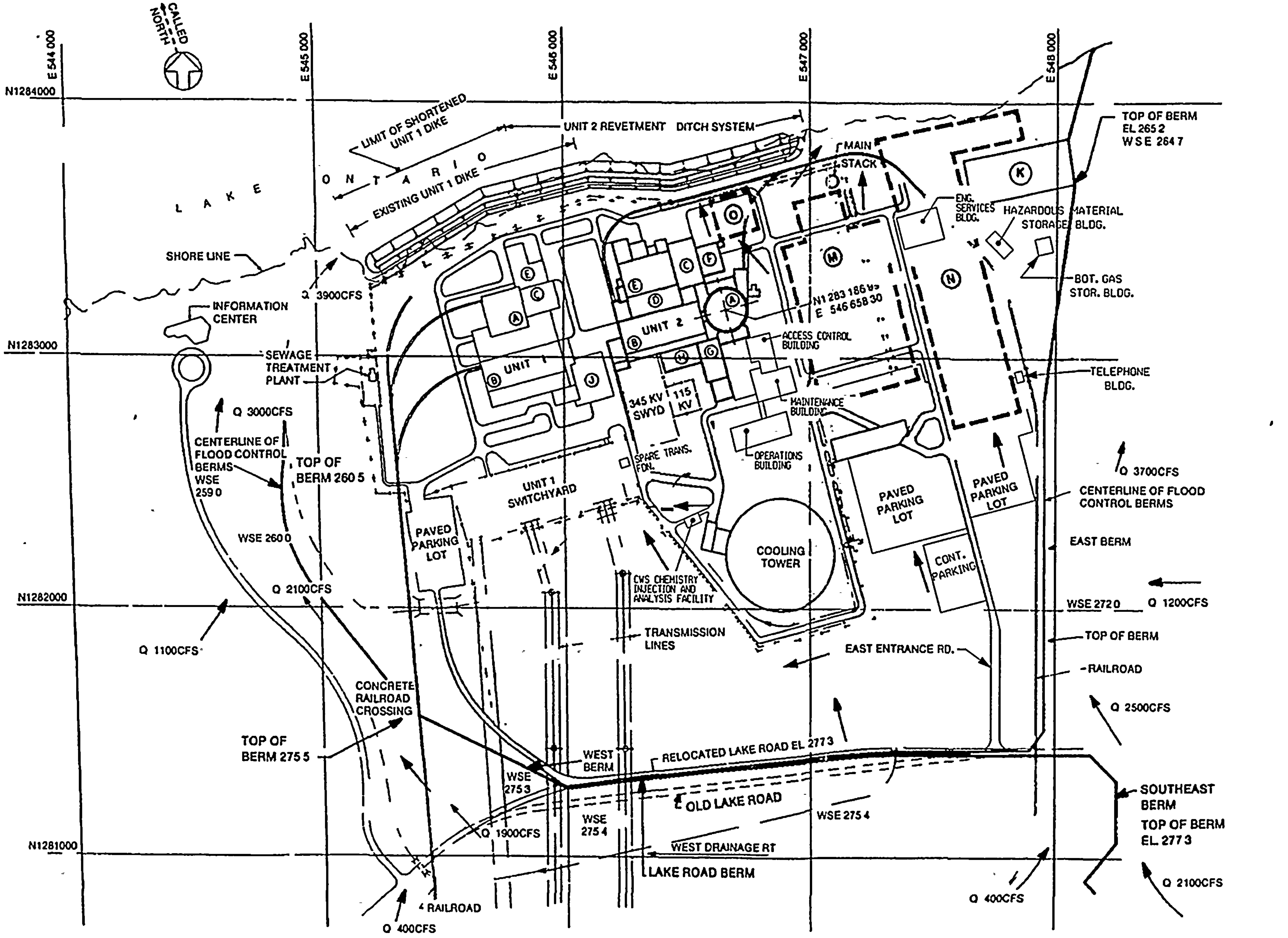
Dams on the St. Lawrence River, under the authority of the International St. Lawrence River Board of Control, are now used to regulate the lake level. The low limit is set for el 74.37 m (244 ft) on April 1 and is maintained at or above that elevation during the entire navigation season (April 1 to November 30). The upper limit of the lake level is el 75.59 m (248 ft).

Water surface setup and seiche are produced by winds and atmospheric pressure gradients. These short-term lake fluctuations are generally less than 0.6 m (2 ft) in amplitude. Winds are directly related to the formation of surface waves, the magnitude of which varies between 0 and 4.6 m (15 ft) in height during a given year. Tide magnitudes amount to less than 2.5 cm (1 in).

The average annual precipitation in the site area is about 92 cm (36 in). It is estimated that approximately 46 cm (18 in) are lost as runoff into stream flow. Of the remaining 46 cm (18 in), approximately 41 cm (16 in) are lost via evaporation from land and water surfaces and transpiration by plants, referred to together as evapotranspiration. The remaining 5 cm (2 in) are available for groundwater recharge. The relatively high runoff can be attributed to the low permeability of the glacial soils and rock formations. The historical maximum precipitation in the vicinity of the site[1] is listed in Table 2.4-1.

Unit 2 is located between two surface water users employing once-through cooling water systems. Unit 1 is located immediately west of Unit 2 and recirculates an average of 1,011 cu m/min (268,000 gpm) of Lake Ontario. James A. FitzPatrick Nuclear Power Plant, located immediately east of Unit 2, recirculates an average of 1,401 cu m/min (370,200 gpm).

The only major public water supplies within a 50-km (30-mi) radius of the site that draw water from the lake through a common intake are the city of Oswego and the OCWD. All water supply systems and industrial users drawing from U.S. waters on Lake Ontario are listed in Table 2.4-11. Data on Canadian water suppliers and industrial users are provided in Table 2.4-12. The 16 U.S. and 10 Canadian municipal water supplies and industrial



IDENTIFICATION LEGEND

A	REACTOR BUILDING
B	TURBINE BUILDING
C	RADWASTE BUILDING
D	HEATER BAYS
E	SCREENWELL BUILDING
F	CONDESATE STORAGE TANK BLDG
G	CONTROL BUILDING
H	NORMAL SWITCHGEAR BUILDING
J	ADMINISTRATION BUILDING
K	WAREHOUSE
LMNO	MAJOR AREAS CONTAINING CONSTRUCTION BUILDINGS CONSIDERED IN THE ANALYSIS

NOTES

1 GRID COORDINATES REFER TO NEW YORK STATE COORDINATE SYSTEM
2 WSE WATER SURFACE ELEVATION
3 PMF DRAINAGE PATH DURING PMP
4 PMF DRAINAGE PATH DURING PMP

FIGURE 2.4-1

PMF DRAINAGE

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TABLE 2.5-28A

QUALITY CONTROL PROGRAM FOR CATEGORY I
STRUCTURAL FILL AND GRANULAR FILL

Test or Inspection	Test Designation	Minimum Frequency
Gradation test	ASTM C136-81	1/1,000 cu yd for structural fill 1/4,000 cu yd for granular fill
Moisture density test	ASTM D1557-78	1/1,000 cu yd for structural fill 1/2,000 cu yd for granular fill
Backfill conditions		Each lift
Backfill and compaction procedure		Each lift
Lift thickness		Each lift
Compaction equipment		Prior to each placement
Passes of compaction equipment		Each lift
In-place density test	ASTM D1556-74 ASTM D2922-81	For open areas, 1/500 cu yd[1] For confined areas, 1/100 cu yd[2]
Relative density test	ASTM D2049-69	Prior to placement, Category I structural fill only

(1) Perform at least one test every lift if the lift is less than 500 cu yd.
(2) Perform at least one test every other lift if the lift is less than 100 cu yd.

TABLE 2.5-48A

TRIAXIAL COMPRESSION TEST DATA SUMMARY
FOR LACUSTRINE SILTY CLAY SPECIMENS

Type of Test: Consolidated Undrained

Type of Specimens: Undisturbed Block Samples

Size of Specimens: 1.4 in Diameter by 3.5 in Height

Soil Description: Silty Clay, Moderately Plastic, Yellow-Brown

Sample Number	82A	82B	82C
Depth (ft)	-	-	-
Specimen Properties			
Initial			
w (%)	21.9	23.7	22.4
r_d (pcf)	102.0	100.1	100.3
e	0.652	0.683	0.681
After Consolidation			
w (%)	25.1	25.1	26.2
r_d (pcf)	101.8	101.3	102.4
e	0.655	0.664	0.647

TABLE 2.5-48B

TRIAXIAL COMPRESSION TEST DATA SUMMARY
FOR LACUSTRINE SILTY CLAY SPECIMENS

Type of Test: Consolidated Undrained

Type of Specimens: Undisturbed Block Samples

Size of Specimens: 1.4 in Diameter by 3.5 in Height

Soil Description: Silty Clay, Moderately Plastic, Gray with Yellow

Sample Number	81G	81H	81I
Depth (ft)	-	-	-
Specimen Properties			
Initial			
w (%)	18.2	19.5	21.3
r_d (pcf)	108.1	106.1	102.3
e	0.560	0.578	0.646
After Consolidation			
w (%)	20.0	20.3	22.2
r_d (pcf)	108.1	108.0	104.6
e	0.560	0.561	0.611

TABLE 2.5-48C

TRIAXIAL COMPRESSION TEST DATA SUMMARY
FOR ORGANIC SILT SPECIMENS

Type of Test:	Consolidated Undrained		
Type of Specimens:	Undisturbed Block Samples		
Size of Specimens:	1.4 in Diameter by 3.5 in Height		
Soil Description:	Organic Silt, Highly Plastic, Dark Gray to Black		
Sample Number	81A	81B	81C
Depth (ft)	-	-	-
Specimen Properties			
Initial			
w (%)	46.7	47.1	45.1
r_d (pcf)	67.0	65.7	69.0
e	1.423	1.469	1.352
After Consolidation			
w (%)	51.3	51.6	48.6
r_d (pcf)	67.1	67.3	71.4
e	1.417	1.413	1.275

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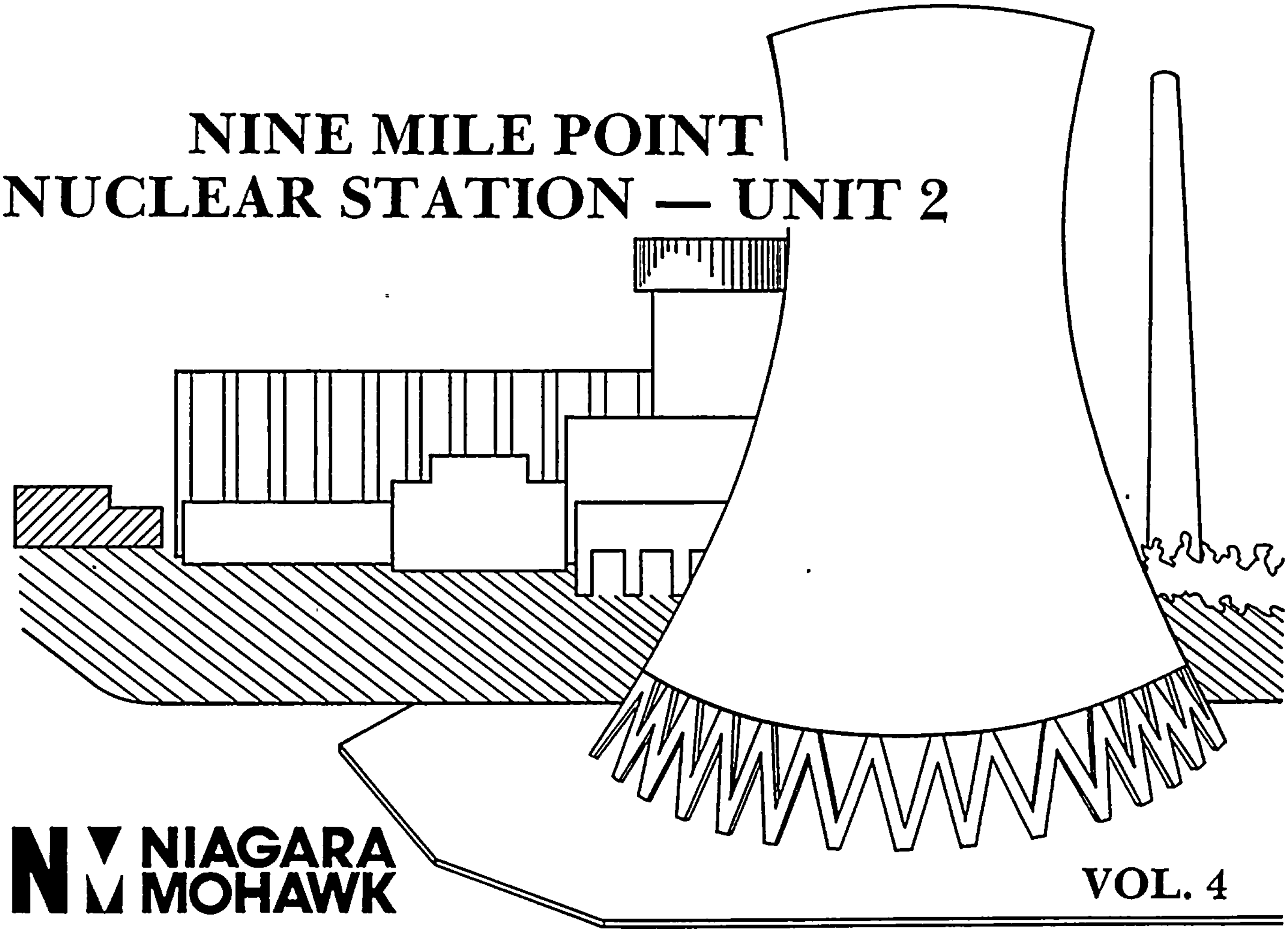

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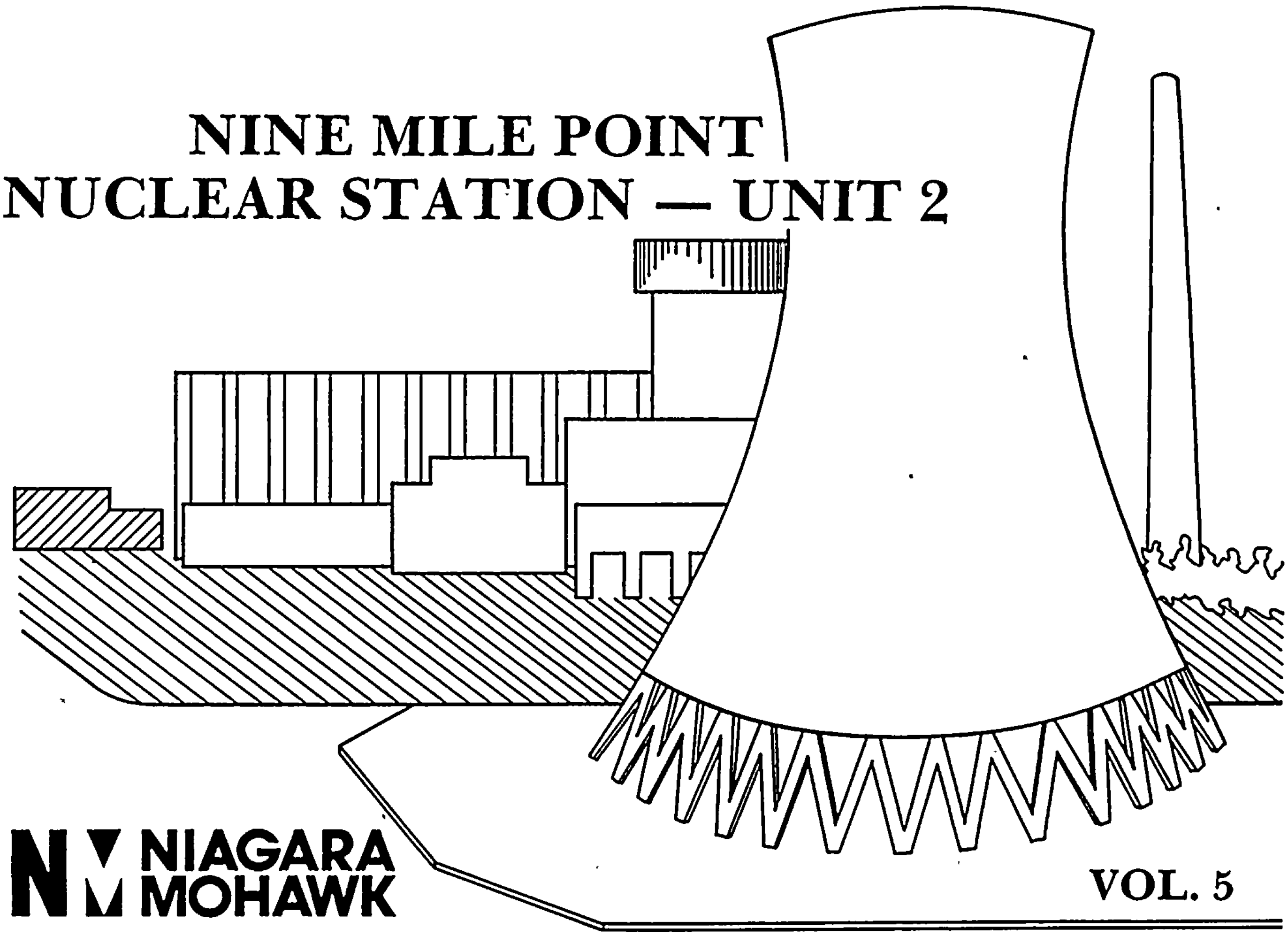

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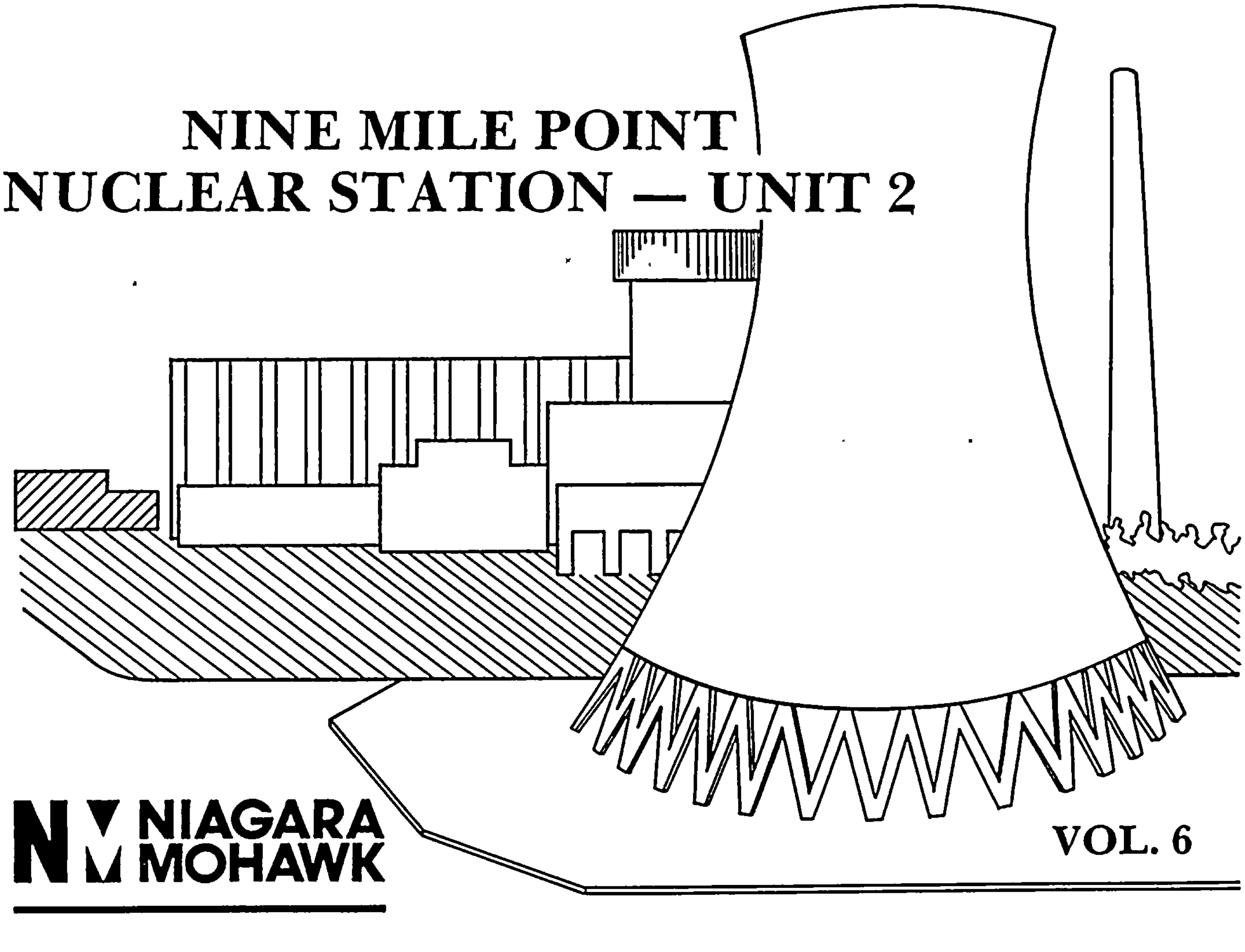

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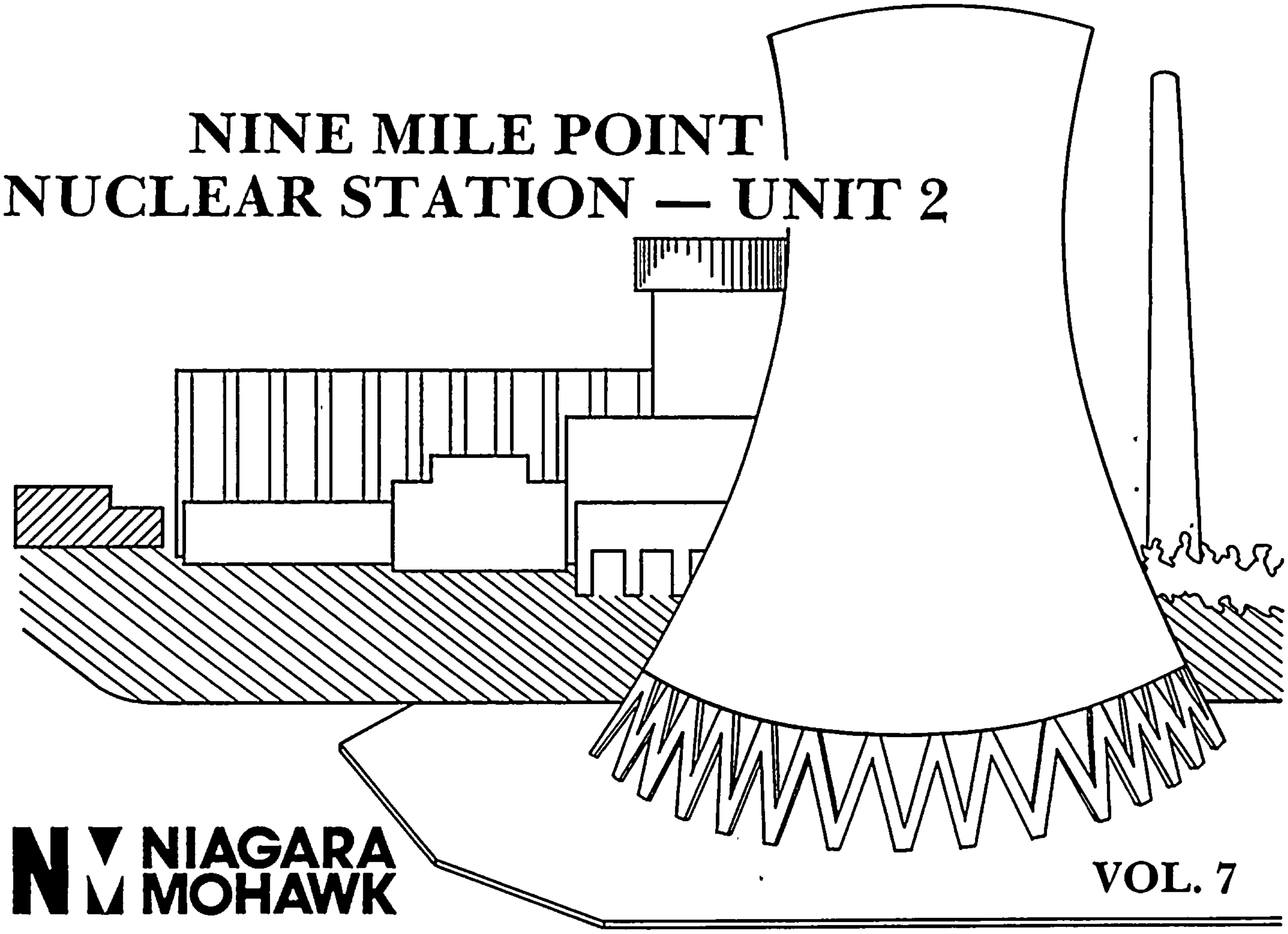

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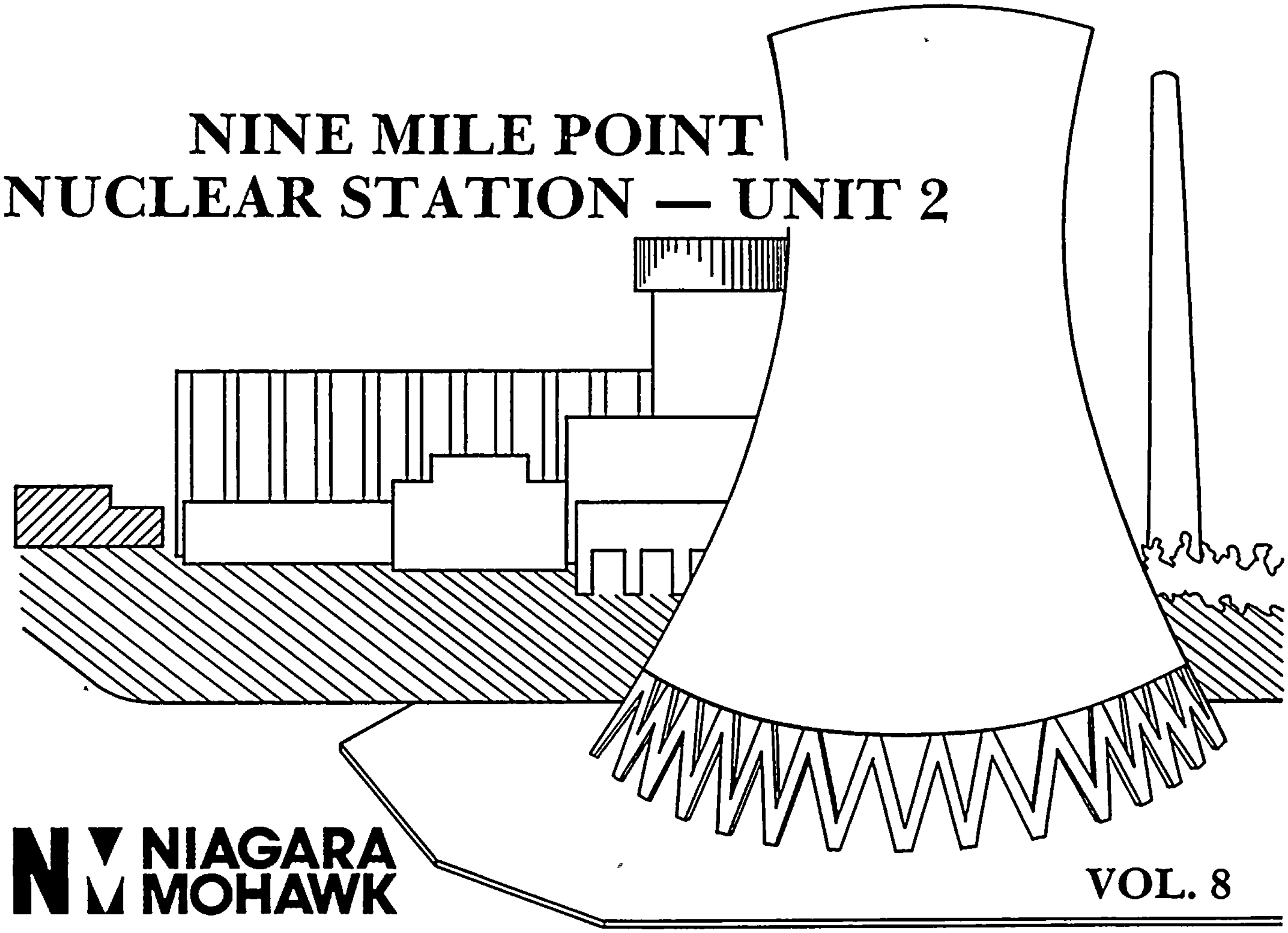

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sensors on a common process tap are run in separate conduits. The system sensors are electrically and physically separated. Only one trip channel actuator logic circuit from each trip system is run in the same conduit.

The RPS is designed to permit maintenance and diagnostic work while the reactor is operating without restricting plant operation or hindering the output of that safety function. Flexibility in design afforded the protection system allows operational system testing by the use of an independent trip channel for each trip logic input. When an essential monitored variable exceeds its scram trip point, it is sensed by at least two independent sensors in each trip system. Maintenance operation, calibration operation, or test, unless manually bypassed, can result in a single channel trip and one trip system trip (half scram). This leaves at least two trip channels per monitored variable of the other trip system capable of initiating a scram. Thus, the arrangement of two trip channels per trip system ensures that a scram occurs as a monitored variable exceeds its scram setting. Only one trip channel in each trip system must trip to initiate a scram.

The protection system meets the design requirements for functional and physical independence, as specified in Criterion 22.

For further discussion, see the following sections:

Principal Design Criteria	1.2.1
Functional Design of Reactivity Control System	4.6
Main Steam Line Isolation System	5.4.5
Residual Heat Removal System	5.4.7
Emergency Core Cooling System	6.3
Reactor Protection (Trip) System	7.2
Engineered Safety Feature Systems	7.3
All Other Instrumentation Systems Required for Safety	7.6
Accident Analysis	15

3.1.2.23 Protection System Failure Modes (Criterion 23)

<u>Criterion</u>

"The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced."

Design Conformance

The RPS is designed to fail into a safe state. Use of an independent trip channel for each trip logic allows the system to sustain any trip channel failure without preventing other sensors monitoring the same variable from initiating a scram. A single sensor or trip channel failure causes a channel trip. Only one trip channel in each trip system must be actuated to initiate a scram. Maintenance operation, calibration operation, or test, unless manually bypassed, can result in a single channel trip and one trip system trip (half scram). A failure of any one RPS input or subsystem component produces a trip in one of two channels and therefore in one trip system. This condition is insufficient to produce a reactor scram, but the system is ready to perform its protective function upon another channel trip in the other trip system.

Environmental conditions in which the instrumentation and equipment of the RPS must operate were considered in establishing component specifications. Instrumentation is designed to function in the worst expected ambient conditions in which the instruments must operate.

Failure modes of the protection system are such that it will fail into a safe state, as required by Criterion 23.

For further discussion, see the following sections:

Principal Design Criteria	1.2.1
Equipment Qualification	3.11
Emergency Core Cooling System	6.3
Reactor Protection (Trip) System	7.2
Engineered Safety Feature Systems	7.3

3.1.2.24 Separation of Protection and Control Systems (Criterion 24)

Criterion

"The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the

TABLE 3.2-1 (Cont'd.)

	Scope of Supply	Location	Electrical Classification	Seismic Category	Quality Group Classification	QA Requirement (31,32,33,34)	Tornado Protection	Notes
Auxiliary service building, superstructure	NA	M	NA	NA	NA	NA	NR	
Demineralized water storage and waste neutralizing tank building	NA	M	NA	NA	NA	NA	NR	
Shorefront revetment ditch	NA	O	NA	NA(23)	NA	I	NR	
PMP exterior flood protection berms	NA	O	NA	NA	NA	I	NR	
Roof and storm drainage systems	P	RB,S,T,W, C,N,P,M,O	NA	NA	NA	NA	NR	(34)
Spent fuel pool and liner	NA	RB	NA	I	NA	I	T	(29)
<u>Miscellaneous Radiation Protection Equipment and Programs</u>								
Portable radioactivity monitoring equipment	P	M	Non-1E	NA	NA	NA	NR	(34a)
Radioactivity sampling equipment	P	M	Non-1E	NA	NA	NA	NR	(34a)
Radioactivity contamination measurement and analysis equipment	P	M	Non-1E	NA	NA	NA	NR	(34a)
Personnel monitoring equipment	P	M	Non-1E	NA	NA	NA	NR	(34a)
Instrument storage, calibration, and maintenance program	P	M	NA	NA	NA	NA	NR	(34a)
Decontamination facilities	P	TB,W,M	NA	NA	NA	NA	NR	(34a)
Respiratory protection equipment	P	M	NA	NA	NA	NA	NR	(34a)
Contamination control equipment	P	M	Non-1E	NA	NA	NA	NR	(34a)
In-plant I_2 monitoring equipment (NUREG-0737, Item III.D.3.3)	P	M	Non-1E	NA	NA	NA	NR	(34a)
<u>Crack Arrest Verification System</u>								
Piping, Other	GE,P	M	NA	NA	D	NA	P	
Valves, Other	GE,P	M	NA	NA	D	NA	P	
Pressure Vessel	GE	M	NA	NA	D	NA	P	
Controls/Instruments	GE,P	M	Non-1E	NA	NA	NA	P	
<u>Oxygen Feedwater Injection System</u>								
Piping, Other	GE,P	M	NA	NA	D	NA	P,NR	
Valves, Other	GE,P	M	NA	NA	D	NA	P,NR	
Controls/Instruments	GE,P	M	Non-1E	NA	NA	NA	P	

TABLE 3.2-2

CODE GROUP DESIGNATIONS, INDUSTRY CODES, AND STANDARDS FOR MECHANICAL COMPONENTS[1]

ASME Section III Code Applicable Sections

Quality Group Classification	ASME Section III Code Class 8	Pressure Vessels and Heat Exchangers	Pumps, Valves and Piping	Metal Containment Components	Storage Tanks 0-15 (psig)	Storage Tanks Atmospheric
A[2]	1	NA or NCA & NB TEMA C	NA or NCA & NB[2,3]	--	--	--
B	2	NA or NCA & NC TEMA C	NA or NCA & NC[2,3]		NA or NCA & NC[4]	NA or NCA & NC[5]
	MC			NA or NCA & NE		
C	3	NA or NCA & ND TEMA R,C	NA or NCA & ND[2,3]	--	NA or NCA & ND[4]	NA or NCA & ND[5]
D		ASME VIII Div. 1 TEMA C,R	Piping & valves B31.1.0 pumps[6]	--	ASME VIII or equivalent[7]	ASME VIII, NBS-PS15-69 API-650 or equivalent[7]

[1] Components required to be stamped to ASME Boiler and Pressure Vessel Code are stamped with the applicable ASME Code symbol and third-party inspected by a qualified inspector.

[2] Components of the RCPB comply with the requirements of 10CFR50.55a codes and standards. All other components satisfy codes and addenda in effect at the time of the component order.

[3] For pumps classified in A, B, or C, the applicable subsection NB, NC, or ND, respectively, of ASME Section III is used as a guide in calculating the thickness of pressure-retaining portions of the pump and in sizing cover bolting.

[4] 100% volumetric examination of the sidewall and roof weld joints for plates over 3/16-in thick and 100% surface examination of weld joints for plates 3/16-in thick or less of the sidewall-to-bottom and sidewall roof joints. These examination requirements are performed in accordance with the rules of ASME Section III, Code Class 2 and 3.

[5] 100% volumetric examination of the sidewall weld joints for plates over 3/16-in thick and 100% surface examination of the sidewall-to-bottom joints. These examination requirements are performed in accordance with the rules of ASME Section III, Safety Class 2 and 3.

[6] For GE-supplied pumps classified D, the ASME Section VIII, Division I pump design for the intended service is utilized. For other pumps classified D, the manufacturers' standard pump design for the intended service is utilized.

[7] Storage tanks are designed to meet the intent of API, NBS-PS, and/or ASME Section VIII standards as applicable.

[8] In the case that material cannot be purchased to meet the specified ASME III Code, then material that meets subsequent ASME III Code Editions/Addenda up to and including 1980 Edition/Summer 1982 Addenda may be substituted after a review and reconciliation of related requirements of the ASME III Code are performed and documented.

TABLE 3.2-3

SUMMARY OF SAFETY CLASS DESIGN REQUIREMENTS

Design Requirements	Safety Class 1	Safety Class 2	Safety Class 3	Safety Class Other
Quality group classification[1,2]	A	B	C	D
Quality assurance requirement[3]	I	I	I	NA
Seismic category[4]	I	I	I	NA

(1) Equipment is constructed in accordance with the indicated code group listed in Table 3.2-1 and defined in Table 3.2-2.

(2) As indicated in Table 3.2-1, for QA Category I components, the quality group classification and/or seismic category is not applicable (NA) in certain cases, e.g., reactor internal structures and the shorefront revetment ditch.

(3) I = Equipment meets the QA requirements of 10CFR50 Appendix B.
NA = Conformance with 10CFR50 Appendix B is not required.

(4) I = Equipment is constructed in accordance with the requirements for Category I components (Section 3.7).
NA = The requirement to withstand the SSE is not applicable to this equipment.

TABLE 3.2-4

REACTOR COOLANT PRESSURE BOUNDARY
CLASS I EQUIPMENT CODE APPLICATION

Equipment	MPL/Mark	Code* Edition	Addenda
Reactor pressure vessel	B13-D003	1971	Winter 1972
Main steam system			
Piping		1974	No Addenda
Containment isolation valves	B22-F022A	1977	Summer 1977
	B22-F022B	1977	Summer 1977
	B22-F022C	1977	Summer 1977
	B22-F022D	1977	Summer 1977
	B22-F016	1974	Winter 1975
	B22-F019	1974	Winter 1975
	B22-F028A	1977	Summer 1977
	B22-F028B	1977	Summer 1977
	B22-F028C	1977	Summer 1977
	B22-F028D	1977	Summer 1977
	2MSS*MOV208	1974	Winter 1975
Manual block valve	2MSS*MOV207	1974	Winter 1975
Safety/relief valves	B22-F013	1974	Summer 1976

TABLE 3.2-4 (Cont'd.)

Equipment	MPL/Mark	Code[1] Edition	Addenda
Recirculation system			
Piping[3]	B35-G001	1977	Summer 1977
Pumps	B35-C001	1971	Summer 1973
Gate valves	B35-F023 B35-F067	1974	Winter 1974
Flow control valves	B35-F060A B35-F060B	1974	Summer 1976
High-pressure core spray system			
Isolation valve	E22-F004	1971	Winter 1973
Piping		1974	No Addenda
Standby liquid control system			
Explosive valve	C41-F004	1977	Summer 1977

(1) Code invoked in purchase order. The reference construction permit docket date was June 15, 1972.
(2) Deleted.
(3) See Section 5.4.1.3.

TABLE 3.3-1

DYNAMIC WIND PRESSURE FOR CATEGORY I STRUCTURES

Height Above Grade (ft)	Basic Wind Velocity Corresponding to Height (mph)	Dynamic Wind Pressures (psf)
0 to 50	90	26
50 to 150	115	42
150 to 400	145	66
400 to 600	175	95

TABLE 3.4-1

FLOOD PROTECTION FOR SAFETY-RELATED STRUCTURES AND SYSTEMS

Safety-Related Structures	Safety-Related Systems	Ground Water Level		Average Plant Grade	Elevation of Lowest Exterior Access Openings Below DBFL[5]	Elevation of Penetrations Through Exterior Walls Below DBFL[5]	Elevation of Electrical Duct Bank Penetrations Through Exterior Walls Below DBFL[5]
		Normal	DBFL[5]				
Reactor building including auxiliary bays	Primary containment, RHR, RCIC, LPCS, HPCS, ADS, service water reactor protection and standby liquid control	255	260.6	260	None	See Table 3.4-2	Top of duct el 253'
Control building	Control room with PGCC, emergency switchgear rooms, battery rooms	255	260.6	260	None	See Table 3.4-3	Top of duct el 234'-2"
Diesel generator building	Standby diesel generators and related support systems	255	260.6	260	None	See Table 3.4-6	-
Screenwell building	Service water pumps and related piping	255	260.6	260	None	See Table 3.4-4	Top of duct el 257' Two duct lines at same elevation
Main stack	Standby gas treatment system	255	260.6	260	None	See Table 3.4-6	-
Standby gas treatment building	Standby gas treatment system	255	260.6	260	None	None	-
Turbine building (main steam tunnel area)	Main steam and feedwater isolation valves and safety-related instruments	255	260.6	260	None	See Table 3.4-3	-
Piping and electrical tunnels	Service water system and electrical systems necessary for reactor control	255	260.6	260	None	See Table 3.4-5	-

TABLE 3.4-1 (Cont'd.)

NOTES:

1. All dimensions are lake survey datum (LSD) elevations in feet and inches.

2. The tunnels housing Category I systems and components are accessible only from the adjoining buildings.

3. Pipe penetrations through exterior walls of Category I structures have watertight seals designed to withstand the flood loads.

4. Where electrical ducts penetrate Category I structures, waterstops are provided to prevent any adverse effect from flooding.

5. The DBFL of 260.6 is based on the PMP as determined from NOAA Hydrometeorological Report No. 33 and is the original basis for the plant design. See Sections 2.4.10 and 3.4 for further discussion concerning DBFL.

TABLE 3.4-2

PENETRATIONS THROUGH EXTERIOR WALLS
OF REACTOR BUILDING BELOW DBFL*

Sleeve No.	Elevation	System
W3888	210'-8"	RHR
W31670	227'	CRD, Electrical
W3227R	235'-1"	HVR
W3226R	225'-11"	HVR
W3225R	235'-7"	HVR
W3455C	235'-3"	SWP
W4288C	224'-6 1/2" HWP	Spare
W4025C	236'-2"	Instrument
W3224R	230'-3"	HVR
W4241C	256'	CMS
W3596C	247'-3"	FPL
W3177C	251'-4"	SWP
W4374C	251'-4"	SWP
W3190C	247'	RCIC
W3820C	248'-3"	FPW
W3546C	256'-3 3/8"	FPW
W4364C	254' HWP	6" Drain
W3943C	254'	DFR
W3944C	254'	DFR
W3945C	254'	DFR
W3946C	254'	DFR
W3947C	254'	DFR
W3545C	246'-3"	FPW
W3102C	208'-8" HWP	RHR
W3890C	210'-8"	RHR
W3625C	203'-4"	Instrument
W3082C	208'	HPCS
W3083C	208'	HPCS
W3084C	210'-4"	RHR
W3045C	208'	SWP
W3046C	208'	SWP
W3042C	206'	SWP
W3821C	208'-11" HWP	DFR
W3041C	206'	SWP
W3079C	203'	SWP
W3108C	208'-9 9/16" HWP	RHR
W3550C	253'-11 1/2"	DFR
W3815C	254'-6"	FPW
W3544C	256'	FPW
W4242C	256'-6"	CMS
W4243C	256'-6"	CMS
W3598C	255'	FPL
W34820	256'-9"	SWP
W3582C	249'-11"	SWP

TABLE 3.4-2 (Cont'd.)

Sleeve No.	Elevation	System
W3583C	248'-5"	SWP
W3584C	252'-11"	SWP
W3585C	251'-5"	SWP
W3570S	248'-8 1/4"	MWS
W3185C	246'-6"	CNS
W4240C	256'	CMS
W3008R	194'-3"	RHR
W3065C	182'-6"	RBCLCW
W3066C	183'-6"	RBCLCW
W4121S	177'-1 7/16"	O_2 Mapp argon
W4122S	178'-5 1/16"	O_2 Mapp argon
W4123S	179'-7 5/16"	O_2 Mapp argon
W4120S	178'-5 1/16"	Electrical
W4118S	179'-7 5/16"	Compressed air
W4119S	182'-4"	Construction water
W4115S	179'-7 5/16"	Electrical
W4114S	178'-5 1/16"	Electrical
W4112S	179'-7 5/16"	Electrical
W4113S	177'-1 7/16"	Electrical
W4110S	182'-4"	Spare
W4111S	182'-4"	Spare
W3062C	192'-3"	RCIC
W3074S	193'-2"	SWP
W3075S	193'-2"	SWP
W3018C	189'-9"	RHR
W3019C	186'-9"	RHR
W3615C	193'-2"	Instrument
W4150S	181'-4 9/16"	Spare
W4152S	178'-7 1/16"	O_2 Mapp argon
W4153S	177'-2 1/16"	Compressed air
W4151S	179'-7 9/16"	Construction water
W4154S	178'-7 1/16"	O_2 Mapp argon
W4155S	179'-7 9/16"	O_2 Mapp argon
W4156S	177'-2 1/16"	Sump discharge
W4157S	179'-7 9/16"	Electrical
W4158S	179'-7 9/16"	Electrical
W4159S	177'-2 1/16"	Electrical
W3616C	190'	Instrument
W3622C	189'	Spare
W4160S	178'-7 1/16"	Electrical
W4161S	179'-7 9/16"	Electrical
W4162S	178'-7 1/16"	Electrical
W3023R	194'	AAS, IAS, SAS, RHR
W4163S	181'-4 9/16"	Spare
W3040C	182'-6"	SWP
W3068C	176'-6"	RBCLCW
W3067C	176'-3"	RBCLCW

TABLE 3.4-2 (Cont'd.)

Sleeve No.	Elevation	System
W3022R	194'	RHR
W3617C	190'	Instrument
W3787C	185'	DFR
W3037C	190'-3"	SWP
W3038C	185'-9"	SWP
W3568C	184'-10"	MWS
W3051C	192'-6"	RBCLCW
W3048O	192'-9"	SWP
W3606C	187'-9"	Spare
W3605C	189'	MWS
W3623C	188'-2"	Spare
W3052C	186'	RBCLCW
W3565S	183'	Instrument

KEY:
AAS = Breathing air
CMS = Containment atmosphere monitoring
CNS = Condensate makeup and drawoff
CRD = Control rod drive hydraulic
DFR = Reactor building floor drains
FPL = Fire protection - low-pressure CO_2
FPW = Fire protection - water
HPCS = High-pressure core spray
IAS = Instrument air
HVR = Reactor building ventilation
HWP = High work point
MWS = Makeup water
RBCLCW = Reactor building closed loop cooling water
RCIC = Reactor core isolation cooling
RHR = Residual heat removal
SAS = Service air
SWP = Service water

* The table is based on a DBFL of 260.6 ft, the original basis for plant design. See Sections 2.4.10 and 3.4 for further discussion concerning DBFL.

TABLE 3.4-3

PENETRATIONS THROUGH EXTERIOR WALLS OF CONTROL AND TURBINE BUILDINGS BELOW DBFL[1]

Sleeve No.	Elevation	System
Turbine Building[2]		
W5335C	253′-9"	DWS
W0504C	255′-11 1/16"	15" Roof drain
W0505C	252′-9"	FPW
W5045C	254′	4" Drain
W0508C	254′-6"	FPW
W0509C	256′	FPW
W0024C	257′	FWS
W5275C	257′	FWS
W0023C	257′	FWS
W5274C	257′	FWS
W0507C	254′	FPW
W0506C	254′	FPW
Control Building		
W6185R	230′-7"	Vent duct
W6186R	230′-7"	Vent duct

KEY: DWS = Domestic water
FPW = Fire protection - water
FWS = Feedwater

(1) The table is based on a DBFL of 260.6 ft, the original basis for plant design. See Sections 2.4.10 and 3.4 for further discussion concerning DBFL.

(2) The turbine building is Category I in the following areas only:

a. Electrical bay area between column lines AM and AK, up to el 261′.
b. Main steam tunnel area and area underneath main steam tunnel between column lines 10 and 12, when providing support to main steam tunnel.
c. Pipe tunnels between column lines 10 and 12 up to el 248′.

TABLE 3.4-4

PENETRATIONS THROUGH EXTERIOR WALLS OF RADWASTE BUILDING AND SCREENWELL BUILDING BELOW DBFL[1]

Sleeve No.	Elevation	System
Radwaste Building[2]		
W8001C	248'-9"	TBCLCW
W8002C	250'-9"	TBCLCW
Screenwell Building		
W2250C	253'-7 3/8"	DFM
W2247C	252'-1 1/4"	10" Roof drain

(1) The table is based on a DBFL of 260.6 ft, the original basis for plant design. See Sections 2.4.10 and 3.4 for further discussion concerning DBFL.

(2) The radwaste building is designed as a Category I structure. However, Category I classification is not used in construction of the radwaste building.

KEY: TBCLCW = Turbine building closed loop cooling water
DFM = Miscellaneous floor and equipment drains

TABLE 3.4-5

PENETRATIONS THROUGH EXTERIOR WALLS
OF PIPE TUNNELS BELOW DBFL*

Sleeve No.	Elevation	System
W1251S	256'-1 3/4"	LWS
W1252S	256'-1 3/4"	MWS
W1253S	255'-1 3/4"	LWS
W1254S	256'-1 3/4"	LWS
W1255S	255'-1 3/4"	LWS
W1256S	256'-2 1/4"	LWS
W1257S	255'-1 3/4"	LWS
W1258S	249'-5"	Drain
W1132R	251'-10 1/4"	Spare
W1134R	250'-6"	Spare
W1133C	256'-2 1/4"	OFG
W1215C	255'-10 7/8"	4" Drain
W1020C	257'-0 3/4"	4" Drain
W1285C	247'-4"	CCS
W1208C	254'	DFM
W1077C	254'	4" Sump discharge
W1143C	246'-11 3/8"	SWP
W1145C	245'-11 7/16"	SWP
W1147C	244'-11 7/16"	SWP
W1207C	254'	DFM
W1006C	252'-7 3/16"	AAS
W1141C	247'-10 3/4"	SWP
W1078C	253'-8 5/8"	4" Sump discharge
W1137C	250'-0 5/8"	SWP
W1139C	248'-11 5/16"	SWP
W1286C	251'-0 1/8"	LWS
W1140C	248'-11 9/16"	SWP
W1144C	246'-11 7/16"	SWP
W1148C	244'-11 1/4"	SWP
W1138C	249'-11 15/16"	SWP
W1142C	247'-11 1/16"	SWP
W1146C	245'-11 3/8"	SWP
W1081C	255'-3 1/2"	Spare
W1259C	251'	LWS
W1276C	249'	DFT
W1243C	256'-6"	Abandoned
W1244C	251'-8 1/4"	Abandoned
W1109S	252'-5"	8" Drain line
W1219S	251'-3"	CCS
W1220S	248'-9"	CCS
W1069C	251'-4"	SWP
W1283C	251'-4"	SWP
W1070C	247'	RCIC
W1057C	254'-6"	SWP

TABLE 3.4-5 (Cont'd.)

Sleeve No.	Elevation	System
W1058C	254'-6"	CWS
W1056C	247'-7"	SWP
W1097C	249'-5"	Spare
W1099C	249'-5"	FPW
W1100C	249'-5"	FPW
W1101C	249'-5"	FPW
W1102C	249'-5"	FPW
W1103C	249'-5"	FPW
W1104C	249'-5"	FPW
W1105C	249'-5"	FPW
W1106C	249'-5"	FPW
W1107C	249'-5"	FPW
W1108C	249'-5"	FPW
W1186C	245'-3 15/16"	MSS
W1187C	245'-3 15/16"	MSS
W1188C	245'-3 15/16"	MSS
W1238C	245'-3 15/16"	MSS
W1239C	245'-3 15/16"	MSS
W1240C	245'-3 15/16"	MSS
W1241C	245'-3 15/16"	MSS
W1242C	245'-6 1/16"	MSS
W1189C	245'-3 15/16"	MSS
W1190C	245'-6 1/16"	MSS
W1249C	247'	IAS
W1152C	252'-7"	SWP
W1153C	249'-10"	SWP
W1154C	247'-9 3/8"	SWP
W1155C	246'-0 3/8"	SWP
W1195C	252'-7"	WTS
W1196C	252'-7"	WTS
W1216C	252'-4"	FPW

KEY:
AAS = Breathing air
CWS = Circulating water
DFT = Turbine building floor drains
DM = Miscellaneous floor and equipment drains
FPW = Fire protection - water
IAS = Instrument air
LWS = Radioactive liquid waste
MSS = Main steam
MWS = Makeup water
OFG = Offgas
RCIC = Reactor core isolation cooling
SWP = Service water
CCS = Turbine building closed loop cooling water
WTS = Water treating

TABLE 3.4-5 (Cont'd.)

* The table is based on a DBFL of 260.6 ft, the original basis for plant design. See Sections 2.4.10 and 3.4 for further discussion concerning DBFL.

TABLE 3.4-6

PENETRATIONS THROUGH EXTERIOR WALLS OF
DIESEL GENERATOR BUILDING AND MAIN STACK BELOW DBFL*

Sleeve No.	Elevation	System
Diesel Generator Building		
W1294C	258'-3"	6" drainage
W1295C	256'-6"	12" drainage
W1328C	255'-10"	6" drainage
W1329C	255'-10"	6" drainage
Main Stack		
W2275C	256'-6"	DFM
W2276C	254'-0 1/8"	ARC

KEY: ARC = Condenser air removal
DFM = Miscellaneous floor and equipment drain

* The table is based on a DBFL of 260.6 ft, the original basis for plant design. See Sections 2.4.10 and 3.4 for further discussion concerning DBFL.

TABLE 3.4-7

PERFORMANCE OF WATER STOP MATERIAL IN EXPECTED ENVIRONMENT

Material	Expected Environment				Expected Performance of Material[2][3]			
	Temperature Range[1]	Chemicals	Radiation Level	Aging	Temperature Range	Chemicals	Radiation Level	Aging
Styrene-Butadiene synthetic rubber waterstops	-20°F to +325°F	Unit 2 site has average pH -8.0-8.4. No acidic environment expected within the walls below grade area.	Below 1.4×10^7 rads	40 yr at normal operating temperature (109°F)	-35°F to +176°F	Unaffected by acidic or alkaline soils or soil bacteria.	2×10^6 rads before threshold damage. 1×10^7 rads before 25% damage. 6.0×10^7 rads before 50% damage.	40 yr at 109°F

(1) Temperature range varies from -26°F minimum outside at Site, 109°F normal operating inside secondary containment, to 325°F maximum accident inside secondary containment. The worst-case design conditions during which the water stop must function do not exceed the expected performance temperature of the material.

(2) Safety-related water stops are insulated and sealed from ambient environmental conditions to establish 40-yr qualified life.

(3) Water stop systems are required to contain long-term flooding from cracks in low temperature (<175°F) systems which have a large inventory of water, e.g., systems connected to the suppression pool. Cracks in these systems could remain undetected for long periods of time assuming failure of nonredundant leak detection systems. Under these conditions, watertight cubicles, which employ several water stops, prevent the spread of flooding to redundant safe shutdown equipment. This is discussed in Appendix 3C.5.

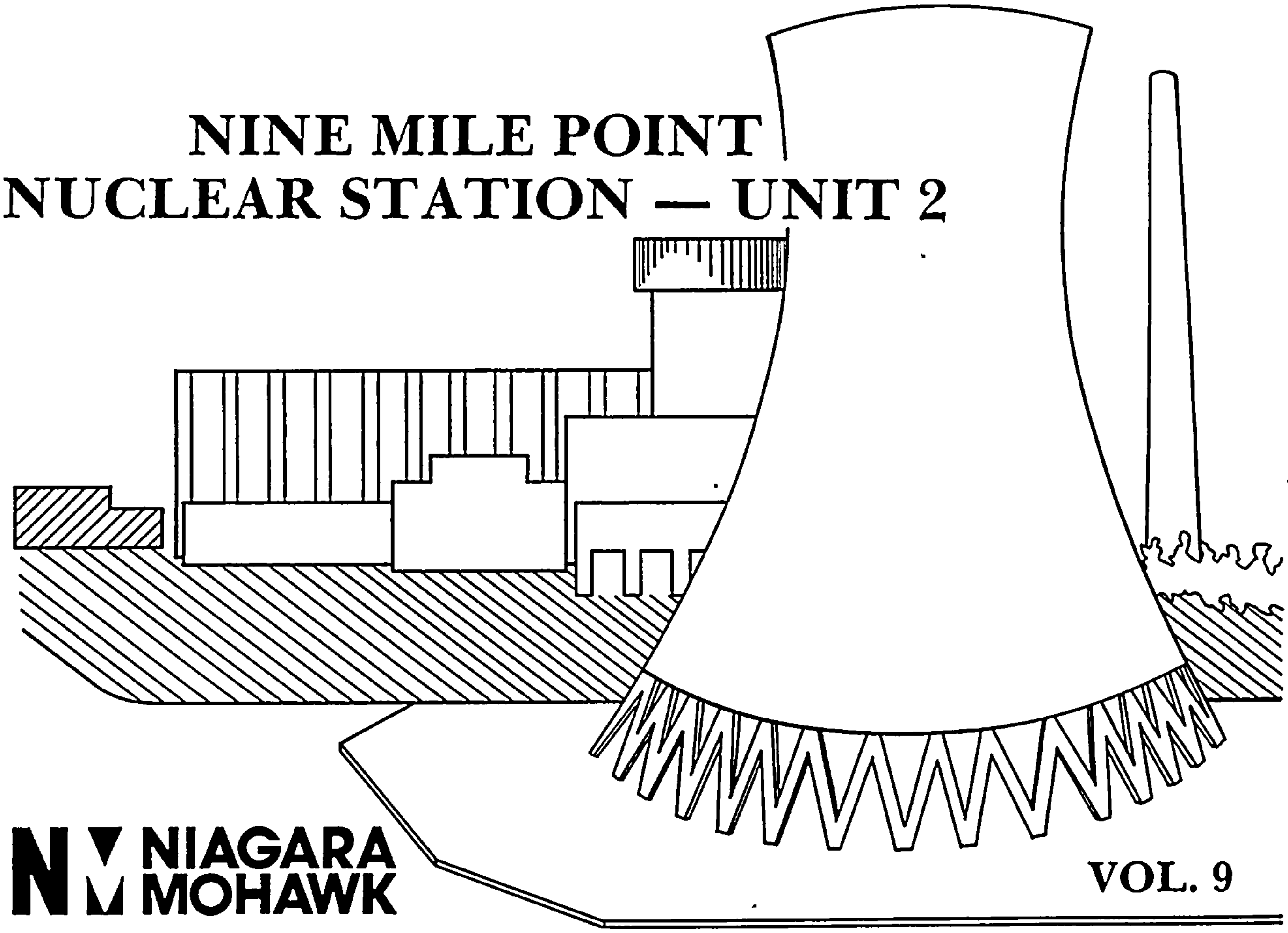
Water stops are not required to contain flooding from high temperature (>175°F) systems, e.g., RCIC, RWCU, or RHR (shutdown cooling) systems. Loss of water from a HELB or moderate-energy line crack in these systems is quickly detected and isolated either by the respective system instrumentation or by redundant leak detection system. Under these conditions, only a limited quantity of water is released, and flooding of redundant safe shutdown equipment is not a concern. Thus, the water stop function is not required.

Required service conditions have been evaluated in establishing the 40-yr qualified life of the water stops. In addition, the water stops are protected with layers of insulation and caulking material for added assurance that the system remains functional under all conditions.

Expected environmental radiation level of 1.4×10^7 rads is consistent with the manufacturer's design data.

UPDATED SAFETY ANALYSIS REPORT

NINE MILE POINT NUCLEAR STATION — UNIT 2



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Consequently, it is not considered a probable missile source.

2. Valves of ANSI 900-psig rating and above, constructed in accordance with ASME Section III, are pressure seal bonnet-type valves. For pressure seal bonnet valves, valve bonnets are prevented from becoming missiles by the retaining ring, which would have to fail by the yoke, capturing the bonnet or reducing bonnet energy. Because of the highly conservative design of the retaining ring of these valves, bonnet ejection is highly improbable. Hence, bonnets are not considered credible missiles.

3. Most valves of ANSI 600-psig rating and below are valves with bolted bonnets. Valve bonnets are prevented from becoming missiles by limiting the stresses in the bonnet-to-body bolting material by the rules set forth in ASME Section III, and by designing the flanges in accordance with the applicable code requirements. Even if bolt failure were to occur, the likelihood of all bolts experiencing a simultaneous complete severance failure is very remote. The widespread use of valves with bolted bonnets and the low historical incidence of complete severance failure of the valve bonnets confirm that bolted valve bonnets need not be considered as credible missiles.

4. Valve stems are not considered potential missiles if at least one feature in addition to the stem threads is included in their design to prevent ejection. Valves with backseats are prevented from becoming missiles by this feature. In addition, air- or motor-operated valve stems are effectively restrained by the valve operators. No credible valve stem missiles were identified at Unit 2.

5. Pressurized compressed gas cylinders are manufactured to Department of Transportation standards and stamped with a "DOT-3AA (Pressure)" designation. These cylinders have stringent manufacturing controls which meet the requirements of 10CFR50 when used in missile-sensitive areas. Controls such as receipt inspection and safe handling requirements in accordance with Compressed Gas Association, Inc., Pamphlets C-6 and P-1 are imposed by the requirements of the procurement specification. Under these conditions, the causes of failures are virtually eliminated, and such cylinders need not be considered as credible missiles.

6. Nuts, bolts, nut and bolt combinations, and nut and stud combinations have only a small amount of stored energy and thus are of minimal concern as potential missiles.

3.5.1.2 Internally-Generated Missiles (Inside Containment)

Location of Structures, Systems, or Components

All safety-related systems and components inside containment are listed in Table 3.2-1. They are protected against damage from internally-generated missiles as described in Sections 3.5.1.1.4 and 3.5.1.1.5.

3.5.1.3 Turbine Missiles

3.5.1.3.1 Turbine Placement and Orientation

Turbine placement and orientation for the three units affecting the turbine missile evaluation for Unit 2 are shown on Figure 3.5-1. They are the turbines of the Nine Mile Point Nuclear Station - Unit 1 (Unit 1), Unit 2, and James A. FitzPatrick Power Station. The spin axes of the Unit 1 and Unit 2 turbine generators are oriented in an east-west direction. The James A. FitzPatrick Power Station turbine generator has its spin axis oriented with a 15-deg clockwise rotation from a north-south direction. Figure 3.5-1 also indicates the ±25-deg missile ejection zone for low-trajectory turbine missiles resulting from low-pressure turbine discs(2).

A plan view of plant regions located at Unit 2 is shown on Figure 3.5-1 along with the turbine generators of Unit 1, Unit 2 and the James A. FitzPatrick plant. Note that applicable low-trajectory targets are those within the ±25-deg missile ejection zones. For high-trajectory missiles, target areas are all aboveground, Category I structures.

Tables 3.5-3 through 3.5-16 provide the probabilities of turbine missile strikes in various events. Due to very low probability of turbine missile strikes, as demonstrated by the above-referenced data, it is not necessary to design the safety-related structures for turbine missiles.

3.5.1.3.2 Missile Identification, Characteristics, and Target Description

Missile Identification and Characteristics

The turbine generators located at Unit 1, Unit 2 and James A. FitzPatrick Power Station are manufactured by General Electric Company (GE). The turbine type for Unit 1 and Unit 2 is a 38-in last-stage bucket, while the turbine type for the James A. FitzPatrick plant is a 43-in last-stage bucket.

At Unit 2, the original built-up type rotor design has been replaced with a monoblock type rotor design which eliminates the probability of missile generation from Unit 2.

For turbine missile evaluation at Unit 1 and the James A. FitzPatrick plant, a hypothetical missile is considered generated in the disc plane. As it penetrates the stationary turbine parts, the missile is deflected from the vertical plane. It has been determined that the deflection angle is a maximum of 5 deg on each side of the plane of the disc for inner-stage buckets. For last-stage buckets, the deflection angles may be up to 25 deg on each side of the plane of the disc. The missile characteristics used for this turbine missile strike probability evaluation are provided in Tables 3.5-17 and 3.5-18[3,4].

Target Description

Systems, equipment, and components required for safe shutdown and to maintain cold shutdown of the reactor, or to prevent the release of radiation to within allowable limits, are housed in the following structures:

1. Reactor building.
2. Control building.
3. Diesel generator building.
4. Screenwell building service water pump room.
5. Standby gas treatment building and railroad access lock.
6. Radwaste building.
7. Auxiliary service building and north and south auxiliary bays.
8. Intake structure, pipe, and shaft.
9. Main steam tunnel.

These targets are considered to be the safety-related regions for turbine missile evaluation.

3.5.1.3.3 Probability Analysis

For Unit 2, the probability of generation and ejection (P1) value from a Unit 2 postulated missile is insignificant (~0), due to the replacement of original built-up type rotor with monoblock type rotor. Therefore, the overall probability (P_4) for a Unit 2 postulated missile is insignificant (~0).

The evaluation of a postulated turbine missile from Unit 1 and the James A. FitzPatrick plant is based on the probability of missile generation and on the effects attributed to it. The overall probability of unacceptable damage to the critical plant regions, P_4, is the product of three contributing factors:

$$P_4 = P_1 \times P_2 \times P_3 \qquad (3.5\text{-}1)$$

Where:

P_1 = The probability of generation and ejection of a high-energy missile

P_2 = The probability that the missile will strike a safety-related region

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P_3 = The probability that the missile strike will damage its target in a manner leading to unacceptable consequences

P_4 = The overall probability

Probability of Generation and Ejection (P_1)

A turbine missile can be caused by brittle fracture of a rotating turbine part at or near turbine operating speed or by ductile fracture upon runaway after extensive, highly-improbable control system failures[2]. The operating experience of GE turbines clearly demonstrates that the structural integrity record of discs and rotor has been excellent. This excellent operational record can be attributed to:

1. Careful control of alloy chemistry and forging heat-treating cycles.

2. Improved steel mill practices in vacuum pouring and alloy addition resulting in more uniform and defect-free forgings.

3. Improved ultrasonic and magnetic particle testing techniques that ensure sound discs, which equal or exceed the specified design standard.

4. Redundancy in the control systems. As a minimum, the GE turbine is equipped with two separate and redundant overspeed protection systems.

These factors minimize the probability of missile generation.

For a postulated turbine failure, two speed failures were considered: the design overspeed failure (120 percent rated speed) and the destructive overspeed failure (180 percent rated speed). The GE estimate for the turbine failure rate is 8.67×10^{-9}/yr/turbine for design overspeed (120 percent rated speed) for 43-in last-stage buckets[3], whereas turbine failure rate is statistically insignificant for 38-in last-stage buckets[4] for design overspeed (120 percent rated speed). For destructive overspeed (180 percent rated speed), failure rate has been estimated to be 5.0×10^{-9}/yr/turbine for both 43-in last-stage buckets[3] and 38-in last-stage buckets[4]. Bush has obtained a failure rate of 3.3×10^{-5} to 3.1×10^{-4}/turbine yr for a turbine population corrected to be relevant to nuclear reactors[5].

RG 1.115, which is based on Bush's results, recommends a failure rate of 1.0×10^{-4} for design overspeed and for destructive overspeed turbine failures. However, Standard Review Plan (SRP) 3.5.1.3 allows an annual turbine failure rate of 10^{-4} (subdivided as 6×10^{-5}/turbine yr for design speed failures and 4×10^{-5}/turbine yr for destructive overspeed failures). Hence, in addition to using the turbine manufacturer's estimated failure

$$V_r = (V_i^2 - V_p^2)^{1/2} \tag{3.5-16}$$

Where:

V_r = Residual missile velocity after perforation

V_i = Incident missile velocity

V_p = Incident missile velocity required to just perforate the barrier, calculated by conservative use of penetration data

The probability of penetrating a concrete structure, P_3, can be calculated based on this residual velocity, taking into account the perforation of missile barriers.

Probability Calculation and Acceptance Criteria

In RG 1.115, the NRC considers the value of 10^{-7}/yr an acceptable risk rate for the loss of an essential system from a single event due to low-trajectory turbine missiles. Also, SRP 2.2.3 indicates that an "expected rate of occurrence of potential exposures in excess of the 10CFR100 guidelines of approximately 10^{-6}/yr is acceptable if, when combined with reasonable qualitative arguments, the realistic probability can be shown to be lower."

The probability calculation was based on estimates of the individual buildings considering both low and high trajectories and both design and destructive overspeed failures from the turbine generators of Unit 1, Unit 2 and the James A. FitzPatrick plant. These were then combined to determine the overall probability of damage (P_4) for Unit 2. The overall probability of damage (P_4) has been estimated using the turbine failure probabilities (P_1) suggested by both the turbine manufacturer (GE) and the NRC.

The turbine manufacturer's missile data (missile ejection velocity, missile weight, characteristic dimensions, etc.) have been used to calculate the strike probability (P_2) on the critical plant regions for both low- and high-trajectory turbine missiles. These data have also been used to evaluate the penetration probability (P_3) defined in Section 3.5.1.3.3.

Tables 3.5-3 through 3.5-8 show the calculated probabilities of damage on the Unit 2 safety-related regions due to low-trajectory turbine missiles for design overspeed and destructive overspeed turbine failures. Tables 3.5-3 through 3.5-5 show the calculated results using the turbine manufacturer's failure rates, while Tables 3.5-6 through 3.5-8 show the calculated results using the turbine failure rates suggested by the NRC. It should be noted that the information in Tables 3.5-3 through 3.5-8 was determined by considering the entire front surface areas and roof areas of all buildings containing the essential systems, as shown on

Figure 3.5-1. This is conservative since it is much greater than the actual projected areas of the essential systems.

Similarly, Tables 3.5-9 through 3.5-14 present the calculated probabilities of damage to the Unit 2 safety-related regions due to high-trajectory turbine missiles for design overspeed and destructive overspeed turbine failures, respectively. The entire front surface areas and roof areas of the buildings containing essential systems were used to determine the strike probabilities rather than the areas of the essential systems. These probabilities were obtained from the sum of the probabilities due to various missile ejection velocities and due to all fragments for turbine generators of Unit 1, Unit 2 and the James A. FitzPatrick plant.

The probability of damage for all Unit 2 buildings due to turbine missiles generated from Unit 1, Unit 2 and the James A. FitzPatrick plant for the two failure modes is presented in Tables 3.5-15 and 3.5-16. It can be observed that the overall probability of damage by turbine missiles is 0.962 x 10^{-7}/yr for Unit 2 if the probability of turbine failure rate of 1.0 x 10^{-4}/yr recommended by the NRC is used for Unit 1 and the James A. FitzPatrick plant. These results are within the acceptance value of 10^{-6}/yr as outlined in SRP 2.2.3, and the acceptance value of 10^{-7}/yr as specified in RG 1.115. These calculated figures are conservative. The overall probability for damage by turbine missiles for Unit 2, when estimated on a more realistic basis with manufacturer's probability, is much lower.

3.5.1.3.4 Turbine Overspeed Protection

The turbine is equipped with a redundant, testable overspeed trip system to minimize the possibility of a turbine overspeed event. The system and its test program are described in Section 10.2.2.2.

3.5.1.3.5 Turbine Valve Testing

Turbine valve testing and test frequency are described in Section 10.2.

3.5.1.3.6 Turbine Characteristics

Turbine characteristics, design, and operation are described in Section 10.2.

TABLE 3.5-3

DAMAGE PROBABILITY DUE TO LOW-TRAJECTORY TURBINE MISSILES FROM UNIT 2 STRIKING PLANT REGIONS AT UNIT 2

Manufacturer's Probability

Safety-Related Regions	Design Overspeed Failure			Destructive Overspeed Failure		
	p_1	$p_2 x p_3$	$p_1 x p_2 x p_3$	p_1	$p_2 x p_3$	$p_1 x p_2 x p_3$
Reactor building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Control building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Diesel generator building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Screenwell building service water pump room	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Standby gas treatment and RR access lock	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Radwaste area	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Auxiliary service building and north and south auxiliary bays	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Intake and discharge shaft area	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Main steam tunnel	Statistically Insignificant (Reference 4)			Statistically Insignificant		

TABLE 3.5-6

DAMAGE PROBABILITY DUE TO LOW-TRAJECTORY TURBINE MISSILES FROM UNIT 2
STRIKING PLANT REGIONS AT UNIT 2

NRC Probability

Safety-Related Regions	Design Overspeed Failure			Destructive Overspeed Failure		
	p_1	p_2xp_3	$p_1xp_2xp_3$	p_1	p_2xp_3	$p_1xp_2xp_3$
Reactor building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Control building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Diesel generator building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Screenwell building - service	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Standby gas treatment and RR access lock	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Radwaste building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Auxiliary service building and north and south auxiliary bays	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Intake and discharge shaft area	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Main steam tunnel	Statistically Insignificant (Reference 4)			Statistically Insignificant		

TABLE 3.5-9

DAMAGE PROBABILITY DUE TO HIGH-TRAJECTORY TURBINE MISSILES FROM UNIT 2
STRIKING PLANT REGIONS AT UNIT 2

Manufacturer's Probability

Safety-Related Regions	Design Overspeed Failure			Destructive Overspeed Failure		
	p_1	$p_2 x p_3$	$p_1 x p_2 x p_3$	p_1	$p_2 x p_3$	$p_1 x p_2 x p_3$
Reactor building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Control building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Diesel generator building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Screenwell building - service water pump room	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Standby gas treatment and RR access lock	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Radwaste area	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Auxiliary service building and north and south auxiliary bays	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Intake and discharge shaft area	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Main steam tunnel	Statistically Insignificant (Reference 4)			Statistically Insignificant		

TABLE 3.5-12

DAMAGE PROBABILITY DUE TO HIGH-TRAJECTORY TURBINE MISSILES FROM UNIT 2
STRIKING PLANT REGIONS AT UNIT 2

NRC Probability

Safety-Related Regions	Design Overspeed Failure			Destructive Overspeed Failure		
	p_1	p_2xp_3	$p_1xp_2xp_3$	p_1	p_2xp_3	$p_1xp_2xp_3$
Reactor building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Control building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Diesel generator building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Screenwell building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Standby gas treatment and RR access lock	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Radwaste building	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Auxiliary service building and north and south auxiliary bays	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Intake and discharge shaft area	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Main steam tunnel	Statistically Insignificant (Reference 4)			Statistically Insignificant		

TABLE 3.5-15

SUM OF DAMAGE PROBABILITY DUE TO LOW- AND HIGH-TRAJECTORY TURBINE MISSILES GENERATED FROM TURBINES AT UNITS 1 AND 2 AND JAMES A. FITZPATRICK TO PLANT REGIONS OF UNIT 2

Manufacturer's Probability

Trajectory and Turbine	Design Overspeed Failure			Destructive Overspeed Failure		
	p_1	$p_2 x p_3$	$p_1 x p_2 x p_3$	p_1	$p_2 x p_3$	$p_1 x p_2 x p_3$
Low trajectory from Unit 2	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Low trajectory from Unit 1	Statistically Insignificant (Reference 4)			1.37×10^{-8}	0	0
Low trajectory from FitzPatrick	1.37×10^{-8}	5.373×10^{-4}	7.361×10^{-12}	1.37×10^{-8}	4.930×10^{-5}	6.754×10^{-13}
High trajectory from Unit 2	Statistically Insignificant (Reference 4)			Statistically Insignificant		
High trajectory from Unit 1	Statistically Insignificant (Reference 4)			1.37×10^{-8}	3.862×10^{-5}	5.291×10^{-13}
High trajectory from FitzPatrick	1.37×10^{-8}	2.299×10^{-4}	3.150×10^{-12}	1.37×10^{-8}	1.064×10^{-4}	1.459×10^{-12}
Total	1.37×10^{-8}	7.672×10^{-4}	1.051×10^{-11}	1.37×10^{-8}	1.943×10^{-4}	2.66×10^{-12}

The total p_2p_3 for design and destructive overspeed failure is 0.962×10^{-3}.
The total $p_1p_2p_3$ for design and destructive overspeed failure is 1.078×10^{-11}.

NOTE: Manufacturer's probability for Unit 1 and Fitzpatrick: $p_1 = 1.37 \times 10^{-8}$

TABLE 3.5-16

SUM OF DAMAGE PROBABILITY DUE TO LOW- AND HIGH-TRAJECTORY TURBINE MISSILES GENERATED FROM TURBINES AT UNITS 1 AND 2 AND JAMES A. FITZPATRICK TO PLANT REGIONS OF UNIT 2

NRC Probability

Trajectory and Turbine	Design Overspeed Failure			Destructive Overspeed Failure		
	p_1	$p_2 x p_3$	$p_1 x p_2 x p_3$	p_1	$p_2 x p_3$	$p_1 x p_2 x p_3$
Low trajectory from Unit 2	Statistically Insignificant (Reference 4)			Statistically Insignificant		
Low trajectory from Unit 1	Statistically Insignificant (Reference 4)			1×10^{-4}	0	0
Low trajectory from FitzPatrick	1×10^{-4}	5.373×10^{-4}	5.373×10^{-8}	1×10^{-4}	4.930×10^{-5}	4.930×10^{-9}
High trajectory from Unit 2	Statistically Insignificant (Reference 4)			Statistically Insignificant		
High trajectory from Unit 1	Statistically Insignificant (Reference 4)			1×10^{-4}	3.862×10^{-5}	3.862×10^{-9}
High trajectory from FitzPatrick	1×10^{-4}	2.299×10^{-4}	2.299×10^{-8}	1×10^{-4}	1.064×10^{-4}	1.064×10^{-8}
Total	1×10^{-4}	7.672×10^{-4}	7.672×10^{-8}	1×10^{-4}	1.943×10^{-4}	1.943×10^{-8}

The total p_2p_3 for design and destructive overspeed failure is 0.962×10^{-3}.
The total $p_1p_2p_3$ for design and destructive overspeed failure is 0.962×10^{-7}.

NOTE: NRC probability (required for Unit 1 and FitzPatrick only): $p_1 = 1 \times 10^{-4}$.

TABLE 3.5-17

TURBINE MISSILE INFORMATION

43-Inch Last Stage Bucket Unit

Stage Group	I[1]				II[1]				III[2]			
Stage numbers in group:	1 - 3				4 - 6				7 (Last)			
Number of representative stage	2				5				7			
Missile Dimensions												
Fragment group	a	b	c	d	a	b	c	d	a	b	c	d
Number of fragments in group	2	1	3	10	2	1	3	10	2	1	3	10
Sector angle, deg	120	60			120	60			120	60		
Fragment weight, lb	2,000	1,000	300	100	4,000	2,000	600	150	8,200	4,100	1,400	200
Radius, in*												
R_1 Bore	20	20	NA	NA	18	18	NA	NA	17	17	NA	NA
R_2 Hub	27	27			27	27			28	28		
R_3 Vane root	48	48			47	47			45	45		
Thickness, in*												
T_1 Hub	9	9			12	12			27	27		
T_2 Web	3	3			5	5			12	12		
Approximate rectangular dimensions, in*			19x19x3	11x11x3			20x20x5	10x10x5			20x20x14	8x8x12

* See Figure 3.5-6.

TABLE 3.5-17 (Cont'd.)

HYPOTHETICAL TURBINE MISSILE INFORMATION

43-In Last Stage Bucket, 1,800 rpm Low-Pressure Turbine

Low Speed Burst
Postulated speed: 2,160 rpm (120%)
Lifetime probability: 2.6×10^{-7}

	Stage Group I	Stage Group II	Stage Group III[3]				
Conditional probability of occurrence in stage group	Not statistically significant	Not statistically significant	1				
Probability of occurrence in stage group			2.6×10^{-7}				
Fragment group	Not statistically significant	Not statistically significant	a			b	
			Energy	Velocity		Energy	Velocity
Minimum			10	280	0	0	
Maximum			22	420	18	530	
Midpoint			16	350	9	380	
Fragment group	Not statistically significant	Not statistically significant	c			d	
			Energy	Velocity		Energy	Velocity
Minimum			0	0	0	0	
Maximum			8	610	2	800	
Midpoint			4	430	1	560	

TABLE 3.5-17 (Cont'd.)

43-Inch Last Stage Bucket

High Speed Burst
Postulated speed: 3,240 rpm (180%)
Lifetime probability: 1.5 x 10^{-7}

	Stage Group I[3]				Stage Group II[3]				Stage Group III[4]			
Conditional probability of occurrence in stage group	3/7				3/7				1/7			
Probability of occurrence in stage group	6.4 x 10^{-8}				6.4 x 10^{-8}				2.1 x 10^{-8}			
Fragment group[5]	a		b		a		b		a		b	
	Energy	Velocity	Energy	Velocity	Energy	Velocity	Energy	Velocity	Energy	Velocity	Energy	Velocity
Minimum	0	0	0	0	0	0	0	0	26	450	0	0
Maximum	8	510	8	720	17	520	16	720	53	650	38	770
Midpoint	4	360	4	510	8.5	370	8	510	39.5	560	19	550
Fragment group[5]	c		d		c		d		c		d	
	Energy	Velocity	Energy	Velocity	Energy	Velocity	Energy	Velocity	Energy	Velocity	Energy	Velocity
Minimum	0	0	0	0	0	0	0	0	0	0	0	0
Maximum	5	1,040	2	1,130	8	930	2	930	16	860	3	980
Midpoint	2.5	730	1	800	4	660	1	660	8	610	1.5	690

(1) For interior disc, δ_1 and δ_2 are 5 deg, respectively.
(2) For end disc, δ_1 and δ_2 are 25 deg and 0 deg or 0 deg and 25 deg, respectively.
(3) The deflection angles δ_1 and δ_2 are 5 deg for inner stage buckets.
(4) For last stage buckets, the deflection angles are 25 deg and 0 deg or 0 deg and 25 deg, respectively.
(5) Missiles in four size classes, a, b, c, and d, are postulated to occur per burst.

NOTES:

a. Energy of ejected missiles is given in million ft lb; velocity in fps.

b. Energies are postulated to be uniformly distributed over stated ranges.

SOURCE: General Electric Memo Report. Hypothetical Turbine Missile Data, 43-inch Last Stage Bucket Units. Stone & Webster Engineering Corporation, Document No. CD7912100015, March 15, 1973.

TABLE 3.5-18

TURBINE MISSILE INFORMATION

38-Inch Last Stage Bucket Units

Stage Group	I[1]				II[1]				III[2]			
Stage numbers in group:	1 - 3				4 - 6				7 (Last)			
Number of representative stage	2				5				7			
Missile Dimensions												
Fragment group	a	b	c	d	a	b	c	d	a	b	c	d
Number of fragments in group	2	1	3	10	2	1	3	10	2	1	3	10
Sector angle, deg	120	60			120	60			120	60		
Fragment weight, lb	2,000	1,000	300	100	3,000	1,500	500	150	6,500	3,200	1,000	200
Radius, in*												
R_1 Bore	18	18			17	17			16	16		
R_2 Hub	24	24			25	25			25	25		
R_3 Vane root	45	45			45	45			45	45		
Thickness, in*												
T_1 Hub	10	10			12	12			21	21		
T_2 Web	3	3			5	5			10	10		
Approximate rectangular dimensions, in*	NA	NA	19x19x3	11x11x3	NA	NA	17x19x5	10x10x5	NA	NA	19x19x10	8x8x10

* See Figure 3.5-6.

TABLE 3.5-18 (Cont'd.)

HYPOTHETICAL TURBINE MISSILE INFORMATION

38-Inch Last Stage Buckets, 1,800 rpm Low Pressure Turbine

Low Speed Burst
Postulated speed: 2,160 rpm (120%)
Lifetime probability: not statistically significant

	Stage Group I	Stage Group II	Stage Group III[1]
Conditional probability of occurrence in stage group	Not statistically significant	Not statistically significant	Not statistically significant
Probability of occurrence in stage group			
Fragment group Minimum Maximum Midpoint	Not statistically significant	Not statistically significant	Not statistically significant
Fragment group Minimum Maximum Midpoint	Not statistically significant	Not statistically significant	Not statistically significant

TABLE 3.5-18 (Cont'd.)

38-Inch Last Stage Buckets, 1,800 rpm Low Pressure Turbine

<u>High Speed Burst</u>
Postulated speed: 3,240 rpm (180%)
Lifetime probability: 1.5×10^{-7}

	Stage Group I[3]				Stage Group II[3]				Stage Group III[4]			
Conditional probability of occurrence in stage group	3/7				3/7				1/7			
Probability of occurrence in stage group	6.4×10^{-8}				6.4×10^{-8}				2.1×10^{-8}			
Fragment group[5]	a		b		a		b		a		b	
	Energy	Velocity	Energy	Velocity	Energy	Velocity	Energy	Velocity	Energy	Velocity	Energy	Velocity
Minimum	0	0	0	0	0	0		0	16	400	0	0
Maximum	7	470	6	620	14	550	13	750	38	610	30	780
Midpoint	3.5	340	3	440	7	390	6.5	530	27	520	15	550
Fragment group[5]	c		d		c		d		c		d	
	Energy	Velocity	Energy	Velocity	Energy	Velocity	Energy	Velocity	Energy	Velocity	Energy	Velocity
Minimum	0	0	0	0	0	0	0	0	0	0	0	0
Maximum	4	930	2	1,130	6	880	2	930	13	910	3	980
Midpoint	2	660	1	800	3	620	1	660	6.5	650	1.5	690

(1) For interior disc, δ_1 and δ_2 are 5 deg, respectively.
(2) For end disc, δ_1, and δ_2 are 25 deg and 0 deg or 0 deg and 25 deg, respectively.
(3) The deflection angles δ_1 and δ_2 are 5 deg for inner stage buckets.
(4) For last stage buckets, the deflection angles are 25 deg and 0 deg or 0 deg and 25 deg, respectively.
(5) Missiles in four size classes are postulated to occur per burst.

<u>NOTES</u>:

a. Energy of ejected missiles is given in million ft lb; velocity in fps.

b. Energies are postulated to be uniformly distributed over stated ranges.

SOURCE: General Electric Memo Report. Hypothetical Turbine Missile Data, 38-inch Last Stage Bucket Units. Stone & Webster Engineering Corporation, March 16, 1973.

TABLE 3.6A-72

ESSENTIAL SYSTEMS/COMPONENTS/EQUIPMENT
EVALUATED FOR PIPE FAILURES

PART 1

Inside Containment
(QA Category I Portions Only Unless Otherwise Noted)

1. Reactor coolant pressure boundary (up to and including the containment isolation valves)
2. Containment isolation system and containment boundary (including liner plate)
3. Reactor protection system (instruments associated with SCRAM signals)
4. Emergency core cooling systems (for LOCA only)
    a. HPCS (CSH)
    b. LPCS (CSL)
    c. LPCI (RHS)
    d. ADS, including SVV valves and discharge lines (SVV)
5. Core cooling systems (other than ECCS)
    a. RCIC (ICS)
    b. RHR shutdown cooling mode (RHS)
6. Control rod drive system
7. Containment heat removal systems
    a. RHR suppression pool cooling mode (RHS)
    b. RHR containment spray mode (RHS)
8. Neutron monitoring system (NMS)
9. Reactor recirculation system including hydraulic lines to hydraulic actuators (includes Class 4 portions) (RCS)
10. Hydrogen recombiner system (HCS)
11. Containment atmosphere monitoring system (CMS)

TABLE 3.6A-72 (Cont'd.)

12. Reactor vessel instrumentation (ISC)

13. The following equipment/systems, or portions thereof, are required to ensure the proper operation of those essential items listed in Items 1 through 12:

    a. Class 1E electrical system

    b. Reactor plant component cooling water (CCP) (excluding Class 4 portions)

    c. Instrument air to ADS (IAS)

    d. Safety-related instrumentation and instrumentation piping

TABLE 3.6A-72 (Cont'd.)

PART 2

Outside Containment
(QA Category I Portions Only Unless Otherwise Noted)

1. Containment isolation system and containment boundary up to and including the outermost containment isolation valves

2. Reactor protection system (instruments associated with SCRAM signals)

3. Core and containment cooling systems

    a. HPCS (CSH)

    b. RCIC (ICS)

    c. LPCS (CSL)

    d. LPCI (RHS)

    e. RHR shutdown cooling mode (RHS)

    f. RHR containment spray mode (RHS)

    g. RHR suppression pool cooling mode (RHS)

4. Containment atmosphere monitoring system (CMS)

5. Control rod drive system (including Class 4 portions) (RDS)

6. Spent fuel pool cooling system (SFC)

7. Reactor water cleanup (ASME Classes 1, 2, and 3 only) (WCS)

8. Reactor vessel instrumentation system (ISC)

9. The following equipment/systems, or portions thereof, are required to ensure the proper operation of those essential items listed in Items 1 through 8:

    a. Class 1E electrical systems, ac and dc (including diesel generator system, emergency buses, motor control centers, switchgear, batteries, auxiliary shutdown control panel, and distribution systems)

TABLE 3.6A-72 (Cont'd.)

b. Service water system (SWP) including reactor plant component cooling water (CCP) (excluding Class 4 portions)

c. Safety-related environmental systems (HVK, HVN, HVR, HVP, HVY, HVC)

d. Safety-related instrumentation and instrument piping/tubing, including leak detection instrumentation

e. Instrument air (IAS) and neutron monitoring (GSN) supply to ADS

10. Reactor recirculation system hydraulic lines to hydraulic actuators (includes Class 4 portion) (RCS)

TABLE 3.6A-73

HIGH-ENERGY PIPING

PART I

Inside Containment

Piping System
Main steam system (MSS)
Main steam drains
Reactor core isolation cooling (RCIC) system - steam
Reactor core isolation cooling (RCIC) system - head spray
Feedwater system (FWS)
Recirculation system (RCS)
High-pressure core spray system (HPCS) (reactor pressure vessel (RPV) to first check valve)
Low-pressure core spray system (LPCS) (reactor pressure vessel (RPV) to first check valve)
Reactor water cleanup (RWCU) system
Main steam vent lines
Residual heat removal (RHR) system (shutdown suction)
Residual heat removal (RHR) system - low-pressure core injection (LPCI)
Control rod drive (CRD) system
Standby liquid control system (SLCS)
Main steam safety relief valve (SRV) piping (between the main steam line and the safety relief valve)

TABLE 3.6A-73 (Cont'd.)

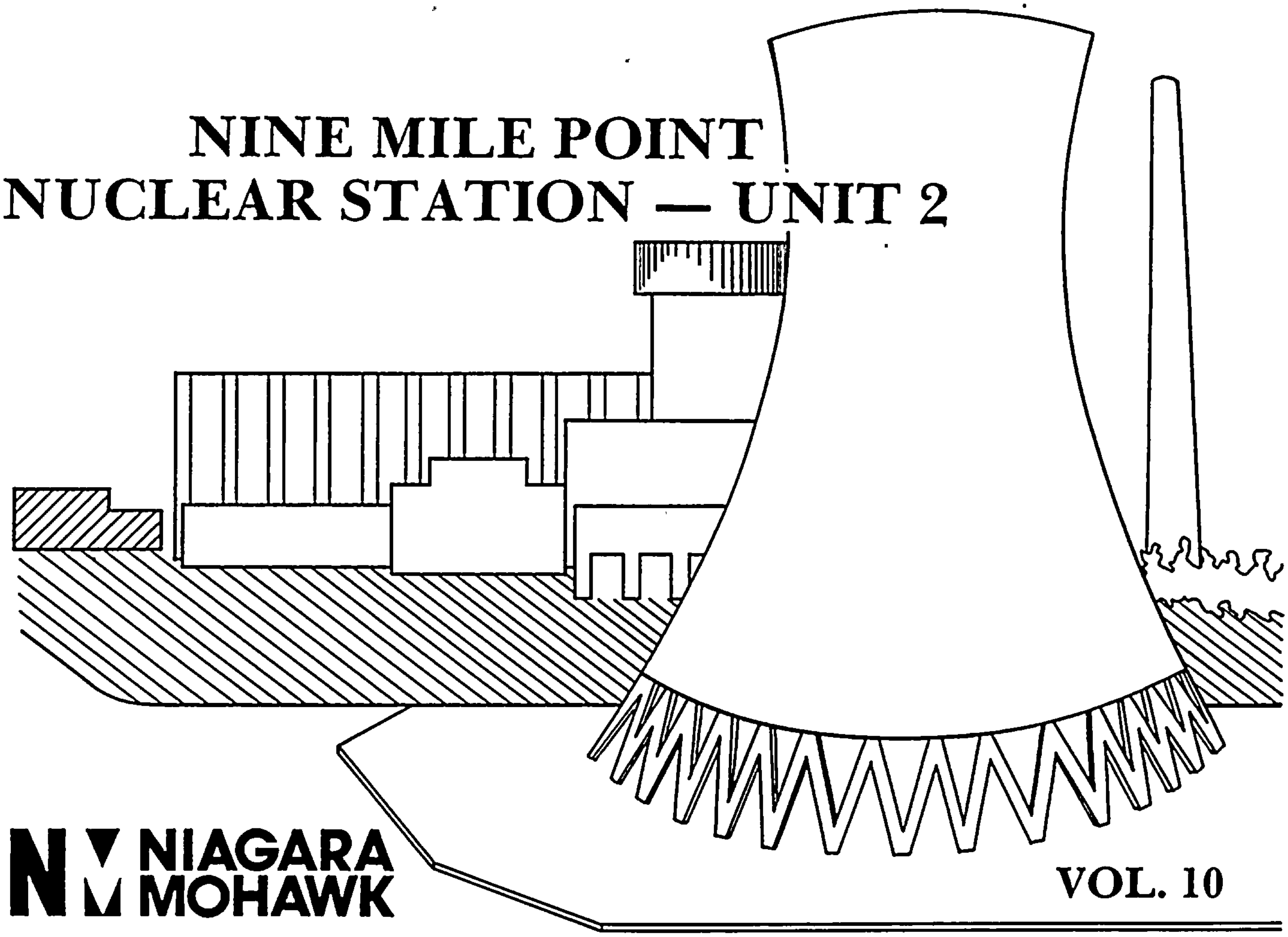
PART II

Outside Containment

Piping System
Main steam system (MSS)
Main steam drains
Feedwater system (FWS)
Reactor core isolation cooling (RCIC) system (steam to RCIC turbine)
Control rod drive (CRD) system
Reactor water cleanup (RWCU) system (including residual heat removal (RHR) feed to feedwater)

UPDATED SAFETY ANALYSIS REPORT

NINE MILE POINT NUCLEAR STATION — UNIT 2



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3.6B PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH POSTULATED RUPTURE OF PIPING (GE SCOPE OF SUPPLY)

See Section 3.6A for an explanation of GE/SWEC scope of supply. The following high-energy systems are in the scope of Section 3.6B.

3.6B.1 Postulated Piping Failures in Fluid Systems Outside Containment

See Section 3.6A.1.

3.6B.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

Information concerning break and crack location criteria and methods of analysis is presented in this section. The location criteria and methods of analysis are needed to evaluate the dynamic effects associated with postulated breaks and cracks in high- and moderate-energy fluid system piping inside and outside the primary containment. This information confirms that the requirements for the protection of structures, systems, and components relied upon for safe reactor shutdown or to mitigate the consequences of a postulated pipe break have been met.

3.6B.2.1 Criteria Used to Define Break and Crack Location and Configuration

The following section establishes the criteria for the location and configuration of postulated breaks and cracks.

3.6B.2.1.1 Criteria for Recirculation Piping System Inside Containment

3.6B.2.1.1.1 Definition of High-Energy Fluid System

High-energy fluid systems are defined as those systems, or portions of systems, that during normal plant conditions* are either in operation or are maintained pressurized under conditions where either or both of the following are met:

Maximum operating temperature exceeds 200°F.

Maximum operating pressure exceeds 275 psig.

3.6B.2.1.1.2 Definition of Moderate-Energy Fluid System

Moderate-energy fluid systems are defined as those systems, or portions of systems, that during normal plant conditions are

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* Normal plant conditions are defined as the plant operating conditions during reactor startup, power, operation hot standby, or reactor cold shutdown.

either in operation or are maintained pressurized under conditions where both of the following are met:

Maximum operating temperature is 200°F or less, and

Maximum operating pressure is 275 psig or less.

Piping systems are classified as moderate-energy systems when they operate as high-energy piping for only short operational periods in performing their system function, but for the major operational period qualify as moderate-energy fluid systems. An operational period is considered "short" if the total fraction of time that the system operates within the P-T conditions specified for high-energy fluid systems is less than 2 percent of total operating time the system is designed for.

3.6B.2.1.1.3 Postulated Pipe Breaks and Cracks

A postulated pipe break is defined as a sudden, gross failure of the pressure boundary either in the form of a complete circumferential severance (guillotine break) or as development of a sudden longitudinal crack (longitudinal split), and is postulated for high-energy fluid systems only. For moderate-energy fluid systems, pipe breaks are confined to postulation of leakage cracks in piping and branch runs. These cracks affect the surrounding environmental conditions only and do not result in whipping of the cracked pipe.

The following high-energy piping systems (or portions of systems) are considered to have potential for initiation of a postulated pipe break during normal plant conditions, and are analyzed for potential damage due to dynamic effects:

1. All piping that is part of the RCPB and subject to reactor pressure continuously during plant operation.

2. All piping that is beyond the second isolation valve but is subject to reactor pressure continuously during plant operation.

3. In addition to piping under 1 and 2, all other piping systems or portions of piping systems considered high-energy systems.

Portions of piping systems that are isolated from the source of the high-energy fluid during normal plant conditions are exempted from consideration of postulated pipe breaks. This would include portions of piping systems beyond a normally closed valve. Pump and valve bodies are also exempted from consideration of pipe break because of their greater wall thickness.

A high-energy piping system break is not postulated simultaneously with a moderate-energy piping system crack, nor is any pipe break or crack outside containment postulated concurrently with a postulated pipe break inside containment. Only the high-energy piping system break is postulated except where a postulated leakage crack in the moderate-energy fluid system piping results in more severe environmental conditions than the break in the approximate high-energy fluid piping system.

3.6B.2.1.1.4 Exemptions from Pipe Whip Protection Requirements

Protection from pipe whip need not be provided if any one of the following conditions exists:

1. Piping is classified as moderate-energy piping.

2. Following a single postulated pipe break, piping for which the unrestrained movement of either end of the ruptured pipe in any feasible direction about a plastic hinge, formed within the piping, cannot impact any structure, system, or component important to safety.

3. Piping for which the internal energy level associated with whipping is insufficient to impair the safety function of any structure, system, or component to an unacceptable level. Any line restrictions (e.g., flow limiters) between the pressure source and break location, and the effects of either a single-ended or double-ended flow condition are accounted for, in the determination of the internal fluid energy level associated with the postulated pipe break reaction. The energy level in a whipping pipe is considered insufficient to rupture an impacted pipe of equal or greater nominal pipe size and equal or heavier wall thickness.

3.6B.2.1.1.5 Location for Postulated Pipe Breaks (ASME Safety Class 1 Piping)

Postulated pipe break locations are selected in accordance with the intent of RG 1.46, BTP APCSB 3-1, Appendix B, and as expanded in BTP MEB 3-1. For ASME Section III, Safety Class 1 piping systems classified as high energy, the postulated break locations are:

1. The terminal ends of the pressurized portions of the run. (Terminal ends are extremities of piping runs that connect to structures, components, or pipe anchors that act as rigid constraints to piping motion and thermal expansion. A branch connection to a main piping run is a terminal end for a branch run except when the branch run is modeled as a part of the piping system in the stress analysis and is shown to have a significant effect on the main run behavior.)

2. At intermediate locations between the terminal ends where the maximum stress range between any two load sets (including zero load set) according to ASME Section III Subarticle NB-3600 for upset plant conditions and an independent OBE event transient, exceeds the following:

   a. If the stress range calculated using Equation 10 of the Code exceeds 2.4 S_m but is not greater than 3 S_m, no breaks are postulated unless the CUF exceeds 0.1.

   b. The stress ranges, as calculated by Equations 12 or 13 of the Code, exceed 2.4 S_m or if the CUF exceeds 0.1 when Equation 10 exceeds 3 S_m.

3.6B.2.1.1.6 Other High-Energy Piping and Moderate-Energy Piping

There are no piping components in this section other than ASME Safety Class 1.

3.6B.2.1.1.7 Regulatory Guide 1.46

RG 1.46 describes an acceptable basis for selecting the design locations and orientations of postulated breaks in fluid system piping within the reactor containment and for determining the measures that should be taken for restraint against pipe whipping that may result from such breaks.

GE-supplied NSSS analysis, design, and/or equipment utilized in this facility is in compliance with the intent of RG 1.46 through the incorporation of the following alternate approach. See the regulatory guide commitment matrix in Section 1.8 for commitment, revision number, and scope.

The recirculation piping also has been analyzed for the effects of hydrodynamic loads and the pipe break criteria of NUREG-0800. The analysis shows that the SRP criteria do not result in any additional pipe breaks beyond those using the design basis criteria (Section 3.6B.2.1.1.5).

3.6B.2.1.1.8 Types of Breaks to Be Postulated in Fluid System Piping

The following types of breaks are postulated in high-energy fluid system piping:

1. No breaks need be postulated in piping having a nominal diameter less than or equal to 1 in.

2. Circumferential breaks are postulated in piping exceeding a 1-in nominal pipe diameter.

3. Longitudinal splits are postulated only in piping having a nominal diameter equal to or greater than 4 in.

4. Circumferential breaks are to be assumed at all terminal ends and at intermediate locations chosen to satisfy the minimum break location criteria (Section 3.6B.2.1.1.5) for Safety Class 1 piping systems. At each of the intermediate postulated break locations identified to exceed the stress and usage factor limits of the criteria for Safety Class 1 piping systems, either a circumferential or a longitudinal break, or both, are postulated in accordance with the following:

    a. Circumferential breaks are postulated at fitting joints.

    b. Longitudinal breaks are postulated in the center of the fitting at two diametrically opposed points (but not concurrently) located so that the reaction force is perpendicular to the plane of the piping and produces out-of-plane bending.

    c. Consideration is given to the occurrence of either a longitudinal or circumferential break. Examination of the state of stress in the vicinity of the postulated break location may be used to identify the most probable type of break. If the maximum stress range in the longitudinal direction is greater than 1.5 times the maximum stress range in the circumferential direction, only the circumferential break may be postulated, and conversely if the maximum stress range in the circumferential direction is greater than 1.5 times the stress range in the longitudinal direction, only the longitudinal break may be postulated. If no significant difference between the circumferential and longitudinal stresses is determined, then both types of breaks are considered.

5. For design purposes, a longitudinal break area is assumed to be the equivalent of one circumferential pipe area.

6. For both longitudinal and circumferential breaks, after assessing the contribution of upstream piping flexibilities, pipe whipping is assumed to occur in the plane defined by the piping geometry and configuration for circumferential breaks and out-of-plane for longitudinal breaks, and to cause pipe movement in the direction of the jet reaction.

7. For a circumferential break, the dynamic force of the jet discharge at the break location is based upon the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. Justifiable line restrictions, flow limiters, and the absence of energy reservoirs are used, as applicable, in the reduction of the jet discharge.

The through-wall leakage cracks are postulated in moderate-energy fluid systems (or portions of systems). There are no moderate-energy piping components in this section.

3.6B.2.1.2 Criteria for Piping System in Area of Containment Isolation Valves

There are no containment penetrations associated with this section on the reactor recirculation piping system.

3.6B.2.2 Analytical Methods to Define Blowdown Forcing Functions and Response Models

3.6B.2.2.1 Analytical Methods to Define Blowdown Forcing Functions

Rupture of a pressurized pipe causes the flow characteristics of the system to change, creating reaction forces that can dynamically excite the piping system. The reaction forces are a function of time and space and depend upon fluid state within the pipe prior to rupture, break flow area, frictional losses, plant system characteristics, piping system, and other factors. The methods used to calculate the reaction forces for various piping systems are presented in the following sections.

3.6B.2.2.1.1 Recirculation Piping System

The criteria used for calculation of fluid blowdown forcing functions include:

1. Circumferential breaks are assumed to result in pipe severance and separation amounting to at least a one pipe diameter lateral displacement of the ruptured piping sections unless physically limited by piping restraints, structural members, or piping stiffness as may be demonstrated by the inelastic pipe whip analysis (Section 3.6B.2.2.2).

2. For a circumferential break, the dynamic force of the jet discharge at the break location is based upon the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. Justifiable line restrictions, flow

limiters, and the absence of energy reservoirs are used, as applicable, in the reduction of the jet discharge.

For both longitudinal and circumferential breaks, after assessing the contribution of upstream piping flexibilities, pipe whip is assumed to occur in the plane defined by the piping geometry and configuration for circumferential breaks and out-of-plane for longitudinal breaks, and to cause pipe movement in the direction of the jet reaction.

3. All breaks are assumed to attain full area instantaneously. A rise time not exceeding 1 msec is used for the initial pulse.

Blowdown forcing functions are determined by the following method: The predicted blowdown forces on pipes fed by a pressurized vessel can be described by transient (time dependent) and steady-state forcing functions. The forcing functions used are based on methods described in Reference 1. These may be simply described as follows:

1. The transient forcing functions occur at points along the pipe from the propagation of waves (wave thrust) along the pipe, and at the broken end from the reaction force due to the momentum of the fluid leaving the end of the pipe (blowdown thrust).

2. The waves cause various sections of the pipe to be loaded with time-dependent forces. It is assumed that the pipe is one-dimensional, in that there is no attenuation or reflection of the pressure waves at bends, elbows, and the like. Following the rupture, a decompression wave is assumed to travel from the break at a speed equal to the local speed of sound within the fluid. Wave reflections occur at the break end and the pressure vessel end, until a steady blowdown condition is established. Free space and vessel conditions are used as boundary conditions. The blowdown thrust that is caused by fluid acceleration from the break and static pressure in the break itself causes a time-dependent reaction force perpendicular to the pipe break, reaching a final steady-state value.

3. The initial blowdown force on the pipe is taken as the sum of the wave and blowdown thrusts and is equal to the vessel pressure (Po) times the break area (A). After the initial decompression period (i.e., the time it takes for a wave to reach the first change in direction), the force is assumed to drop off to the value of the blowdown thrust (i.e., 0.7 PoA).

4. Time histories of transient pressure, flow rate, and other thermodynamic properties of the fluid can be used to calculate the blowdown force on the pipe using the following equation:

$$F = \left[(P - P_a) + \frac{\rho V^2}{g_c} \right] A$$

(3.6B-1)

Where:

F	=	Blowdown force
P	=	Pressure at exit plane
P_a	=	Ambient pressure
V	=	Velocity at exit plane
ρ	=	Density at exit plane
A	=	Area of break
g_c	=	Newton's constant

5. Following the transient period, a steady-state period is assumed to exist. Steady-state blowdown forces are calculated considering frictional effects. ANS-58.2[1] is the base document used for determining thrust coefficients in evaluating the dynamic force due to jet discharge.

For frictionless flow, the theoretical maximum value of thrust coefficient for subcooled water is 2.0. Frictional effects are then considered to calculate the blowdown forces from this theoretical maximum value.

The steady-state thrust coefficient for frictionless flow of subcooled water based on the Henry-Fauske model[5] results in the following expression:

$C_T = 3.0 - 0.861h^{*2}$; $0 \leq h^* \leq 0.75$

$C_T = 3.22 - 3.0h^* + 0.97h^{*2}$; $0.75 < h^* \leq 1.0$

Where:

h^*	=	$(h_o - 180)/(h_{saturated} - 180)$
h_o	=	stagnation enthalpy (Btu/lbm)

Where:

$h_{saturated}$ = saturated water enthalpy at the stagnation pressure (Btu/lbm)

This model was confirmed by the experimental comparison work of Hanson.[6] For all values of h*, C_T is no greater than 2.0. For recirculation line break, in general, C_T = 2.0 is used for conservatism unless otherwise justified by documented evaluation of the empirical equation.

3.6B.2.2.2 Pipe Whip Dynamic Response Analyses

The prediction of time-dependent and steady-thrust reaction loads caused by blowdown of subcooled, saturated, and two-phase fluid from a ruptured pipe is used in design and evaluation of dynamic effects of pipe breaks. A detailed discussion of the analytical methods employed to compute these blowdown loads is given in Section 3.6B.2.2.1. A detailed discussion of analytical methods used to account for this loading is discussed below.

3.6B.2.2.2.1 Recirculation Piping System

The criteria used for performing the pipe whip dynamic response analyses include:

1. A pipe whip analysis is performed for each postulated pipe break. However, a given analysis can be used for more than one postulated break location if the blowdown forcing function, piping and restraint system geometry, and piping and restraint system properties are conservative for other break locations.

2. The analysis includes the dynamic response of the pipe in question and the pipe whip restraints that transmit loading to the structures.

3. The analytical model adequately represents the mass/inertia and stiffness properties of the system.

4. Pipe whip is assumed to occur in the plane defined by the piping geometry and configuration, and to cause pipe movement in the direction of the jet reaction.

5. Piping contained within the broken loop is no longer considered part of the RCPB. Plastic deformation in the pipe is considered a potential energy absorber. A limit of strain is imposed similar to that on the pipe whip restraint material (Section 3.6B.2.3.3.1, Type 1 restraint design limits).

6. Components such as vessel safe ends and valves that are attached to the broken piping system and do not serve a

safety function, or whose failure would not further escalate the consequences of the accident, are not designed to meet ASME Code imposed limits for essential components under faulted loading. However, if these components are required for safe shutdown, or if they serve a safety function to protect the structural integrity of an essential component, then these components are designed to ASME Code limits for faulted conditions and for limits necessary to ensure operability.

The pipe whip analysis was performed using the PDA (pipe dynamic analysis) computer program to determine the response of a pipe subjected to the thrust force occurring after a pipe break[2]. The program treats the situation in terms of generic pipe break configuration, which involves a straight, uniform pipe fixed at one end and subjected to a time-dependent thrust force at the other end. A typical restraint used to reduce the resulting deformation is also included at a location between the two ends. Nonlinear and time-dependent stress-strain relations are used for the pipe and the restraint. Similar to the popular plastic-hinge concept, bending of the pipe is assumed to occur only at the fixed end and at the location supported by the restraint.

Shear deformation is also neglected. The pipe bending moment-deflection (or rotation) relation used for these locations is obtained from a static nonlinear cantilever beam analysis. Using the moment-rotation relation, nonlinear equations of motion of the pipe are formulated using an energy consideration and the equations are numerically integrated in small time steps to yield time-history information about the deformed pipe.

A comprehensive verification program has been performed to demonstrate the conservatisms inherent in the PDA pipe whip computer program and the analytical methods utilized. Part of this verification program includes an independent analysis by Nuclear Services Corporation (NSC), under contract to GE, of the recirculation piping system for the 1969 Standard Plant Design. The recirculation piping system was chosen for study due to its complex piping arrangement and assorted pipe sizes. The NSC analysis included elastic-plastic pipe properties, elastic-plastic restraint properties, and gaps between the restraint and pipe, and is documented in Reference 3. The piping/restraint system geometry and properties and fluid blowdown forces were the same in both analyses. However, a linear approximation was made by NSC for the restraint load-deflection curve supplied by GE. This approximation is demonstrated on Figure 3.6B-1. The effect of this approximation is to give lower energy absorption of a given restraint deflection. Typically, this yields higher restraint deflections and lower restraint-to-structure loads than the GE analysis. The deflection limit used by NSC is the design deflection at one-half of the ultimate uniform strain for the GE restraint design. The restraint properties used for both analyses are provided in Table 3.6B-1.

A comparison of the NSC analysis with the PDA analysis (Table 3.6B-2) shows that PDA predicts higher loads in 15 of the 18 restraints analyzed. This is due to the NSC model including energy-absorbing effects in secondary pipe elements and structural members. However, PDA predicts higher restraint deflections in 50 percent of the restraints. The higher deflections predicted by NSC for the lower loads are caused by the linear approximation used for the force-deflection curve rather than by differences in computer techniques. This comparison demonstrates that the simplified modeling system used in PDA is adequate for pipe rupture loading, restraint performance, and pipe movement predictions within the meaningful design requirements for these low-probability postulated accidents.

A comprehensive test program was performed to develop the restraint properties such as the load-deflection power relationships shown in Table 3.6B-1. A series of static and dynamic deformation tests of model restraints were conducted. The model restraints were scaled down from the restraints suitable for 26-in size pipe. Also, the static and dynamic material properties were obtained from tensile tests of bar specimens. The results of these tests were studied and analyzed for use in the development of an analytical model that predicts the behavior of a restraint when loaded by a moving pipe. Tests were performed on some full-scale restraints that showed that the pipe whip restraints will perform their designated functions adequately.

3.6B.2.3 Dynamic Analysis Methods to Verify Integrity and Operability

3.6B.2.3.1 Jet Impingement Analyses and Effects on Safety-Related Components

The methods used to evaluate the jet effects resulting from the postulated breaks of high-energy piping are presented in Section 3.6A.

3.6B.2.3.2 Pipe Whip Effects on Safety-Related Components

This section provides the criteria and methods used to evaluate the effects of pipe displacements on safety-related structures, systems, and components following a postulated pipe rupture.

3.6B.2.3.2.1 Pipe Whip Effects Following a Postulated Rupture of the Recirculation Piping System

Pipe whip (displacement) effects on safety-related structures, systems, and components (nozzles, valves, tees, etc.) that are in the same piping run as the one in which the break occurred are determined by:

1. The criteria used for determining the effects of pipe displacements on the in-line components are as follows:

    a. Components such as vessel safe ends and valves that are attached to the broken piping system and do not serve a safety function, or whose failure would not further escalate the consequences of the accident, need not be designed to meet ASME Section III imposed limits for essential components under faulted loading.

    b. If these components are required for safe shutdown, or serve a safety function to protect the structural integrity of an essential component, limits to meet the Code requirements for faulted conditions and limits to ensure operability, if required, are met.

2. The methods used to calculate the pipe whip loads on piping components in the same run as the postulated break are described in Section 3.6B.2.2.2.

3.6B.2.3.3 Load Combinations and Design Criteria for Pipe Whip Restraints

Pipe whip restraints, as differentiated from piping supports, are designed to function and carry load for an extremely low probability gross failure in a piping system carrying high-energy fluid. The piping integrity does not usually depend on the pipe whip restraints for any load combination. When the piping integrity is lost because of a postulated break, the pipe whip restraint acts to limit the movement of the broken pipe to an acceptable distance. The pipe whip restraints (i.e., those devices that serve only to control the movement of a ruptured pipe following gross failure) are subjected to once-in-a-lifetime loading. The pipe break event is considered to be an abnormal condition for the ruptured pipe, its restraints, and the structure to which the restraint is attached. The design and analysis of these components for this event are described in Section 3.6B.2.2 and the following sections.

3.6B.2.3.3.1 Recirculation Piping System Pipe Whip Restraints

The pipe whip restraints designed, tested, and fabricated by GE for the recirculation loop piping utilize energy-absorbing U-rods to attenuate the kinetic energy of a ruptured pipe. A typical pipe whip restraint is shown on Figure 3.6B-2. A principal feature of these restraints is that they are installed with several inches of annular clearance between them and the process pipe. This allows for installation of normal piping insulation and unrestricted pipe thermal movements. Select critical locations inside primary containment are also monitored during hot functional testing to provide verification of adequate clearances prior to plant operation.

The specific design objectives for the restraints are:

1. The restraints will in no way increase the RCPB stresses by their presence during any normal mode of reactor operation or condition.

2. The restraint system will function to stop the movement of a pipe failure (gross loss of piping integrity) without allowing damage to critical components or missile development.

3. The restraints should provide minimum hindrance to ISI of the process piping.

For the purposes of design, the pipe whip restraints are designed for the following dynamic loads:

1. Blowdown thrust of the pipe section that impacts the restraint.

2. Dynamic inertia loads of the moving pipe section that is accelerated by the blowdown thrust and subsequent impact on the restraint.

3. Design characteristics of the pipe whip restraints are included and verified by the pipe whip dynamic analysis described in Section 3.6B.2.2.2.

4. Since the pipe whip restraints are not in contact with the piping during normal plant operation, the postulated pipe rupture event is the only design loading condition.

The recirculation loop pipe whip restraints are composed of several components, each of which performs a different function. These components are categorized as Types I, II, III, and IV, as follows:

Type I — Restraint energy absorption members - Members that, under the influence of impacting pipes (pipe whip), absorb energy by significant plastic deformation (e.g., U-rods).

Type II — Restraint connecting members - Components that form a direct link between the restraint plastic members and the structure (e.g., clevises, brackets, pins).

Type III — Restraint connecting member structural attachments - Fasteners that provide the method of securing the restraint connecting members to the structure (e.g., weld attachments, bolts).

Type IV Structural and civil components - Steel and concrete structures that ultimately must carry the restraint load (e.g., sacrificial shield, trusses).

Each of these components is typically constructed of a different material, with a different design objective in order to perform the overall design function. Therefore, the material and inspection requirements and design limits for each are somewhat different. These requirements for each component are given below:

1. Type I Restraint (e.g., U-rods)

   a. Materials All materials used to absorb energy through significant plastic deformation conform to:

      (1) ASME Section III, Subsection NB, Boiler and Pressure Vessel Code for Safety Class I components, or

      (2) ASTM Specifications with consideration for brittle fracture control, or

      (3) ASME Section III, Subsection NF, Boiler and Pressure Vessel Code, if applicable.

      (4) GE Material Specifications.

   b. Inspection Inspection and identification of materials conform to:

      (1) ASME Section III, Subsection NB, Boiler and Pressure Vessel Code for Safety Class I components (Section V, Nondestructive Examination Methods), or

      (2) ASTM Specifications procedures including volumetric and surface inspection, or

      (3) ASME Section III, Subsection NF, Boiler and Pressure Vessel Code, if applicable.

      (4) GE Methods and Acceptance Standards.

   c. Design Limits

      (1) Design local strain The permanent strain in metallic ductile materials is limited to:

(a) 50 percent of the minimum actual ultimate uniform strain (at the maximum stress on an engineering stress-strain curve) based on restraint material tests, or

(b) One-half of minimum percent elongation as specified in the applicable ASME Section III Boiler and Pressure Vessel Code or ASTM Specifications, when demonstrated to be as or more conservative than the above.

(2) <u>Design steady-state load</u> The maximum restraint load is limited to 80 percent of the minimum calculated static ultimate restraint strength at the drywell design temperature. This strain is less than 50 percent of the ultimate uniform strain for all materials used for Type I components.

(3) <u>Dynamic material mechanical properties</u> The material selected exhibits tensile and impact properties not less than:

(a) 70 percent of the static percent elongation, or

(b) 80 percent of the statically determined minimum total energy absorption.

2. <u>Type II Restraint</u> (e.g., clevises, brackets, pins)

a. <u>Materials</u> Material selection conforms to:

(1) ASTM Specifications including consideration for brittle fracture control, or

(2) ASME Section III, Subsection NF, Boiler and Pressure Vessel Code, if applicable

(3) GE Material Specifications

b. <u>Inspection</u> Inspection conforms to:

(1) ASME/ASTM requirements or process qualification and finished part surface inspection in accordance with ASTM methods, or

(2) ASME Section III, Subsection NF, Boiler and Pressure Vessel Code, if applicable.

(3) GE Methods and Acceptance Standards.

c. Design Limits Design limits are based on the following stress limits:

(1) Primary stresses (in accordance with definitions in ASME Section III) are limited to the higher of:

(a) 70 percent of S_u where S_u = minimum ultimate strength by tests or ASTM specification

(b) S_y+ 1/3 (S_u-S_y) where S_y = minimum yield strength by test or ASTM specification, or

(2) Recommended stress limits in accordance with ASME Section III, Subsection NF, for faulted conditions, if applicable.

3. Type III Restraint (fasteners)

a. Materials Fastener material conforms to ASTM, ASME, or MIL requirements.

b. Inspection All fasteners are inspected or certified in accordance with applicable ASTM, ASME, or MIL specifications.

c. Design limits Same as Type II.

4. Type III Restraint Material (welds)

a. Materials Weld materials for attachment to carbon steel structures are limited to low hydrogen types, or processes that are inherently low hydrogen.

b. Inspection Liquid penetrant surface inspection is performed in accordance with:

(1) ASTM Specification E165, or

(2) AWS Structural Welding Codes, AWS-D1.1.

c. Design limits Design limits are based on the following stress limits: the maximum primary weld stress intensity (two times maximum shear stress) is limited to three times AWS or AISC building allowable weld shear stress.

d. Procedures Procedures and welders are qualified in accordance with the latest AWS Code for welding in building structures.

5. Type IV Restraint (structural and civil components) Material, inspection, and design requirements for the structural and civil components are provided by industry standards such as AISC, ACI, and ASME Section III, Division II, along with appropriate requirements imposed for similar loading events. These components are also designed for other operational and accident loadings, seismic loadings, wind loadings, and tornado loadings.

The design basis approach of categorizing components is consistent in allowing less stringent inspection requirements for those components subject to lower stresses. Considerable strength margins exist in Type II through IV components even to the limit of load capacity (fracture) of a Type I component. Impact properties in all components are considered since brittle type failures could reduce the restraint system effectiveness.

In addition to the design considerations discussed above, strain rate effects and other material property variations have been considered in the design of the pipe whip restraints. The material properties utilized in the design have included one or more of the following methods:

1. Code minimum or specification yield and ultimate strength values for the affected components and structures are used for both the dynamic and steady-state events.

2. Not more than a 10-percent increase in Code or specification values is used when designing components or structures for the dynamic event. Code minimum or specification yield and ultimate strength values are used for the steady-state loads.

3. Representative or actual test data values are used in the design of components and structures.

4. Representative or actual test data are used for any affected component(s) and the minimum Code or specification values for the structures for the dynamic and the steady-state events.

3.6B.2.4 Material to Be Submitted for the Operating License Review

3.6B.2.4.1 Implementation of Criteria for Pipe Break and Crack Location and Orientation

Postulated Pipe Breaks in Recirculation Piping System

The criteria for selection of postulated pipe breaks in the recirculation piping system inside containment are provided in Section 3.6B.2.1. The postulated pipe break locations and types selected in accordance with these criteria are shown on Figure 3.6B-3.

3.6B.2.4.2 Implementation of Special Protection Criteria

Pipe Whip Restraints for Recirculation Piping System

The pipe whip restraint locations for the recirculation piping system are shown on Figure 3.6B-3. This system of restraints is provided to prevent unrestrained pipe whip at break locations postulated in Section 3.6B.2.4.1.

3.6B.2.4.3 Summary of Jet Effects Analyses Results

Jet Effects for Postulated Ruptures of Recirculation Piping System

The fluid jet thrust for each of the recirculation piping postulated break locations shown on Figure 3.6B-3 is calculated in accordance with Section 3.6B.2.2. The jet effects will be evaluated in accordance with Section 3.6A.2.3.1 and results will be presented in this section.

3.6B.3 References

1. American National Standards Institute (ANSI). ANS-58.2 (ANSI N176), Proposed American National Standard Design Basis for Protection of Light Water Nuclear Power Plants Against Effects of Postulated Pipe Rupture, 1980.

2. GE Report NEDE-10813A. PDA - Pipe Dynamic Analysis Program for Pipe Rupture Movement (Proprietary Filing).

3. Nuclear Services Corporation Report No. GEN-02-02, Final Report Pipe Rupture Analysis of Recirculation System for 1969 Standard Plant Design.

4. Shapiro, A. H. The Dynamics and Thermodynamics of Compressible Fluid Flow, Vol. 1. Ronald Press, New York, 1965.

5. Webb, S. W. Evaluation of Subcooled Water Thrust Forces. Nuclear Technology, Vol. 31, October 1976.

6. Hanson, G. H. Subcooled-Blowdown Forces on Reactor System Components: Calculation Method and Experimental Confirmation. Idaho Nuclear Corporation Report IN-1354, June 1970.

3.7 SEISMIC DESIGN

This section is composed of two parts. Section 3.7A is applicable to the seismic design applied to structures, systems, and components within the SWEC scope of supply. Section 3.7B is applicable to the seismic design of structures, systems, and components within the GE scope of supply.

3.7A SEISMIC DESIGN (SWEC SCOPE OF SUPPLY)

3.7A.1 Seismic Input

3.7A.1.1 Design Response Spectra

The design response spectra are developed in accordance with published procedures[1] and RG 1.60 (Section 3.7A.2.5). In this method, the critical parameter is the maximum expected ground acceleration. Associated with the maximum ground acceleration, the maximum ground displacement is determined by linear scaling from Figures 1 and 2 of RG 1.60 in proportion to the maximum expected ground acceleration. Detailed smooth spectra for any given value of damping are obtained by locating critical points and joining them by straight lines in a tripartite logarithmic plot. These critical points are obtained from given amplification factors and cutoff frequencies. The design value of the maximum ground acceleration is 0.15 g for the SSE and 0.075 g for the OBE. Figures 3.7A-1 and 3.7A-2 show the smooth design response spectra for horizontal and vertical earthquakes associated with the SSE. These design response spectra are not related to any site-dependent ground motion time history.

As shown in Figures 3.7A-1 and 3.7A-2, the ratios of vertical design response spectral values to the horizontal design response spectral values comply with the position of NRC RG 1.60. The ratio varies for different frequencies as required by RG 1.60.

3.7A.1.2 Design Response Spectrum Derivation

An artificial earthquake is generated to give response spectra enveloping the design response spectra. The artificial earthquake is generated and checked by comparing spectral values at 80 periods between 0.02 and 5.0 sec. The periods are spaced according to the rule:

$$T_n = \lambda T_{n-1} \qquad (3.7A\text{-}1)$$

Where:

T_n = Period n in spectrum computation

T_{n-1} = Period (n-1) in spectrum computation

$$\lambda = \left(\frac{T_f}{T_i}\right)\left(\frac{1}{n-1}\right) = 1.0724$$

T_i = Initial period = 0.02 sec

T_f = Final period = 5.0 sec

N = Total number of periods = 80

The acceleration time history yields ground response spectra at damping values of 1, 2, 5, 7, and 10 percent that envelop the smoothed site design ground response spectra (SSE) for damping values as shown on Figures 3.7A-3 through 3.7A-17. The calculated response spectra and design response spectra of RG 1.60 are compared. Based on this comparison, the artificial earthquake is used as the design time history for structural analysis.

Details of the artificial acceleration record and its development are presented in Section 3.7A.2.5.

3.7A.1.3 Critical Damping Values

3.7A.1.3.1 Structures

Seismic analysis is performed using total system damping characterized by modal damping. The modal damping value is calculated as a ratio of the sum of the energy dissipated in each component element (based upon the assigned damping ratio of each element) to the total available modal energy. Further discussion of modal damping appears in Section 3.7A.2.15.

In determining the modal damping ratios, component damping values consistent with the stress intensities are used. For example, component damping for welded structural steel is assigned a value of 2 percent for OBE and 4 percent for SSE.

The damping ratios in RG 1.61 and Table 3.7A-1 for various components are used in the design.

3.7A.1.3.2 Equipment

The percentages of critical damping values assigned to Category I systems and components are in accordance with RG 1.61 and are presented in Table 3.7A-1.

3.7A.1.3.3 Piping

The percentage of critical damping values used for the analysis of all piping are consistent with RG 1.61 and are presented in Table 3.7A-1.

The alternative damping values used for Unit 2 will be those described in ASME Code Case N-411 (Figure 3.7A-34). The expected increased piping displacements (resulting from using damping values in ASME Code Case N-411) will be verified when piping supports are moved, modified, or eliminated. The verification will ensure that there will be no adverse interaction with adjacent structures, components, and equipment.

3.7A.1.4 Support Media for Category I Structures

Major Category I structures are founded on sound rock. The top of bedrock is encountered at elevations ranging from 246 to 240 ft. The foundation/support media information for Category I structures is summarized in Table 3.7A-2. Rock properties including wave propagation velocities, densities, and shear modulus can be found in Sections 2.5.2.5 and 2.5.4.2. The static properties of Category I structural backfill are discussed in Section 2.5.4.5. The dynamic properties of Category I structural backfill are presented in Section 2.5.4.8.

3.7A.2 Seismic System Analysis

3.7A.2.1 Seismic Analysis Methods

The structural responses of the reactor building and other Category I structures to the application of horizontal and vertical earthquake ground motions are determined by the response spectrum modal analysis method. Seismic responses for all Category I structures are determined from an application of two orthogonal horizontal and one vertical earthquake ground motions, assumed to act simultaneously. The earthquake ground motions are established in the form of response spectra for the SSE and OBE as described in Section 3.7A.1. The computer program STRUDL (Appendix 3A) is used to obtain the mode shapes, natural frequencies, and responses for the major Category I structures. The response spectra for the floor levels are obtained using the TIMHIS6 computer program (Appendix 3A). The combination of design loading conditions with seismic loading and the allowable stress levels are given in Section 3.8.

The dynamic models of the Category I structures consist of generalized systems of lumped masses, each with 6 deg of freedom, connected by massless, linearly elastic springs. The masses consist of floors, tributary walls, columns, equipment, and piping. The lumped mass models of the structures are shown on Figures 3.7A-18 through 3.7A-20.

Since the major structures are founded on rock, the seismic analyses are performed using fixed bases (Section 3.7A.2.4.1). The secondary containment and primary containment structures are connected only at the foundation; therefore, each structure was modeled separately. Each model is constructed so that it properly represents the free vibration of a cantilevered structure in shear, flexure, and torsion. Generally, mass locations are selected at points with a concentration of mass (e.g., floor elevations), or where there is a special interest in the response (equipment locations).

In the dynamic modeling of Category I structures, the floors are treated as rigid plates or diaphragms that transfer earthquake inertia forces to frames and diaphragm walls, which in turn transfer the loads to the foundation mat. Beam theory, combining the effects of shear, flexure, torsion, and axial deformation, is used to establish the stiffness characteristics of the frame-wall systems.

The criteria used to determine an adequate number of masses in dynamic modeling of all Category I structures are, in general, dictated by the number of floor elevations and the roof elevations of a structure. It is at these points that masses are lumped and include half the walls above and below the floor, the floor itself, and major pieces of equipment resting on the floor or supported from the walls. This is done because this mass distribution closely approximates the real mass distribution of the actual structure. Additional mass points may also be included where mass distribution dictates.

The criterion for establishing the minimum number of masses is one that provides sufficient accuracy for natural frequencies below 33 cps, since above this value there is little amplification of earthquake excitation. Equipment stiffer than 33 cps responds at the mass response, and larger errors in structural frequencies can be tolerated. The number of masses in a model are considered adequate if the number of degrees of freedom is equal to twice the number of modes with frequencies less than 33 cps. A sufficient number of modes were considered so that the inclusion of any additional mode will not result in a 10-percent increase in responses.

The seismic motion of all Category I structures is determined by applying the earthquake ground motions to the appropriate dynamic models. Where non-Category I structures are attached to Category I structures, the effects are analyzed by including the non-Category I structure in the seismic model of the Category I structure. In general, interaction between Category I and non-Category I structures is eliminated by providing structural gaps and separate foundations for the structures. The structural gaps are designed so that seismic motion between the structures is unimpeded and algebraic summation of maximum relative displacements of adjacent structures under most critical conditions is less than the structural gap.

A tabulation of the structural gaps surrounding Category I structures is shown in Table 3.7A-10. To determine worst computed gaps between the structures, out-of-phase deflection (displacement) of the structures (i.e., structures leaning toward each other) is assumed during a SSE event. Allowable construction tolerance, when added to cumulative displacements, will not exceed structural gap provided. See Figure 1.2-2 for the arrangement of plant structures. As can be seen from the tabulation, the cumulative deflection (displacement) under a SSE event does not exceed the structural gap provided in each case. A minimum of 3 in structural gap is provided between the structures as a common design practice.

For the effects of hydrodynamic loads on the structures, refer to the Design Assessment Report for Hydrodynamic Loads (DAR) (Appendix 6A). For the effects of piping on the seismic analysis of the structures, refer to Section 3.7A.2.3.

3.7A.2.2 Natural Frequencies and Response Loads

A summary of natural frequencies of vibration for Category I structures and the corresponding mode shapes is given in Tables 3.7A-3 through 3.7A-6. Typical modal responses for the primary containment at several selected elevations are given in Tables 3.7A-7 and 3.7A-8. Structural response characteristics in the form of building acceleration profiles for the primary containment, reactor building, control building, and diesel generator building are shown on Figures 3.7A-21 through 3.7A-32.

Amplified response spectra (ARS) are generated for all Category I structures at major Category I equipment elevations and points of support to define the seismic environment for the subsystem analyses (Section 3.7A.2.5).

3.7A.2.3 Procedures Used for Modeling

The dynamic models of Category I structures consist of systems of generalized spring-connected lumped masses. The lumped masses and connecting springs of the dynamic model are determined in order to obtain a satisfactory representation of the dynamic behavior of the actual structure. In general, masses are located at floor elevations and include the floor system, a portion of the walls and columns both above and below the floor system, and major components, equipment, and piping. In addition, masses are located at elevations where response values are required. The structural spring elements that connect mass points represent the stiffness characteristics of the walls and column sections. These characteristics are determined from beam theory for concrete sections, which includes the effects of shear, flexure, and torsion, and from frame analysis for steel frameworks.

In the course of analysis, a comparison of relative mass and stiffness properties between connected components is performed to determine whether coupling effects should be considered in the analysis of the supported components. This consideration is pertinent to all Category I systems and equipment. It is a design goal to decouple (dynamically) equipment from its attached components.

Some basic relationships are used to allow the conclusion of decoupling. Failure to meet any one of these requires that either (1) additional restraints are provided to suitably alter stiffness parameters and thereby dynamically uncouple the system, or (2) the analytical model be formulated to include both the supporting and supported components in order to actually determine the coupling effects of the combined system. The principal purpose for such considerations is to define component interface loads for inclusion in component adequacy documentation, as well as seismic inputs to supported components. The basic relationships used as a guide to conclude decoupling are:

1. If R_m <0.01, decoupling is acceptable for any R_f.

2. If 0.01 $\leq R_m \leq$0.1, decoupling is acceptable if 0.8 $\geq R_f$ $\geq$1.25.

3. If R_m >0.1, an approximate model of the subsystem is included in the primary system model.

Where:

$$R_m = \frac{\text{Total mass of the supported subsystem}}{\text{Total mass of the supporting system}}$$

$$R_f = \frac{\text{Fundamental frequency of the supported subsystem}}{\text{Dominant frequency of the support motion}}$$

If the subsystem is comparatively rigid and also rigidly connected with the primary system, only the mass of the subsystem is included at the support point in the primary system model. On the other hand, in the case of a subsystem supported by very flexible connections (e.g., pipe supported by hangers), the system is not included in the primary model. In most cases, the equipment and components, which come under the definition of subsystems, are analyzed as a decoupled system from the primary structure, and the seismic input for the subsystem is obtained by the analysis of the primary system.

3.7A.2.4 Soil/Structure Interaction

All Category I structures are supported on natural bedrock except that electrical ductline 907, part of ductline 922, and electrical manhole no. 1 are supported by Category I structural fill (Table 3.7A-2). Certain portions of the floor slabs in the diesel generator building beneath which granular fill was placed are shown on Figure 3.7A-33. This fill is used solely as a construction form and, as shown on Figure 3.7A-33, was placed in limited areas. The static and dynamic design of these slabs assumed no load transfer to, or bearing support from, the underlying backfill. Wall design, however, considered lateral soil pressures as shown on Figure 2.5-110. The electrical ductline is a continuously supported underground structure consisting of conduit totally encased in reinforced concrete. The electrical ductlines are evaluated for relative motion between the portions of duct supported on rock and structural fill using ASCE procedures[5]. The electrical ductlines are capable of withstanding the stresses and strains induced by the relative motion between the portions of duct supported on rock and structural fill. Manhole no. 1 is designed for seismic loads using the procedures described in Section 3.7A.3.12 for Category I tunnels. See Figure 3.7A-33 for details of the manhole and ductline.

All Category I structural fill provides a factor of safety against liquefaction as listed in Section 2.5.4.8. For static properties of the Category I fill see Section 2.5.4.5.

3.7A.2.4.1 Rock/Structure Interaction

Dynamic analyses for structures founded on rock are performed using fixed base models because shear wave velocity exceeds 3,500 fps. The details of the geophysical survey are discussed in Sections 2.5.4.2 and 2.5.4.4.

3.7A.2.5 Development of Floor Response Spectra

ARS are defined as plots of the maximum response versus period for single degree-of-freedom systems at various locations in structures subjected to dynamic loading. In the analysis of equipment with masses that are small compared to the masses of the dynamic model of the supporting structure, the response of the structure is independent of the response of the equipment. The problem can then be solved in two parts: the response of the structure due to ground acceleration can be determined, and that response can be applied as support accelerations to the equipment. In such cases, the use of ARS is an acceptable approach to the problem of determining the dynamic loads on equipment. The time history method of analysis, using the TIMHIS6 computer program (Appendix 3A), is used to generate the ARS for Category I piping and equipment.

The undamped equations of motion for an n degree-of-freedom structural model are solved to determine modal eigenvalues, eigenvectors, and participation factors. The modal equation of motion for structural response in mode i can be written:

$$\ddot{X}_i(t) + 2D_i\omega_i\dot{X}_i(t) + \omega_i^2X_i(t) = \Gamma_i\ddot{U}_g(t) \qquad (3.7A\text{-}2)$$

Where:

$X_i(t)$ = Time-dependent modal amplitude

D_i = Modal damping

ω_i	=	Modal circular frequency
Γ_i	=	Modal participation factor
$\ddot{U}_g(t)$	=	Ground acceleration time history

Equation 3.7A-2 can be solved numerically for $X_i(t)$. The solution for the structural response in mode i is then:

$$[U_i(t)] = [a_i]\, X_i(t) \tag{3.7A-3}$$

and:

$$[\ddot{U}_i(t)] = [a_i]\, \ddot{X}_i(t) \tag{3.7A-4}$$

Where:

$[U_i(t)]$	=	Time-dependent displacement vector in mode i for the n degree-of-freedom system
$[\ddot{U}_i(t)]$	=	Time-dependent acceleration vector in mode i for the n degree-of-freedom system
$[a_i]$	=	Eigenvector for mode i

The significant structural responses may be added numerically to obtain the time history of acceleration which may be applied to the supports of damped single degree-of freedom systems. The maximum values of response of the single degree-of-freedom systems produce the ARS. ARS are developed for two horizontal and one vertical excitation.

The ground acceleration $\ddot{U}_g(t)$ is an artificial time history with a total duration of 15 sec whose ground response spectrum envelops the response spectrum as specified in RG 1.60. An artificial accelerogram of ground excitation that reproduces the frequency content displayed in a response spectrum is simulated statistically in a digital computer by using a multi-P stochastic model. In this model, the earthquake motion is considered to be a wide-band stationary process whose spectral density function, duration, and maximum acceleration are specified.

The artificial motion is generated by matching the target spectrum for several specified percentages of critical damping at 80 oscillator periods distributed logarithmically from 0.02 sec (50 Hz) to 5.0 sec (0.2 Hz). These statistically independent orthogonal ground accelerations (two horizontals and one vertical) are applied simultaneously to the support. The particular response in a direction of interest is obtained by algebraic summation of the response in that direction at each time interval due to each of the three ground accelerations.

3.7A.2.6 Three Components of Earthquake Motion

The spatial components from seismic response analysis are combined in accordance with RG 1.92. All Category I structures are analyzed from the three orthogonal component motions (two horizontal and one vertical) of the prescribed earthquake. When response spectrum analysis is performed, the representative maximum value of a particular response of interest for design (e.g., stress, strain, moment, shear, or displacement) of a given element of a structure, system, or component subjected to a simultaneous action of the three components of the earthquake is obtained by taking the square root of the sum of the squares (SRSS) of corresponding representative maximum values of the spectrum response to each of the three components calculated independently.

In cases where time history dynamic analysis is used, three statistically independent (maximum correlation factor of 0.2) orthogonal ground accelerations (two horizontal and one vertical) of the prescribed earthquake are input simultaneously. The particular response in a direction of interest is obtained by algebraic summation of the response in that direction at each time interval due to each of the three ground accelerations.

3.7A.2.7 Combination of Modal Responses

In the response spectrum modal analyses, the modal responses are combined using the grouping method or double sum method as described in RG 1.92.

3.7A.2.8 Interaction of Non-Category I Structures with Category I Structures

When Category I and non-Category I structures are integrally connected, the non-Category I structures are included in the model when determining the response of the Category I structures.

All non-Category I structures meet one of the following requirements:

1. The collapse of a non-Category I structure will not cause the non-Category I structure to strike a Category I structure or component.

2. The collapse of a non-Category I structure will not impair the integrity of Category I structures or components.

3. The non-Category I structure is analyzed and designed to prevent its failure under SSE conditions in such a

manner that the margin of safety of the structure is equivalent to that of the Category I structures.

3.7A.2.9 Effects of Parameter Variations on Floor Response Spectra

The effects on the calculated value of fundamental structural periods due to expected variations in damping and the structural material properties are taken into account. SWEC-supplied Category I equipment and piping systems designed using floor response spectra and having natural periods within ±15 percent of the peak resonant period(s) are assigned the peak response value. Outside this range, the broadened peak is bounded on each side by lines that are parallel to lines forming the original spectrum peak. Damping values are assigned to systems as outlined in Section 3.7A.1.3.

In addition to the broadening of peaks, two dynamic models were used in the seismic analysis of the primary containment. This was done to account for the variation in the stiffness of the primary containment members that occur during a LOCA. For one model, the concrete containment is assumed to be completely uncracked, while the other model utilizes cracked section properties for the primary containment wall.

Since two different models are used, two sets of response spectra in each direction are generated at each floor level. The design floor response spectra are obtained from an envelope of the two floor spectra for each direction of excitation. The response spectrum analysis is also performed on each model. The maximum value of a particular response of interest (e.g., moment, shear, axial force, displacement) is used for the design of the primary containment structure.

3.7A.2.10 Use of Constant Vertical Static Factors

Vertical seismic system multimass dynamic models are used to obtain vertical response loads for the seismic design of Category I structures. Therefore, constant vertical load factors are not used to account for vertical response to earthquakes of Category I structures.

3.7A.2.11 Method Used to Account for Torsional Effects

Category I structures may have natural torsional modes of vibration due to eccentricities between the centers of rigidity and centers of mass of the structural elements. The presence of eccentricities generates coupling between translational directions of motion, resulting in torsion. Thus, general three-dimensional models are set up, followed by complete dynamic analyses as described previously. The results of these analyses include torsional effects.

Since the three-dimensional model accounts for the torsional effects, including the effects of eccentricities between the centers of rigidity and centers of mass of the structural components, the additional eccentricity of 5 percent of the maximum building dimension is not considered in the analyses. Additionally, the design of the control building, which is considered representative of Category I structures, was reviewed for an additional torsional moment resulting from additional eccentricity of 5 percent of the maximum building dimension. It was shown that the additional shear stresses resulting from this analysis were not significant and were within the design capacities.

3.7A.2.12 Comparison of Responses

A typical comparison of structural response obtained by response spectrum and time history analyses for selected points throughout the plant is given in Table 3.7A-9.

3.7A.2.13 Methods for Seismic Analysis of Dams

There are no dams that will impact Unit 2.

3.7A.2.14 Determination of Category I Structure Overturning Moments

The overturning moments induced by seismic loading are computed by the spectrum method of analysis (Section 3.7A.2.1.1) for each direction of excitation separately. The applied overturning moment is computed from the SRSS of maximum responses from three individual directions of excitation.

3.7A.2.15 Analysis Procedure for Damping

In order to use modal analysis, damping values in different elements of a coupled system are accounted for by using stiffness as a weighting function in generating the modal damping values[2]. According to this method, for a system vibrating in its ith mode, the ith modal damping can be estimated by evaluating the ratio of the total energy dissipated due to the presence of damping in different elements of the system to the total strain energy stored in the system in its ith mode. The limiting values of damping factors are given in Table 3.7A-1 for various systems and components.

Two types of damping are generally recognized: (1) viscous, in which the energy dissipated per cycle is proportional to frequency, and (2) hysteretic, in which no frequency dependence is seen. Most structural elements display hysteretic behavior, while supporting soils appear to combine both hysteretic and viscous damping mechanisms. Since the dynamic analysis is performed using a fixed base model, only hysteretic damping need be considered. Whitman's Seismic Design for Nuclear Power Plants[3] gives a useful approximation for the damping of each mode when material damping varies from element to element. This expression for the (equivalent viscous) modal damping is obtained by a strain-energy weighting of element damping:

$$B^j_{eqv} = \frac{\sum_{i=1}^{N} D_i E^j{}_i}{\sum_{i=1}^{N} E^j{}_i}$$

(3.7A-5)

Where:

N	=	Number of elements
D_i	=	Hysteretic damping ratio for element i
E^j_i	=	Strain energy in element i when deflected into mode shape j
B^j_{eqv}	=	Equivalent viscous damping ratio (fraction of critical) for structure vibrating in mode j

In particular, when damping is uniform, i.e., $D_i = D$, then:

B^j_{eqv} = D for all modes

When damping is not uniform, modal damping is weighted toward those elements that make the largest contribution to the energy of each mode.

3.7A.3 Seismic Subsystem Analysis

The design of Category I subsystems (i.e., components, equipment, piping, supports) includes OBE and SSE seismic loading conditions.

The SSE produces the maximum vibratory ground motion for which Category I systems and components are designed to remain functional. These systems and components are those necessary to ensure:

1. The integrity of the RCPB.

2. The capability to shut down the reactor and maintain it in a safe shutdown condition.

3. The capability to prevent or mitigate the consequences of accidents that could result in potential offsite

exposures comparable to the guideline exposures of 10CFR100.

The OBE produces the vibratory ground motion for which those features of the nuclear power plant that are necessary for continued operation without undue risk to the health and safety of the public are designed to remain functional. System seismic classification is provided in Table 3.2-1.

3.7A.3.1 Seismic Analysis Methods

3.7A.3.1.1 Seismic Qualification of Components

This section provides the qualification methods for equipment affected by seismic loads. The methods for the qualification of equipment affected by hydrodynamic loads associated with SRV discharge and the postulated LOCA are provided in the DAR, Appendix 6A, Subsection 6A.9.

All Category I equipment is qualified for seismic adequacy. Depending upon equipment location, the basic source of seismic design data is either the ground response spectra or the ARS, derived through a dynamic analysis of the structure. The four principal methods of documenting adequacy for Category I components are static analysis, dynamic analysis, dynamic testing, and static deflection testing. These methods are used singly or in combination to qualify equipment.

Static Analysis

Static analysis is used for equipment that can be modeled as relatively simple structures. This type of analysis involves the multiplication of the component weights by the specified seismic accelerations (direction-dependent loadings) to produce forces that are applied at the centers of gravity in the horizontal and vertical directions. A stress analysis of critical items, such as support points, hold-down bolts, and other structural members, is performed to determine their adequacy. The deflections of critical components are also calculated and compared with specified tolerances.

In the specification of equipment for static analysis, two ranges of acceleration data are provided: a resonant range distinguished by lower frequencies with amplified response accelerations, and a rigid range characterized by higher frequencies and essentially nonamplified response. The division between the two ranges is termed the cutoff frequency. Selection of the appropriate range depends upon the fundamental natural frequency of the equipment. If this value is beyond the resonant range (i.e., higher than the cutoff frequency), the equipment is analyzed to rigid range response accelerations.

Equipment having a fundamental frequency in the resonant range of the ARS is analyzed by using the peak resonant acceleration, increased by a static coefficient of 1.3. This factor accounts for potential multimode response (Section 3.7A.3.5).

Each of the three defined directions of earthquake input (two horizontal and one vertical taken orthogonally) are evaluated separately. The calculated results of the three analyses are superimposed using the SRSS criterion. The particular response values (e.g., acceleration, force, stress) to be combined are optional, but the option selected remains consistent throughout, following the guidelines of RG 1.92.

Dynamic Analysis

A detailed dynamic analysis is performed when component complexity or dynamic interaction precludes static analysis, or when static analysis is too conservative. An infinite number of coordinates would be required to fully describe the behavior of a component subjected to dynamic loads. Since calculation at every point of a complex model is impractical, the analysis is simplified by the selection of a limited number of mass points. The lumped mass approach is employed in which the main structure is represented in a model with masses interconnected by flexible elements. The nature of the component and the stiffness properties of the corresponding modeling elements determine the minimum spacing of the mass points and the degrees of freedom associated with each point. In cases where some dynamic degrees of freedom do not contribute to the total response, static or kinematic condensation is employed in the analysis.

The normal mode approach is employed for dynamic analysis of components. Natural frequencies, eigenvectors, participation factors, and the required component dynamic responses, such as modal member-end forces and moments of the undamped structure, are calculated. The basis for combination of modal responses is discussed in Section 3.7A.2.7. Documented computer programs in the public domain are used for performing dynamic analysis. However, if proprietary computer programs are used, qualification of the programs is required.

Each of the three defined directions of earthquake input (two horizontal and one vertical taken orthogonally) are evaluated separately. The calculated results of the three analyses are combined by using the SRSS method. The selected response values (e.g., acceleration, force, stress) are combined following the guidelines of RG 1.92.

Testing

Equipment that is too complex to analyze or whose operability cannot be adequately demonstrated by analysis is qualified by dynamic testing. The equipment specification testing requirements supplement the testing methods and acceptance criteria of applicable industry standards (such as IEEE-344-1975, Section 3.10), or provide guidance for testing where no such codes are available.

The minimum acceptance criteria for equipment adequacy are:

1. No loss of function, or ability to function, before, during, or directly after completion of the proposed test.

2. No structural/electrical failure (i.e., connections and anchorages) that would compromise component integrity.

3. No adverse operation or faulty operation before, during, or after completion of the test that could result in an improper safety action.

Equipment vendors and suppliers are required to formulate programs for qualifying the equipment in accordance with the specified seismic requirements.

The base motions used to simulate the seismic loadings consist of either a single frequency or multiple frequencies and are applied either along one axis or along horizontal and vertical axes simultaneously. The choice of the input motion, i.e., frequency and axis, depends on the dynamic characteristics of the equipment and on the frequency content of the seismic loading. The criteria for selecting these specific input test motions are in accordance with IEEE-344-1975 and RG 1.100.

Exploratory tests are run to determine the response characteristics of the equipment and to aid in selecting the method of testing. The exploratory test consists of a low-level sinusoidal sweep over the frequency range of seismic loading (1 to 33 Hz). The sweep rate is 2 octaves/min or lower to excite all the resonances. If the equipment is shown to be nonresonant in the frequency range of seismic loading, it is considered a rigid body and tested accordingly. If the equipment exhibits multiple resonant response, further testing programs, based on multifrequency input, are considered more appropriate and are used in qualifying the equipment.

Multifrequency Testing Multifrequency input, applied biaxially, is the preferred method used for seismic qualification. Other methods are used as justified. Input motion for testing is applied to the vertical and one of the two principal horizontal axes simultaneously, unless it is demonstrated that the equipment response along the vertical direction is not sensitive (coupled) to the vibratory motion along the horizontal direction and vice versa. Phase-incoherent (i.e., statistically independent) inputs in the vertical and horizontal directions are use to avoid purely rectilinear motion. When the test facility limitations do not allow the use of independent inputs, two tests are performed: (1) vertical and horizontal inputs in-phase, and (2) vertical and horizontal inputs 180 deg out-of-phase. This test is repeated with the equipment rotated 90 deg in the horizontal plane. The test setup simulates as closely as possible the actual in-service installation. Equipment is tested in the mode (energized or de-energized) that reflects its design safety function. Equipment operability is verified by performing the appropriate functional cycle during and after the dynamic tests.

The basic objective of qualification or proof testing is to produce a test response spectrum (TRS) that envelops the required response spectrum (RRS). ARS, when properly broadened to account for variations in structural properties, become the RRS for qualification.

For the multifrequency input applied, the testing machine input must, as a minimum, equal the maximum floor acceleration of the RRS. The TRS is adjusted in successive test runs so that it envelops the RRS over the required frequency range. Curves for identical damping are used in comparing TRS and RRS information. Five OBE-level tests are performed prior to SSE qualification testing following the recommendations of IEEE-344-1975.

Multifrequency testing provides broad band test input motion which produces simultaneous response from all the modes of the equipment. Multifrequency motions are derived using any of the following techniques.

1. Time History An acceleration motion in the time domain, at the equipment mounting location, obtained from dynamic analysis of the structure.

2. Random Motion An electrically generated random noise signal that is selectively amplified, or attenuated, in one-third or smaller frequency band-widths. The motion resulting from this modified signal is arranged so that it envelops the TRS. This is the most commonly used input motion for multifrequency testing. The peak acceleration amplitude of this motion equals or exceeds the zero-period acceleration (ZPA) of the RRS. The random motion signal is applied for a minimum duration of 15 sec.

3. Complex Wave A complex wave is a sum of a group of decaying sinusoidal signals spaced at 1/3 octave or narrower frequency intervals over the frequency range of the RRS.

Single Frequency Testing Following the recommendations in IEEE-344-1975, single frequency input for testing is used only if one of the following conditions is met:

1. The characteristics of the required input motion indicate that the motion is dominated by one frequency (i.e., by structural filtering effects).

2. The anticipated response of the equipment is adequately represented by one mode.

The objective is to produce a TRS acceleration at the test frequency at least equal to that given by the RRS. The test table input equals or exceeds the maximum floor acceleration of the RRS. The single-frequency test consists of an exploratory test and a dwell test. In the exploratory test, the table input motion equals or exceeds the maximum floor acceleration, i.e., the ZPA of the RRS. Dwell testing is performed at the natural frequency identified during the exploratory test. The dwell test consists of applying a continuous sinusoidal input motion at the maximum floor acceleration for a minimum duration of 20 sec. Dwell testing is also performed using a sine beat input instead of a continuous sine input. A sine beat consists of a continuous sinusoid at the test frequency, amplitude modulated by a sinusoid of a lower frequency. The duration and peak amplitude of the beat for each particular test frequency are chosen to generate a magnitude of equipment response that is at least equal to that imposed by the RRS at the appropriate damping level. As a minimum, the peak amplitude of the beat should equal the rigid range acceleration of the RRS. Ten cycles per beat are used, following the recommendations of IEEE-344-1975.

Static Deflection Testing

A static deflection test consists of applying a sustained static load on critical sections of the component in such a way that the deflection caused by this load duplicates or exceeds the calculated SSE deflection. Concurrently, the component is operated in the required manner, and all applicable design loads are superimposed during the test.

3.7A.3.1.2 Seismic Qualification of Piping Systems

This is described in Section 3.7A.3.8.

3.7A.3.1.3 Other Dynamic Loads

Loading combinations and stress limits, including loads due to hydrodynamic effects, are described in Section 3.9A.3.1. A further discussion of hydrodynamic phenomena is included in the DAR (Appendix 6A).

3.7A.3.2 Determination of Number of Equivalent Stress Cycles

The following criteria are applied to all Category I subsystems:

1. A total of five OBE and one SSE are considered.

2. For subsystems, except piping, 20 cycles (full sign reversals) per seismic event, i.e., a total of 120 cycles, are considered.

3. For all piping systems, 10 maximum stress cycles per OBE (i.e., a total of 50 cycles) are postulated.

4. Where time history analysis is performed, a minimum duration of 10 sec is assumed.

3.7A.3.3 Procedure Used for Modeling

The procedure described in the following sections is specifically written for piping systems. Other subsystems are seismically qualified as described in Section 3.7A.3.1.1.

3.7A.3.3.1 Summary

Portions of piping systems that are bounded by anchors or equipment are statically and dynamically independent from the remainder of the piping. Generally, a piping system consists of several such subsystems. The analytical model and its geometric boundaries are described in detail in the following sections.

3.7A.3.3.2 Geometric Boundaries of Analytical Models

For the purpose of analysis, piping systems are subdivided into smaller units (referred to as problems) that are bounded by structural anchors (6 degree-of-freedom constraints) or by other virtually rigid points such as equipment, penetrations, and piping of much larger diameter.

A branch line with a moment of inertia of 1/10 or less of the run pipe may be ignored in the model. However, if the branch line needs to be analyzed, its model includes the effect of the run pipe. Where Category I piping is connected to nonseismic piping, the adjoining portion of the nonseismic piping up to the first anchor is included in the analytical model of the Category I piping, and all supports, up to and including this anchor, are designed seismically (Section 3.7A.3.13).

3.7A.3.3.3 Model

The basic method of analysis used is a finite element computer program (Appendix 3A). In accordance with this method, the continuous piping is mathematically idealized as an assembly of elastic structural members connecting discrete nodal points. Nodal points are placed in such a manner as to isolate particular types of piping elements such as straight runs of pipe, elbows, valves, etc., for which force-deformation characteristics can be categorized. Nodal points are also placed at all discontinuities such as piping supports, concentrated weights, branch lines, and changes in cross section. System loads such as weights, equivalent thermal forces, fluid transient dynamic forces, and inertia forces are applied at the nodal points. Stiffness characteristics of the interconnecting members are related to the effective shear area and moment of inertia of the pipe. The stiffness of piping elbows and certain branch connectors is modified to account for local deformation effects by the flexibility factors specified in ASME Section III, 1974, Subarticles NB-3600 (Safety Class 1 piping analysis) and NC-3600 (Safety Class 2 and 3 piping analysis). The increased stiffness of valve bodies is taken into consideration.

3.7A.3.3.4 Selection of Mass Points

The lumped masses are located to adequately represent the dynamic properties of the piping system. Mass points are generally selected in accordance with the following guidelines:

1. At each node where a concentrated weight is placed (valves, flanges, or other in-line piping components).

2. At each intersection where three or more piping elements are connected (branch connections, tees, and y-fittings).

3. At the end of elbows and turn of direction.

4. At nodes subjected to input of dynamic force excitation.

5. At each terminal (node where only one element is connected such as end caps, valve operators).

6. At least one mass point between two restraints acting in the same direction.

7. Unless there is a concentrated weight, a lumped mass point should not be placed at a dynamic restraint.

When these guidelines are used the number of degrees of freedom in the dynamic model is greater than twice the number of modes with frequencies less than 33 Hz.

3.7A.3.3.5 Number of Modes and Cutoff Frequency

The number of modes used in the dynamic analysis of piping depends upon the number of mass points, dynamic degrees of freedom, and cutoff frequency.

The cutoff frequencies used for different dynamic loads are as follows:

Load Type	Cutoff Frequency (Hz)
Seismic	33
SRV	100
LOCA	100
Hydraulic transients	$\geq$100

3.7A.3.4 Basis for Selection of Frequencies

3.7A.3.4.1 Components

ARS developed for the two orthogonal and vertical direction earthquakes are the basic source of seismic design accelerations. Seismic accelerations are selected from the ARS based on the natural frequency calculations of the components with proper consideration of the frequency characteristics of the component supports. Appropriate amplification factors are included in the seismic loads to insure the adequacy of the design of the components.

3.7A.3.4.2 Piping

Piping systems are generally supported in such a way that the lowest natural frequency of analytical subsystems (piping bounded by components and/or structural anchors) does not occur in the peak range of the applicable ARS. For small size Category I piping, subsystems are supported as outlined in Section 3.7A.3.8.2. The lowest natural frequency for this piping is above the applicable spectrum peak range.

3.7A.3.5 Use of Equivalent Static Load Method of Analysis

Those components that are considered relatively simple or rigid are designed, by virtue of natural frequency calculations, to withstand the effects of amplified seismic acceleration values dependent upon frequency and amplitude ranges associated with the relevant ARS. Analysis of components to the peak value of resonant response is considered conservative, since fundamental natural frequencies do not generally coincide with the frequency at resonance of the relevant response curve. Components having fundamental natural frequencies less than the cutoff frequency (Section 3.7A.3.1.1) are designed to peak acceleration values, increased by a factor of 1.3, or as justified, to account for the contribution of all significant dynamic modes under a resonant condition. Justification for the use of 1.3 as a static coefficient can be found in ASME Paper 74-NE-6[4].

The validity of the use of the 1.3 static coefficient factor is evaluated for equipment where a large number of modes of vibration within the seismic load frequency range are anticipated.

An example justifying the use of a 1.3 factor is demonstrated through a comparison of results obtained for a typical equipment. The equipment selected is an instrument panel, 2DFM-PNL102, for Unit 2. This panel was analyzed by static analysis using the 1.3 static coefficient, and also by a response spectrum modal analysis. The input response spectra used in the modal analysis were modified from the actual Unit 2 spectra, as shown on Figure

3.7A-36 (Sheets 1 and 2). This yields conservative results and allows a wide applicability of the conclusion.

Static Analysis

Reference: Technical Report No. 18609-83N-2, Specification NMP2-C062G, Acton Environmental Testing Corporation/Electro-Mechanics, Inc.

Fundamental Natural Frequency = 18.8 Hz

This frequency is less than 33 Hz; therefore, the static analysis was performed using the following accelerations:

X-1 Direction (Horizontal) = 5.51g (4.24g x 1.3)
X-2 Direction (Horizontal) = 5.51g (4.24g x 1.3)
X-3 Direction (Vertical) = 4.10g (3.15g x 1.3)

Dynamic Analysis

The response spectra modal analysis was performed using the spectra of Figure 3.7A-36 (Sheets 1 and 2). All the modes up to a frequency of 70 Hz were included in this analysis.

Results and Conclusion

The maximum responses (displacements, stresses) from the two analyses and a ratio of these responses are presented in Table 3.7A-13. The margins evident from the ratio of these responses conclude the adequacy of the 1.3 static coefficient.

3.7A.3.6 Three Components of Earthquake Motion

The maximum structural responses (displacements, acceleration, forces, and moments) due to each of the three components of earthquake motion are combined by taking the SRSS of the maximum codirectional responses, caused by each of the three components of earthquake motion at a particular point of the structure or of the mathematical model. This is in conformance with RG 1.92.

3.7A.3.7 Combination of Modal Responses

The basis for computing combined response for use in subsystem analysis is presented in Section 3.7A.2.7.

3.7A.3.8 Analytical Procedures for Piping

3.7A.3.8.1 Introduction

Piping classified as Category I is designed to withstand levels of loading imposed by the OBE and the SSE. The piping systems are classified as:

1. Those governed by the ASME Code as Safety Class 1, 2, or 3 piping.

2. Those governed by the ANSI B31.1 Code and requiring seismic analysis.

The seismic response of piping systems is analyzed by the response spectrum method or the time history method. The response spectrum method requires that seismic loading be combined from the dynamic response of the system based on an ARS, and from the response to a quasi-static differential support movement, also called seismic anchor movement, which represents the out-of-phase movement of portions of the structure to which the system is attached. Computer analysis considers all vibration modes up to at least the mode beyond which the contribution to the overall seismic dynamic response is insignificant.

The structural damping is the same for all modes of the piping system and varies only with pipe size (Section 3.7A.3.15). A response spectrum curve contains a certain damping value implicitly. In time history analysis, the damping value is an input parameter to the analysis. The number of earthquake cycles needed for fatigue analysis is given in Section 3.7A.3.2. Pipe stress analysis classifications are given in Table 3.9A-3.

All safety-related piping systems that have been seismically analyzed are reviewed to verify that engineering input information and as-installed configurations are consistent with the design requirements required by IE Bulletin 79-14. The process that governs this is a part of the design verification program for all Category I piping. The review consists of two parts: one that examines design inputs such as ARS and anchor motion, and one that compares as-built drawings against the as-analyzed calculations of record.

For large bore piping, the large bore piping as-built drawings are developed based on the installation control drawing for piping. The large bore piping as-built drawings are marked up and checked by an independent organization to show the as-installed configuration in accordance with requirements. For small bore piping, the as-built drawings are the piping design drawings marked up to show the as-installed configuration in accordance with the specification requirements. For both large and small bore pipe supports, the as-built drawings are the engineering pipe support design drawings and associated change documents which have been verified in accordance with the specification requirements. In all cases, the information is compiled by groups responsible for the final analysis where as-built, as-analyzed comparisons are performed. Either the differences in configuration or input information are justified on a case-by-case basis or the necessary changes are issued to the field. The engineering small bore piping design drawings and large bore piping as-built drawings are revised to incorporate as-built information.

The design attributes that are reviewed and the source documents that provide these attributes are provided in Table 3.7A-11 for large bore piping and Table 3.7A-12 for small bore piping. A list of applicable safety-related piping systems is provided in Table 3.2-1. Load combinations and stress criteria are described in Section 3.9A.1.5.

The final documentation of this program occurs at the time of N-5 signoff, when a review is conducted to ensure that all input information is still valid and that any revisions that have taken place do not change the basis for the final analysis.

3.7A.3.8.2 Analytical Techniques

General Criteria

Piping systems are rigidly supported, where possible, to assure a first mode natural frequency above the peak frequency after peak spreading.

Qualification of Small Size Piping

The scope of small size piping is limited to:

1. ASME Safety Class 1 piping of 1-in NPS and smaller, which can be analyzed by Safety Class 2 rules in accordance with Subsubarticle NB-3630.

2. ASME Safety Class 2 and 3 piping of 6-in NPS and smaller.

3. ANSI B31.1 piping (Safety Class 4 piping) of 6-in NPS and smaller.

In general, the analysis of small-size ASME Safety Class 1 piping (1-in NPS and smaller) and ASME Safety Class 2 and 3 piping (6-in NPS and smaller) is performed by means of simplified seismic analysis without computer application.

For ASME Safety Class 1 piping (1-in NPS and smaller) and ASME Safety Class 2 and 3 piping (2-in NPS and smaller), the maximum support spans are determined by limiting the stresses to within the code allowables.

For ASME Safety Class 2 and 3 piping with 2 1/2- to 6-in NPS, the support spacing is selected so that the fundamental frequency (fp) of the piping section will always be beyond the resonant frequency of the structure, as determined from applicable seismic ARS. The peak of the floor response spectra for design of piping supported between two points is used for the simplified seismic analysis. Deadload and thermal responses are also calculated in accordance with Subarticle NC-3600.

The simplified analytical approach is to perform stress calculations for consecutive sections of piping, bounded at support points, without using computer application. This is justifiable because a rigid system with sufficient pipe supports represents many one-dimensional, straight-beam problems wherein the coupling effects of the three-dimensional piping systems are eliminated. Constraints are placed near elbows, tees, and concentrated masses, such as valves, so that coupling effects are negligible. These calculations of maximum combined stresses provide sufficient and conservative data to satisfy the requirement of Subarticle NC-3600.

3.7A.3.8.3 Dynamic Analysis

Model

The modeling procedure, including the selection of mass points and the adequacy of number of degrees of freedom, is described in Section 3.7A.3.3.

Response Spectrum Method

When a piping system is analyzed by means of the response spectrum method, one of the piping analysis computer programs described in Appendix 3A is used to calculate the modal response at each node point in the piping system due to the ARS excitation applied to the system. Generation or selection of the appropriate set of ARS for a subsystem supported at different elevations, and consideration of the effect of seismic differential displacements between restraints, are discussed in Section 3.7A.3.9. The damping values for piping depend on pipe size and are given in Table 3.7A-1. The equations of motion and their solution are as previously described for structures (Section 3.7.2).

Time History Method

The applicable base motion time history is the structural response at a representative mass point of the structure to the ground motion time history. The equations of motion and their solution are the same as in Section 3.7A.2.5, but the scalar acceleration term in the excitation function is now the amplitude of the acceleration of the base of the subsystem (points of attachment), not of the ground. The effect of parameter variations on the floor response spectra are taken into account (Section 3.7A.2.9).

Dynamic Analysis Formulation

The basic equations of motion and their solutions are the same as for structures (Section 3.7.2). Absolute accelerations at points on the piping system are sometimes needed for qualification of critical, safety-related equipment. With the response spectrum method, the maximum absolute acceleration at a mass point in mode i is obtained from Newton's law by dividing the effective inertia force by the mass at the mass point:

$$\{a\} = [M]^{-1} \{Q\} \tag{3.7A-6}$$

Where:

$[M]$ = Diagonal mass matrix of the system

$\{Q\}$ = Effective inertia forces in mode i

With the time history method, the absolute accelerations are obtained by adding the base acceleration to the relative accelerations of the mass points.

Seismic Differential Displacements

Description of Input The seismic differential displacements are also called seismic anchor movements. This effect is analyzed in a separate static load case for OBE anchor movements. The anchor movements are obtained from the seismic differential displacements of the structural nodes.

The displacements are obtained in the following form, one set for each mass point, N, of the building model:

	Earthquake Direction		
Mass Node	X (East-West)	Y (Vertical)	Z (North-South)
1	D_{1_X}	D_{1_Y}	D_{1_Z}
2	D_{2_X}	D_{2_Y}	D_{2_Z}
3	D_{3_X}	D_{3_Y}	D_{3_Z}
	⋮	⋮	⋮
N	D_{N_X}	D_{N_Y}	D_{N_Z}

Where:

$D_{N_X} (D_{N_Y}, D_{N_Z})$ = SRSS of displacements in X (Y, Z) direction due to earthquake excitation in X, Y, and Z directions at Node N

These are the movements of points on the walls relative to the foundation of the building. Relative displacements between mass points are used to determine the movements of support points. Support displacements are imposed on the system in a conservative manner.

Combination of Anchor Movement Loads The individual X, Y, and Z anchor movement components of OBE are analyzed as three separate static anchor movement load cases. These load cases are then combined by the SRSS method and the resultant load case is used in the code stress evaluation. Anchor movement load cases are analyzed using one of the piping analysis computer programs described in Appendix 3A.

Combined Seismic Response

The system response to the response-spectrum excitation (i.e., displacements, internal forces and moments, stresses, and support reactions) is obtained by first combining the modal contributions for each earthquake component. In conformance with RG 1.92, the effect of closely-spaced modes is taken into account by the procedure described in Section 3.7A.2.7. The contributions of each of the three components are then combined by the SRSS methods.

When the response spectrum method is used, response to the differential support motion is considered. In Safety Class 1 piping analysis, this motion is combined with the inertial response; the result is then combined with other load cases. In the analysis of other piping classes, the seismic anchor movement is combined with secondary loads. Seismic load cases are combined with other load cases (thermal, weight, pressure, other occasional loads) in accordance with ASME Section III, 1974. The load combinations are given in Section 3.9A.3.1.

Fatigue Considerations

For ASME Safety Class 1 piping, if Equation (10) of Subparagraph NB-3653.1 is not satisfied, a fatigue analysis is performed in accordance with Subparagraph NB-3653.2 or Subarticle NB-3200. This analysis uses the total number of stress cycles of all OBEs. The number of earthquake cycles is discussed in Section 3.7A.3.2.

Computer Programs Used for Seismic Analysis

All analyses are performed using one of the piping analysis computer programs described in Appendix 3A. These programs handle response spectrum and support motion time history analyses. Programs used for generating ARS input curves to a piping analysis are also described in Appendix 3A.

Development of Relative Displacements and Their Application to Piping Analysis

The relative displacement between a point on the reactor building and a point on the primary containment is obtained as follows:

1. The displacement time history at each point is calculated using as input the three-directional synthetic time history of ground motion.

2. The displacement time history of the first point is subtracted from that of the second to get a time history of relative displacement.

Some examples of displacements are included on Figure 3.7A-35.

The effect of relative displacement between the supports is considered in the piping analysis.

One support is selected as a reference point. Dynamic displacements at this reference point are taken either as zero or as displacements relative to a given structural node. At the first (adjacent) support from the reference point, relative displacements between this support and the reference point are taken and added to the dynamic displacement of the reference point (the direction of displacements is determined and applied to the supports in relation to its function). The resultant displacements become the dynamic displacements of the first support. The first support then becomes the new reference point. This process is repeated until dynamic displacements at all supports are obtained.

3.7A.3.9 Multiply Supported Equipment Components with Distinct Inputs

When a subsystem is attached to different parts of a structure, such as separate elevations on one wall or several walls, the response spectra of all structural nodes for which response spectra exist and which lie nearest to the support elevation at the subsystem, both below and above the support elevation, are enveloped, and this envelope spectrum is applied to the subsystem. In cases where a subsystem runs between two different buildings, a single ARS enveloping the spectra associated with all support points is used.

In conjunction with the response spectrum loading, the loading from differential support displacements is calculated, and the two load cases are combined as described in Section 3.9A.3.1. Components and equipment generally have localized supports and the effect can be ignored. The application to piping is discussed in detail in Section 3.7A.3.8.3.

3.7A.3.10 Use of Constant Vertical Static Factors

Constant vertical static factors are not used.

3.7A.3.11 Torsional Effects of Eccentric Masses

For Category I piping systems, concentrated loads in the piping system, such as valves and valve operators, are modeled as massless members with the mass of the components lumped at the center of gravity. A rigid member is modeled connecting the center of gravity to the piping so that the torsional effects of the eccentric masses are considered. The stress produced at the pipe connection is given in the NUPIPE output.

3.7A.3.12 Buried Category I Piping Systems and Tunnels

There are no buried Category I piping systems at Unit 2. The design of Category I tunnels is performed in one of two ways.

1. Tunnels which are physically connected to other Category I structures are founded on rock and are analyzed and designed for seismic events as parts of these structures.

2. Tunnels which are physically separated from other structures are founded on rock (including buried Category I tunnels) and are analyzed using the methods of Section 3.7 for seismic events or are conservatively designed for acceleration values of 0.5g for SSE and 0.25g for OBE. A typical dynamic analysis was performed to demonstrate and verify that the acceleration values chosen were conservative.

Partially or completely buried tunnels are also designed to consider the static and dynamic effects of the backfill using the pressure distribution shown on Figure 2.5-110.

The design of Category I tunnels is described in Section 3.8.4.4.7.

3.7A.3.13 Interaction of Other Piping with Category I Piping

In order to prevent propagation of failure from the seismically induced effect of nonseismic class piping on Category I piping, each nonseismic class piping system is generally designed to be isolated from any Category I piping system. If it is not feasible or practical to isolate the Category I piping system from the nonseismic class piping system, then adjacent nonseismic class piping is seismically designed according to the same criteria that are applicable to the Category I piping system. For the nonseismic class piping systems interfacing with Category I piping systems, the seismic analysis encompasses the nonseismic class system to the first anchor point.

Isolation anchors that separate QA Category I from nonsafety-related piping are designed considering seismic loads from both sides of the anchor. If the nonsafety-related side has not been seismically analyzed, the seismic load from that side of the anchor is assumed to be three times the seismic load from the QA Category I side. The Engineer will evaluate the piping and support design of the nonsafety-related piping to ensure that the seismic load will not exceed the assumed loads. In the event that the assumption cannot be justified, the anchor will be designed to sustain the maximum moment which the nonsafety-related piping can impose on the anchor.

The loading conditions and load combinations are in accordance with Table 3.9A-14. Allowable stresses designated in the AISC specification are used for the design of structural members and welds. When the maximum moment is used, the allowable stresses in the structural member will be increased to 0.9Sy.

Allowable stresses for induced local pipe wall stresses are in accordance with ASME III, Subsection NC, 1974. When the maximum moment is used, the allowable local stress will be increased to 2Sy.

3.7A.3.14 Seismic Analyses for Reactor Internals

See Section 3.7B.3.14.

3.7A.3.15 Analysis Procedure for Damping

The percentages of critical damping values assigned to Category I subsystems and components are in accordance with RG 1.61 (Table 3.7A-1).

In the dynamic analysis of any particular item of equipment, the same percentage of critical damping is used for all modal responses. In cases where pipe size dictates the use of two sets of damping values for the same analysis, the damping values corresponding to the pipe size of the majority of the system are used for the entire analysis.

3.7A.4 Criteria for Seismic Instrumentation Program

3.7A.4.1 Comparison with Regulatory Guide 1.12

A seismic instrumentation program has been implemented to monitor and record input motion and behavior of the plant in the event of an earthquake. This instrumentation program complies with the requirements of RG 1.12 and ANSI Standard ANSI/ANS-2.2-1978.

3.7A.4.2 Location and Description of Instrumentation

3.7A.4.2.1 Triaxial Time History Accelerograph

Strong motion triaxial time history accelerometers are installed in three locations. Two accelerometer sensor packages are installed in the reactor building outside of the primary containment. One sensor package is located on the reactor mat in the secondary containment adjacent to the exterior reactor building wall at el 175 ft, and the second is located approximately 178 ft above the first on the refueling floor at el 353 ft. The third sensor package is located on top of the control building mat at el 214 ft.

The strong motion triaxial time history accelerograph has the following physical characteristics:

1. Accelerometers are of the force-balance type, with the capability of recording a maximum of 1.0g at full scale.

2. Accelerometers are sensitive to frequencies in the range of 0 to 50 Hz.

3. The seismic instrumentation and recording system is in a quiescent state until activated by seismic triggers set at 0.01g. These seismic triggers (both horizontal and vertical) activate the recording system in less than 100 msec. The recording system will operate continuously during the period in which the earthquake exceeds the 0.01g threshold, plus 10 sec minimum beyond the last seismic trigger signal. The system will be capable of a minimum of 30 min total recording time.

4. Each sensor package contains three orthogonal accelerometers. All strong motion sensor packages are oriented to the same azimuths.

3.7A.4.2.2 Triaxial Peak Accelerograph

Also in accordance with RG 1.12, triaxial peak accelerographs are installed on other selected Category I structures, equipment, and components to verify the seismic response determined analytically by using the traces recorded by the accelerographs. The accelerographs are located as follows:

1. Reactor pedestal.

2. Reactor building high-pressure core spray (CSH) piping.

3. Service water piping in diesel generator building.

These instruments detect and record peak amplitudes of accelerations in a minimum frequency range of 0 to 26 Hz.

Triaxial peak accelerographs have the following physical characteristics:

1. Accelerographs are of the short period torsional type, with a sensitivity from 0.01 to 10g full scale.

2. The accelerograph records by scribing excursions of a diamond stylus on a replaceable metal plate.

3. No power is required to operate the instrument.

4. Air damping is used to 60 percent of critical with an accuracy of ±5 percent of critical.

5. Operating temperature range is -40°F to 185°F.

3.7A.4.2.3 Triaxial Response Spectrum Recorder

The triaxial response spectrum recorder senses and permanently records information defining a response spectrum. It is a completely passive device, covering the range from 1 to 32 Hz in 1/3 octave increments. Sixteen reeds of different lengths and weights, 1 for each frequency, have attached to their free end a diamond-tipped stylus that inscribes a permanent record of its deflection on 16 respective record plates. A calibration sheet lists the resonant frequency and g-sensitivity of each reed and allows a plot of acceleration versus frequency to be made. These instruments have the following physical characteristics:

1. Damping for the oscillators is not less than 2 percent nor more than 5 percent of critical.

2. Operating temperature range is -40°F to 185°F.

3. Accuracy: frequency is ±1 percent; acceleration is ±3 percent of full scale; damping is ±0.15 percent of nominal.

4. The dynamic range for acceleration is 100 to 1 minimum.

Four triaxial response spectrum recorders are installed at the following locations:

1. Reactor building mat (el 175 ft).

2. Control building mat (el 214 ft).

3. Refueling floor (el 353 ft 10 in).

4. Primary containment wall penetration for RHR piping (el 294 ft 6 in).

The basis for selection of these locations is to provide some measure of redundancy to the strong motion accelerographs and also to provide additional data to verify the seismic response determined analytically by using the traces recorded by the strong motion accelerographs.

3.7A.4.2.4 Triaxial Seismic Switch

One triaxial seismic switch is installed on the reactor building mat to provide an immediate signal to the main control room to indicate if specified design accelerations (OBE) have been exceeded.

The seismic switch has the following physical characteristics:

1. The package is composed of three orthogonal acceleration transducers.

2. Setpoint is adjustable from 0.025 to 0.25g.

3. Switch remains closed for 6 to 20 sec (adjustable) after detection of an acceleration over the preset value.

4. Operating temperature range is from 0°F to 130°F.

3.7A.4.3 Main Control Room Operator Notification

Recording equipment, seismic annunciators, and response spectrum annunciators are located in the relay and computer room, el 288. The annunciators have both visual and audible alarms to notify the Control Room Operator when a seismic event causes the OBE design accelerations to be exceeded.

The seismic trigger on the reactor building mat activates the triaxial accelerograph and the magnetic tape recording system when the acceleration exceeds 0.01g.

The response spectrum recorder on the reactor building mat activates the response spectrum annunciator when preset g levels at corresponding frequencies are exceeded.

The triaxial seismic switch on the reactor building mat sets off the seismic annunciator when the acceleration exceeds the OBE limits of 0.075 g horizontal and 0.050 g vertical.

The seismic trigger and, hence, the triaxial accelerograph and the magnetic tape recording system will be activated by accelerations from nonseismic events such as SRV blowdown loads and hydrodynamic loads due to LOCA events. However, since the normal operating SRV acceleration level (i.e., the acceleration value corresponding to zero period in the reactor building mat design response spectra, as defined in RG 1.12) is less than the OBE acceleration level, the seismic switch will not be activated by the normal operating nonseismic events. Therefore, these events will not interfere with normal plant operations.

3.7A.4.4 Comparison of Measured and Predicted Responses

To determine if a nuclear power plant can continue to operate safely following a seismic event, comparisons are made between the seismic response as measured by the seismic instrumentation and the computed response used as a design basis. Such comparisons are performed only after OBE or more severe seismic conditions occur. To make such comparisons, the following procedure is implemented:

1. Magnetic tape records are digitized and corrected for time signal variations and baseline deviations.

2. Time-history records from triaxial sensors located on the reactor building mat are used to directly calculate ARS at appropriate critical damping values.

3. Time-history records from the reactor building mat sensor are used as input ground motion for the reactor building dynamic model. ARS are then calculated at the locations of the other two sensors in the containment structure for comparison and correlation with the response spectra determined in Item 2. Reasonable correlation between the spectra is accomplished on an iterative basis by varying the physical properties of the models (stiffnesses and damping characteristics) to calibrate the dynamic model. Once the dynamic model has been calibrated, additional verification of its correctness is made using the acceleration readings from the peak recording accelerographs.

The results of the comparison will be used to evaluate the seismic effects on the structures and equipment by forming the basis for remodeling, detailed reanalysis, and physical inspection. The reanalysis and inspection results will be used to determine the appropriate actions that are required as a result of the earthquake.

3.7A.4.5 In-service Surveillance Requirements

In-service surveillance will be performed on the seismic instrumentation at the intervals specified in the Technical Specifications.

3.7A.5 References

1. Newmark, N. M., Blume, J. A., and Kapur, K. K. Design Response Spectra for Nuclear Power Plants, Journal of Power Division, Proceedings of American Society of Civil Engineers, Vol. 99, No. P02, November 1973.

2. Roesset, J. M., Whitman, R. V., and Dobry, R. Modal Analysis for Structures with Foundation Interaction. Journal of the Structural Division, Proceedings ASCE, pp 399-416, March 1973.

3. Whitman, R. V. Soil Structure Interaction. Seismic Design for Nuclear Plants. MIT Press, Cambridge, MA, pp 241-269, 1970.

4. Gwinn, J. M. and Goldstein, N. A. Equivalent Static Loads from Amplified Response Curves. ASME Paper 74-NE-6, presented at the American Society of Mechanical Engineers Pressure Vessel and Piping Conference, Miami, FL, June 1974.

5. ASCE Structural Division Committee on Nuclear Structures and Materials, Seismic Response of Buried Pipes and Structural Components, Report Prepared by Committee on Seismic Analysis.

TABLE 3.7A-1

DAMPING FACTORS

Item, Equipment, or Structure	Damping (Percent Critical) OBE	SSE
Equipment and large diameter piping systems, pipe diameter greater than 12 in	2	3
Small diameter piping systems, diameter less than or equal to 12 in	1	2
Welded steel structures	2	4
Bolted steel structures	4	7
Prestressed concrete structures	2	5
Reinforced concrete structures	4	7

TABLE 3.7A-11

DESIGN ATTRIBUTES TO BE REVIEWED FOR VERIFICATION OF LARGE BORE SAFETY-RELATED PIPING AS REQUIRED BY IE BULLETIN 79-14

Design Attribute	Source Document
Pipe run configuration and geometry	L/B as-built drawing
Pipe support location	L/B as-built drawing
Valve location	L/B as-built drawing
Support design and function	Engineering design drawing
Embedment plate, base plate, and structural steel	Engineering design drawing and pipe support installation Specification No. NMP2-P301J
Pipe clearance at supports	Engineering design drawing
Other pipe clearances	Construction Site Instructions CSI 2.11
Attachments to pipe	L/B as-built drawing and engineering design drawing
Attachments to supports	Engineering design drawing
Valve weights	Vendor drawing
Amplified response spectra	Piping Design Specification No. NMP2-P301A
Support seismic anchor motion	Piping Design Specification No. NMP2-P301A
Other design attributes	Piping Design Specification No. NMP2-P301A

TABLE 3.7A-12

DESIGN ATTRIBUTES TO BE REVIEWED FOR VERIFICATION OF SMALL BORE SAFETY-RELATED PIPING AS REQUIRED BY IE BULLETIN 79-14

Design Attribute	Source Document
Pipe run configuration and geometry	Engineering design drawing
Pipe support location	Engineering design drawing
Valve location	Engineering design drawing
Support design and function	Engineering design drawing
Embedment plate	Engineering design drawing and S/B pipe support installation Specification No. NMP2-P301F
Pipe clearance at supports	Engineering design drawing
Other pipe clearances	Construction Site Instructions CSI 2.11
Attachments to pipe	Engineering design drawing
Attachments to supports	Engineering design drawing
Valve weights	Vendor drawing
Amplified response spectra	Piping Design Specification No. NMP2-P301A
Support seismic anchor motion	Piping Design Specification No. NMP2-P301A
Other design attributes	Piping Design Specification No. NMP2-P301A

3.7B SEISMIC DESIGN (GE SCOPE OF SUPPLY)

As discussed at the beginning of Section 3.7, the following input is applicable to the design of systems, components and equipment within the NSSS scope of supply by GE.

All systems, components and equipment of the NSSS are defined as either Category I or non-Category I. The requirements for Category I identification are given in Section 3.2 along with a list of systems, components, and equipment that are so identified.

All systems, components and equipment important to plant safety are designed to withstand potential earthquakes defined as follows. The SSE is an earthquake based upon an evaluation of the maximum earthquake potential considering the regional and local geology, seismology, and specific characteristics of local subsurface material. This earthquake produces the maximum vibratory ground motion for which Category I systems and components are designed to remain functional. These systems and components are those necessary to ensure:

1. The integrity of the RCPB.

2. The capability to shut down the reactor and maintain it in a safe shutdown condition.

3. The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines exposures of 10CFR100.

The OBE is the earthquake which, considering the regional and local geology, seismology, and specific characteristics of local subsurface material, could reasonably be expected to affect the plant site during the operating life of the plant. This earthquake produces the vibratory ground motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public are designed to remain functional.

3.7B.1 Seismic Input

3.7B.1.1 Design Response Spectra

See Section 3.7A.1.1.

3.7B.1.2 Design Time History

See Section 3.7A.1.2.

3.7B.1.3 Critical Damping Values

The damping factors indicated in Table 3.7B-1 are used in the response analysis of various systems, components, and equipment, and in preparation of floor response spectra used as forcing inputs for piping and equipment analysis or testing. These factors are in compliance with RG 1.61.

3.7B.1.4 Supporting Media for Category I Structures

See Section 3.7A.1.4.

3.7B.2 Seismic System Analysis

3.7B.2.1 Seismic Analysis Methods

Analysis of Category I GE-supplied systems and components is accomplished, where applicable, using the response spectrum or time history approach. Both utilize the natural period, mode shapes, and appropriate damping factors of the particular system. Certain pieces of equipment having very high natural frequencies may be analyzed statically if the fundamental frequency of the component is greater than the ZPA frequency of the excitation. In some cases, dynamic testing of equipment may be used for seismic qualification.

The time history analyses involve the solution of the equations of dynamic equilibrium (Section 3.7B.2.1.1) by means of the methods discussed in Section 3.7B.2.1.2. In this case, the duration of motion is of sufficient length to ensure that the maximum values of response have been obtained.

A response spectrum analysis involves the solution of the equations of motion (Section 3.7B.2.1.1) by the method discussed in Section 3.7B.2.1.3. The total seismic structural response will be predicted by combining the response calculated from the two horizontal and the one vertical analyses. When the response spectrum is used, the methods for combining the loads from the three analyses will be based on the method described in Section 3.7B.3.6.

3.7B.2.1.1 The Equations of Dynamic Equilibrium

Assuming velocity-proportional damping, the dynamic equilibrium equations for a lumped mass, distributed stiffness system are expressed in matrix form as:

$$[M][\ddot{u}(t)] + [C][\dot{u}(t)] + [K][u(t)] = 0 \qquad (3.7B\text{-}1)$$

$$u_t(t) = u(t) + u_s(t) \qquad (3.7B\text{-}2)$$

$$[M][\ddot{u}(t)] + [C][\dot{u}(t)] + [K][u(t)] = [P(t)] \qquad (3.7B\text{-}3)$$

Where:

$[M]$ = Lumped mass matrix (nxn)

$[\ddot{u}(t)]$ = Time-dependent acceleration vector (1xn) of nonsupport points relative to the base support

$[C]$ = Damping matrix (nxn)

$[\dot{u}(t)]$ = Time-dependent velocity vector (1xn) of nonsupport points relative to the base support

$[K]$ = Stiffness matrix (nxn)

$[u(t)]$ = Time-dependent displacement vector (1xn) of nonsupport points relative to the base support

$$[u_t(t)] = [u(t)+u_s(t)] \qquad (3.7B\text{-}2)$$

$[P(t)]$ = Time-dependent inertial force vector $(-[M]\ [\ddot{u}_s(t)])$ acting at nonsupport points (1xn)

The manner in which a distributed mass, distributed stiffness system is idealized into a lumped mass distributed stiffness system representation of the NSSS component is shown on Figure 3.7B-1, along with a schematic representation of relative acceleration $[\ddot{u}(t)]$, support acceleration $[\ddot{u}_s(t)]$, and total acceleration $[\ddot{u}_t(t)]$.

3.7B.2.1.2 Solution of the Equations of Motion by Mode Superposition

The technique used for the solution of the equations of motion is the method of mode superposition in which the equations of motion are decoupled by the eigen transform.

The set of homogeneous equations represented by the undamped free vibration of the system is:

$$[M][\ddot{u}(t)] + [K][u(t)] = [0] \qquad (3.7B\text{-}4)$$

Since the free oscillations are assumed to be harmonic, the displacement vector $[u(t)]$ can be written:

$$[u(t)] = [\phi]\ e^{i\omega t} \qquad (3.7B\text{-}5)$$

Where:

$[\phi]$ = Column matrix of the amplitude of displacements $[u]$

ω = Circular frequency of oscillation

t = Time

$i = \sqrt{-1}$

Substituting Equation 3.7B-5 and its derivatives in Equation 3.7B-4 and noting that $e^{i\omega t}$ is unequal to zero for all values of ωt yields:

$$[-\omega^2 [M] + [K]] [\phi] = [0] \qquad (3.7B\text{-}6)$$

Equation 3.7B-6 is the characteristic equation for the classical eigenvalue problem in which the eigenvalues are the frequencies of vibrations ω_i, and the eigenvectors are the mode shapes, $[\phi_i]$, $(i = 1,2,....,n)$.

For each frequency ω_i, there is a corresponding solution vector $[\phi_i]$. It can be shown that the eigenvectors are orthogonal with respect to the weighted stiffness matrix [K] in the n-dimensional vector space. The eigenvectors are also orthogonal with respect to the weighted mass matrix [M].

The orthogonality of the eigenvectors is used to effect a coordinate transformation to the generalized coordinate system in which the governing equations of motion are decoupled. Thus, the problem becomes one of solving n independent differential equations rather than n simultaneous differential equations, and since the system is linear, the principle of superposition holds and the total response of the system oscillating simultaneously in n modes is determined by direct addition of the responses in the individual modes.

3.7B.2.1.3 Analysis by Response Spectrum Method

The response spectrum method is based on the fact that the modal responses can be expressed in terms of a set of convolution integrals that satisfy the governing differential equations. The advantage of this form of solution is that for a given ground motion the only variables under the integral are damping factor and frequency. Thus, for a specified damping factor, it is possible to construct a curve that gives a maximum value of the integral as a function of frequency. This curve is called a response spectrum for the particular input motion and the specified damping factor. The integral has units of velocity; consequently, the maximum of the integral is called the spectral velocity.

Using the calculated natural frequencies of vibration of the system, the maximum values of the modal responses are determined directly from the appropriate response spectrum. The modal maxima are then combined as discussed in Section 3.7B.3.7.

When the equipment is supported at more than two points located at different elevations in the building, response spectrum analysis is performed using the envelope response spectrum of all attachment points. In some cases, the worst single floor response spectrum selected from a set of floor response spectra obtained at various floors may be applied identically to all floors provided there is no significant shift in frequencies of the spectra peaks.

Alternatively, multiple support excitation analysis methods may be used where individual acceleration time-histories or response spectra are applied at all the equipment attachment points.

3.7B.2.1.4 Multisupport Excitation Analysis of Systems, Components, and Equipment

Analytical procedures for obtaining force and displacement responses engendered by time-dependent, base support excitation are discussed in preceding sections. In a multisupported system, the relative motion among the individual multisupport points gives rise to time-varying displacements at the nonsupport points.

The governing equations of motion of a multisupported system, component, or equipment undergoing individual multisupport excitations may be expressed in the following matrix form:

$$[M]\{\ddot{u}\} + [C]\{\dot{u}\} + [K]\{u\} = \{F\} \qquad (3.7B\text{-}7)$$

Where:

$\{u\} = \{u(t)\}$ = The corresponding dynamic model nodal displacement vector of absolute displacements

The general case is considered in which k of the total n degrees-of-freedom corresponds to the individual multisupport points which undergo known time-history motions. The nodal displacement vector of absolute displacements can be partitioned and written as:

$$\{u\} = \left\{\frac{u_a}{\bar{u}_s}\right\} = \left\{\frac{u_a^{\,d} + u_a^{\,s}}{\bar{u}_s}\right\} \qquad (3.7B\text{-}8)$$

Where:

$\{u_a\}$ = Absolute displacement vector of the active (unsupported) degrees-of-freedom

$\{\bar{u}_s\}$ = Known absolute displacement vector corresponding to the multisupported degrees-of-freedom

The vector $\{u_a\}$ in Equation 3.7B-8 has been further separated into a dynamic part and a pseudo-static part where:

$\{u^d_a\}$ = Dynamic part of $\{u_a\}$

$\{u^s_a\}$ = Pseudo-static part of $\{u_a\}$

Multisupport excitation may require the utilization of all modes which span the $\{u_a\}$ space of active (unsupported) degrees-of-freedom in the modal superposition in order to obtain reliable solutions of Equation 3.7B-7. Substitution Equation 3.7B-8 enables the circumventing of that very costly requirement. Only the dynamic part $\{u^d_a\}$ is obtained by modal superposition which does not require all modes. The pseudo-static part $\{u^s_a\}$ is obtained from the known multisupport excitation.

The partitioned equations of motion are obtained by substituting Equation 3.7B-8 into Equation 3.7B-7 to yield:

$$\left[\begin{array}{c|c} M_a & O \\ \hline O & M_s \end{array}\right] \left\{\begin{array}{c} \ddot{u}_a^d + \ddot{u}_a^s \\ \hline \ddot{u}_s \end{array}\right\} + \left[\begin{array}{c|c} C_{aa} & C_{as} \\ \hline C_{aa} & C_{ss} \end{array}\right] \left\{\begin{array}{c} \dot{u}_a^d + \dot{u}_a^s \\ \hline \dot{u}_s \end{array}\right\} + \left[\begin{array}{c|c} K_{aa} & K_{as} \\ \hline K_{sa} & K_{ss} \end{array}\right] \left\{\begin{array}{c} u_a^d + u_a^s \\ \hline U_s \end{array}\right\} = \left\{\begin{array}{c} F_a \\ \hline F_s \end{array}\right\}$$

(3.7B-9)

Where:

$\{u^d_a\}$ = Dynamic part (as defined by Equation 3.7B-8) of the absolute displacement vector of the active (unsupported) degrees-of-freedom

$\{u^s_a\}$ = Pseudo-static part (as defined by Equation 3.7B-8) of the absolute displacement vector of the active (unsupported) degrees-of-freedom

$[M_a]$ and $[M_s]$ = Lumped diagonal mass matrices associated with the active degrees-of-freedom and the multisupport points, respectively

$[C_{aa}]$ and $[K_{aa}]$ = Damping matrix and elastic stiffness matrix, respectively, relating the forces developed in the active degrees-of-freedom to the motion of the active degrees-of-freedom

$[C_{ss}]$ and $[K_{ss}]$ = Support forces due to unit velocities and displacements, respectively, of the multisupport points

$[C_{as}]$ and $[K_{as}]$	=	Damping and stiffness matrices denoting the coupling forces developed in the active degrees-of-freedom due to the motion of the supports, or vice versa
$\{F_a\}$	=	Prescribed time-dependent applied load vector corresponding to the active degrees-of-freedom
$\{\overline{F}_s\}$	=	Reaction force vector corresponding to the system multisupport points
$(\cdot)$	=	Total differentiation with respect to time when appearing over a time variable

The procedure utilized to construct the damping matrix is discussed in Section 3.7B.2.15. The mass matrix and elastic stiffness matrix are formulated by standard procedures.

Since the components of $\{\ddot{\bar{u}}_s\}$, hence of $\{\dot{\bar{u}}\}$ and $\{\bar{u}_s\}$, are known functions of time, only the first partitioned portion of Equation 3.7B-9 is of interest.

$$[M_a]\{\ddot{u}_a^d\} + [M_a][\ddot{u}_a^s\} + [C_{aa}]\{\dot{u}_a^d\} + [C_{aa}]\{\dot{u}_a^s\} + [C_{as}]\{\dot{\bar{U}}_s\} + [K_{aa}]\{u_a^d\} + [K_{ag}]\{u_a^s\} + [K_{as}]\{\bar{U}_s\} = \{F_a\}$$

(3.7B-10)

The pseudo-static displacement vector is written in terms of the multisupport displacement vector by taking:

$$[K_{ag}]\{u^s_a\} + [K_{as}]\{\bar{u}_s\} = \{0\} \quad (3.7B\text{-}11)$$

Therefore:

$$\{u^s_a\} = -[K_{aa}]^{-1}[K_{as}]\{\bar{u}_s\} \quad (3.7B\text{-}12)$$

It follows from Equation 3.7B-11 that:

$$[C_{aa}]\{\dot{u}^s_a\} + [C_{as}]\{\dot{\bar{u}}_s\} = \{0\} \quad (3.7B\text{-}13)$$

The partitioned portion of the equation is reduced to its final form by substituting Equations 3.7B-11, 3.7B-12, and 3.7B-13 into Equation 3.7B-10 to yield:

$$[M_a]\{\ddot{U}^d_a\} + [C_{aa}]\{\ddot{U}^d_a\} + [K_{ag}]\{u^d_a\} = \{F_a\} + [M_a][K_{aa}]^{-1}[K_{as}]\{\ddot{\bar{u}}_s\} \quad (3.7B\text{-}14)$$

The solution in time of Equation 3.7B-14 for $\{u^d_a\}$ is readily obtained by the standard normal mode solution methodology. Once $\{u^d_a\}$ is obtained, the total solution for the absolute displacement vector $\{u_a\}$, corresponding to the active degrees-of-freedom, is given by substituting $\{u^d_a\}$ from Equation 3.7B-14 and $\{u^s_a\}$ from Equation 3.7B-12 into Equation 3.7B-8.

After obtaining the absolute displacement vector response of the active degrees-of-freedom, $\{u_a\}$, the second partitioned portion of Equation 3.7B-9 can be used to calculate the reaction force vector $\{F_s\}$ corresponding to the multisupport degrees-of-freedom; i.e.:

$$\{\overline{F}_s\} = [M_s]\{\ddot{u}_s\} + [C_{sa}]\{\dot{u}_a\} + [C_{ss}]\{\dot{\bar{u}}_s\} + [K_{sa}]\{u_a\} + [K_{ss}]\{\bar{u}_s\} \qquad (3.7B\text{-}15)$$

Note that $\{\overline{F}_s\}$ is the total external force vector applied to the multisupport degrees-of-freedom required to produce the given multisupport excitation $\{\ddot{u}_s\}$. The interaction force vector $\{F_s\}$ corresponding to the reaction of the active degrees-of-freedom portion of the dynamic model on the multisupport points is given by:

$$\{F_s\} = [C_{sa}]\{\dot{u}_a\} + [K_{sa}]\{u_a\} \qquad (3.7B\text{-}16)$$

The interaction force vector $\{F_s\}$ can also be expressed in terms of the multisupport excitation input motion $\{\bar{u}_s\}$ by substituting Equations 3.7B-12 and 3.7B-8 into Equation 3.7B-16 to yield:

$$\{F_s\} = [C_{sa}]\{\dot{u}^d_a\} + [K_{sa}]\{u^d_a\} - [C_{sa}][K_{aa}]^{-1}[K_{as}]\{\dot{\bar{u}}_s\} - [K_{sa}][K_{aa}]^{-1}[K_{as}]\{\bar{u}_s\} \qquad (3.7B\text{-}17)$$

3.7B.2.1.5 Dynamic Analysis of Category I Systems, Components, and Equipment

Time-history and response spectrum techniques are used as applicable for the dynamic analysis of Category I systems, components, and equipment that are sensitive to dynamic seismic events.

Dynamic Analysis of Piping Systems

Each pipe line is idealized as a mathematical model consisting of lumped masses connected by elastic members. The stiffness matrix for the piping system is determined using the elastic properties of the pipe. This includes the effects of torsion, bending, shear, and axial deformations as well as change in stiffness due to curved members. Next the mode shapes and the undamped natural frequencies are obtained. When the piping system is anchored and supported at points with different excitations, the response spectrum analysis is performed using the envelope response spectrum of all attachment points. Alternately, the multiple excitation analyses may be used where acceleration time-histories or response spectra are applied to piping system attachment points.

The relative displacement between anchors is determined from the dynamic analysis of the structures. The results of the relative anchor point displacement are used for a static analysis to determine the additional stresses due to relative anchor point displacements.

Dynamic Analysis of Equipment

Each component of equipment is idealized as a mathematical model consisting of lumped masses connected by elastic members or springs. When the equipment is supported at more than two points located at different elevations in the building, the response spectrum analysis is performed using the envelope response spectrum of all attachment points. Alternatively, the multiple excitation analysis methods may be used where individual acceleration time-histories or response spectra are applied at each of the equipment attachment points. The relative displacement between supports is determined from the dynamic analysis of the structure. The relative support point displacements are used for a static analysis to determine the secondary stresses due to support displacements. Further details are given below.

Differential Seismic Movement of Interconnected Components

The procedure for considering differential displacements for equipment anchored and supported at points with different input motion is as follows: The relative displacements between the supporting point induces additional stresses in the equipment supported at these points. These stresses can be evaluated by performing a static analysis where each of the supporting points is displaced a prescribed amount. The time-history of displacement at each supporting point is readily obtained from the corresponding multisupport acceleration time-history which is provided as input for the dynamic analysis of the total component. These displacements are used to calculate stresses by determining the peak nodal responses.

In the static calculation of the stresses due to relative displacements in the response spectrum method, the maximum value of the nodal displacement is used. Therefore, the mathematical model of the equipment is subjected to a maximum displacement vector of its supporting points obtained from the nodal displacements. This procedure is repeated for the significant modes (modes contributing most to the total displacement response at the supporting point) of the component. The total stresses due to relative displacements are obtained by combining the modal results using the SRSS method. Since the maximum displacements for different modes do not occur at the same time, the SRSS method is a reasonable method. When a component is covered by the ASME Boiler and Pressure Vessel Code, the stresses due to relative displacement as obtained above are treated as secondary stresses.

3.7B.2.1.6 Seismic Qualification by Testing

For certain Category I equipment and components where dynamic testing is necessary to ensure functional integrity, test performance data and results reflect the following:

1. Performance data of equipment which, under the specified conditions, has been subjected to dynamic loads equal to or greater than those to be experienced under the specified seismic in-service conditions.

2. Test data from previously tested comparable equipment which, under similar conditions, has been subjected to dynamic loads equal to or greater than those specified.

3. Actual testing of equipment in accordance with one of the methods described in Sections 3.9B.2 and 3.10B.

3.7B.2.2 Natural Frequencies and Response Loads

See Section 3.7A.2.2.

3.7B.2.3 Procedure Used for Modeling

3.7B.2.3.1 Modeling Techniques for Category I Systems, Components, and Equipment

An important step in the seismic analysis of Category I systems, components, or equipment is the procedure used for modeling. The techniques currently being used for modeling are represented by lumped masses and a set of spring dashpots idealizing both the inertial and stiffness properties of the system. The details of the mathematical models are determined by the complexity of the actual system and the information required from the analysis.

3.7B.2.3.2 Modeling of Reactor Pressure Vessel and Internals

The seismic loads on the RPV and reactor internals are based on a dynamic analysis of the reactor building with the appropriate forcing function supplied at ground level. The mathematical model of the RPV and internals is shown on Figure 3.7B-2.

The RPV and internals mathematical model consists of lumped masses connected by linear elastic beam element members. Using the elastic properties of the structural components, the stiffness properties of the model are determined and the effects of bending shear, torsion, and axial loading are included.

Mass points are located at all points of critical interest such as anchors, supports, and points of discontinuity. In addition, mass points are chosen so that the mass distribution in various zones is as uniform as practicable and the full range of frequency of response of interest is adequately represented. Further, in order to facilitate hydrodynamic mass calculations, several mass points (fuel, shroud, vessel) are selected at the same elevation. The various lengths of CRD housings are grouped into the two representative lengths shown on Figure 3.7B-2. These lengths represent the longest and shortest housings in order to adequately represent the full range of frequency response of the housings.

The high fundamental frequencies of the CRD housings result in very small seismic loads. Furthermore, the small frequency differences between the various housings due to the length differences result in negligible differences in dynamic response. Hence, the modeling of intermediate length members becomes unnecessary. Not included in the mathematical model are the stiffness of light components, such as jet pumps, in-core guide tubes and housings, spargers, and their supply headers. This is done to reduce the complexity of the dynamic model and is justified because dynamic interaction is not significant. For the seismic responses of these components, floor response spectra generated from the system analysis are used.

The presence of the fluid and other structural components (e.g., fuel within the RPV) introduces a dynamic coupling effect. Dynamic effects of water enclosed by the RPV are accounted for by introduction of a hydrodynamic mass matrix that will serve to link the acceleration terms of the equations of motion of points at the same elevation in concentric cylinders with a fluid entrapped in the annulus. The details of the hydrodynamic mass derivation are given in Seismic Analysis of the Boiling Water Reactor[1]. The seismic model of the RPV and internals has 6 degrees-of-freedom for each mass point considered in the analysis.

The shroud support plate in its own plane is extremely stiff and therefore is modeled as a rigid link in the translational direction. The shroud support legs and local flexibilities of the vessel and shroud contribute to the rotational flexibilities and are modeled as an equivalent torsional spring.

3.7B.2.4 Soil-Structure Interaction

See Section 3.7A.2.4.

3.7B.2.5 Development of Floor Response Spectra

See Section 3.7A.2.5.

3.7B.2.6 Three Components of Earthquake Motion (NSSS)

Details are the same as those given in Section 3.7B.3.6.

3.7B.2.7 Combination of Modal Responses (NSSS)

All the modal responses are combined as described in Section 3.7B.3.7.

3.7B.2.8 Interaction of Non-Category I Structures with Category I Structures

See Section 3.7A.2.8.

3.7B.2.9 Effects of Parameter Variations on Floor Response Spectra

See Section 3.7A.2.9.

3.7B.2.10 Use of Constant Vertical Static Factors (NSSS)

Constant vertical static factors are not used for systems such as the RPV, internals, and large piping. See Section 3.7B.3.10 for subsystems and components.

3.7B.2.11 Method Used to Account for Torsional Effects (NSSS)

The RPV is an axisymmetric model with no built-in eccentricity.

3.7B.2.12 Comparison of Responses (NSSS)

Either the time-history method or the response spectrum approach may be used for the seismic analysis of NSSS components. Generally, the responses computed by both methods are comparable in magnitude, with the loads determined by the response spectrum method being somewhat more conservative. As both of these approaches are acceptable, additional comparison of results is unnecessary.

3.7B.2.13 Methods for Seismic Analysis of Dams

See Section 3.7A.2.13.

3.7B.2.14 Determination of Category I Structure Overturning Moments

See Section 3.7A.2.14.

3.7B.2.15 Analysis Procedure for Damping

In a linear dynamic analysis the procedure utilized to properly account for damping in different elements of a coupled system model is as follows:

1. The structural percent critical damping of the various structural elements of the model are first specified. Each value is referred to as the damping ratio (C_j) of a particular element that contributes to the complete stiffness of the system.

2. An eigenvalue analysis of the linear system model is performed. This results in the eigenvector matrices

(ϕ_i) that are normalized and satisfy the orthogonality conditions:

$$[{}^{\prime}\omega_i^2] = [\phi^T]\,[K]\,[\phi]$$
$$[{}^{\prime}I] = [\phi^T]\,[M]\,[\phi]$$

(3.7B-18)

Where:

[K] = Stiffness matrix

['ω_i] = Diagonal matrix of circular natural frequency of mode i

[ϕ^T] = Transpose of ϕ, which is a column vector of ϕ corresponding to the mode shape of mode i

The matrix ϕ contains all translational and rotational coordinates.

3. Using the strain energy of the individual components as a weighting function, the following equation is used to obtain a suitable damping ratio (β_i) for mode i:

$$\beta_i = \frac{1}{\omega_i} \sum_{j=1}^{N} \left[C_j \left(\phi_i^T\, K\, \phi_i\right)\right]$$

(3.7B-19)

Where:

N = Total number of structural elements

ϕ_i = Components of mode i eigenvector corresponding to the beam element j

β_i = Modal damping coefficient for mode i

K_j = Stiffness contribution of element j

ω_i = Circular natural frequency of mode i

ϕ_i^T = Transpose of ϕ defined above

C_j = Percent critical damping associated with element

3.7B.3 Seismic Subsystem Analysis

3.7B.3.1 Seismic Analysis Methods

The seismic system analysis methods described in Section 3.7B.2.1 are applicable to the Category I subsystems, components, and equipment. A description is given of the following methods by which Category I subsystems and components are qualified to ensure the functional integrity of the specific operating requirements that characterize their Category I designations.

In general, one of the following five methods of seismically qualifying the equipment is chosen based on the characteristics and complexities of the subsystem:

1. Dynamic analysis.

2. Testing procedures.

3. Equivalent static load method of analysis.

4. Combination of 1 and 2, or

5. Combination of 2 and 3.

The equivalent static load method of subsystem analysis is described in Section 3.7B.3.5.

Appropriate design response spectra (OBE and SSE) are furnished to the manufacturer of the equipment for seismic qualification purposes. Additional information such as input time-history is also supplied only when necessary.

When analysis is used to qualify Category I subsystems and components, the analytical techniques must conservatively account for the dynamic nature of the subsystems or components. Both the SSE and OBE, with their different damping values, are considered when the dynamic analysis is performed.

The general approach employed in the dynamic analysis of Category I equipment and component design is based on the response spectrum technique. The time-history technique described in Section 3.7B.2.1.1 generates time-histories at various support elevations for use in the analysis of subsystems and equipment. The structural response spectrum curves are subsequently generated from the time-history accelerations.

At each level of the structure where vital components are located, three orthogonal components of floor response spectra (two horizontal and one vertical) are developed. The response spectra are peak broadened ± 15 percent.

For vibrating systems and their supports, multidegree-of-freedom models are used in accordance with the lumped-parameter modeling techniques and normal mode theory described in Section 3.7B.2.1.1 and the references listed in Section 3.7B.5. Piping analysis is described in Sections 3.7B.3.3.1 and 3.7B.2.1.5.

When testing is used to qualify Category I subsystems and components, all the loads normally acting on the equipment are simulated during the test. The actual mounting of the equipment is also simulated or duplicated. Tests are performed by supplying input accelerations to the shake table to such an extent that generated TRS envelop the required response spectra.

Section 3.7B.2.1.6 discusses qualification requirements for certain Category I equipment and components where dynamic testing is necessary to ensure functional integrity.

The methodology used to account for the effects of differential support motion of multisupported equipment is described in Subsection 3.7B.2.1.5. The secondary stresses due to the support motions are combined with the corresponding inertia stresses by the SRSS method.

Also see Section 3.7A.3.1.

3.7B.3.2 Determination of Number of Earthquake Cycles

3.7B.3.2.1 Piping Systems

A total of 50 peak OBE stress cycles are postulated for fatigue evaluation.

3.7B.3.2.2 Other Equipment and Components

To evaluate the number of cycles caused by a given earthquake, a typical BWR building-reactor dynamic model was excited by three different recorded time-histories (May 18, 1940, El Centro NS component, 29.4 sec; 1952, Taft N 69° W component, 30 sec; and March 1957, Golden Gate S 80° E component, 13.2 sec). The modal response was truncated so that the response of three different frequency bandwidths could be studied: 0 to 10 Hz, 10 to 20 Hz, and 20 to 50 Hz. This was done to give a good approximation to the cyclic behavior expected from structures with different frequency content. Enveloping the results from the three earthquakes and averaging the results from several different points of the dynamic model, the cyclic behavior as given in Table 3.7B-2 was formed.

Independent of earthquake or component frequency, 99.5 percent of the stress reversals occur below 75 percent of the maximum stress level, and 95 percent of the reversals lie below 50 percent of the maximum stress level.

In summary, the cyclic behavior number of fatigue cycles of a component during an earthquake is found in the following manner:

1. The fundamental frequency and peak seismic loads are found by a standard seismic analysis (i.e., from eigenvalue extraction and a forced response analysis).

2. The number of cycles that the component experiences are found from Table 3.7B-2 according to the frequency range within which the fundamental frequency lies.

3. For fatigue evaluation, 0.5 percent of these cycles are conservatively assumed to be at the peak load, 4.5 percent at or above three-quarter peak. The remainder of the cycles have negligible contribution to fatigue usage.

The SSE has the highest level of response. However, the encounter probability of the SSE is so small that it is not necessary to postulate the possibility of more than one SSE during the 40-yr life of a plant. Fatigue evaluation due to the SSE is not necessary since it is a faulted condition and thus not required by ASME Boiler and Pressure Vessel Code, Section III.

The OBE is an upset condition and, therefore, must be included in fatigue evaluations according to ASME Boiler and Pressure Vessel Code, Section III. Investigation of seismic histories for many plants shows that during a 40-yr life, it is probable that five earthquakes with intensities 10 percent of the SSE intensity, and one earthquake with approximately 20 percent of the proposed SSE intensity, will occur. To cover the combined effects of these earthquakes and the cumulative effects of even lesser earthquakes, 10 peak OBE stress cycles are postulated for fatigue evaluation.

3.7B.3.3 Procedure Used for Modeling

Also see Section 3.7A.3.3.

3.7B.3.3.1 Modeling of Piping Systems

Summary

To predict the dynamic response of a piping system to the specified forcing function, the dynamic model must adequately account for all significant modes. Careful selection must be made of the proper response spectrum curves and proper location of anchors in order to separate Category I from non-Category I piping systems.

Selection of Mass Points

When performing a dynamic analysis, a piping system is idealized as a mathematical model consisting of lumped masses connected by weightless elastic members. The elastic members are given the properties of the piping system being analyzed. The mass points are carefully located to adequately represent the dynamic properties of the piping system. A mass point is located at the beginning and end of every elbow or valve, at the extended valve operator, and at the intersection of every tee. On straight runs, mass points are located at spacings no greater than the span length corresponding to 33 Hz. A mass point is located at every extended mass to account for torsional effects on the piping system. In addition, the increased stiffness and mass of valves are considered in the modeling of a piping system.

Selection of Spectrum Curves

In selecting the spectrum curves to be used for dynamic analysis of a particular piping system, curves are chosen that most closely describe the accelerations existing at the end points and restraints of the system. The procedure employed for decoupling the NSSS recirculation piping systems when establishing the analytical models to perform seismic analysis are as follows:

1. The small branch lines (6-in diameter and less) are decoupled from the recirculation piping systems and analyzed separately except for the bypass lines around 2RHS*AOV39A,B, which are not decoupled from the recirculation piping analysis.

2. The stiffness and mass of all the anchors and their supporting steel are large enough to effectively decouple the piping on either side of the anchor for analytic and code jurisdictional boundary purposes. The RPV is very stiff and massive compared to the piping system and, thus, during normal operating conditions, the RPV is also assumed to act as an anchor. Penetration assemblies (flued head fittings) are also very stiff compared to the piping system and are assumed to act as anchors. The stiffness matrix at the attachment location of the process pipe (i.e., RCIC, RHR supply, or RHR return) head fitting is sufficiently high to decouple the penetration assembly from the process pipe. GE analysis indicates that a satisfactory minimum stiffness for this attachment point is equal to the stiffness in bending and torsion of a cantilevered pipe section of the same size as the process pipe and equal in length to three times the process pipe outer diameter.

For a piping system supported at more than two points located at different elevations in the building, the response spectrum analysis is performed using the envelope response spectrum of all attachment points. The worst single floor response spectrum selected from a set of floor response spectra obtained at various floors may be applied identically to all floors provided it envelops the other floor response spectra in the set. Alternatively, the multiple support excitation analysis methods may be used where acceleration time-histories or response spectra are applied to all the piping attachment points.

3.7B.3.3.2 Modeling Equipment

For dynamic analysis, Category I equipment is represented by lumped mass systems that consist of discrete masses connected by weightless springs. The criteria used to lump masses are:

1. The number of modes of a dynamic system is controlled by the number of masses used. Therefore, the number of masses is chosen so that all significant modes are included. The modes are considered significant if the corresponding natural frequencies are less than the ZPA frequency of the excitation, and the stresses calculated from these modes are greater than 10 percent of the total stresses obtained from lower modes. The number of degrees-of-freedom are taken more than twice the number of modes with frequencies less than 33 Hz.

2. Mass is lumped at any point where a significant concentrated weight is located. Examples are the motor in the analysis of the pump motor stand and the impeller in the analysis of the pump shaft.

3. If the equipment has a free-end overhang span with flexibility significant compared to the center span, a mass is lumped at the overhang span.

4. When a mass is lumped between two supports, it is located at a point where the maximum displacement is expected to occur. This tends to lower the natural frequencies of the equipment, which results in a conservative analysis because the equipment frequencies are in the higher spectral range of the response spectra. Similarly, in the case of live loads (mobile) and a variable support stiffness, the location of the load and the magnitude of support stiffness are chosen so as to yield the lowest frequency content for the system. This ensures conservative dynamic loads since the equipment frequencies are always higher than the frequencies at which the spectral peaks occur. If not, the model is adjusted to give more conservative results.

3.7B.3.3.3 Field Location of Supports and Restraints

The final location of seismic supports and restraints for Category I piping, piping system components, and equipment, including the placement of snubbers, is checked against the drawings and instructions issued by the Engineer. An additional examination of these supports and restraining devices is made to assure that their location and characteristics are consistent with the dynamic and static analyses of the system. The final analyses of the as-built systems are performed as necessary, and the final certified as-built design reports are issued.

3.7B.3.4 Basis of Selection of Frequencies

All frequencies in the range of 0.25 to 33 Hz are considered in the analysis and testing of systems, components, and equipment. These frequencies are excited under the seismic excitation.

If the fundamental frequency of a component is greater than or equal to 33 Hz, it is treated as seismically rigid and analyzed accordingly. Frequencies less than 0.25 Hz are not considered as they represent very flexible structures and are not encountered in this plant. The frequency range between 0.25 and 33 Hz covers the range of the broad band response spectrum used in the design.

The number of modes used in the dynamic analysis of piping depends upon the number of mass points, dynamic degrees-of-freedom, and cutoff frequency.

The cutoff frequencies used for different dynamic loads are as follows:

1. For SRV and LOCA loads, the cutoff frequency is determined by evaluating the loading and structural characteristics so that the dynamic responses of interest are not significantly affected by the omission of modes with frequencies higher than the cutoff value. To ensure that considered modes include at least 90 percent of the response (in compliance with NUREG-0800, Criterion 3.7.2.II.1.a(5)), the following rules are used:

    a. For systems with fundamental frequency in the direction of excitation less than 20 Hz, the cutoff frequency is 60 Hz.

    b. For systems with fundamental frequency in the direction of excitation greater than 20 Hz, the cutoff frequency is 100 Hz.

2. If hydraulic transients are postulated, the system response will be obtained by direct integration of the equation of motion, rather than modal integration.

3.7B.3.5 Use of Equivalent Static Load Method of Analysis

When the natural frequencies of a system, component, or equipment are unknown, it may be analyzed by applying an equivalent static coefficient analysis. This procedure allows a simpler technique for added conservatism. The static acceleration of a component is conservatively assumed to be the peak spectral acceleration of the RRS which envelops the multisupport input spectra. The oscillation damping associated with the enveloping RRS must be representative of the actual component damping.

The equivalent static acceleration is then obtained by multiplying the static acceleration by a static coefficient, C_s, which takes into account the effects of both multifrequency excitation and multimode response. For verifying the structural integrity of frame-type components physically similar to beams and columns, the static coefficient, C_s, is taken as 1.5. For equipment having other than a frame-type configuration, justification is provided for the static coefficient used.

The equivalent static forces on each subcomponent of the equipment are obtained by multiplying the subcomponent masses by the equivalent static acceleration. The resulting static load vector is distributed over the equipment in a manner proportional to its mass distribution. The static stress analysis is then performed in a normal manner.

3.7B.3.6 Three Components of Earthquake Motion

3.7B.3.6.1 Response Spectrum Method

The total seismic response is predicted by combining the response calculated from the two horizontal and the vertical excitations. When the response spectrum method is used, the methods for combining the responses due to the three orthogonal components of seismic excitation are as follows:

$$R_i = \left[\sum_{K=1}^{3} R_{ij}^2\right]^{1/2}$$

(3.7B-20)

Where:

R_{ij} = Maximum, coaxial seismic response of interest (e.g., displacement, moment, shear, stress, strain) in directions i (due to earthquake excitation) and j (j = 1,2,3)

R_i = Seismic response of interest in i direction for design (e.g., displacement, moment, shear, stress, strain) obtained by the SRSS rule to account for the nonsimultaneous occurrence of the R_{ij}'s

3.7B.3.6.2 Time-History Method

When the time-history method of analysis is used, the time-history responses from each of the three components of the earthquake motion are combined algebraically at each time step. The maximum response is obtained from this combined time solution. The earthquake motions specified in the three different directions are verified to be statistically independent.

3.7B.3.7 Combination of Modal Responses

The requirements of RG 1.92 are satisfied as follows. In a response spectrum modal dynamic analysis, if the modes are not closely spaced (i.e., if the frequencies differ from each other by more than 10 percent of the lower frequency), the modal responses are combined by the SRSS method as described in Section 3.7B.3.7.1 and RG 1.92. If some or all of the modes are closely spaced, a double sum method (Section 3.7.3.7.2) is used to evaluate the combined response. The use of the time-history analysis method precludes the need to consider closely-spaced modes.

3.7B.3.7.1 Square Root of the Sum of the Squares Method

Mathematically, the SRSS method is expressed as follows:

$$R = \left(\sum_{i=1}^{n} (R_i)^2 \right)^{1/2}$$

(3.7B-21)

Where:

R = Combined response

R_i = Response due to mode i

n = Number of modes considered in the analysis

3.7B.3.7.2 Double Sum Method

This method, as defined in RG 1.92, is expressed mathematically:

$$R = \left(\sum_{k=1}^{N} \sum_{s=1}^{N} \left| R_k R_s \right| \epsilon_{ks} \right)^{0.5}$$

(3.7B-22)

Where:

R = Representative maximum value of a particular response of a given element to a given component of excitation

R_k = Peak value of the response of the element due to mode k

n = Number of significant modes considered in the modal response combination

R_s = Peak value of the response of the element attributed to mode s

Where:

$$\epsilon_{ks} = \left[1 + \left\{\frac{(\omega'_k - \omega'_s)}{(\beta'_k\omega_k + \beta'_s\omega_s)}\right\}^2\right]^{-1}$$

(3.7B-23)

In which:

$$\omega_k' = \omega_k\left[1 = \beta_k^2\right]^{0.5}$$

$$\beta'_k = \beta_k + \frac{2}{t_d\omega_k}$$

(3.7B-24)

Where:

ω_k = Modal frequency in mode k

β_k = Damping ratio in mode k

t_d = Duration of the earthquake

3.7B.3.8 Analytical Procedure for Piping

The analytical procedures for piping analysis are described in Sections 3.7B.2.1.5 and 3.7B.3.3.1.

3.7B.3.9 Multiply Supported Equipment Components with Distinct Inputs

The procedure and criteria for analysis are described in Sections 3.7B.2.1.5 and 3.7B.3.3.2.

3.7B.3.10 Use of Constant Vertical Static Factors

Constant vertical static factors in analysis for subsystems and components are used as described in Section 3.7B.3.5.

3.7B.3.11 Torsional Effects of Eccentric Masses

Torsional effects of eccentric masses are included for Category I subsystems (Section 3.7B.3.3.1).

3.7B.3.12 Buried Category I Piping Systems and Tunnels

See Section 3.7A.3.12.

3.7B.3.13 Interaction of Other Piping with Category I Piping

When other (non-Category I) piping is attached to Category I piping, the other piping is analytically coupled sufficiently so as not to significantly degrade the accuracy of the analysis of the Category I piping. Furthermore, the other piping is designed to withstand the SSE sufficiently to prevent failure of the Category I piping.

3.7B.3.14 Seismic Analysis for Reactor Internals

The modeling of the RPV and internals is discussed in Section 3.7B.2.3.2. The damping values are given in Table 3.7B-1. The seismic model is shown on Figure 3.7B-2, and a summary of loading conditions (including seismic and hydrodynamic), evaluation criteria, calculated maximum stresses in the selected locations, and the allowable stresses is given in Table 3.9B-2.

3.7B.3.15 Analysis Procedures for Damping

Analysis procedures for damping are discussed in Section 3.7B.2.15.

3.7B.4 Seismic Instrumentation

See Section 3.7A.4.

3.7B.5 Reference

1. Liu, L. K. Seismic Analysis of the Boiling Water Reactor, Symposium on Seismic Analysis of Pressure Vessel and Piping Components, First National Congress on Pressure Vessel and Piping, San Francisco, CA, May 1971.

TABLE 3.7B-1

CRITICAL DAMPING RATIOS FOR DIFFERENT MATERIALS

Item	Percent Critical Damping OBE Condition	Percent Critical Damping SSE Condition
Welded structural assemblies	2.0	4.0
Steel frame structures	2.0	4.0
Equipment	2.0	3.0
Bolted or riveted structural assemblies	4.0	7.0
Vital piping systems		
Diameter greater than 12 in	2.0	3.0
Diameter less than or equal to 12 in	1.0	2.0
Reactor pressure vessel, support skirt, shroud, shroud head/separator	2.0	4.0
CRD housings and guide tubes	1.0	2.0
Fuel assembly	6.0	6.0
CRD support springs, shroud support spring, and stabilizer	2.0	4.0
Primary containment	4.0	7.0
Shield wall	4.0	7.0
Pedestal	4.0	7.0

TABLE 3.7B-2

NUMBER OF DYNAMIC RESPONSE CYCLES EXPECTED
DURING A SEISMIC EVENT

Frequency Band (Hz)	0-10	10-20	20-50
Total number of seismic cycles	168	359	643
No. seismic cycles - 0.5% cycles between 75 and 100% of peak loads	0.8	1.8	3.2
No. seismic cycles - 4.5% cycles between 50 and 75% of peak loads	7.5	16.2	28.9

3.8 DESIGN OF SEISMIC CATEGORY I STRUCTURES

3.8.1 Concrete Containment

The concrete containment structure is designed to house the reactor vessel and RCPB and is part of the containment system whose functional requirement is to control the release of radioactivity. The major components of this pressure suppression-type containment system are the primary containment steel liner and the primary containment concrete structure. This section describes the structural design considerations for the primary containment.

Since the original design of this containment preceded the issuance of ASME Section III, Division 2, the reinforced concrete primary containment is designed and constructed to the requirements of the American Concrete Institute, Building Code Requirements for Reinforced Concrete ACI 318-71. Except for regions around penetrations that are designed to meet the requirements of ASME Section III, Division 2, the primary containment steel liner is designed following the requirements of ASME Section III, Division 1.

3.8.1.1 Description of the Containment

The primary containment is a reinforced concrete structure that consists of a drywell chamber located above a suppression chamber, and a drywell floor which separates the drywell chamber from the suppression chamber. The primary containment structure is supported on a 10-ft thick reinforced concrete mat which also supports the reactor building. A series of 24-in diameter downcomer vent pipes penetrates the drywell floor.

The drywell is a steel-lined reinforced concrete vessel in the shape of a frustum of two cones. It is enclosed at the top by a drywell head dome. The steel liner is attached to the inside face of the wall and functions primarily as a leak-tight membrane. The inside diameter of the drywell is 91 ft at the drywell floor level (el 240 ft 7 in) and 34 ft at the top of the primary containment (el 326 ft 10 in).

The suppression chamber is a stainless clad steel-lined cylindrical shell with an inside diameter of 91 ft. It is located directly below the drywell and is supported on a 10-ft thick reinforced concrete mat at el 175 ft. The suppression chamber contains a large reservoir of water called the suppression pool, which serves as a heat sink to absorb energy released into the suppression pool as a result of SRV blowdown or a LOCA.

The suppression pool is composed of an inner and outer pool, the inner pool being located inside the cylindrical reactor pedestal and the outer pool being located between the pedestal outside diameter and the primary containment wall. The inner and outer pools are connected by six vent openings located in the reactor pedestal wall.

The drywell floor is a 4-ft thick annular reinforced concrete slab separating the drywell from the suppression chamber. It is anchored at the reactor support pedestal and the containment wall. Its primary function is to separate the drywell from the suppression pool. It also provides the primary support for the downcomer vent lines and also supports other penetrations and embedments (Section 3.8.3).

The transfer of loads from the drywell floor to the primary containment concrete wall (Figure 3.8-1) is made through thickened liner sections. The force in the reinforcing steel is transmitted to the liner by Cadweld sleeves which are attached, in line, to each side of the thickened liner plate (Figure 3.8-1). Continuity is thus provided to the reinforcing steel without perforating the liner boundary.

The primary containment wall contains penetrations for process piping, instrument piping, and electrical conductors. It supports floor beam seats, and supports embedments for pipe restraints, pipe supports, conduit, and duct supports. Five access hatches and one airlock penetrate the containment wall and provide for personnel and equipment egress (Section 3.8.1.1.2). One of the larger hatches contains an additional airlock as described in Section 3.8.1.1.2.

The containment wall is designed to withstand anticipated loads without participation of the liner as a structural component.

3.8.1.1.1 Reinforcing Steel Arrangement

The main reinforcing steel in the primary containment wall consists of inside and outside layers of hoop and meridional reinforcement and diagonal reinforcement. The diagonal reinforcement is placed in two orthogonal directions near the outside wall face to form a helix with an angle of 45 deg from the vertical axis of the shell. Hoop and longitudinal tension forces along with the tangential shears will be resisted by the hoop, meridional and diagonal reinforcing steel.

To resist the large radial shear near the base of the wall, flat steel bars inclined at 45 deg to the horizontal, and welded to the vertical reinforcing steel, are installed (Figure 3.8-3). The welded flat bars are terminated at a level above the mat, and single-leg radial shear reinforcing steel is placed as required by the ACI-318 code.

Supplementary reinforcing steel, normal to the face of the wall, is provided in the lower portion of the containment structure wall to resist splitting of the concrete in the vertical plane. Minimum concrete cover for all principal reinforcing steel of the primary containment structure either equals or exceeds the requirements of ACI-318-71.

Figures 3.8-3 and 3.8-3a show typical details of reinforcing steel in the containment wall. Section 3.8.1.4 describes the reinforcing steel arrangement at hatch openings, as shown on Figure 3.8-4.

3.8.1.1.2 Steel Liner and Penetrations

Steel Liner

Except at various penetrations and access openings through the walls, the primary containment liner is a continuous steel membrane, backed by reinforced concrete. The function of the primary containment liner is to act as a leak-tight membrane to provide a barrier to the release of fission products.

Generally, the liner is 3/8-in thick except at discontinuities such as corners, penetrations, and beam seat and pipe restraint attachment regions, where it is adequately thickened. In the drywell, the steel liner plate material is carbon steel. In the suppression chamber it is carbon steel clad with stainless steel, except at embedments or penetrations which are stainless steel throughout. The portion of the primary containment liner that functions as the suppression pool floor is 1/4 in thick except in regions where load transfers through the floor require that it be thicker. The 1/4-in thick floor liner is welded to the wall liner through a corner junction embedment. It is also welded to the reactor pedestal embedment plates, the SRV T-quencher embedments, and the bridging bar seam embedments that anchor the membrane to the reinforced concrete reactor building floor mat. Approximately 12 in of insulation concrete is installed over the floor liner to protect it from the thermal effects of the DBA and to minimize the corresponding loads on the concrete containment wall. Also, an additional liner is provided as described below.

Other Liners Inside the Primary Containment

An additional liner is provided over the insulation concrete to serve as a waterproof membrane for that concrete. This floor insulation liner is stainless steel as is all other suppression pool steel in contact with its atmosphere. This liner is 1/8 in thick and is anchored to the insulation concrete through a rectangular grid of channel and headed concrete anchors, and is anchored to the 18 T-quencher supports, the primary containment liner, and the reactor pedestal liner. It is installed after the concrete is poured.

The reactor pedestal (Section 3.8.3) is also lined with stainless steel. This liner serves as a concrete form during construction and functions as a waterproof membrane for the pedestal concrete during plant operation. It is anchored to the concrete by headed concrete anchors.

The bottom surface of the drywell floor is formed with corrugated stainless steel and the top is covered with carbon steel liner. The stainless corrugated steel on the bottom surface functions as a concrete form. The steel liner on the top surface functions as a positive gas-tight membrane between the drywell and the suppression chamber to ensure that steam can enter the suppression chamber only through the SRV vent lines or the downcomers. This liner is anchored and seal-welded to the pedestal wall, the primary containment liner knuckle, and all of the drywell floor penetrations and embedments.

Attachments to the Primary Containment Wall

The loads from the structural equipment supports inside the containment are transferred directly to the concrete wall through thickened and stiffened liner insert embedment plates. Loads are distributed on these insert plates through gussets and shear bars which are welded on the back side in an orthogonal grid pattern to provide stiffness to the plate and to provide a means to transfer shear loads to the concrete. The liner insert plates are attached to the concrete wall by either standard headed concrete anchors or larger fabricated anchors, depending upon the load to be transferred. The fabricated anchors are made from 1 1/2-in plate stock and are generally 3 in wide and 18 to 24 in long with a head or flat plate welded at the end to develop the concrete shear cone necessary for load transfer directly to the concrete. This ensures that the loads are not transferred to the steel liner.

Equipment and floor loads are transferred to the wall by floor beam seats, pipe restraint seat/embedments, pipe support embedments, and overlay plates.

Floor beam seat embedments are provided to support the structural steel framing inside the containment. A typical design is shown on Figure 3.8-5.

Pipe restraint embedments are provided to transfer piping restraint loads from postulated accidents. They can either be of a design similar to the floor beam seats or of a design that accepts the welding of a restraint beam to the embedment. A typical pipe whip restraint embedment is shown on Figure 3.8-5.

The reactor stabilizer embedments are provided to transfer the load from the reactor stabilizer structure (the star truss) to the concrete wall.

Pipe support embedments are provided to transfer moderate pipe and duct support loads to the wall.

When the loads from the equipment supports are comparatively small, overlay pads are attached to the liner to distribute and transfer loads through the liner and the liner anchorage into the concrete wall.

Anchorage

The primary containment liner is in intimate contact with the concrete wall through a series of anchor studs welded to the back side of the liner and embedded in concrete on a controlled nodal point pattern (Section 3.8.1.4.2 and Figure 3.8-1). The size of these anchor studs is 5/8 in diameter by 6 1/2 in long. The anchor studs have been designed to accommodate the loads due to both operating and postulated accident conditions.

The primary containment suppression pool floor liner at el 175 ft is anchored to the mat through a pattern of 1/2- by 7-in bridging bars which are embedded on edge and located at each floor liner plate seam (Figure 3.8-2). The bridging bars employ fabricated anchors to anchor the insulation concrete as well as the floor liner to the mat.

Equipment Hatch

A 13 ft 3-in diameter equipment hatch (Figure 3.8-6) is provided in the drywell at el 266 ft 4 in and azimuth 315 deg to service the 261 ft floor and nearby areas during reactor shutdown. The hatch barrel has a floor at el 261 ft and has a bolted-flange type closure mounted outside the containment which contains a double O-ring seal with a leak test tap between the O-rings to accommodate the periodic Type B leak rate test requirements. The hatch barrel is 5/8-in thick and the thickness increases to 3 in in the region of the closure. The hatch cover is a 1 1/2-in thick spherical cap and is bolted to the hatch barrel with 1 1/4-in diameter swing bolts.

Combination Equipment Hatch and Personnel Airlock

A combination equipment hatch and personnel airlock (Figure 3.8-6) is provided in the drywell at el 266 ft 4 in and azimuth 135 deg to service the 261 ft floor and nearby areas located on the opposite side of the drywell from the 315-deg equipment hatch. It provides access for large equipment during reactor shutdown and access for personnel when required. The equipment hatch geometry is the same as the hatch at azimuth 315 deg except that the hatch cover that contains the personnel airlock is conical.

The personnel airlock is located in and welded to the hatch cover. It is 9 ft 7 1/2-in outside diameter, 15 ft 3 in long, and contains a 3 ft 6-in by 6 ft 8-in door at each end. The doors are located in reinforced bulkheads and each is sealed with a double-gasket compression seal with a leak test tap for leak rate testing between the seals. Both doors can be leak rate tested from outside the containment. The mechanical and electrical bulkhead penetrations also have provisions for leak rate testing. Both doors swing inward so as to seat under the positive accident pressure. To ensure containment integrity, they are mechanically interlocked so that if one door is open, the other cannot be activated. Each door can be closed automatically from outside the opposite door to facilitate entry and egress from the containment when the opposite door has been left open by the last person to enter or exit. The doors are equipped with valves for equalizing air pressure prior to opening. Limit switches are provided on the doors to enable their operation to be monitored from outside each door and in the control room.

Escape Airlock

A 5 ft 6-in diameter emergency airlock (Figure 3.8-6) is provided in the drywell at el 264 ft 9 in and azimuth 236 deg. This airlock serves as a personnel alternate escape route to the main airlock should the need arise. It also provides additional service accessibility during shutdown. It is similar to the personnel airlock in that it also has interlocked doors, equalizing valves, and double-gasket compression seals have provisions for leak testing from outside the containment. Door operation is manual. Limit switches are provided on the doors to enable their operation to be monitored from outside each door and in the control room.

Suppression Chamber Access Hatches

Two 3 ft 9-in diameter hatches provide access to the suppression chamber (Figure 3.8-7). The hatches are located at el 225 ft and at azimuths 130 deg 30 min and 310 deg 30 min. Each access opening has a bolted steel flanged closure mounted outside the containment structure. Each hatch has a double O-ring seal with a leak test tap provided between the seals for periodic testing.

Control Rod Drive Removal Hatch

A 2 ft 1-in diameter CRD removal hatch provides access for the CRD assemblies in the drywell at a centerline location of el 263 ft and azimuth 221 deg (Figure 3.8-7). The hatch has a bolted steel closure mounted outside the containment structure. It has a double O-ring seal with a leak test tap provided between the seals for periodic testing.

Drywell Head

The drywell has a removable 34-ft diameter top cover called a drywell head (Figure 3.8-8), which provides accessibility to the top of the reactor during refueling. It is a freestanding 1 1/8-in thick torispherical head that utilizes a finger pin quick-disconnect closure joint designed as shown on Figure 3.8-8. The finger pin connection represents an improvement over earlier bolted flanges in that stress discontinuities are reduced, and it results in saving time required to remove and install the head. The head is welded to a cylindrical skirt that is, in turn, welded to a thickened joint closure. The upper portion of the joint closure fits inside the U-shaped bottom closure.

Forty-eight 3-in diameter horizontal pins engage the top and bottom to join the mating parts as shown on Figure 3.8-8. The flange design incorporates a flat-faced, double O-ring seal with a leak test tap between the O-rings to accommodate the Type B leak rate test requirements described in Section 6.2.6.

Penetrations

Services and communications between the inside and outside of the primary containment are effected through penetrations. All penetrations consist of a basic wall insert pipe or sleeve that is welded to a liner reinforcing plate and any additional items such as pressure piping, insulation, electrical, or mechanical components required for the individual service.

There are three basic types of penetrations: electrical, instrument, and piping. Electrical penetration sleeve assemblies (Figure 3.8-9) consist of a seamless pipe wall insert sleeve that is welded to a liner reinforcing plate. The sleeve design is similar to the thermally cold piping penetration discussed below. A weld neck flange is welded to the outside end of the sleeve to accommodate the electrical penetration bolted flange. Sealing is effected by a double O-ring seal and a leak test tap is provided between the O-rings to accommodate the Type B leak rate tests.

The basic instrument-type penetration (Figure 3.8-9) is similar to the thermally hot piping penetrations described below in that the process pipe portion is located inside the wall insert sleeve. The process pipe is socket-welded to a transition forging that is in turn welded to the outboard end of the wall insert sleeve. The instrument-type penetration includes: instrument penetrations, piping penetrations for the CRD system, and mechanical penetrations for the TIP drive. Also there is a series of temperature-monitoring instrument penetrations (Figure 3.8-9) located symmetrically around the outer wall of the suppression pool. The wall insert is a heavy wall stainless steel pipe that is welded to the liner reinforcing plate. A special stainless thermowell tip is welded to the end of the sleeve to accommodate the temperature sensor.

Piping penetrations fall into two categories: thermally hot and thermally cold (Figure 3.8-10). Thermally hot penetrations are provided when the design temperature exceeds 200°F. For thermally hot penetrations the wall insert pipe acts as a sleeve. For thermally cold penetrations the wall insert serves as a section of the process piping. For thermally hot penetrations the annular space between the wall insert pipe and the process pipe provides thermal protection to the wall. Each thermally hot penetration is designed with sufficient space between the insert pipe and the process pipe for the required pipe insulation to ensure that the temperature of the concrete in contact with the sleeve remains within 200°F for normal conditions. However, a short term maximum of 350°F is allowed for accident conditions. The process pipe portion of the penetration assembly is a continuous forging. It is attached to the sleeve through a flued-head transition that is an integral part of the forging (Figure 3.8-10).

Provisions for a heat exchanger were originally designed into each hot penetration sleeve but these were found to be unnecessary considering the geometry, insulation, and current temperature limits.

All penetrations are designed to be capable of accommodating the stresses imposed on them by the design conditions. The design is discussed in detail in Section 3.8.1.4.2.

3.8.1.2 Applicable Codes, Standards, and Specifications

The design and construction of the primary containment, steel liner, equipment and personnel access hatches, and penetrations meet or exceed the requirements of the codes, standards and regulations listed in Section 3.8.4.2 for the design of Category I structures.

3.8.1.3 Loads and Load Combinations

3.8.1.3.1 Primary Containment Structure

The reinforced concrete structure of the primary containment is designed to withstand the loadings and stresses anticipated during the 40-yr operating life of the unit which arise from normal operation and other postulated loads such as earthquake, LOCA, jet impingement, and suppression pool hydrodynamic loads as defined in Table 3.8-1. The steel liner which is attached to and supported by the concrete also transmits loads to the concrete. The load combinations for which the structure is designed are defined in Table 3.8-1.

Design load criteria conform to current containment design practice. The criteria contain varying load factors for combining dead, pressure, temperature, and earthquake forces. The total loading resulting from the summation of any one of the combinations may cause a maximum stress condition depending on the type of stress and member under consideration.

The primary containment is designed to withstand the applicable loads and reaction forces due to postulated pipe rupture (Section 3.6A) in combination with other loads. The primary containment is also designed for the postaccident environments.

3.8.1.3.2 Steel Liner and Penetrations

Table 3.8-2 lists the load and symbol definitions for the following portions of the primary containment liner.

Wall Liner, Floor Liner, and Embedments

The wall liner, floor liner plate, and corner transition section are designed for the loads and load combinations described in Table 3.8-2 so that either the resulting stress levels do not exceed the allowable limits given in ASME Section III, Division 1, Subsection NE, 1971 Edition through 1973 Summer Addenda, or that the resulting strain levels do not exceed the allowable strain levels given in ASME Section III, Division 2, Subsection CC-3700, 1975 Edition. The welded attachments to the primary containment liner are designed to meet the requirements of Subsection NE, 1974 Edition, with Winter 1976 Addenda and Code Case N-224-1. The liner is shielded from the application of internally-generated concentrated missile loads (Section 3.5). Jet impingement effects from postulated pipe rupture inside primary containment are applied to the liner. Structural attachment positions of the liner embedments that are beyond the scope of ASME III, Division 1, are designed using the load combinations for structural steel design (Section 3.8.4.3).

Access Locks and Hatches

The equipment hatch, equipment hatch/personnel airlock, escape airlock, suppression pool hatches, and CRD removal hatch are subjected to the loads and load combinations described in Table 3.8-3.

Drywell Head

The drywell head is subjected to the loads and load combinations described in Table 3.8-4.

Penetrations

Load combinations and stress allowables for the penetrations are summarized in the following tables:

ASME III Safety Class 1 piping	Table 3.8-5
ASME III Safety Class 2 piping	Table 3.8-6
ASME III Class MC components	Table 3.8-7

3.8.1.4 Design and Analysis Procedure

3.8.1.4.1 Concrete Primary Containment Structure

The reinforced concrete primary containment structure is analyzed for the various loading conditions using the SHELL 1 computer code (Appendix 3A) for thin shells of revolution that are symmetrically as well as asymmetrically loaded. The effects of dead and live loads, internal pressure, temperature, earthquake, pipe rupture, and hydrostatic loads are considered. For the purpose of analysis and design, the concrete is assumed to be fully cracked in the hoop direction over the entire height. In the vertical direction, the concrete is also assumed to be fully cracked everywhere except for a small region adjacent to the foundation mat. The stiffness properties of the wall are calculated accordingly.

Additional reinforcing steel is provided around penetrations and hatches in order to minimize the effects from localized load concentrations and to provide continuity of the load-carrying capability of the wall. These local effects, which do not affect the overall analysis, are considered on an individual basis for design.

After the internal loads in the shell are determined, the reinforcing steel is checked for adequacy. Special attention is given to the effects of shear forces in the shell, with additional shear reinforcing being provided as required. As a result, diagonal shear bars are used to resist radial shear at the top of the upper cone, the junction of the two cones, the cone-to-cylinder junction, and at the base of the wall adjacent to the foundation mat. Typical arrangement of reinforcing is shown on Figure 3.8-3.

Certain areas of the wall require special analysis due to geometric conditions. One region requiring special analysis is the top of the concrete containment. The design of this area is based on a supplementary analysis using a finite element approach, which addresses the geometric and nonlinear material effects, i.e., concrete cracking. The particular model used for this analysis is extended well into the membrane zone of the shell so that membrane conditions are assumed at the boundaries. Figure 3.8-3 shows the reinforcing detail in this area.

Another area that requires special analysis using the finite element approach is in the vicinity of the hatches. Typical reinforcing in the area of the equipment hatch is shown on Figure 3.8-4.

These regions in the wall are analyzed by means of the three-dimensional, finite element capability of the computer program STRUDL II (Appendix 3A). Because of geometrical symmetries in the area of the hatches, the finite element models selected for these analyses are semicircular. In order to eliminate the effects of the hatch opening and thickened ring beam at the grid's outer boundary, the semicircular finite element grid is assumed to have a maximum radius approximately equal to 2.5 times the radius to the outer edge of the thickened ring beam. A geometry of quadrilateral shapes emanating radially from the center of the opening is used to provide a fine grid in the meridional and circumferential shell directions. Three elements are taken through the typical wall section, with additional elements added on each side of the typical wall section to account for the thickened ring beam section.

The STRUDL II program allows the modeling of the structural characteristics of the wall (cracked versus uncracked) and the inclusion of the liner for the various loadings including temperature. During an accident the liner will normally be in a state of compression due to the sharp temperature rise within the containment structure, and therefore will add load to the ring beam. The liner, however, is assumed not to contribute to the structural capacity of the ring beam or wall for any loading condition other than the primary containment structural acceptance test.

To obtain a more realistic assessment of the strains, displacements, and stresses in the area of the hatches, the ring beam and cylindrical wall are assumed to be fully cracked and to have only the stiffness of the reinforcing bars.

The STRUDL II program yields both the discontinuity effects between the cylinder wall and the thickened ring beam and the pattern of membrane forces in the region of the hatch openings. Additional reinforcement (circumferential, meridional, and diagonal) is provided in regions where a significant increase over the typical membrane forces is observed. The principal circumferential and meridional reinforcing bars are extended to the inner face of the ring beam and are either bent at right angles or Cadwelded to each other.

Seismic analysis of the primary containment structure provides the acceleration to which the primary containment would be subjected during an earthquake. These accelerations, when multiplied by the associated masses, are applied as static loads to the structure. Tangential shears caused by asymmetric loads are resisted by the concrete and the steel reinforcement in the wall. Strains, compatible with the effects of internal pressure-induced stresses, are assumed in calculating the reinforcing steel requirements for earthquake loads. The steel liner plate is conservatively assumed to provide no shear capacity to the wall section.

3.8.1.4.2 Steel Liner and Penetrations

Steel Liner

The liner was analyzed using the computer program KALNINS. The orthotropic capability is used to model the reinforcing bar array as an equivalent orthotropic shell. Temperature variations within the containment are considered by applying an equivalent pressure that satisfies the equilibrium equations of the liner and concrete vessel. The primary containment is conservatively assumed to be completely cracked to obtain the maximum liner deformation. The boundary condition at the wall-to-mat junction is assumed to be completely fixed to obtain the maximum liner stress intensity range.

The liner structural integrity against buckling is obtained from the results of an analysis performed using the ANSYS computer program. Results show that the anchor studs have a safety factor of at least 2.0 against progressive failure.

Floor Liner Plate and Corner Transition Section

The outer edge of the suppression pool floor liner plate and the corner transition section are analyzed using the computer code SHELL 1 (Appendix 3A) for thin shells of revolution. The liner model extends to the first row of wall liner plate anchor studs. At this anchor stud location, the forces obtained from the containment liner analysis are imposed on the corner junction.

Temperature distributions following a DBA are calculated using the finite difference computer code TAC-2D (thermal analysis code - two-dimensional) (Appendix 3A). The temperatures determined by TAC-2D are used as input for SHELL 1 to determine thermal stresses.

Hot Piping Penetrations (Sleeved)

The hot (sleeved) piping penetration assemblies are analyzed for stress using the finite element computer code ASAAS (asymmetric stress analysis of axisymmetric solids) (Appendix 3A). This program is capable of evaluating the effects of symmetric and asymmetric loads as well as pressure and temperature loads.

Temperature distributions in the penetrations are evaluated using TAC-2D. The temperatures as determined by TAC-2D are used as input for ASAAS to determine thermal stresses. For the ASME Code Class 1 penetrations, pressure and temperature transients are analyzed for all modes of operation. For the ASME Code Class 2 penetrations, only steady-state temperatures are considered. The stresses are limited to the allowables in Table 3.8-5 for Code Class 1 piping penetrations and Table 3.8-6 for Code Class 2 piping penetrations. For the portions of the penetrations that form part of the primary containment boundary and are classified MC, the stresses are limited to the allowables in Table 3.8-7.

Cold Piping Penetrations (Unsleeved)

The cold (unsleeved) piping penetrations through the primary containment are analyzed using the program ASAAS. Stress concentration factors for the junction of the pipe and the containment wall are developed using this program. The allowable stresses for these Safety Class 2 penetrations are given in Table 3.8-6.

Beam Seat and Pipe Restraint Embedments

The embedments for beam seats and pipe restraints are analyzed for the loads imposed by the design and operating conditions. The anchor-concrete interface pattern is assumed analogous to a reinforced concrete beam, and anchor loads and stresses in reinforcing steel are determined. The front face of the embedment is designed by conventional steel design methods using AISC Specification for Design, Fabrication and Erection of Structural Steel for Buildings. The anchor concrete interface is designed to the guidelines outlined in ACI-318-71.

Access Openings

The equipment hatches, suppression pool access hatches, and CRD removal hatch are analyzed using classical equations and the ANSYS finite element computer program (Appendix 3A) for axisymmetric shells. Where required, natural frequencies are determined and seismic loadings are applied accordingly. The suppression pool and CRD hatches are assumed to be rigidly supported by the concrete wall. The equipment hatch barrels are supported by the concrete around a 90-deg arc at the bottom and rigidly restrained at the liner reinforcing plate-to-barrel junction.

Analysis of the escape lock and the personnel lock is performed using hand calculations and by the NASTRAN (Appendix 3A) computer program. The escape lock is assumed to be rigidly restrained by the concrete wall. The personnel lock is assumed to be supported by the equipment hatch cover welded to its barrel. The NASTRAN finite element models use bar and plate elements for analysis of the bulkheads.

Drywell Head

The drywell head is analyzed as a finite element shell of revolution using axisymmetric conical shell elements. The structural analysis program used is ANSYS (Appendix 3A). The analysis is performed for pressure, deadweight, seismic, thermal, and jet impingement loadings.

The finger pin closure is analyzed using the same ANSYS program used for the drywell head. It is also modeled as an axisymmetric shell and includes a portion of the shell above and below the closure. The finger pins which join the upper and lower closure fingers are modeled as two equivalent rings in the two 1/8-in gaps between the fingers. Loads obtained from the head analyses are applied at the upper boundary of the closure analysis. The concrete is conservatively assumed to provide a fixed support at the lower boundary of the straight portion of the shell at the top of the containment.

3.8.1.5 Structural Acceptance Criteria

3.8.1.5.1 Primary Containment Structure

The design of the primary containment structure follows ACI-318-71. The basic criterion for concrete strength design is expressed as:

Required strength $\leq$ calculated strength

All members of the primary containment structure are proportioned to meet this criterion. The required strength is expressed in terms of design loads or their related internal moments and forces. Design loads are defined as loads that are multiplied by their appropriate load factors (Section 3.8.1.3). Calculated strength is computed by provisions of the ACI-318-71 Code, including the appropriate capacity reduction factors. The modulus of elasticity of reinforcing steel is 29,000,000 psi. A Poisson ratio of 0.167 is used for reinforced concrete. Tangential shear (v_u) resulting from asymmetric loadings will be resisted by either the concrete or the concrete and steel reinforcing bars.

The maximum allowable stress for tangential shear (v_u) in the concrete alone is 40 psi. The actual and allowable stresses for the containment structure are given in Table 3.8-8.

3.8.1.5.2 Steel Liner and Penetrations

The allowable stresses and strains for the primary containment liner are limited by either the stress criteria given in ASME Section III, Division 1, 1971 Edition through the Summer 1973 Addenda, or the strain criteria given in ASME Section III, Division 2, 1975 Edition. The allowable stresses and strains for the primary containment penetrations are limited by the criteria given in ASME Section III, Division 1, 1971 Edition through the Summer 1973 Addenda. In particular:

1. Wall liner, floor liner, corner junction transition and liner insert plate embedments (Table 3.8-2 and Section 3.8.1.3.2).

2. Access locks and hatches (Table 3.8-3 and Section 3.8.1.3.2).

3. Drywell Head (Table 3.8-4 and Section 3.8.1.3.2).

4. Penetrations (Tables 3.8-5 through 3.8-7 for the ASME Safety Class 1, 2, and MC portions, respectively, and Section 3.8.1.3.2).

5. The minimum allowable stresses for ASTM A500 Grade B tube steel (for welded attachments to the liner) are as specified in Code Case N-224-1.

3.8.1.6 Materials, Quality Control, and Construction Techniques

3.8.1.6.1 Concrete Primary Containment Structure

The materials used in the construction of the primary containment are the same as those used for other Category I structures (Section 3.8.4.6). The quality control program for the primary containment structure meets or exceeds the requirements outlined in Sections 3.8.4.2 and 3.8.4.6 for procurement, fabrication, testing, and construction of Category I structures. There are no special techniques used in the construction of the concrete primary containment structure.

3.8.1.6.2 Steel Liner and Penetrations

Steel Liner

All materials used in the fabrication of the primary containment liner and liner components are in accordance with approved materials listed in Section III Division 1 Appendixes of the ASME Boiler and Pressure Vessel Code for Nuclear Power Plant Components, 1971 Edition through Summer 1973 Addenda. The welded attachments to the liner meet the material requirements of Subsection NE, 1974 Edition, with Winter 1976 Addenda, and Code Case N-224-1. All liner materials are tested and certified in accordance with ASME Boiler and Pressure Code Section II, 1971 Edition through Summer 1973 Addenda. The steel liner is fabricated (but not stamped) in accordance with ASME Section III, Subsection NE. Exceptions are discussed in Section 3.8.1.6.3.

Material selected for construction of the liner is ferritic steel with appropriate ductility. All ferritic materials forming part of the primary containment boundary are Charpy V-notch tested at -10°F (except bolting at +40°F) and conform to ASME Section III, Subarticle NE-2300. The welded attachments to the liner are impact-tested in accordance with ASME Section III, 1974 Edition, with Winter 1976 Addenda, Subarticle NE-2300. All ferritic materials that form part of the pressure boundary and are more than 5/8 in thick, except as noted below under access locks, are also dropweight tested in accordance with Section 15 of ASTM E208 at a set temperature of 10°F or less to verify that the NDTT is equal to or less than 0°F.

All welding procedures and tests required by ASME Section IX are adhered to for the selection of weld filler material, heat treatment, the performance of welding machines, and the qualifying of welding procedures and Welding Operators who construct the primary containment liner, except as noted in Table 1.8-1, RG 1.19, Item 4. The welding qualification includes 180-deg bend tests of weld material. These procedures ensure that the ductility of welded seams will be comparable to the ductility of the containment liner plate material.

Liner surfaces exposed to the suppression chamber atmosphere are made from the following materials: stainless steel plate for embedments, carbon steel plate clad with stainless steel for the liner, and carbon steel plate overlaid with stainless steel sheet for the knuckle. All carbon steel weld seams in the suppression pool wall liner are weld overlaid with stainless steel. Stainless steel plate is furnished by the supplier to the requirements of SA-240 Type 304L in the solution-annealed condition.

Surfaces of the liner exposed to the drywell atmosphere inside the containment are coated in accordance with RG 1.54. The backside of the liner in contact with the concrete has adequate corrosion protection and is not painted.

The primary containment wall liner plate and suppression pool floor liner plate material is SA-537 Class 2, quenched and tempered with a specified minimum tensile strength of 80,000 psi, a minimum guaranteed yield strength of 60,000 psi, and a guaranteed minimum elongation of 22 percent in a standard 2-in specimen. The wall liner has a 3/8-in nominal base thickness, and the floor liner is 1/4 in thick. The wall liner in the suppression pool is SA-537 Class 2 that has been roll-bonded under heat and pressure with stainless steel clad plate, or has been wallpapered with stainless steel to ensure adequate protection from corrosion. All cladding is SA-240 Type 304L stainless steel made to SA-264, and is 20 percent of nominal base metal thickness with a 0.0375-in minimum thickness and a bond shear strength of 30,000 psi. All wallpapered cladding is 1/8-in thick SA-240 Type 304 stainless steel. Thickened embedment plates inserted into the liner which transfer structural equipment loads to the reinforced concrete in the drywell are 1 1/2-in thick SA-537 Class 2 Lukens Lectrifine to enhance the cross-transverse (Z-axis) properties. Insert plates in the suppression pool are 1 1/2-in thick SA-240 Type 304L stainless steel. Liner cone transitions are thickened plates of 1 1/4-in thick SA-537 Class 2 material. The primary containment liner plates were ordered to conform to mill practice with regard to thickness and tolerance in accordance with ASTM-A20.

The construction tolerances specified for the liner are:

1. The maximum difference in cross-sectional diameters of the steel liner shall be in accordance with ASME Section III, Subparagraph NE-4221.1.

2. The out-of-round tolerance measured at the top of the knuckle or starter plate shall not exceed ±0.25 percent of the nominal inside diameter. The top of the knuckle plate shall be level within ±3/8 in.

3. The maximum deviation from a straight line or from circular form measured anywhere in the liner shall not exceed ±1/4 in in a 14-in span.

4. The cylindrical portion of the liner shall be plumb within 2 3/4 in at any height of the liner measured from an established vertical line extending up from the base of the mat liner. For conical sections, the liner shall be within 2 3/4 in of basic as a function of elevation.

5. The maximum misalignment between liner plates shall be in accordance with ASME Section III, Paragraph NE-4232.

Nondestructive testing for the steel liner was performed as required by RG 1.19 (or acceptable alternative as described in Table 1.8-1) or the applicable code for noncode-stamped components. Procedures for the steel liner were in accordance with the requirements of ASME Section V, 1971 Edition through Summer 1973 Addenda, except that radiographic examination was also performed to the requirements of the 1977 Edition through the Summer 1978 Addenda. Acceptance standards for radiographic examination are in accordance with ASME Section III, 1971 Edition through Summer 1973 Addenda, or 1977 Edition through Summer 1978 Addenda. For magnetic particle and liquid penetrant examination, the evaluation of indications and the acceptance standards are in accordance with ASME Section III, 1971 Edition through Summer 1973 Addenda. For these examinations conducted after November 1, 1978, the evaluation of indications and acceptance standards is in accordance with Paragraphs NE-5341, NE-5342, NE-5351, and NE-5352 of ASME Section III, 1977 Edition, and Code Case N-339. Acceptance requirements for all other nondestructive testing procedures are in accordance with ASME Section III, 1971 Edition through Summer 1973 Addenda, Subarticle NE-5300.

Leak-tightness testing of all wall and floor liner seam welds was completed during construction. When leakage was detected, repair welding was performed and the weld was retested by the same methods applied to the original weld.

The Cadweld sleeves welded to the liner to accommodate the drywell floor and pedestal interfaces (Section 3.8.1.1.2) were qualified by sister splices which were tensile-tested to failure. Cadweld sister splices were tested to substantiate the structural integrity of the Cadweld sleeve-to-plate joint by demonstrating that failure occurred each time in the rebar. A sister splice plate is composed of material, thickness, and configuration identical to the production piece.

Anchor Studs

Anchor stud material is ASTM A108 Grades 1010-1020, with a minimum tensile strength of 60,000 psi. The anchor studs are welded as an ASME Section IX P-1, Group 1 material. Welding of anchor studs is not performed when the base metal temperature is below 50°F. The areas to which the studs are to be welded are brushed or ground free of scale or rust.

The normal tolerance for locating the centerline of a concrete anchor stud on the 3/8-in liner is 1 1/2 in horizontally and meridionally. Additional anchor studs are added to the liner if interferences exist with reinforcing steel, and are documented to assure that they do not exceed the minimum spacing requirements. An anchor stud may be bent up to 10 deg to avoid reinforcing steel. However, an anchor stud may be bent up to 20 deg in order to avoid interference with reinforcing steel if no other anchor stud within an 18-in radius is bent more than 10 deg. Any anchor stud that is not acceptable by the preceding criteria will be replaced by locating an additional anchor stud within a 3 1/2-in radius.

For each shift, stud welders were qualified by successfully welding two studs to a sister plate. After the weld cooled, each stud was bent 30 deg by striking the stud with a hammer. If failure occurred in the weld of either stud, the procedure was corrected and two successive studs were successfully welded and tested before work proceeded. Sister plates were of material similar to that of the production piece and had the same thickness as the production piece, except that a 1-in sister plate was acceptable for production items larger than 1 in. All studs were visually inspected. Studs on which a full 360-deg weld flash was not obtained were replaced or repaired as a minimum for compliance to AWS D1.1.

Access Hatches and Locks

The materials used in the construction of the access locks and hatches are in accordance with ASME Section III, Division 1, 1971 Edition through Summer 1973 Addenda, and are tested and certified in accordance with Section II of the above code.

The access hatches, personnel airlock, and escape lock are fabricated but not stamped in accordance with ASME Section III, Subsection NE.

The major components (panel, cover, and flanges) for the suppression chamber hatches are built with SA-240 Type 304L stainless steel with a minimum tensile strength of 70,000 psi and a minimum yield strength of 25,000 psi. The major components (panel, cover, and flanges) of the equipment hatch, equipment hatch/personnel airlock, escape airlock, and CRD hatch are built with SA-516 Grade 70 with a minimum tensile strength of 70,000 psi, and a minimum yield strength of 38,000 psi. SA-516 Grade 70 material 2 1/2 in and larger is used in the hatch closer joints in the quench and tempered condition without dropweight tests. Bolting material is SA-193 Grade B7 and the O-ring is material ethylene propylene-diene-monomer (EPDM). The plates conform to standard mill practice with regard to thickness and tolerance in accordance with ASTM A-20.

All welding, welding procedure qualification, and welder qualifications are in accordance with ASME Section III. The methods of nondestructive examination of access opening welds are in accordance with ASME Section III, Subarticle NE-5200. Nondestructive examinations are in accordance with RG 1.19 (except as noted in Table 1.8-1, RG 1.19, Item 5) and are qualified in accordance with the methods and techniques described in ASME Section III, Division 1, Appendix X, 1971 Edition through

Summer 1973 Addenda with acceptance requirements in accordance with ASME Section III, Subarticle NE-5300.

Drywell Head

The drywell head is fabricated to the requirements of ASME Section III, Division 1, 1971 Edition through Summer 1973 Addenda. The torispherical head is built with SA-516 Grade 70 with a minimum tensile strength of 70,000 psi, minimum yield strength of 38,000 psi, and a guaranteed minimum elongation of 21 percent in a standard 2-in specimen. The closure flange is SA-240 Type 304. The closure pin material is SA-564 Grade 630 with Code Case 1388-2. The O-ring material is Ethylene-Propylene-Diene-Monomer (EPDM).

All welding procedures for the fabrication of the drywell head are in accordance with ASME Section III, Division 1, Class MC requirements. Nondestructive testing procedures are qualified in accordance with the methods and techniques as described in ASME Section III, Division 1, Appendix X, 1971 Edition through Summer 1973 Addenda with acceptance criteria according to ASME Section III, Subarticle NE-5300.

The elevation tolerance of the drywell head flange is +2 to 0 in and the levelness tolerance is 1/8 in. Out-of-roundness does not exceed 0.75 in. Variation in closure circumference does not exceed ±0.375 in. The centerline of the installed closure joint with respect to the containment centerline is concentric within a 2-in diameter circle. The maximum gap between the sealing surfaces without the O-rings in place does not exceed 0.031 in after shop fabrication and 0.051 in after field installation.

Penetrations

The penetration pipe sleeves (Class MC boundary) are fabricated (but not stamped) in accordance with ASME Section III, Subsection NE except for Paragraphs NE-4121 and NE-4125 and Subparagraph NE-4621.1. The process pipes are fabricated and stamped in accordance with ASME Section III, Subsection NB or NC as applicable.

The materials used for the penetration assemblies are discussed below.

Hot Penetrations (Figure 3.8-10) All carbon steel integral flued head forgings forming a portion of a piping system are either SA-105 Grade 2, or SA-508 Class 1, including ASME Code Case 1332-6, except that the integral forgings for the feedwater system are SA-508 Class 2. Stainless steel flued head forgings for applicable piping systems are SA-182 Type F304L. The carbon steel sleeves in the drywell assemblies are SA-333 Grade 6 and SA-106 Grade B, except for the main steam and feedwater sleeves which are SA-155 Grade CMSH80. Sleeves for the stainless steel forgings are SA-312 Type 304L and SA-240 Type 304L for seamless pipe up to 10 in in diameter, and are SA-312 Type 304 for welded pipes over 10 in in diameter.

The ASME Section III Safety Class 1 and 2 penetration pipe ferritic forging with process pipe regions thicker than 5/8 in will be Charpy V-notch toughness tested for acceptance at +40°F and dropweight tested at 10°F in accordance with ASME Section III Subarticle NB-2300.

All welding, welding procedure qualifications, and welder qualifications for ASME Section III piping Safety Class 1 or 2 components are in accordance with the requirements of ASME Section III. For fabrications of the Class MC portions, the welding requirements are the same as those for the steel plate liner. The methods of nondestructive examination of Safety Class 1 and 2 piping penetration welds are in accordance with ASME Section III Subarticles NB-5200 and NC-5200, respectively. Nondestructive testing procedures are qualified in accordance with ASME Section V with acceptance requirements to Subarticle NE-5300 for the Class MC items.

<u>Thermally Cold Penetrations (Figure 3.8-10)</u> The carbon steel service pipe forming a portion of the piping system is SA-333 Grade 6 or SA-106 Grade B, fine-grained and normalized. Stainless steel service piping for applicable piping systems is SA-312 Type 304L or 316L for pipe sizes up to 10 in diameter and SA-312 Type 304 for pipe sizes over 10 in diameter.

Carbon steel process pipes with thickness greater than 5/8 in are impact tested and dropweight tested as described above under thermally hot penetrations.

<u>CRD and Instrument Penetrations (Figure 3.8-9)</u> The inner pipe forming a portion of the piping system pressure boundary is SA-312 Type 304L. The forged adapter which also forms part of the piping system boundary in addition to attaching the inner pipe to the containment sleeve is SA-182 Type 304L. The containment sleeves in the suppression chamber are SA-312 Type 304L. The containment sleeves in the drywell are SA-333 Grade 6. The requirements of welding and testing (examination) of these items are the same as those described under Thermally Hot Penetrations.

<u>Electrical Penetration Sleeves (Figure 3.8-9)</u> The material for the electrical penetrations in the drywell is SA-333 Grade 6, fine-grained and normalized. The stainless steel electrical penetrations in the suppression chamber are SA-312 Type 304L. The weld neck flange that accommodates the bolted electrical insert is SA-105 in the drywell and SA-182 Type 304L in the suppression chamber.

Primary containment liner reinforcement plates in the drywell are SA-537 Class 2 material. Liner reinforcement plates in the suppression chamber are SA-240 Type 304L.

The meridional slope and the circumferential slope for the penetrations are specified as ±1/2 in at the end of the penetration farthest from the plane of the reinforcement plate-to-penetration weld. The specified slope tolerances for the main steam and feedwater penetrations are ±1 in. Tolerances for service pipe and sleeve wall thicknesses for forged penetrations are in accordance with ASTM A530 except that the maximum thickness is more tightly controlled to accommodate machining operations and to limit eccentricity between the outside and inside diameters.

3.8.1.6.3 Exceptions and/or Clarifications to ASME Code

The exceptions and/or clarifications to the ASME Boiler and Pressure Vessel Code, Division 1, 1971 Edition through Summer 1973 Addenda, are as follows for noncode-stamped items on the containment:

NA-1120 Definition of Nuclear Power System Components and Containment Vessels

NA-3500 Responsibilities of Inspection Agencies Engineering Specialists and Inspectors

NA-4000 Quality Assurance

NA-5000 Inspection

NA-8000 Name Plates, Stamping, and Reports

NE-1110 Aspects of Construction Covered by These Rules

NE-1120 Rules for Class MC Containment Vessels

NE-4121 Means of Certification - NE-4121 has a requirement for the application of the code symbol to the containment structure, but if the containment liner is backed by concrete, it does not qualify as a MC vessel.

NE-4125 Testing of Welding Materials - NE-4125 has a requirement that defines a "lot" of weld material based on heats of metal or dry batch. Code Cases 1567 and 1568 were invoked to define a "lot" of weld material to be based on time and quantity of production runs.

NE-4311 Studwelding Restrictions

NE-4322 Maintenance and Certification of Records

NE-4421 Backing Rings

NE-4429 Welding of Clad Parts - This section requires that any weld-deposited cladding be examined by a liquid penetrant method. In lieu of 100-percent liquid penetrant examination, some categories of weld-deposited cladding on the suppression chamber lower knuckle were visually examined. A small portion of the cladding in one of these categories was inaccessible for visual examination and is accepted based on the satisfactory visual examination results for the accessible cladding.

NE-4621.1 Materials Exempted from Post-Weld Heat Treatment - NE-4621.1 contains a requirement in Subparagraph NE-4621.1(b) to postweld heat treat all subassemblies prior to insertion into the shell. Subparagraph NE-4622.1, which exempts weldments up to 1 1/2-in thick (embedments) from postweld heat treatment, is used in place of Subparagraph NE-4621.1(b) since the liner insert portion of the embedment is tapered down to the liner thickness.

NE-5211 Category A and B Welds

NE-5212 Examination of Embedded Welds of Vessels

NE-5231 Butt Welds - This section requires that for butt welds in nonpressure-retaining structural parts attached to pressure-retaining parts, the portion of the butt weld within 16t, where t is the thickness of the structural part, shall be examined by a radiographic method in accordance with Appendix X. In lieu of the radiographic examination in butt welds within this 16t, the welds will be examined by an ultrasonic method, a magnetic particle method, or a liquid penetrant method all in accordance with ASME V.

NE-5232 Nonbutt Welds in Non-Category Joints - This section requires that for nonbutt welds in nonpressure-retaining structural parts attached to pressure-retaining parts, the nonbutt welds within the limit of 16t, where t is the thickness of the structural part, shall be examined by an ultrasonic method, a magnetic particle method, or a liquid penetrant method, all in accordance with Appendix X. The nonbutt weld of the structural part to the pressure-retaining part will be examined in accordance with the requirements of NE-5232 using the methods of ASME V in lieu of Appendix X. The nonbutt welds in nonpressure-retaining structural parts within the limit of 16t will be examined in accordance with the governing structural fabrication code requirements.

NE-5522 Verification by Inspector of Personnel Qualification

NE-5610 Vessel Materials - Examination

NE-6000 Testing

NE-8000 Nameplates, Stamping, and Reports

The exceptions and/or clarifications to the ASME Boiler and Pressure Vessel Code, Division 1, 1974 Edition through Winter 1976 Addenda, are as follows for noncode-stamped attachments to the primary containment liner:

NA-1120 Definition of Nuclear Power System Items

NA-3500 Responsibilities of Inspection Agencies, Inspection Specialists, and Inspectors

NA-4000 Quality Assurance

NA-5000 Inspection

NA-8000 Certificates of Authorization, Nameplates, Stamping, and Reports

NE-1110 Aspects of Construction Covered by These Rules

NE-1120 Rules of Class MC Containment Vessels

NE-4121 Means of Certification - NE-4121 has a requirement for the application of the code symbol to the containment structure, but the containment liner is backed by concrete. It does not qualify as a MC vessel.

NE-4125 Testing of Welding Materials - NE-4125 refers to NE-2400 which has a requirement that defines a "lot" of weld material based on heats of metal or dry batch. Code Cases 1567 and 1568 were invoked to define a "lot" of weld material to be based on time and quantity of production runs.

NE-4311.1 Studwelding Restrictions

NE-4322 Maintenance and Certification of Records

NE-4421 Backing Rings

NE-5210 Category A Welds

NE-5220 Category B Welds

NE-5261 Butt Welds - This section requires that for butt welds in nonpressure-retaining structural parts attached to pressure-retaining parts, the portion of the butt weld within 16t, where t is the thickness of the structural part, shall be examined by a radiographic method. In lieu of the radiographic examination in butt welds within this 16t, the welds will be examined by an ultrasonic method, a magnetic particle method, or a liquid penetrant method, all in accordance with ASME V.

NE-5262 Nonbutt Welds in Non-Category Joints - This section requires that for nonbutt welds in nonpressure-retaining structural parts attached to pressure-retaining parts, the nonbutt weld within the limit of 16t, where t is the thickness of the structural part, shall be examined by an ultrasonic method, a magnetic particle method, or a liquid penetrant method, all in accordance with ASME V. The nonbutt weld of the structural part to the pressure-retaining part will be examined in accordance with the requirements of NE-5262. The nonbutt welds in nonpressure-retaining structural parts within the limit of 16t will be examined in accordance with the governing structural fabrication code requirements.

NE-5522 Verification by Inspector

NE-5610 Vessel Materials

NE-6000 Testing

NE-8000 Nameplates, Stamping, and Reports

The above exceptions/clarifications, for the most part, reflect the fact that the containment liner, being backed up by concrete, does not qualify as an ASME Section III Class MC vessel and the examination requirements of RG 1.19 apply. Therefore, it could not be stamped MC and thus the portions of the code that address definitions, stamping, third party inspection, and data reports do not apply.

The exceptions and/or clarifications to the ASME Boiler and Pressure Vessel Code Section III, Division 1, 1974 Edition through Winter 1976 Addenda, for the installation of code-stamped electrical penetrations mechanically attached to a noncode-stamped item on the containment liner are as follows:

NA-1280 - Installation

NA-3300 - Responsibilities of an N Certificate Holder

NA-3400 - Responsibilities of an NPT Certificate Holder

NA-3500 - Responsibilities of an NA Certificate Holder

NA-4000 - Quality Assurance

NA-5000 - Inspection

NA-8000 - Certificates of Authorization Nameplates, Stamping, and Reports

NE-8000 - Nameplates, Stamping, and Reports

The electrical penetrations have been designed, manufactured, and shop-tested in accordance with the applicable portions of the ASME Boiler and Pressure Vessel Code Section III, Division 1, 1974 Edition through Winter 1976 Addenda.

The electrical penetrations are installed, inspected, and field-tested in accordance with SWEC Quality Standards (QA and EA procedures relating to Category I installation of safety-related equipment) which conform to 10CFR50 Appendix B.

Acceptability of the installation of the electrical penetrations will be based on a field-performed leak rate test as described in Section 8.3 of IEEE-317-1976.

3.8.1.7 Testing and In-service Inspection

3.8.1.7.1 Concrete Containment

Primary Containment Test

The primary containment is subjected to a structural acceptance test in which it is pressurized internally to 1.15 times the 45 psi design pressure. It is pressurized in increments of 10 psi from atmospheric pressure to its peak value of 51.75 psi and then depressurized in a similar manner to atmospheric pressure. At the end of each 10 psi step change, the pressure is held for at least 1 hr. Strain and deflection measurements are recorded and compared to predicted values.

The structural acceptance test for primary containment equals or exceeds the requirements of RG 1.18 for nonprototype containments. The strain measurements are made by using strain gauges that have been installed at two azimuths on the inside and outside meridional and hoop rebars at various elevations. The radial displacements of the containment are measured using direct current differential transducers (DCDTs) at several points along six meridians spaced around the containment. In addition to this, additional strain gauges and DCDTs are mounted around the personnel hatch to record the strains and displacements, respectively. These measurements are compared to the predicted values which are shown in Table 3.8-15. The predicted response of the structure is based on the analysis procedures described in Section 3.8.1.4. The measured maximum deflections at points of maximum predicted deflection are acceptable if they do not exceed the predicted values by more than 30 percent. This requirement may be waived if the deflection recovery within 24 hr after complete depressurization is greater than 80 percent. These tolerances are based on the tolerances allowed in ASME III, Division 2. Crack patterns exceeding 0.01 in in width before, during, or after the test are mapped.

Drywell Floor Test

The drywell floor is subjected to the design differential pressure of 25 psi. The structural adequacy of the floor will be determined by a visual inspection for signs of permanent damage to the concrete after return to the atmospheric pressure. The response of the containment is measured during the test using procedures similar to the ones used for the primary containment test.

3.8.1.7.2 Steel Liner and Penetrations

Steel Liner

The primary containment liner is required to withstand the pressure of 115 percent of the design pressure by the acceptance test described in Section 3.8.1.7.1. The primary containment leakage rate test (Section 6.2.6) is conducted after the structural acceptance test in compliance with the requirements of Appendix J of 10CFR50. Test channels are provided for all seams covered by concrete or otherwise made inaccessible after completion of construction and all basic liner seam welds in the suppression chamber. This will enable leak-tightness testing of these local areas to be performed at any time during the life of the plant should the need arise.

Access Hatches and Drywell Head

The personnel airlocks and the escape airlocks were subject to a shop acceptance test in accordance with Article NE-6000 of ASME Section III, Division 1. They were also subject to a shop pressure decay test. The seals for the airlocks were subject to a halogen leak test, and the seals for the drywell head were subject to a pressure decay test. The seals for the drywell hatches were subject to a halogen leak test, and the stainless steel suppression pool hatch seals were pressure decay tested. Test channels are provided for the convenience of leak testing during construction for all containment hatch-to-liner reinforcement plate welds. This also will enable leak-tightness testing of these weld joints to be performed at any time during the life of the plant should the need arise.

Penetrations

All penetrations were subject to a shop hydrostatic acceptance test in accordance with NB-6000 or NC-6000, as applicable, of ASME III, Division 1, 1971 Edition through Summer 1973 Addenda.

Additionally, penetrations Z-32 (neutron monitoring - GSN), Z-46A and Z-47 (closed loop cooling - CCP), Z-46C (fire protection - FPW), Z-99A through D and Z100A through D (reactor core cooling hydraulic system - RCS) were hydrostatically tested after installation in accordance with Paragraph NC-6129 of ASME III, 1977 Edition, Winter 1978 Addenda.

Test channels are provided only for all containment penetration insert-to-liner reinforcement plate welds which are welded from one side with a backing strip to enable leak-tightness testing of these weld joints to be performed at any time during the life of the plant should the need arise. The integral flued head forging design of the thermally hot piping penetration permits ISI of the process pipe to transition region. The ends of the piping forging are dimensioned to facilitate ISI of the process pipe welds.

3.8.2 Not Applicable

3.8.3 Concrete and Steel Internal Structures of the Primary Containment

3.8.3.1 Description of Internal Structures

The internal structures support the RPV, provide shielding, and form the pressure suppression system. The internal structures include the following:

1. Drywell floor.

2. Reactor vessel pedestal.

3. Biological shield wall.

4. Star truss.

5. Floors.

Steel linear supports for RCS systems are described in Section 3.9.3.4A.

The containment internal structures are Category I and are shown on Figure 3.8-11. The BSW is a composite structure of pressure vessel quality steel and heavy-density fill material. The drywell floor and pedestal are heavily reinforced concrete structures. The concrete structures are designed for DBA conditions and provide radiation shielding. The DBA radiation source terms have been evaluated and do not adversely affect these structures (Section 12.3). The star truss is constructed and designed to withstand DBA conditions, including temperature effects, along with seismic and other applicable loads.

3.8.3.1.1 Drywell Floor

The drywell floor (Figures 3.8-12 and 3.8-13) is a reinforced concrete annular slab, 4 ft thick, having an inner diameter of 30 ft supported by the pedestal, and an outer diameter of 91 ft supported by the primary containment wall.

The drywell floor serves both as a pressure barrier between the drywell and suppression chamber and as the lateral support structure for the reactor pedestal and anchor support for the downcomers.

The drywell floor is rigidly connected to the containment wall thereby preventing relative motion between the two structures. This alternative approach is used as a substitute for the requirement of a drywell floor seal. A full moment and shear connection is provided by cadwelding the reinforcing bars to the reinforced liner plate (Figure 3.8-13). Thermal expansion is considered in the containment design, and the resulting forces and moments on the floor are accommodated within the allowable stress limits.

The drywell floor at el 240 ft is penetrated by 118 24-in diameter stainless steel downcomer pipes, and four 24-in diameter truncated downcomers for vacuum breakers. A description of the downcomer vents is given in Section 6.2.1. Figure 3.8-12 shows the arrangement of the downcomers. The drywell floor is also penetrated by 18 12-in diameter SRV lines, 18 2 1/2-in diameter SRV vent lines, and 20 4-in diameter floor drains. SRV line penetrations 12 in in diameter are flued head-type penetrations as shown on Figure 3.8-13. SRV vent line penetrations 2 1/2 in in diameter are collar-type penetrations similar to instrument penetrations as shown on Figure 3.8-9.

The seal for the drywell slab is provided by a 3/16-in thick liner that is placed on top of the floor slab (Figure 3.8-13). A minimum of 6-in thick reinforced concrete insulation slab, sloped for drainage, is placed on the liner.

The cavity floor slab, located at el 232 ft, is 3 ft thick and is within the pedestal shell walls. This slab is an extension of the drywell floor but is 8 ft lower than the drywell floor to avoid interference with the CRD assemblies. The slab is penetrated by eight downcomer pipes and one floor drain. A full moment and shear connection is provided between the slab and pedestal. Similar to the drywell floor, this slab provides a pressure barrier between the drywell and suppression chamber.

3.8.3.1.2 Reactor Vessel Pedestal

The reactor vessel pedestal is a 91-ft 6 1/2-in high, heavily reinforced concrete right vertical cylindrical shell with a 20-ft 3-in inside diameter and a thickness that varies from 4 ft at the base to 5 ft 1 1/2 in above the drywell floor. The wall thickness increases to 7 ft 9 1/2 in at the top portion which supports the RPV and the BSW. At lower elevations the pedestal also supports the drywell floor, the cavity floor slab, various pipe rupture restraints, radial beams, and pipe supports. The pedestal is located concentric with the RPV centerline and is supported on the reactor building foundation mat (Figure 3.8-14). A full moment and shear connection at the junction of the pedestal and mat is provided by reinforcement extending into the mat.

The top of the pedestal contains embedded anchor bolts for anchorage of the RPV and BSW. The reactor vessel is anchored by two 3-in diameter bolts at every 6 deg of pedestal circumference; the BSW is anchored by two 2 1/4-in diameter bolts at approximately every 5 deg of pedestal circumference. All pedestal concrete surfaces in the suppression chamber are lined with a 1/4-in stainless steel liner plate. The lined surface includes the bottom of the cavity floor on the inside of the pedestal and extends to the drywell floor on the outside of the pedestal. The pedestal liner plate is anchored to the concrete by headed concrete anchor studs.

3.8.3.1.3 Biological Shield Wall

The BSW consists of two concentric steel cylinders connected by internal horizontal and vertical stiffeners (Figure 3.8-15). The shield wall is 48 ft 4 in high and has an inner radius of 14 ft 3/4 in and an outer radius of 15 ft 9 1/4 in. The shield wall is supported by the reactor pedestal and is attached to the pedestal by embedded anchor bolts (Figure 3.8-15).

The inner and outer walls and the stiffeners are 1 1/2 in thick, ASTM A537 Class 1 steel plates connected by full penetration welds. The space between the walls is filled with nonstructural heavy-density fill material (a combination of cement and iron ore) for radiation shielding purposes. The shield wall is penetrated by air duct openings, inspection and access openings, instrumentation lines, and piping penetrations for various systems. Attached to the BSW are pipe restraints, star truss and stabilizer supports, insulation support brackets, and miscellaneous supports for structural steel floor beams. The BSW protects the RPV from pipe whip, jet impingement, and missile loads, and protects drywell structures from the effects of recirculation and feedwater line breaks in the annular region.

3.8.3.1.4 Star Truss

The star truss (Figure 3.8-16) is a tubular structure spanning between the BSW and the primary containment wall (Figure 3.8-16), whose function is to provide lateral support to the BSW, which in turn provides lateral support to the RPV. It is welded to the ring girder extension on top of the BSW at six equally spaced, circumferential locations. This welded connection provides a fixed support for the star truss extending to the primary containment wall. The star truss connection at the primary containment wall allows unrestrained radial and vertical translation but prevents tangential translation.

3.8.3.1.5 Floors

Floors are located within the containment to provide support for and access to equipment. The floors generally are constructed of steel framing with steel grating or checkered plate decks. The steel grating or checkered plate flooring is supported by steel beams which, in turn, are supported by radial or nonradial steel girders spanning from the reactor pedestal or the BSW to the primary containment wall. This floor framing system is braced against lateral movement. The steel girders are simply supported at the primary containment wall by beam seat arrangements and are pin connected to the embedment plates at the reactor pedestal end.

3.8.3.2 Applicable Codes, Standards, and Specifications

The design codes, standards, specifications, and regulations used in the design, procurement, fabrication, and construction of the steel and concrete containment internal structures, meet or exceed the requirements outlined in Section 3.8.4.2 for the design of Category I structures.

The procurement specification requirements for concrete, reinforcing steel, and other materials used in the drywell floor and the RPV pedestal are described in Section 3.8.4.2. The summary of the procurement and erection specifications for the BSW and the star truss are given in Sections 3.8.3.2.1 and 3.8.3.2.2, respectively.

3.8.3.2.1 Biological Shield Wall

Design and construction criteria for the BSW are:

1. <u>Design Stresses</u> AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, 1969.

2. <u>Quality Control</u> Testing, inspection, and documentation, ANSI N45.2.5 and 10CFR50 Appendix B.

3. <u>Welder Qualification</u> ASME Section IX or AWS D1.1.

4. Welding AWS D1.1 with the exception to Section 3.3 that an offset of 1/2 in maximum is permitted.

5. NDE Examination for Welding Ultrasonic inspection or progressive magnetic particle method of inspection.

6. Packaging, Shipping, Storage, and Handling of Materials ANSI N45.2.2 and RG 1.38.

7. Traceability Full traceability for inner and outer wall plates, stiffeners, base plate, sole plate, doors, and bolts over 1 in diameter.

8. Painting RG 1.54.

9. Erection AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, 1969, and AWS D1.1.

3.8.3.2.2 Star Truss

Design and construction criteria for the star truss are:

1. Design Stresses ASME Section III, Subsection NF, 1977, unstamped.

2. Quality Control Testing, inspection, and documentation - 10CFR50 Appendix B, ASME Section III, Subsection NF.

3. Fabrication and Welding ASME Section III, Subsection NF, unstamped, AWS D1.1.

4. NDE Examination for Welding ASME Section III, Subsection NF, ASME Section V, progressive magnetic particle inspection.

5. Packaging, Shipping, Storage, and Handling of Materials ANSI N45.2.2 and RG 1.38.

6. Painting RG 1.54.

7. Erection AWS D1.1 and ASME Section III, Subsection NF, unstamped.

3.8.3.3 Loads and Loading Combinations

3.8.3.3.1 Drywell Floor and Reactor Vessel Pedestal

The loads imposed on these concrete structures, the notations used, and the load combinations are the same as those listed in Section 3.8.1.3 for concrete structures.

3.8.3.3.2 Biological Shield Wall

For normal operating loading combinations, the design stresses for the BSW are in accordance with the AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings; however, for abnormal and/or extreme environmental design load combinations, the allowable elastic stresses are increased in accordance with factored (e.g., 1.6, 1.8, 2.0) sectional strengths as shown in load combination equations 3 through 6 of Table 3.8-9.

The BSW is analyzed and designed for a variety of individual pipe rupture forces and associated pressures. Each section of the BSW is designed to satisfy the maximum pipe loading condition for that area.

The load combination equations used for the design of the shield wall are presented in Table 3.8-9. The BSW is also designed for hydrodynamic load combinations listed in Table 3.8-10, Part II.

3.8.3.3.3 Star Truss

The star truss loads, notations, and load combinations are the same as those listed in Section 3.8.4.3 for steel structures. Additionally, the star truss is checked for the effect of SRV loading resulting from suppression pool hydrodynamics, in addition to other concurrent loading as listed in Table 3.8-10, Part II.

3.8.3.4 Design and Analysis Procedures

3.8.3.4.1 Drywell Floor

The drywell floor is analyzed and designed for the load combinations outlined in Section 3.8.4.3. The analysis procedure used is outlined in Section 3.8.3.4.2. The following effects of a potential LOCA occurring within the drywell are considered in the design:

1. Jet impingement forces on the drywell floor.

2. Impact loads transmitted to the drywell floor by any attached pipe whip restraints.

3. Differential pressure across the drywell floor of:

    a. 25 psi (drywell pressure greater than suppression chamber).

    b. 10 psi (suppression chamber pressure greater than drywell).

Additionally, the drywell floor is designed for dead load, seismic, thermal, and hydrodynamic loading (Appendix 6A).

All loads appropriately combined will not compromise the intended function of the drywell floor to provide a barrier between the drywell and suppression chamber. Stress resultants are calculated using elastic methods in accordance with ACI-318 and the AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, as applicable. Reinforcement arrangements in the drywell floor are shown on Figure 3.8-13.

3.8.3.4.2 Reactor Pedestal

The pedestal is analyzed and designed for the load combinations outlined in Section 3.8.4.3. Inertia loadings from earthquakes are obtained from the dynamic analyses of the reactor building, as outlined in Section 3.7A.2.

The reactor pedestal and drywell floor are analyzed using SHELL 1, a finite element computer program (Appendix 3A). The structures are modeled using axisymmetric shell elements. The asymmetric loadings (e.g., seismic loads) are represented by a series of Fourier coefficients.

The boundary conditions for the pedestal are considered to be fixed at the junction with the foundation mat. For the boundary conditions at the top of the pedestal, forces at the base of the BSW and RPV resulting from the loads defined in Sections 3.8.3.3.2 and 3.8.4.3, respectively, were applied to the SHELL 1 model. The boundary conditions for the drywell floor at the junction with the primary containment reflect the stiffness of the primary containment at the junction.

Design of the pedestal conforms to the requirements of ACI-318-71. Arrangements of reinforcement at the mat-pedestal and pedestal-BSW interface and in the pedestal wall are shown on Figure 3.8-14.

The top portion of the pedestal is designed to resist all seismic and pipe rupture forces transmitted from the reactor vessel skirt and the base of the BSW. Pipe rupture forces and discontinuity forces at the base of the BSW resulting from pressurization of the annulus between the reactor vessel and BSW during a recirculation or feedwater line break are used to analyze and design the pedestal in combination with the seismic forces determined from the dynamic analysis of the reactor building. In addition, the pedestal is analyzed and designed for jet impingement and pipe rupture loads.

To ensure that the pedestal cracking problem that occurred in the James A. FitzPatrick Nuclear Power Plant does not occur in the Unit 2 pedestal, the construction technique and the anchorage detail for the BSW are modified. Because of the BSW construction technique described in Section 3.8.3.6.2, the weld shrinkage and distortion usually encountered during fabrication and assembly have stabilized prior to the BSW's attachment to the reactor pedestal. In addition, the BSW is designed to allow radial movement due to thermal effects with limited effect on the reactor pedestal. Therefore, the reactor pedestal is adequately designed and will maintain its structural integrity for the effects of BSW loading.

3.8.3.4.3 Biological Shield Wall

The BSW is analyzed and designed for the load combinations described in Section 3.8.3.3.2. Analysis of the wall is performed using the two-dimensional finite element capability of the STRUDL computer code. Both plane stress and plate bending elements were used. As the structure is sufficiently symmetrical, only one-half of the structure (i.e., 180 deg) need be modeled. The boundary condition at the top of the structure approximates the effect of the star truss structure. The boundary condition at the bottom of the structure is considered an equivalent of springs connected to the pedestal in the circumferential and vertical directions and unrestrained in radial direction. The use of a 180-deg model allows for the analysis of asymmetric loadings by applying half of the load to the model with symmetric boundary conditions and half with asymmetric boundary conditions. The results of the two analyses are then superimposed for the net results. In addition, classical beam theory and plate and shell theory are used for the analysis and design of local areas. The following loads are considered in the analysis and design for LOCA effects:

1. Jet impingement forces on the BSW.

2. Impact loads transmitted to the BSW by any attached pipe rupture restraints.

3. Pressurization of the annulus between the BSW and reactor vessel.

4. Thermal effects.

These loads are combined in accordance with Section 3.8.3.3.2, taking account of the postulated failure locations and types. It has been concluded, because of the dynamic characteristics of the BSW, that the peak restraint impact loads are local impulsive loads on the BSW. These impact loads occur in the first milliseconds after rupture. The shield wall is allowed to yield locally at regions of impact loads provided that:

1. Overall capability of the shield wall to resist elastically the other forces listed is not affected.

2. Local yielding does not produce effects that jeopardize the safety of other components.

3.8.3.4.4 Star Truss

The star truss is a cantilevered plane truss that transmits horizontal force between the top of the BSW and the primary containment wall (Section 3.8.3.1.4). The truss joints at the BSW end are considered fixed. The support at the primary containment wall end is constrained in the circumferential direction so that only tangential horizontal force can be transmitted to the primary containment wall.

The star truss is analyzed using the STARDYNE computer program (Appendix 3A) and is designed for the load combinations listed in Section 3.8.3.3.

The complete structure is modeled using a combination of beam and plate elements for the finite element model. The beam part of the model represents the star truss part of the structure, while the plate elements represent the remaining parts of the structure.

The structure is designed using the STARDYNE computer program for each individual loading condition, and the stresses are calculated by superimposing the applicable loads with appropriate load factors for each load combination described in Section 3.8.3.3.

3.8.3.4.5 Floors

The structural steel framing system for floors within the primary containment are analyzed using the STRUDL computer program, as necessary, for the loads and load combinations outlined in Section 3.8.4.3. The floor framings are supported vertically at the containment end as well as at the reactor pedestal end. The connection at the pedestal end is treated as a pinned connection whereas the connection at the containment end provides sliding support to the framing members. The STRUDL computer program (Appendix 3A) is used to analyze the floors and platforms. The effects of dynamic loading, such as SRV, seismic, and LOCA, are considered while analyzing and designing these members. The design parameters, allowable stresses, and material properties, as described in Section 3.8.4, are selected in accordance with the requirements of the AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings.

3.8.3.5 Structural Acceptance Criteria

For concrete structures, the allowable stresses, load factors, and capacity reduction factors are in accordance with strength design methods of ACI-318-77, with the following exception: design for horizontal shear forces is in accordance with the requirements of ACI-318-71 which incorporates the combined effects of shear and tensile stresses into the nominal allowable shear stress, v_c, to be carried by the concrete.

Tangential shear stress (v_u) resulting from earthquake loading will be resisted by the concrete or by the concrete and steel reinforcing bars. The tangential shear stress (v_c) carried by the concrete is taken equal to one of the following values:

$v_c = 12{,}000\ p$ for $p < 0.01$, or

$v_c = 93 + 2{,}700\ p$ for $0.01 \leq p \leq 0.025$

Where:

p = Lesser of the steel-to-concrete ratios in the vertical or circumferential direction

v_c = Tangential shear stress, psi

For steel structures, the allowable stresses and safety factors are in accordance with the AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, with the following exceptions:

1. No credit is taken for the 33 1/3-percent increase in allowable stresses, S, permitted in the AISC Code when earthquake and wind loads are present in the load combinations.

2. The structural members of the star truss are designed (unstamped) in accordance with ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NF, 1977 Edition including Summer 1978 addenda. The allowable stress limits set forth in Appendix XVII of this code are used in designing the structures.

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

The construction materials and the quality control requirements used for the containment internal structures are the same as those used for other Category I structures, as described in Section 3.8.4.6. When necessary, the special materials and the special construction techniques are used as described in Sections 3.8.3.6.1 through 3.8.3.6.3.

3.8.3.6.1 Reactor Vessel Pedestal and Drywell Floor

Precast concrete beams and stainless steel beams are used as stay-in-place forms to support the drywell floor during construction. The support system for the drywell floor between the reactor vessel pedestal and primary containment at el 240 ft consists of 20 precast beams spanning radially. Stainless steel beams span between the precast beams. Inside the reactor vessel pedestal, the support system for the slab at el 232 ft consists of two precast concrete beams supporting stainless steel beams.

The liner on both sides of the reactor vessel pedestal in the suppression chamber serves as the form for pouring concrete. A regular form is used for the upper portion of the pedestal.

3.8.3.6.2 Biological Shield Wall

The BSW inner and outer walls, stiffeners, and base plate are manufactured in accordance with the ASTM A537 Class 1 specification. In addition, the stiffeners and base plate contain a maximum 0.01 percent sulfur to increase the through-thickness direction properties. The ASTM A537 Class 1 steel plates are 100-percent ultrasonically tested to the ASTM A578 Level I specification.

The BSW anchor bolts are manufactured to ASME SA-193 Grade B7 specification and are nondestructively examined by either magnetic particle or liquid penetrant methods in accordance with ASME Section III, Subparagraph NF-2581.1.

The BSW is filled with a heavy-density fill material that is installed using a pressure grouting technique. The technique consists of pumping heavy-density fill material at the base of the shield wall and allowing it to flow upward through the compartments to a predetermined lift height forcing any trapped air ahead of the fill material. This process continues circumferentially and vertically around the BSW until the entire structure is filled.

The BSW is shop fabricated in three rings, each approximately 16 ft high. Each ring is fabricated in three 120-deg sections. The sections are assembled at the site to form three 360-deg rings which are then stacked and welded together to form the BSW.

3.8.3.6.3 Star Truss

The material used in the star truss assembly is SA-537, Class 1, Lukens Lectrefine, except that the diagonal members of the truss system are tubular in cross section, conforming to SA-333 Grade 6. The ring girder at the top of the BSW is made of ASTM A537, Class 1, Lukens Lectrefine material. The truss connections are welded connections with full penetration welds.

3.8.3.7 Testing and In-service Surveillance Requirements

Drywell pressurization tests are performed in accordance with the requirements of Technical Specifications and Appendix J to 10CFR50. Testing of the construction materials referenced in this section is described in Section 3.8.4.6. A drywell floor structural test is performed and is described in Section 3.8.1.7.1.

3.8.4 Other Seismic Category I Structures

3.8.4.1 Description of the Structures

As shown in Table 3.2-1, other seismic Category I structures (e.g., diesel generator building, control building) that contain or support safety-related systems and/or equipment are designed to withstand the SSE. These structures, except for the main stack, are also designed to withstand tornado loads including tornado-generated missiles. Seismic loads are not considered to act simultaneously with tornado loads. Table 3.2-1 identifies seismic Category I equipment and structures that are tornado protected.

In general, seismic Category I structures, herein called Category I structures, are completely independent of adjacent structures. Adequate space is provided between structures to retain their independent functional characteristics, and to allow for rotation, translation, and deformation under seismic loading. The spaces have flexible seals (Figure 3.8-17). Structures having both Category I and nonseismic category elements are designed using the Category I criteria for the Category I portions of the structures. The Category I structures or portions of the structures are also investigated to determine the effect of failure of nonseismic category structures or portions of the structures. If it is determined that their failure (under Category I loading conditions) could endanger the integrity of the Category I portions of the structures, then the nonseismic structures or portions are also designed to Category I criteria. In these instances, where Category I structures are integrally connected to the other structures, the Category I structures are analyzed and designed considering the effect of the interconnection and modeled with the connecting structure(s) as a unit. Those portions of the Category I structures located below the DBFL are provided with waterstops (Figure 3.8-17). Vermiculite, vermiculite concrete, or compressible materials are used between the exterior face of substructure concrete walls and the excavated face of rock surfaces, as described in Section 2.5.4.10. No unique materials or features are used in the design or construction of the structures described in this section.

The relative locations of the Category I structures are shown on Figures 1.2-1 and 1.2-2. The general arrangement of operating personnel access between the buildings is shown on Figures 1.2-3 through 1.2-5. The general arrangements of the Category I buildings are illustrated in the following figures:

Reactor building including auxiliary bays, Figures 1.2-6 through 1.2-12

Auxiliary service building, Figures 1.2-7 and 1.2-8

Radwaste building, Figures 1.2-13 and 1.2-14

Standby gas treatment building, Figures 1.2-35 and 1.2-36

Control room building, Figures 1.2-15 and 1.2-16

Screenwell building including service water pump room, Figures 1.2-26 through 1.2-28

Intake tunnels, Figures 1.2-29 and 1.2-30

Main stack, Figure 1.2-31

Diesel generator building, Figures 1.2-17 and 1.2-18

Offgas rooms, Figures 1.2-19 through 1.2-25

Intake structure, Figure 1.2-29 and 1.2-30

The general arrangements of the major buildings that are not Category I are illustrated in the following figures:

Turbine building, Figures 1.2-19 through 1.2-25

Natural-draft cooling tower, Figures 1.2-38 and 1.2-39

Service building, Figure 1.2-19 and 1.2-22

Regeneration and condensate demineralizer regenerative rooms, Figures 1.2-19 through 1.2-25

Auxiliary boiler building, Figure 1.2-34

Normal switchgear building, Figures 1.2-32 and 1.2-33

Hydrogen storage area, Figure 1.2-40

CST building, Figure 1.2-37

The Category I structures other than the primary containment and its internal structural components are described in the following sections. For the primary containment and its internal structural components, refer to Sections 3.8.1 and 3.8.3, respectively.

3.8.4.1.1 Reactor Building

The reactor building completely encloses the reactor and the primary containment. The reactor building houses the refueling and reactor servicing equipment, new and spent fuel storage facilities, and other reactor auxiliary or service equipment, including the RCIC system, reactor water cleanup (RWCU) system, SLC system, CRD system equipment, core standby cooling systems, RHR systems, and electrical equipment components. The primary purposes for the secondary containment are to minimize ground level release of airborne radioactive materials and to provide means for a controlled elevated release of the building atmosphere if an accident should occur.

The reactor building wall is a 166-ft ID reinforced concrete cylinder with varying wall thickness, extending from the top of the mat at el 175 ft to the polar crane level at el 386 ft 10 in. The wall from the crane rail elevation to the roof at el 430 ft 1 in (approximately 40 ft) is steel framing with insulated metal siding. The metal siding panels have sealed joints to minimize air leakage.

Roof construction consists of steel trusses, which support metal decking, covered with insulation, and asphalt and gravel built-up roofing. The structural steel frame of the reactor building upper superstructure is designed to withstand the tornado wind with the metal siding remaining in place during a tornadic event. The reactor building roof decking is designed for normal wind loading. When the design velocity is appreciably exceeded, the decking may blow off. The reactor building metal siding is designed to withstand wind loads generated during a tornadic event.

Various floor levels form diaphragms within the secondary containment to provide support for and access to equipment. The floor and wall thicknesses meet both shielding and structural requirements. The reactor building superstructure floors and walls are entirely separated from the primary containment structure by a space that prevents the reactor building from restraining the primary containment under any condition.

The refueling floor level contains the fuel pool and is supported by the reactor building wall and the fuel pool girders.

The arrangement details of the reactor building wall and reinforced concrete floor levels are shown on Figures 1.2-6 through 1.2-12.

A portion of the main steam tunnel is an integral part of the reactor building between the primary containment and the reactor building wall.

The reactor building, including the auxiliary bays, is founded on a rock-bearing, reinforced concrete mat and is designed for the load combinations in Section 3.8.4.3. The mat acts to support the reactor building, auxiliary bays, and the primary containment. The auxiliary bays are rigidly attached to the reactor building and considered part of the secondary containment structure.

The drainage system around the reactor building is described in Section 2.5.4.6.

3.8.4.1.2 Control Room Building

The control room building is a Category I structure. It is a five-story concrete and steel structure 96 by 134 ft in plan and is designed for earthquake and tornado loads. The exterior walls and roof are constructed of a minimum of 2-ft thick reinforced concrete and are designed to provide tornado missile protection. The interior floors are concrete decking supported by steel framing. The building is founded on bedrock and is supported by a reinforced concrete mat at el 209 ft 6 in.

The control room building houses the control room, safety-related switchgear, batteries, and associated equipment. The upper four floors are reinforced concrete slabs on steel deck supported by structural steel. Two steel-framed grating stairways are provided for access to the upper floors. In addition, access to the turbine building is provided at each floor.

Underground concrete tunnels connect the control room building to three points of entry into the reactor building. These tunnels are designed to resist both tornado and earthquake loads. The diesel generator building is located south of and adjacent to the control room building.

3.8.4.1.3 Diesel Generator Building

The diesel generator building is a single-story, Category I structure 87 by 98 ft in plan, enclosing the three diesel generators and their associated equipment with a ground floor slab at el 261 ft. The diesel generators are supported on reinforced concrete pedestals. The building is divided into three rooms separated by fire walls, each housing one diesel generator. Three fuel oil storage tanks are located below the building, with their fuel oil pumps housed in the individual diesel generator rooms.

The diesel generator building is a reinforced concrete structure founded on bedrock and supported by wall footings. The exterior walls and roof are a minimum of 2 ft thick and are designed to provide tornado missile protection. All ventilation intakes are arranged to preclude penetration from tornado-generated missiles. The diesel generator exhaust silencer is not protected by a missile hood; however, an exhaust relief valve is provided to maintain the diesel generator function, and this exhaust relief valve is protected against missiles (see Sections 9.5.8.1 and 9.5.8.2). The building is located south of the control room building. Personnel access to the building is provided from the control room building.

3.8.4.1.4 Screenwell Building

The screenwell building consists of a concrete substructure and a steel frame superstructure. The substructure, including the service water pump room, is designated Category I, whereas the steel frame superstructure including the circulating water pump and water treatment area is designed as a non-Category I area. The screenwell building includes the service water pump rooms, the diesel and electric fire pump rooms, the water treatment area, the circulating water pumps, and other associated equipment.

Stop logs, traveling screens, trash rakes, etc., are set in the concrete walls, as required to divert the flow of water, and for maintenance purposes, respectively. These components are built-up structures of steel and concrete guided and supported by the reinforced concrete walls and floors.

The screenwell building concrete substructure consists of walls and slabs arranged to direct the flow of lake water and to support equipment including circulation water pumps, trash racks, and screens. The screenwell building superstructure is a steel building 140 by 220 ft with insulated metal wall panels. The steel roof deck is covered with insulation and four-ply, built-up roofing.

The safety-related service water pumps are enclosed in a tornado-resistant, concrete structure within the screenwell building. The diesel and electric fire pumps are enclosed in separate concrete rooms in the screenwell building.

North of the screenwell building, there are concrete chambers that house the reverse flow gates. These gates are located adjacent to the intake and discharge shafts. The shafts extend vertically down into bedrock to el 120 ft and terminate at the intake and discharge tunnels.

3.8.4.1.5 Intake Structures

The two Category I intake structures are hexagonal-shaped reinforced concrete structures connected to the intake and discharge tunnels. The structures rest on a tremie slab founded on bedrock at the lake bottom and are anchored to the concrete-encased steel tiedowns embedded into the rock. Each hexagonal-shaped intake structure has a face-to-face dimension of approximately 29 ft and a height of 10 ft 6 in. The continuity of waterflow into the intake tunnel is assured by means of electrically heated bar racks, one at each face of the hexagon.

3.8.4.1.6 Intake (and Discharge) Tunnels

The general arrangement and details of intake (discharge) tunnels are shown on Figure 1.2-29. As noted on Sheet 2 of Figure 1.2-29, the tunnels are lined with 3 1/2 in shotcrete.

The two tunnels (13 ft 5 in x 13 ft 5 in) extend from the screenwell shaft about 1,400 and 1,300 ft, respectively, eastward and northward under Lake Ontario to the intake structures. The nonseismic category discharge tunnel extends an additional 500 ft beyond the intake structure to the discharge diffusers. The tunnels are lined with 3 1/2 in shotcrete.

Within Tunnel No. 1, the intake water flows through a 4 ft 6 in ID formed opening in an 8 ft 4 in wide by 6 ft 9 in high Category I concrete encasement. The discharge water flows around the concrete encasement within the tunnel and is eventually discharged into the lake via a discharge diffuser. The tunnels are normally flooded.

3.8.4.1.7 Electrical Tunnels and Piping Tunnels

Category I electrical tunnels and piping tunnels (Figure 1.2-2) contain Category I systems and are constructed of reinforced concrete. The tunnel walls and roof are of sufficient thickness to resist penetration by tornadic missiles, or the tunnels are buried underground as required for missile protection.

Tunnels are isolated from adjoining structures by a space except that they are integrally connected to the adjacent structures when required to prevent sliding overturning and/or flotation.

Category I electrical and piping tunnels are protected from external flooding by:

1. Sealing the space between the tunnels and the adjoining structures using waterstops and flexible seals (Figure 3.8-17).

2. Providing all penetrations below grade with air and water seals, as applicable.

3.8.4.1.8 Main Stack

The Category I unlined concrete main stack located on the northeast side of the power station is designed and constructed to provide elevated release of offgas, standby gas treatment, turbine building ventilation, and other systems.

The operating conditions for the main stack are described in Sections 6.5.1, 9.4.4, and 11.3 as part of the system description. The base of the mat is at el 242 ft. The ground-level slab is at el 261 ft and the top slab is at el 276 ft. The inside base diameter is approximately 26 ft and the inside top diameter is approximately 6 ft. The height of the stack is approximately 430 ft.

3.8.4.1.9 Standby Gas Treatment Building and Railroad Access Lock Area

The standby gas treatment (SGT) building and railroad access lock area are classified Category I structures up to el 286 ft. The portion of the building above el 286 ft is classified as nonseismic.

The SGT building is a two-story, reinforced concrete and steel-framed structure approximately 40 x 90 ft. The structure shares a common wall with the railroad access lock adjacent to the reactor building. The reinforced concrete floor slab is provided at the grade level of el 261 ft. The roof is steel deck with insulation and four-ply, built-up roofing.

A railroad access lock approximately 25 x 90 ft is provided adjacent to the reactor building. This building is a reinforced concrete and steel-framed structure and shares a common wall with the SGT building. The reinforced concrete floor slab is provided at the grade level of el 261 ft. The roof is steel deck with insulation and four-ply, built-up roofing. Interlocking swinging doors are provided at each end of the structure. The exterior door is designed to withstand tornado-generated missiles. Doors are gasketed to minimize air leakage. The interlocks between the sets of doors safely prevent the doors at one end from being opened until the doors at the other end have been closed and sealed. This building is designed to accommodate a 66-ft long railroad car, plus 8 ft for a track-mobile engine.

3.8.4.1.10 Auxiliary Service Building

The auxiliary service building is a two-story, reinforced concrete and steel-framed structure approximately 55 x 76 ft in plan. The auxiliary service building below el 261 ft is classified as a Category I structure. The building is surrounded by the reactor building, turbine building, and control building. The basement floor is a concrete slab poured over electrical tunnels. The floor at el 261 ft is a concrete slab on steel deck supported by structural steel. The roof is steel deck with insulation and four-ply, built-up roofing.

3.8.4.1.11 Radwaste Building

The radwaste building houses the radioactive waste system and is analyzed to seismic conditions. It is a five-story, concrete and steel building, approximately 110 x 150 ft. The exterior walls are reinforced concrete. A rolling steel door is provided in the north wall for truck access into the building. The radwaste building is classified as shown in Table 3.2-1.

Where required for shielding purposes, interior concrete walls are provided. The basement floor is a concrete mat on rock. The upper four floors are concrete supported by steel deck and beams. The roof consists of steel framing with steel deck, insulation, and four-ply, built-up roofing. Two steel-framed grating stairways are provided for access to the upper floors.

The decontamination area is located south of the radwaste building. It is an extension of the turbine building and the radwaste building. The structure is a four-story building of concrete and steel approximately 110 x 105 ft in plan. The exterior walls are of reinforced concrete. The superstructure is covered with insulated steel roof deck with four-ply, built-up roofing.

3.8.4.1.12 Turbine Building

The turbine building complex includes the turbine building, heater bays, main steam tunnel, and condensate demineralizer regenerative and offgas area. The complex houses the turbine generator, condenser, moisture separator, etc., in the turbine building areas, heaters and related pumps and accessories in heater bay areas, and offgas system equipment and tanks in offgas areas. A portion of turbine building, main steam tunnel area, and offgas area are analyzed to seismic conditions, whereas the remaining portions are designed as nonseismic.

The turbine building houses the turbine generator and associated auxiliary systems. The heater bay which houses the feedwater heaters is adjacent to and north of the turbine building. East of the turbine building, the main steam tunnel connects the turbine building with the reactor building. North of the turbine building and west of the heater bay, the regeneration and condensate demineralizer area, offgas room, building service equipment area, and turbine building railroad passage are located.

The turbine building is a concrete and steel, three-story building, approximately 130 x 333 ft. A slab is provided at el 250 ft. All four walls are of reinforced concrete up to el 261 ft. Above this elevation, the structure is steel framed. Where exterior concrete walls are required for shielding purposes, the structural steel is encased in the concrete walls. Insulated metal wall panels are used for exterior walls. The steel deck roof is covered with rigid insulation and four-ply, built-up roofing. Smoke vents are installed on the roof.

The turbine building operating floor is concrete supported by steel deck and beams. Except where concrete floors are required for shielding, the mezzanine floor is steel-supported galvanized steel grating. Access to the upper two floors of the turbine building is provided by three sets of steel-framed galvanized grating stairs and three electric elevators.

The turbine pedestal support is of reinforced concrete and located near the center of the turbine building. The support is isolated from the turbine building by vibration joints at the mezzanine floor el 277 ft 6 in and at the operating floor el 306 ft.

The three-story heater bay is located north of the turbine building and divided into three areas by concrete walls. The exterior walls are insulated metal siding. Roof deck with steel frame construction, rigid insulation, and four-ply, built-up roofing covers the roof. The ground floor is a slab at el 250 ft, with the mezzanine floor being steel-supported galvanized steel grating. The operating floor is concrete on metal deck and steel framing. A set of steel-framed galvanized grating stairs in each of the three heater areas provides access to the mezzanine and operating floors.

The main steam tunnel, which connects the east end of the turbine building with the reactor building, is a concrete structure. The thickness of the wall, slab, and roof concrete meets both structural and shielding requirements.

West of the heater bay and north of the turbine building is the regeneration and condensate demineralizer area, the offgas room, and building service equipment area. The structure is two stories of concrete and steel construction. The wall, floor, and roof concrete meet shielding and structural requirements. Where shielding is not required for the building service equipment area, insulated metal wall panels and steel roof deck are used. The roof is covered with rigid insulation and four-ply, built-up roofing.

There is a steel-framed railroad passage to the turbine building at el 261 ft under the building service equipment area. The exterior walls consist of insulated metal wall panels. A rolling steel door is provided in the outside wall and also where the railroad enters the turbine building. Personnel access doors are also provided.

3.8.4.2 Applicable Codes, Standards, and Specifications

Codes, specifications, standards, and regulatory guides that are used in establishing design methods, analytical techniques, and material properties for Category I structures are listed herein. The criteria for the structural design of Category I structures are developed using the following regulatory guides and Code of Federal Regulations:

RG 1.10, 1.12, 1.15, 1.19, 1.31, 1.37, 1.50, 1.54, 1.55, 1.60, 1.61, 1.66, 1.69, 1.76, 1.85, 1.94, 1.117.

Appendix A of 10CFR50, Criteria 1, 2, 4, and 5 of General Design Criteria for Nuclear Power Plants

The degree of compliance to these documents is discussed in Sections 1.8 and 3.1, respectively.

The codes and standards used in the structural design of concrete and steel components of the Category I structures are as follows:

ACI-211.1-1974, 1977	American Concrete Institute, Recommended Practice for Selecting Proportions for Concrete
ACI-214-1965, 1977	American Concrete Institute, Recommended Practice for Evaluation

	of Compression Test Results of Field Concrete
ACI-301-1972, 1975	American Concrete Institute, Specification for Structural Concrete for Buildings (Exceptions to this code are listed in Section 3.8.4.6.)
ACI-304-1973	American Concrete Institute, Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete
ACI-305-1972	American Concrete Institute, Recommended Practice for Hot Weather Concreting
ACI-306-1966	American Concrete Institute, Recommended Practice for Cold Weather Concreting
ACI-307-1979	American Concrete Institute, Specification for the Design and Construction of Reinforced Concrete Chimneys
ACI-315-1974	American Concrete Institute, Manual of Standard Practice for Detailing Reinforced Concrete Structures
ACI-318-1971, 1977	American Concrete Institute, Building Code Requirements for Reinforced Concrete (Exceptions to this code are listed in Section 3.8.4.6.)
ACI-347-1968	American Concrete Institute, Recommended Practice for Concrete Formwork
AISC 1969, 1978	American Institute of Steel Construction, Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, including Supplements 1, 2, and 3 (November 1, 1970, December 8, 1971, and June 12, 1974)
AISC 1972, 1976	Code of Standard Practice for Buildings and Bridges, AISC Manual
AISI 1968	American Iron and Steel Institute, Specification for the Design of

	Cold Formed Steel Structural Members, including 1972 Printing with Addendum No. 1.
ASME III - 1971 Divisions 1, 5, 9	American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 1971 Edition through Summer 1973 Addenda, Sections II, III, Divisions 1, 5, and 9, including applicable code cases (Exceptions to this code are discussed in Section 3.8.1.)
ASME III - 1974 Division 1	American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 1974 Edition through 1976 Addenda (Exceptions to this code are discussed in Section 3.8.1.)
ASME III - 1977 Subsection NF	American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 1977 Edition, Subsection NF (Exceptions to this code are discussed in Section 3.8.3.)
AWS D1.1-1975 through 1982	American Welding Society, Structural Welding Code (Exceptions to this code are listed in Sections 3.8.4.6 and 3.8.3.2.)
AWS D1.1-83	American Welding Society, Structural Welding Code (This applies to the minimum fillet weld size for SMAW of studs only.)
AWS D12.1-75	American Welding Society, Recommended Practice for Welding Reinforcing Steel Metal Inserts, and Connections in Reinforced Concrete Construction
AWS D1.4-79	American Welding Society, Structural Welding Code - Reinforcing Steel
NCIG-01, Rev. 2, May 7, 1985	Visual Weld Acceptance Criteria (VWAC) for Structural Welding at Nuclear Power Plants, Prepared by Nuclear Construction Issues Group (NCIG)

U.S. Department of Labor, Occupational Safety and Health Administration. Occupational Safety and Health Standards (October 18, 1972).

New York State building codes, as required.

The analysis and design of plant structures follow the current dates of the applicable codes and specifications at the time of design, as listed herein.

3.8.4.3 Loads and Load Combinations

Except when otherwise noted herein and in Sections 3.8.1.3, 3.6.2.3, and 3.8.3.3, the load combinations for Category I reinforced concrete and steel structures are listed in Tables 3.8-11 and 3.8-10, respectively. The load combinations for intake tunnels are listed in Table 3.8-12.

The reinforced concrete structures within the primary containment, the exterior reactor building wall, and the reactor building foundation mat are also designed to withstand SRV and suppression pool hydrodynamic loading, using appropriate SRV load combinations listed in Table 3.8-11, Part II.

Additionally, the steel structures within the primary containment (e.g., BSW, star truss) and steel framing for floor and equipment supports in the reactor building are designed to withstand SRV and suppression pool hydrodynamic loading, using appropriate SRV load combinations listed in Table 3.8-10, Part II.

3.8.4.4 Design and Analysis Procedures

All other Category I structures are analyzed and designed as described herein. The structures are analyzed and designed for the load combinations as outlined in Section 3.8.4.3.

Category I structures are supported on reinforced concrete mat or wall footings. The design of reinforced concrete components of the structures follows ACI-318, whereas the structural steel components of the structures are designed using the AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings.

The exterior walls and roof of the structures are of minimum 2-ft thick reinforced concrete and are designed to withstand the most critical loading, as applicable, including the tornado-generated missile impact loads. The exterior walls below grade are designed for earth pressure, hydrostatic pressure, and surcharge loads, as applicable, including the dynamic effect of these loadings during OBE or SSE events.

The roof and floors of the structures are generally supported on steel framing. The floor systems, including the roof, serve as shear diaphragm to transfer lateral loads to the exterior and interior concrete walls acting as shear walls. These walls are designed to withstand gravity loads, in addition to acting as shear walls, and transmit all loads to the foundation. These walls are designed for in-plane shear forces in accordance with the requirements of Section 11.16 of ACI-318.

Masonry wall construction of solid or hollow concrete blocks, bonded together by a layer of mortar, grout, or concrete to form a rigid wall, is not incorporated in Category I applications. However, for equipment access openings, removable, solid concrete blocks contained in position by structural steel supports and adjacent concrete structures are used in Category I areas for equipment replacement and shielding. These removable walls are not used to support Category I systems or components and are not considered to act as shear walls. These walls are designed so as not to damage any safety-related structure, system, or component by virtue of their being completely enclosed by structural steel elements. The structural steel members and adjacent concrete walls providing containment and lateral support for the concrete blocks are designed to withstand the applicable loads and load combinations listed in Section 3.8.4.3.

Category I structures are essentially considered nonvented structures for tornadic loading. Exterior walls, roof, and exterior doors are conservatively designed to withstand a maximum of 3 psi pressure drop and other tornadic loads, in addition to other applicable gravity loads. The exterior doors are either protected from postulated tornado-generated missile impingement by providing protective structures around them, or designed to withstand the tornado-generated missiles.

3.8.4.4.1 Reactor Building

The reactor building wall, floor levels, and superstructure are analyzed and designed for the load combinations outlined in Section 3.8.4.3.

The exterior wall is analyzed using SHELL 1, a finite difference computer program described in Appendix 3A. The building model consists of a concrete cylindrical shell with seven branches representing the concrete floor slabs. Since the roof support system consists of steel trusses supported by structural steel columns, the roof loads are applied as an external loading to the shell.

The reactor building wall is investigated for both seismic shear and lateral earth pressure loads that are generated during an earthquake by the backfill against the exterior wall. The forces, moments, and shears in the exterior wall are determined using the SHELL 1 computer program. The inertia loadings from the dynamic analysis outlined in Section 3.7.2A are incorporated into the SHELL 1 results to account for the rigid attachment of the auxiliary bays to the cylindrical reactor building wall.

Wind pressure is distributed along the entire height of the reactor building and is analyzed using the SHELL 1 computer program. The wind pressure distribution used in the analysis is outlined in Section 3.3.1. To account for the structural steel roof framing in the SHELL 1 model, the wind pressure exerted on the roof is applied to the top of the concrete shell as an external load.

The equivalent wind pressure from tornado conditions is presented in Section 3.3.2, and the reactor building is analyzed using the SHELL 1 computer program. The building is also analyzed for a pressure drop of 3 psi (Section 3.3.2) and for the impact of tornado-borne missiles using methods described in Section 3.5.3. The tornado loads are distributed on the structural steel roof framing and siding as outlined in Section 3.3.2.

The reactor building wall is not subjected to the direct pressure or temperature loading resulting from a DBA, but the discontinuity forces, moments, and shears at the base of the wall occur as a secondary effect resulting from the deformations of the mat during the DBA. These discontinuity forces are obtained from the mat analysis (Section 3.8.5) and are included in the wall design.

The design of reinforcing steel for the concrete components conforms to ACI-318. Arrangements of reinforcing steel for typical portions of the reactor exterior building wall are shown on Figure 3.8-18.

The structural steel components of the structure are designed using the AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings.

The dead and live loads associated with heavy equipment and cask laydown are provided for in the analysis and design of the reactor building. The dynamic effect of impact caused by dropping a cask is not considered in the analysis and design since the Unit 2 plant makes use of a redundant crane. The consequences of dropping other heavy loads is described in Section 9.1.4.

The spent fuel pool and the reactor internals storage pool are supported by two nonprismatic deep girders that span the reactor building and are in turn supported on four pilasters that transmit vertical loads into the reactor building foundation mat (Figure 3.8-19). The ends of the girders are built integrally with the reactor building wall and thus are partially restrained against rotation at their ends.

The major portion of load on the girders is the contribution from the deadweight of the girders themselves plus various walls and floor slabs that frame into them. Additional sources of load on the girders include:

1. Water, fuel, and fuel racks in the spent fuel pool.

2. Water and live load in the reactor internals pool under refueling conditions.

3. Live loads on floors framing into the girders.

4. Temperature gradients through walls and floors.

5. Earthquake loads.

The sequence of design and analysis procedures followed to ensure the structural adequacy of the girders under the preceding loadings is as follows:

1. Perform a shell analysis of the reactor building using the SHELL 1 computer code to determine the degree of restraint at the ends of the girders. This was accomplished by applying unit loads to a mathematical model of the reactor building shell including various floor slabs. The loads were applied over the area of building shell covered by the girder ends and in this way equivalent springs were generated to represent the restraint provided by the shell. These equivalent springs serve as the boundary conditions for the girder analysis described as follows.

2. Model the girder for the finite element analysis using the finite element capabilities of the STRUDL II computer code. Components of the girders were represented by plane strain finite elements to account for stretching and in-plane shear. In addition, plate bending elements were used to overlay the plane strain elements to account for transverse bending and shear.

3. Perform finite element analyses using the model generated in steps 1 and 2 for the loading cases previously described.

4. Integrate the stresses obtained from step 3 to obtain cross-sectional force resultants along the length of the girder and combine the individual loading conditions to obtain design values. The effects of earthquake-induced loads were incorporated into the design values by an equivalent static method where the applied loading was increased using the ZPA values from the seismic analysis.

5. Provide reinforcement in accordance with ACI-318 using ultimate strength methods to resist shears and moments generated above. Additional horizontal and vertical reinforcing was provided along the faces of the girder to resist shear loads as suggested in the ACI code under Special Provisions for Deep Girders. Bond and

anchorage requirements were also determined based on ACI code allowables.

In addition to the preceding procedures, the effects of certain design variables that could influence the final results were considered. These additional considerations were primarily the effects of shrinkage, creep, and concrete cracking on the resulting concrete strains and deflections and also the effect of the degree of end restraint on final design moments along the span of the girders. To ensure proper clearances, the deflections of the girders were calculated using the finite element results modified with appropriate factors for shrinkage, creep, and concrete cracking.

3.8.4.4.2 Control Room Building

The control room building is designed as a reinforced concrete structure supported on a mat foundation. Since the control building is structurally connected to the diesel generator building, a seismic analysis is performed by modeling both buildings as a unit.

The control building is designed for all postulated events and applicable load combinations outlined in Section 3.8.4.3 using conventional design procedures.

3.8.4.4.3 Diesel Generator Building

The diesel generator building is designed as a reinforced concrete structure supported on wall footings. The diesel generator building is integrally connected to and therefore modeled as a unit with the control building for determination of seismic response of the structures. The diesel generator fuel oil tanks are located underneath the structure and are encased in reinforced concrete.

The diesel generator support pedestals are isolated from the other portions of the structure at grade level and extend approximately 19 ft below grade, supported on rock. In addition to the dead and live loads and resulting moments and forces, the support pedestals are also designed to withstand seismic events and the moments and forces generated during the diesel generator startup.

The diesel generator building is designed for all postulated events and applicable load combinations outlined in Section 3.8.4.3, using conventional design procedures.

3.8.4.4.4 Screenwell Building

The screenwell building is used as a source of water to the recirculating and service water pumps. Reinforced concrete construction with a low water-cement ratio is used to minimize water leakage, and the reinforcing steel has a 3-in minimum protective cover where the concrete surface is in contact with water.

The structural components are designed for the load conditions that include dead loads, live loads, equipment loads, maximum buoyant and uplift forces, crane load, loads during equipment handling and maintenance, and tornadic and seismic loads, as applicable, in accordance with the load combinations described in Section 3.8.4.3.

3.8.4.4.5 Intake Structures

The intake structures are designed for dead loads, live loads, maximum buoyant and uplift forces, forces generated by wave action (Section 3.4.2), and seismic loads.

Reinforced concrete construction with a low water-cement ratio is used to minimize water leakage, and the reinforcing steel has a 3-in minimum protective cover.

The concrete-encased steel tiedowns embedded into the rock are designed for maximum uplift forces generated due to the most critical environmental condition that can be postulated (Section 3.4).

3.8.4.4.6 Intake (and Discharge) Tunnels

The portions of the tunnels that encase the intake system are designed to withstand all possible loading effects including impact/impulse effects due to rock fall and thermal stresses. This encasement is analyzed and reinforced as an infinite beam resting on a continuous elastic foundation. The sides of the encasement are treated as columns with a maximum ductility factor of 1.3. All reinforcing steel has a 3-in minimum protective cover and high-strength concrete with a low water-cement ratio is used. The encasements are also designed for internal and external pressures and OBE and SSE seismic forces in combination with other loading, as described in Section 3.8.4.3.4.

3.8.4.4.7 Electrical Tunnels and Piping Tunnels

Electrical tunnels and piping tunnels housing safety-related systems are constructed of reinforced concrete. The tunnels are designed for gravity loads in addition to other applicable loads, such as hydrostatic loads, seismic loads, and earth pressure, in accordance with the load combinations described in Section 3.8.4.3. In addition to these loadings, the tunnel roof and walls are designed to carry the applicable crane loads resulting from the movement of cranes during and after construction. The tunnel roof is designed for surcharge loading, seismic or tornadic forces, whichever is critical, in addition to other applicable loadings in accordance with the load combinations outlined in Section 3.8.4.3.

3.8.4.4.8 Main Stack

The main stack (Figure 1.2-31) is analyzed in accordance with the methods and procedures outlined in ACI-307. This structure is not designed for a tornado; however, the distance between this and other Category I plant structures is far enough to preclude damage to any Category I structures in the event the main stack structure collapses partially or completely (Figure 1.2-2). The main stack is designed to withstand seismic and other applicable forces from the load combinations of Section 3.8.4.3.

3.8.4.4.9 Standby Gas Treatment Building and Railroad Access Lock Area

The Category I portions of these structures are designed to meet all the load conditions described in Section 3.8.4.3, in addition to meeting shielding requirements for the area. The railroad access lock area is also designed to withstand moving wheel loads from a 66-ft long railroad car and a track-mobile engine.

3.8.4.4.10 Auxiliary Service Building

The auxiliary service building below el 261 ft msl is designed to withstand the loads and load combinations described in Section 3.8.4.3.

3.8.4.4.11 Radwaste Building

The radwaste building is designed as a reinforced concrete structure supported on a mat foundation. The foundation mat is analyzed and designed using the finite element capability of the STRUDL program.

The structural steel and reinforced concrete components of this building are designed in accordance with the AISC Manual of Steel Construction and ACI-318, respectively. The load combinations described in Section 3.8.4.3 are used in designing the structure, except that protection against tornadic events is not provided.

In the event of a postulated tank rupture, the base mat and exterior walls are designed to retain the spillage within the building. The base mat and the exterior concrete walls are lined with a steel liner up to el 242 ft 2 in to contain a spillage.

3.8.4.4.12 Turbine Building

The turbine building complex is constructed partially on spread footings and partially on a mat foundation. Structural steel and reinforced concrete components of this building are designed using load combinations in accordance with the AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, and ACI-318, respectively. This building complex is constructed of reinforced concrete floors and walls up to the operating floor level. The structure above the operating floor level is constructed of a structural steel framing system braced by vertical and horizontal bracing systems up to roof level, enclosed by metal siding. A steel roof deck with roofing is provided at the top of the structure.

To prevent collapse of the steel superstructure on the nearby structures, the structural steel superstructure above the operating level is analyzed and designed using the finite element capability of the STRUDL computer program for the tornadic or seismic loads, whichever is critical, in addition to other gravity loads. The metal siding, roof decking, girts, etc., are assumed to blow away during a tornadic event; however, the main structural steel members, such as columns, beams, and bracing members, are designed to remain in place.

3.8.4.5 Structural Acceptance Criteria

For the steel structures, the allowable stresses and factors of safety are in accordance with the AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, with the following exceptions:

1. Safety-related structures as identified in Table 3.2-1 are capable of withstanding the SSE loads in combination with applicable dead and live loads.

2. These safety-related structures are also checked using OBE loads in combination with applicable dead and live loads. For this loading condition, allowable stresses are the normal working stresses, instead of applying a 33-percent increase in allowable stresses, as allowed by AISC specification.

For concrete structures, the allowable stresses, load factors, and capacity reduction factors are in accordance with strength design methods of ACI-318-77. The required strength is expressed in terms of design loads, or their related internal moments and forces. Design loads are defined as loads that are multiplied by their appropriate load factors (safety factors). Calculated strength is that computed by the provisions of ACI-318, including the appropriate capacity reduction factors. Capacity reduction factors are taken as given in Section 9.3 of ACI-318.

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

The materials for construction of Category I structures are procured, fabricated, and delivered to the site in accordance with the codes, standards, and specifications described in Section 3.8.4.2. The shipping, storage, and handling of materials during construction follow the requirements of ANSI N45.2.2. The major materials for construction of Category I structures are described herein. The major codes and the appropriate American Society for Testing and Materials (ASTM) standards used in procurement, fabrication, and testing of Category I materials are referenced herein, as applicable. The current editions of the ASTM standards adopted by the vendors' fabricating facilities at the time of procurement, fabrication, and testing of these materials are utilized.

3.8.4.6.1 Concrete

ACI-301, Specification for Structural Concrete for Buildings, together with ACI-347, Recommended Practice for Concrete Formwork, and ACI-318, Building Code Requirements for Reinforced Concrete, form the general basis for the concrete specifications. ACI-301 is supplemented as necessary with mandatory requirements relating to types and strengths of concrete, including minimum concrete densities, proportioning of ingredients, reinforcing steel requirements, joint treatments, and testing requirements.

Admixtures, types of cement, bonding of joints, embedded items, concrete curing, additional test specimens, additional testing services, cement and reinforcing steel mill test report requirements, and additional concrete test requirements are specified in detail.

All cement conforms to the Specification for Portland Cement, ASTM C 150, Type II, low alkali. Aggregates are tested initially and monitored throughout the construction phase of the project to assure that there is no adverse reaction between the cement and the aggregates. Initial examination of aggregates by ASTM C295 and tests by ASTM C289 and ASTM C227 (or ASTM C586) are used to make this determination. The ASTM C227 (or ASTM C586) test need not be completed prior to aggregate usage when ASTM C289 results are acceptable and when using low-alkali cement (Na_2O + 0.658 K_2O $\leq$0.6 percent). Certified copies of the mill test report, showing that the cement meets or exceeds the ASTM requirements for portland cement, are furnished by the manufacturer. An independent testing laboratory is retained to perform periodic tests on the cement for compliance with the specifications.

An air-entraining agent is used in the concrete in an amount sufficient to satisfy ACI-301, Section 3.4.1. This agent conforms to the requirements of ASTM C260. Before using an air-entraining admixture, a certificate of compliance from the manufacturer is obtained stating that the admixture conforms to the applicable requirements, when tested in accordance with ASTM C233. The air content of concrete is tested at the site each time a set of concrete compressive strength specimens is made. Air-entrained cement is not used.

Mixing water and/or ice is clean and free from injurious amounts of oils, acids, alkalies, salts, organic materials, or other substances deleterious to concrete or steel. The mixing water is periodically checked and tested for suitability by comparing the results of ASTM C109, ASTM C151, and ASTM C191 with those obtained using distilled water. Each source of mixing water and/or ice is subject to these tests prior to use in production concrete, and every 6 months thereafter, to assure continued acceptability.

Fine and coarse aggregates conform to ASTM C33 when tested to the following ASTM and Corps of Engineers requirements:

ASTM C40	Organic Impurities
ASTM C88	Soundness
ASTM C117	Material Finer than the 200 Sieve
ASTM C123	Lightweight Pieces
ASTM C131 or C535	Los Angeles Abrasion
ASTM C136	Sieve Analysis
ASTM C142	Friable Particles
ASTM C227 and C289	Potential Alkali Reactivity
ASTM C295	Petrographic Examination
CRD C119	Flat and Elongated Particles

An independent testing laboratory tests the aggregates initially for conformance to all the above requirements. ASTM C87 is performed only after failure of ASTM C40 tests. ASTM C666 is performed only after failure of ASTM C88. In addition, during concrete production onsite, the independent testing laboratory performs ASTM C88, ASTM C289, and ASTM C131 or ASTM C535 tests every 6 months. Tests by ASTM C39 are performed only upon failure of ASTM C131 or ASTM C535 tests. When limestone aggregate is tested, ASTM C586 is used as a basis for acceptance.

The following tests are performed at the frequency indicated during concrete production onsite:

ASTM C136	Daily
ASTM C117	Daily
ASTM C566	Daily
ASTM C40	Weekly
ASTM C123	Monthly
ASTM C142	Monthly
ASTM C235	Monthly

CRD C119 At 6-month intervals

Sampling of aggregates conforms to ASTM D75. Tests for unit weight conform to ASTM C29. Tests for specific gravity and absorption conform to ASTM C127 and ASTM C128 and are performed when these data are required. In addition to the above testing, appropriate tests are performed when a new source is to be used.

Proportioning of structural concrete conforms to ACI-301, Chapter 3. In general, structural concrete mixes have a 28-day minimum specified strength of 3,000 psi. When higher strength concrete is required, higher minimum specified strength concrete mixes are used.

Concrete used for shielding purposes, i.e., the majority of concrete used in floors, walls, roofs, and foundations, has a weight not less than 135 lb/cu ft, when air-dried in accordance with ACI-301 Section 3.3. Reference to lightweight concrete in ACI-301 Section 3.3 is also considered applicable to regular structural concrete in determining the unit weight of concrete. Proportions of ingredients for structural concrete mixes are determined and tests conducted in accordance with the method detailed in ACI-301 and ACI-211.1 for combinations of materials to be established by trial mixes.

Concrete protection for reinforcement, preparation, cleaning of construction joints, concrete mixing, delivering, placing, and curing is equal to or exceeds the requirements of ACI-301, with the following exceptions:

1. The maximum slump for massive concrete will be 3 in in general. However, up to 5-in maximum slump is permitted in congested areas to permit placing concrete in the heavily reinforced structures.

2. The minimum curing period is 1 week unless otherwise noted elsewhere in this section.

3. In lieu of Section 14.5.4, use Section 12.3.3.

4. Maximum placing temperature of the concrete when deposited conforms to the requirements of ACI-301 and ACI-305, Recommended Practice for Hot Weather Concreting, except for the placing of mass concrete. The placing temperature of mass concrete does not exceed 80°F. ACI-301 indicates placement of mass concrete sections to 70°F. This limit is based on concrete using standard or common Type I cement. Type II cement, which is used for this project, generates 80 to 85 percent of the heat of hydration of Type I cement. ACI-207 states that the heat-generating characteristic of Type II cement corresponds closely to that of Type I cement at 10°F lower placing temperatures.

5. All mass concrete placed at a temperature above 75°F is water cured in accordance with ACI-301, Chapter 12, and as described elsewhere in this section.

6. Section 4.3 of ACI-347 and Section 2.4 of ACI-347 form the basis for establishing formwork tolerances, except that when a steel plate (liner) is used for formwork, the liner tolerances will govern. Also, when the side of a wall opposite the steel liner is to be formed using something other than a liner, the theoretical form line will be established by measuring the thickness of the wall from the steel liner on the opposite face. Once the theoretical form line is established, the variation in thickness will be governed by the tolerances given in ACI-301 and ACI-347, as applicable.

Batching and mixing conform to ACI-301, Chapter 7, and ACI-304. Concrete ingredients are batched in a batch plant and transferred to transit mix trucks for mixing, agitating, and delivering to the point of placement, or are batched and mixed in a controlled mixer and transferred to a truck for delivery.

Placing of concrete is by bottom dump buckets, chuting, concrete pump, or conveyor belt. The rate of placing concrete is controlled so that concrete may be effectively placed and compacted by vibrating with particular attention given around embedded items and near the forms.

Vertical drops greater than 6 ft for any concrete are not permitted, except where suitable equipment is provided to prevent segregation.

In construction joints where keys are required by design, they are provided before the concrete has reached its final set. When keys are not provided, the surfaces of all construction joints are thoroughly cleaned by satisfactory means to remove laitance and to expose clean, sound aggregate. Excess water from joint cleaning not absorbed by the concrete is removed.

Horizontal construction joints are covered by a minimum 1/2-in thick layer of sand/cement grout, which has a compressive strength that is equal to or exceeds that of the concrete, and new concrete is then placed immediately against the fresh grout.

As an alternate to this procedure, a coating of surface retarder is applied to delay the setting of the concrete surface as described in subsections 6.1.4.2 and 6.1.4.3 of ACI-301. The horizontal surface is then prepared for the next pour by using high-pressure jet spray to remove the retarded mortar. The surface is then cut to expose the aggregates such that an irregular surface at least 1/4 in deep is exposed prior to placing the next layer of concrete.

Curing and protection of freshly deposited concrete conforms to the following:

1. Concrete to be cured with water is kept wet by covering with an approved water-saturated material, or by a system of perforated pipes or mechanical sprinklers, or by any other approved methods that will keep surfaces continuously wet. Water used for curing is generally clean and free from any elements that might cause objectionable effects.

2. The surfaces on which curing compounds may be used are specified. Curing compounds are not used on surfaces to which additional concrete is to be bonded.

3. All concrete is cured for at least 7 days unless otherwise noted in Items 5, 6, and 7 below. For all concrete, when the mean daily temperature of the surrounding air is less than 40°F, the temperature of the concrete is maintained at between 50°F and 70°F for the required curing period (7 days). Changes in temperature of the air immediately adjacent to the concrete during and immediately following the curing period are kept as uniform as possible and do not exceed 5°F in any 1-hr or 50°F in any 24-hr period.

4. For Category I watertight concrete, the reactor containment mat, the turbine pedestal and pedestal mat, and the large foundation mats 7 ft or greater in thickness, curing is with water for 7 days to maintain concrete surface moisture. For the dolosse used in the construction of the revetment ditch, the atmospheric pressure steam curing of concrete method is used in accordance with ACI-517-70.

5. For other Category I massive concrete sections not included in the preceding paragraph, the concrete is water cured for 48 hr following completion of the placement, then either a curing compound may be applied or water curing may be continued, and the temperature of the concrete maintained as above, until at least 7 days after the placement.

   Alternatively, for massive structures less than 7 ft thick, two field-cured cylinders kept adjacent to the placement and cured by the same methods may be tested for compressive strength at any time after the first 48 hr of curing. If the average compressive strength of the two cylinders tested equals or exceeds 70 percent of the specified compressive strength of the mix being used, then curing may be terminated.

6. Porous concrete containing calcium aluminate cement is water cured for a minimum of 24 hr. Two-course

floor topping, deferred placement, is water cured for seven days.

7. All other Category I concrete not designated as massive concrete is cured for 7 days either by water curing or application of a curing compound immediately after removal of forms or finishing of exposed surface.

   However, for all other concrete, curing may be terminated when the average compressive strength of field-cured cylinders kept adjacent to the structure and cured by the same methods have reached 70 percent of the specified strength of the mix.

Concrete Testing

Compressive strength tests of concrete placed in Category I structures are performed in accordance with ACI-301, Technical Specifications, for every 100 cu yd of concrete or a minimum of one set per 8-hr shift, whichever is greater.

The test specimens for compressive strength are 6-in diameter by 12-in long cylinders. Each set consists of at least three specimens. At least one is tested after 7 days and two after 28 days or 60 days age, as applicable.

Concrete strength tests are evaluated in accordance with ACI-214, Recommended Practice for Evaluation of Compression Test Results of Field Concrete, and ACI-301, Chapter 17.

The strength of concrete is considered satisfactory as long as the frequency of occurrence of the following is less than 1 in 100 if:

1. The averages of all sets of three consecutive strength test results of the laboratory-cured specimens at the specified age is equal to or greater than the specified compressive strength, f'_c, of the concrete.

2. No individual strength test result falls below the specified strength, f'_c, by more than 500 psi.

The field tests for slump of concrete are in accordance with ASTM C143. Any batch not meeting specified requirements is rejected. Slump tests are made periodically during concrete placement and each time concrete compressive strength test specimens are taken.

If cylinders should fail to meet the concrete strength requirements at the specified age, strength development and design strength requirements are reviewed. An evaluation is performed and if required, core tests are conducted in accordance with ASTM C42, Method of Obtaining and Testing Drilled Cores and Sawed Beams of Concrete. Should core tests be inconclusive or impracticable to obtain and structural analysis does not confirm the safety of the structure, load tests are performed. Concrete work judged inadequate by structural analysis or by load tests is reinforced with additional construction or is removed and rebuilt.

Statistical quality control of the concrete is maintained by a computer program based on an article in ACI Publication SP-16, Computer Applications in Concrete Design and Technology. This program analyzes compressive strength test results by the testing laboratory in accordance with methods established by ACI-214, Recommended Practice for Evaluation of Strength Test Results of Concrete.

3.8.4.6.2 Reinforcing Steel

Reinforcing steel bars, sizes N3 through N6, conform to Grade 40, and N7 through N11 conform to Grade 40 or Grade 60 of the Standard Specification for Deformed Billet-Steel Bars for Concrete Reinforcement, ASTM A615 and Supplement S-1. For fuel pool beams and certain other areas of the reactor building, special large-size reinforcing bars, N14 and N18, conform to Grade 60 of ASTM A615. Grade 60 N8 reinforcing bars are used for missile enclosure of valves in the screenwell building.

Mill test reports showing actual chemical and physical properties, including bend tests, are furnished for each heat of steel used in making all reinforcing steel.

Except when otherwise noted herein, reinforcing bars N14 and N18 will be controlled chemistry steel of 50,000 psi minimum yield point, conforming to the Standard Specification for Deformed Billet-Steel Bars for Concrete Reinforcement, ASTM A615, as modified to meet the following chemical and physical requirements:

Carbon	0.35 percent maximum
Manganese	1.25 percent maximum
Silicon	0.15 to 0.25 percent
Phosphorous	0.05 percent maximum
Sulfur	0.05 percent maximum
Minimum yield strength	50,000 psi
Elongation	13 percent minimum in an 8-in test sample
Tensile strength	70,000 psi minimum

For these special chemistry bars, all ingots are identified and all billets are stamped with identifying heat numbers. All bundles of bars are tagged with a heat number as they come off the cooling bed of the rolling mill. A special mark is rolled into all bars conforming to this special chemistry to identify them as possessing the chemical and mechanical qualities specified. The chemical variations allowed for special chemistry bars are in accordance with ASTM A29.

Placing of reinforcing steel conforms to the requirements of Chapter 5 of ACI-301, Structural Concrete for Buildings, and Chapter 7 of ACI-318, Building Code Requirements for Reinforced Concrete.

Tack welding of designed reinforcing steel that does not become an integral part of the weldment is not permitted.

Structural ductility is maintained by staggering critical splices wherever possible to ensure that small adverse effects of multiple splices in the same plane will not occur. Full-scale pressure tests conducted in May 1967 on a completed concrete containment structure, in which cadweld splices and welded splices were used in a similar manner to that proposed here, showed no stress concentrations or lack of structural ductility. Locations of splice groups were not discernible from inspection of the test crack patterns.

Reinforcing Steel Inspection and Testing

The engineers' inspectors witness, on a random basis, the pouring of the heats and the physical and chemical tests performed by the manufacturer for the special chemistry reinforcing steel. Bars failing to conform to required chemistry and physical requirements are rejected.

Mill test reports showing actual chemical ladle analysis, physical properties, bend test, and variations in weight will be obtained from the manufacturer for each heat. In addition, confirmatory tensile tests for each 50 tons of every heat of steel for every bar size will be made to determine physical properties.

Full-size test specimens of all rebars are tested on a testing machine using an 8-in gauge length. The loading rate for these tests is as specified in ASTM A370. The acceptance standards are in accordance with ASTM A615. At least one full diameter specimen from each bar size is tested for each 50 tons, or fraction thereof, of reinforcing bars produced from each heat.

The preceding frequency of testing, test procedures, and acceptance standards conform to RG 1.15. The degree of compliance to RG 1.15 is discussed in Section 1.8.

Mechanical Rebar Splicing Systems

The mechanical splice criteria follow the requirements of NRC RG 1.10, Rev. 1, as discussed in Section 1.8. The design basis described in the Preliminary Safety Analysis Report (PSAR) and approved by the NRC in the Safety Evaluation Report (SER) for the construction permit included the use of this regulatory guide. Use of this regulatory guide provides adequate assurance that the Category I structures perform their intended safety function.

Cadweld Splices Cadweld reinforcing steel splices, manufactured by Erico Products, Inc., Cleveland, OH, are used to splice N14 and N18 reinforcing bars. All cadweld splices are made in accordance with the instructions for their use issued by the manufacturer. The degree of compliance to RG 1.10 is discussed in Section 1.8. In areas where space or other requirements make cadweld splices unsuitable, N14 and N18 reinforcing bars are butt-welded in a manner conforming to the requirements of AWS D12.1.

Reinforcing bars No. 11 and smaller are generally lap spliced. Where lap splicing is impractical, and where threaded rebar splices are not used, splicing is accomplished by:

1. Cadweld as manufactured by Erico Products, Inc., or equal, using the sleeves that develop the full tensile strength of the reinforcing bars, or

2. Butt-welding in accordance with the requirements of AWS D12.1.

In order to qualify Operators for making cadweld production joints, each Operator is required to prepare two satisfactory qualification splices for each of the splice positions to be used. Testing is by tensile testing a cadweld that simulates field conditions and uses the same materials as those to be used in the structure.

The ends of the reinforcing steel bars to be joined by the cadweld process are saw cut, flame cut, or shear cut. The ends of the bars are thoroughly cleaned of all rust, scale, grease, oil, water, or other foreign matter before the joints are made.

Cadweld Testing and Inspection Cadweld process splices are visually inspected in accordance with RG 1.10. Visual inspection includes random inspection of the ends of the bars for dryness and cleanliness prior to fitting the sleeve over the ends.

Inspection is made of the completed splice for properly filled joints that have filler metal visible at both ends of the sleeve for T-series splices and the exposed end for B-series splices and at the tap hole in the center of the sleeve. Splices that do not meet all these inspection criteria are rejected.

Randomly selected cadweld splices based on separate test cycles for horizontal, vertical, and diagonal bars, size of rebar, and cadwelder are removed from the structure and tensile tested, or a combination of production and sister splices are tested in accordance with ASTM A370. Testing is in accordance with the following schedule if only production splices are tested:

1 out of first 10 splices.

1 out of next 90 splices.

2 out of next and each subsequent unit of 100 splices.

If combinations of production and sister splices are tested, the sample frequency is as follows:

1 production splice out of the first 10 production splices.

1 production and 3 sister splices out of the next 90 production splices.

3 splices, either production or sister splices, for the next and subsequent units of 100 splices. At least one-fourth of the total number of splices tested are production splices.

The sample frequency for splices in curved bars with a radius of curvature less than 60 ft is as follows:

1 sister splice for the first 10 production splices.

4 sister splices for the next 90 production splices.

3 sister splices for the next and subsequent units of 100 production splices.

Sister splices are made using straight bars.

The tensile strength of each sample tested should equal or exceed 125 percent of the specified minimum yield strength for the grade of reinforcing bar used. Failure of any splice to achieve 125 percent of the specified minimum yield strength is evaluated in accordance with Section 5 of the Procedure for Substandard Tensile Test Results as given in RG 1.10.

<u>Dywidag Threadbar System Splices</u> Dywidag threadbar system splicing is used, on a limited basis, for bar sizes No. 6 through and including No. 11 in structures, and will conform to the mechanical splice criteria of ACI-318 Building Code Requirements for Reinforced Concrete.

All Dywidag splices are made in accordance with the instructions for their use issued by the manufacturer, Dywidag Systems International, Lincoln Park, NJ.

Unstaggered Dywidag splices may be used in the radwaste building. Dywidag threadbar splice system is not used in any QA Category I structure.

Welding of Reinforcing Steel

All welding of reinforcement (i.e., rebar to plate or rebar to rebar) conforms to Recommended Practices for Welding Reinforcing Steel, Metal Inserts, and Connections in Reinforced Concrete Construction, AWS D12.1. Certified material test reports for welding electrodes are obtained from the electrode manufacturer. The ends of the bars to be joined by butt welding are prepared by saw cutting and dressing by grinding, where necessary. In order to qualify welders for work on the reinforcing steel bars, each welder makes test welds in each position he will be required to use during production. Each test weld is tension tested and each is required to meet or exceed the minimum tensile strength of the reinforcing bar. Structural ductility is maintained by staggering critical splices wherever possible to assure that small adverse effects of multiple splices in the same plane do not occur.

Generally, welding of reinforcing steel bars is employed in isolated instances only when mechanical (cadweld) splicing is not feasible due to space restrictions or clearance problems. Specifically, the reinforcing bars in size Nos. 4 to 18 are welded to the steel plates, etc., as required due to space restriction or clearance problems or to provide anchorage.

In all cases, the weld details are prequalified in accordance with AWS D12.1. Welding of reinforcing steel bars to plates is used in construction of primary containment, drywell floor, reactor pedestal, and screenwell building.

Although welding of N14 and N18 rebars is permitted, no rebar to rebar welding has been used on seismic Category I structures.

With the exception of main steam tunnel area, rebar to rebar butt welding has not been used in any other seismic Category I structure. In main steam tunnel area 1-No. 7 and 32-No. 8, rebar dowels at el 239′-0", 244′-0", and 250′-0" were butt welded to the same size bars.

Inspection and Testing of Reinforcing Steel Welds

All welds are visually inspected. Any cracks, porosity, or other defects are removed by chipping or grinding until sound metal is reached, and then repaired by welding. Peening is not permitted. Completed welded joints in reinforcing steel are selected on a random basis from Category I structures and radiographically inspected in accordance with the following schedule:

1 out of first 10 splices.

3 out of next 100 splices.

1 out of next and subsequent units of 100 splices.

Cracks and any excessive amount of contained voids, as specified in AWS D12.1, are cause for repair or removal and replacement. Replaced welds are examined in a similar manner. Reinforcing steel bars welded to steel embedments are tested by sister splice, in accordance with the following schedules:

1 sister splice out of the first 10 production splices.

4 sister splices for the next 90 production splices.

3 sister splices for the next and each subsequent unit of 100.

3.8.4.6.3 Structural Steel

Structural steel material and fabrication tolerances are in accordance with the AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, including the supplements (Section 3.8.4.2). Structural steel erection tolerances are in accordance with Code of Standard Practice for Buildings and Bridges, except that for columns other than crane columns and elevator hoistway columns, 1 1/2 in tolerance at any point in its height is permitted. In general, steel used for structural framing conforms to ASTM A36. In areas where the design indicates that a higher strength steel is required, ASTM A440, A441, A572, A588, or A242 steel is used. In the suppression pool area, stainless structural steel conforming to ASTM A167 is used.

Certified copies of mill test reports showing actual chemical and physical properties are furnished for each heat of steel used in making Category I structural steel.

Welding of structural steel is in accordance with AWS D1.1 with the following clarifications:

AWS D1.1 Section 2.4.3 requires fillers 1/4 in and larger to be extended beyond the edge of the splice plate or connection material. However, fillers greater than or equal to 1/4 in are installed using requirements of AWS D1.1 Section 2.4.2 in the following conditions:

1. Restrictions due to space limitations.

2. Presence of tapered gaps due to fabrication or erection tolerances requiring use of multiple shims.

3. When stresses cannot be transferred through the filler plates.

The structural welding code contains the requirement that undercut will not exceed 0.01 in deep when the direction is transverse to the primary tensile stress in the part that is undercut. Unless so noted, all welding performed under the AWS Code is inspected for a maximum undercut of 1/32 in.

Inspections to AWS D1.1 may be performed with the exception that the inspector need not identify with a distinguishing mark all parts or joints that he has inspected and accepted. These inspections must be documented by the Contractor's QA program.

ASTM A515 GR 65 may be considered an AWS D1.1, prequalified group no. 1 material.

AWS D1.1 Section 3.3.1 requires that the leg of a fillet weld be increased if the separation between the parts to be joined is 1/16 in or greater. Unless noted, all welding performed under the AWS Code shall have the leg of the fillet weld increased if the separation is greater than 1/16 in.

Alternatively, the visual inspection of AWS D1.1 structural welds for the items listed below may be performed using the criteria in Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants (VWAC), issued by the NCIG as described in NCIG-01, Rev. 2, May 7, 1985 (including the corresponding NCIG training). This document is accepted by the NRC in their letter of May 26, 1985, from J. P. Knight of NRC to D. E. Dutton of NCIG. The implementation date of NCIG-01 Revision 2 is documented on the appropriate specification change approval documents. The items are:

1. Structural steel.

2. Cable tray supports.

3. Conduit supports.

4. Duct supports.

5. Instrumentation supports.

6. Equipment supports.

The material installation and inspection of high-strength bolts conform to the requirements of the Specification for Structural Joints using ASTM A325 or A490 Bolts.

Welder performance qualification in accordance with ASME IX shall qualify the welder to perform welding to AWS D1.1. The welder performance qualification in accordance with ASME IX will meet all essential variables of AWS D1.1, with the following exceptions:

1. The performance test coupon with a 37 1/2 deg ± 2 1/2 deg bevel is acceptable in lieu of the 22 1/2 deg ± 2 1/2 deg bevel required by AWS D1.1.

2. In addition to welder performance qualification established on material permitted by AWS D1.1, welder performance qualification on an ASME IX P1-listed material shall qualify the welder to weld AWS D1.1 prequalified materials.

3.8.4.7 Testing and In-service Surveillance Requirements

No full-scale structural testing or in-service surveillance is anticipated for the structures described in Section 3.8.4.1. For testing of the materials used in construction, refer to Section 3.8.4.6.

3.8.5 Foundations and Concrete Supports

3.8.5.1 Description of the Foundation and Supports

Table 3.8-13 lists the foundation systems that are used for major Category I structures. Major Category I structures are founded on or below natural bedrock surface. Foundations with bases on or within the top 10 ft of the natural bedrock are designed for an allowable bearing load of 10 tons/sq ft. Foundations with bases deeper than 10 ft below the natural bedrock surface are designed for an allowable bearing load of 20 tons/sq ft. When Category I structures (such as electrical duct lines) are founded on Category I structural fill, they are designed to satisfy the Category I loading combinations as described in Section 3.8.4.3. Allowable bearing pressure on the Category I structural fill is 2 tons/sq ft.

The normal design level for groundwater is el 255 ft msl. The maximum design flood water level is el 261 ft msl. The foundations of most major Category I structures are below the groundwater level. Waterstops are provided at vertical and horizontal construction joints below the flood level of 261 ft msl to prevent seepage of water through the joints.

All Category I structures are constructed in such a manner that a minimum 6-in space is provided between the extremities of the foundations and the excavated rock surfaces. This space is filled with compressible materials such as vermiculite, vermiculite concrete, or compressible filler material. Additional information is contained in Section 2.5.4.

3.8.5.1.1 Reactor Building

The reactor building, primary containment, reactor support pedestal, and auxiliary bays are founded on a common mat, sitting on rock at el 164 ft. The mat is a reinforced concrete structure 10 ft thick and 183 ft in diameter with two rectangular auxiliary bay wings. Figure 3.8-20 shows the foundation mat configuration. The mat is reinforced with both top and bottom layers of reinforcing steel (Figures 3.8-22 and 3.8-23). Shear reinforcing steel, for radial shear forces, is placed in the vertical direction. The reinforcement for the bottom of the mat is in an orthogonal grid pattern with layers at 90 deg to each other with some additional bars in the radial direction. Reinforcement for the top of the mat consists of concentric circular bars and radial bars. The reinforcing pattern for the top of the mat is arranged to maintain a uniform spacing of the bars which extend into the mat from the vertical walls above. Mat reinforcing bars are not spliced at the junction of the mat and vertical walls.

The reactor support pedestal, the primary containment wall, and the secondary containment wall are adequately connected to the mat to resist discontinuity moments and shears by the use of vertical reinforcing dowels (Figure 3.8-20).

At least 8 in of porous concrete topped by 1/4 in of seal concrete layer is placed under the reactor building mat to intercept the groundwater, which is then channeled through 6-in porous concrete pipes to the sumps located below the mat. In addition to this, an independent water collection piping system, using half-round 8-in diameter pipes, is provided near the top of the mat to collect the leakage into two standpipes in the reactor building. The details of the groundwater drainage system for the reactor building and vicinity area are covered in Section 2.4.13.

3.8.5.1.2 Foundations for Other Structures

Foundations for all major Category I structures and the turbine building are either reinforced concrete mats or spread footings. In some instances the Category I structures are supported by the underlying Category I tunnels, which in turn are founded on bedrock using a reinforced concrete mat system. These building foundations are listed in Table 3.8-13.

Since major Category I structures are founded on natural bedrock, no potential for liquefaction exists. Category I structures, when founded on Category I structural fill, are evaluated against the possibility of any liquefaction (Section 2.5.4.8). Figure 3.8-21 shows a typical reinforcing pattern at the junction of reinforced concrete vertical structural elements and a foundation mat.

3.8.5.2 Applicable Codes, Standards, and Specifications

The design codes, standards, specifications, and regulations that are used for the design and construction of Category I foundations, including the reactor building mat, are listed in Section 3.8.4.2.

3.8.5.3 Loads and Load Combinations

The loads and loading combinations for the reactor building mat are the same as those for the primary containment structure (Section 3.8.1.3). The loads and loading combinations for the foundations of other Category I structures are the same as those used in designing the Category I structures (Section 3.8.4.3).

In addition, the following load combinations are used to check against sliding and overturning due to earthquakes, winds, and tornadoes, and against flotation due to floods:

1. D + H + OBE

2. D + H + W

3. D + H + SSE

4. D + H + W_t

5. D + F′

Where:

D, OBE, W, SSE, and W_t (defined in Section 3.8.4.3)

H = Lateral earth pressure

F′ = Buoyant force of the design basis flood

3.8.5.4 Design and Analysis Procedures

The reactor building mat is analyzed and designed for the loading combinations defined in Section 3.8.5.3. The MAT-6 program, a digital computer program based upon the general methods described in Appendix 3A, is used to determine the stresses in the mat due to statically applied axisymmetric loads. This program analyzes an axisymmetrically loaded circular plate on an elastic foundation and maintains compatibility between the plate and concentric walls supported by the plate. The mat analysis includes the effects of the primary containment pressure loads generated by the DBA, hydrodynamic loads, loads from temperature due to operating conditions and the DBA, stiffness characteristics of the cylindrical shells that are considered as elastic constraints on the mat, dead loads, and characteristics of the supporting media. The subgrade stiffness is based upon the Boussinesq theory, which assumes the subgrade to be a homogeneous isotropic elastic medium. The discontinuity moments and shears at the junctions of the primary containment, secondary containment, and reactor pedestal wall with the mat are computed by the program by applying compatibility conditions at the interface of the mat with each of the above. Appendix 3A presents the design control measures that have been employed to demonstrate the applicability and validity of the MAT-6 program.

Dynamic analysis of the reactor building provides acceleration profiles for the reactor building which are applied as static loads on the structure. Since these loads are asymmetric, the mat is analyzed using SHELL 1, a finite-difference computer program (Appendix 3A).

Since both the MAT-6 and SHELL 1 programs can handle only axisymmetric structures, the effect of auxiliary bays on the mat is investigated by three-dimensional finite element analysis using the ICES STRUDL II computer program. The results of this analysis are incorporated in the mat design. The mat is analyzed for hydrodynamic loads which are described in the Design Assessment Report for Hydrodynamic Loads (DAR) (Appendix 6A).

The foundations of other major Category I structures are analyzed by either hand calculations or using the finite element capability of the STRUDL II/STARDYNE computer programs (Appendix 3A). The foundations of small structures such as tunnels are analyzed as framed structures either using the STRUDL II computer program or by hand calculations. The loads on spread footings for walls and columns are obtained by analyzing the structure using the frame analysis approach. All the foundations of Category I structures are analyzed and designed for the applicable load combinations (Section 3.8.4.3).

3.8.5.5 Structural Design Criteria

Structural design of all foundations, including the reactor building mat, is in accordance with ACI-318, using ultimate strength design. Capacity reduction factors are used as given in Chapter 9 of ACI-318.

The structural analysis of the Category I structures provides factors of safety against overturning, sliding, and flotation and are as follows:

	Minimum Factors of Safety		
Load Combination	Overturning	Sliding	Flotation
1. D + H + OBE	1.5	1.5	NA
2. D + H + W	1.5	1.5	NA
3. D + H + SSE	1.1	1.1	NA
4. D + H + W_t	1.1	1.1	NA
5. D + F′	NA	NA	1.1

Major Category I plant structures are checked for overturning, sliding, and flotation. Table 3.8-14 provides a summary of stability analyses of major Category I structures.

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

Materials, quality control, and special construction techniques used for the construction of foundations are the same as for other Category I structures (Section 3.8.4.6).

3.8.5.7 Testing and In-service Surveillance Requirements

Testing and ISI is not planned for any foundation structure.

3.8.6 Reference

1. Timoshenko, S. P. and Gere, J. M. Theory of Elastic Stability, 2nd Edition. McGraw-Hill Book Company, New York, NY, 1961.

TABLE 3.8-1

LOAD COMBINATIONS FOR CONCRETE PRIMARY CONTAINMENT

PART I

Load Combinations Without SRV Discharge Loading

Service Conditions	Load Combination
Test	S = 1.0 D + 1.0 L + 1.15 Pa + 1.0 To
Construction	S = 1.0 D + 1.0 L + 1.0 To
Design Conditions	
Normal	U = 1.4 D + 1.7 L + 1.3 (To+Ro)
Severe environmental	U = 1.4 D + 1.7 L + 1.9 E + 1.3 (To+Ro)
Extreme environmental	U = 1.0 (D+L+To+Ro+E′)
Abnormal	U = 1.0 (D+L+Ta+Ra) + 1.5 Pa
Abnormal/severe environmental	U = 1.0 (D+L+Ta+Ra+Rj+Rr+Rm) + 1.25 (E+Pa) U = 1.0 (D+DF+L+TF+E)
Abnormal/extreme environmental	U = 1.0 (D+L+Ta+Ra+Rj+Rr+Rm+Pa+E′)

PART II

Load Combinations With SRV Discharge Loading

To account for the effects of SRV discharge loading and suppression pool hydrodynamic loading, the primary containment is checked for the following combinations:

Design Conditions	Load Combination
Normal without temperature	U = 1.4 D + 1.7 L + 1.0 Po + 1.5 SRV_{acq} U = 1.4 D + 1.7 L + 1.0 Po + 1.5 SRV_{asy}

TABLE 3.8-1 (Cont'd.)

Design Conditions	Load Combination
Normal with temperature	$U = 1.0\ D + 1.3\ L + 1.0\ Po + 1.0\ To + 1.0\ Ro + 1.3\ SRV_{seq}$ $U = 1.0\ D + 1.3\ L + 1.0\ Po + 1.0\ To + 1.0\ Ro + 1.3\ SRV_{asy}$
Normal/severe environmental	$U = 1.0\ D + 1.0\ L + 1.0\ Po + 1.0\ To + 1.0\ Ro + 1.25\ E + 1.25\ SRV_{seq}$ $U = 1.0\ D + 1.0\ L + 1.0\ Po + 1.0\ To + 1.0\ Ro + 1.25\ E + 1.25\ SRV_{asy}$
Abnormal	$U = 1.0\ D + 1.0\ L + 1.25\ Pb + 1.0\ Ta + 1.0\ Ra + 1.25\ SRV_{ads}$ $U = 1.0\ D + 1.0\ L + 1.25\ Pb + 1.0\ Ta + 1.0\ Ra + 1.25\ SRV_{asy}$ $U = 1.0\ D + 1.0\ L + 1.25\ Pa + 1.0\ Ta + 1.0\ Ra + 1.0\ SRV_{sngl}$
Abnormal/severe environmental	$U = 1.0\ D + 1.0\ L + 1.1\ E + 1.1\ Pb + 1.0\ Ta + 1.0\ Ra + 1.1\ SRV_{ads}$ $U = 1.0\ D + 1.0\ L + 1.1\ E + 1.1\ Pb + 1.0\ Ta + 1.0\ Ra + 1.1\ SRV_{asy}$ $U = 1.0\ D + 1.0\ L + 1.1\ E + 1.1\ Pa + 1.0\ Ta + 1.0\ Ra + 1.0\ SRV_{sngl}$
Normal/extreme environmental	$U = 1.0\ D + 1.0\ L + 1.0\ Po + 1.0\ To + 1.0\ Ro + 1.0\ E' + 1.0\ SRV_{seq}$ $U = 1.0\ D + 1.0\ L + 1.0\ Po + 1.0\ To + 1.0\ Ro + 1.0\ E' + 1.0\ SRV_{asy}$
Abnormal/extreme environmental	$U = 1.0\ D + 1.0\ L + 1.0\ E' + 1.0\ Pb + 1.0\ Ta + 1.0\ Ra + 1.0\ Rj + 1.0\ Rm + 1.0\ Rr + 1.0\ SRV_{ads}$ $U = 1.0\ D + 1.0\ L + 1.0\ E' + 1.0\ Pb + 1.0\ Ta + 1.0\ Ra + 1.0\ Rj + 1.0\ Rm + 1.0\ Rr + 1.0\ SRV_{asy}$ $U = 1.0\ D + 1.0\ L + 1.0\ E' + 1.0\ Pa + 1.0\ Ta + 1.0\ Ra + 1.0\ Rj + 1.0\ Rm + 1.0\ Rr + 1.0\ SRV_{sngl}$

TABLE 3.8-1 (Cont'd.)

NOTES

1. Normal wind and tornado loads do not have an effect on the structure since it is surrounded by the reactor building wall, which is designed for tornadic loading (including missiles).

2. Variations in the dead load of the structure are provided for by using 5 percent of the dead load coefficient provided in the above formulas.

3. Normal Loads Loads encountered during plant operations and shutdown.

   D = Dead loads, including hydrostatic and permanent equipment loads.

   L = Live loads, including any movable equipment loads and other loads that vary with intensity and occurrence, such as soil pressures.

   To = Thermal effects and loads during normal operating or shutdown conditions as applicable, based on the most critical transient or steady-state condition.

   Ro = Pipe reactions during normal operating or shutdown conditions as applicable, based on the most critical transient or steady-state condition.

   Po = Pressure load during normal operating condition.

4. Construction Loads Loads applied to the structure from start to completion of construction including the loads generated during the erection of the RPV. D, L, and To are defined in Note 3. Normal loads are applicable, but the construction value is used.

5. Test Loads Loads applied to the structure during the structural integrity test. D, L, and To are defined in Note 3. Normal loads are applicable. Pa is the DBA pressure load as defined in Note 8.

6. Severe Environmental Loads Events and the resulting loads occurring only infrequently. The loads associated with flooding the containment with water and the OBE are included in this category. The definitions in Note 3 apply, in addition to the following:

TABLE 3.8-1 (Cont'd.)

E = Loads due to acceleration from the OBE and including the lateral or vertical acceleration, or a combination of both, where the effects (as measured by the stresses resulting from the separate acceleration components) of lateral and vertical ground accelerations are combined algebraically.

DF = Hydrostatic loads associated with flooding the containment with water.

TF = Thermal loads associated with flooding the containment with water.

7. <u>Extreme Environmental Loads</u> Highly improbable events and the resulting loads.

8. <u>Abnormal Loads</u> Loads generated by the DBA.

Pa = Design pressure load within the containment generated by the DBA, including LOCA.

Pb = Design pressure load generated by SBA or IBA.

NOTE: Pa and Pb include the suppression pool hydrodynamic loads from effects of chugging and/or condensation oscillation.

Ta = Thermal effects and loads generated by the DBA including To.

Ra = Pipe reaction from thermal conditions generated by the DBA including Ro.

Rr = Load on the containment generated by the DBA, e.g., reaction of a ruptured high-energy pipe during the postulated event. The time-dependent nature of the load and the ability of the structure to deform beyond yield are considered in establishing the structural capacity necessary to resist the effects of Rr.

Rj = Load on the containment generated by the DBA, e.g., jet impingement from a ruptured high-energy pipe during the postulated event. The time-dependent nature of the load and the ability of the structure to deform beyond yield are considered in establishing the structural capacity necessary to resist the effects of Rj.

TABLE 3.8-1 (Cont'd.)

Rm = Equivalent static missile impact load acting on a structure generated by or during the postulated accident. Load includes an appropriate dynamic load factor applied to the peak of the missile impact-time curve.

E' = Load due to acceleration from the safe shutdown earthquake and including the lateral or vertical acceleration, or combination of both, where the effects (as measured by the stresses resulting from the separate acceleration components) of lateral and vertical ground accelerations are combined algebraically.

9. Abnormal/Severe Environmental Loads Combinations that result from the postulated combined occurrence of abnormal and severe environmental effects, when the occurrence of a specified severe environmental category load condition at the plant site imposes effects that significantly increase the probability of abnormal category load conditions, or when the specified abnormal or severe environmental category load condition is of such extended duration that a significant probability exists that loads in these two categories will occur simultaneously.

10. Abnormal/Extreme Environmental Loads Combinations that result from the postulated combined occurrences of abnormal and extreme environmental effects, when the occurrence of a specific extreme environmental category load condition at the plant site imposes effects that significantly increase the probability of the occurrence of abnormal category load conditions, or when the specified abnormal or extreme environmental category load condition is of such extended duration that a significant probability exists that loads in these two categories will occur simultaneously.

11. U is the required section strength based on the strength design methods described in ACI-318.

In computing the required section strength, actual compressive strength of concrete may be used in place of specified minimum compressive strength of concrete. The actual compressive strength of concrete shall be determined from the compressive strength test reports of the concrete pour under consideration. This provision may be used to establish design adequacy in isolated

TABLE 3.8-1 (Cont'd.)

cases where the specified minimum strength of the material is potentially exceeded.

12. S is the required section strength based on alternate design method and the allowable stress described in ACI-318.

13. SRV discharge loads are defined as shown below:

SRV_{sca} = SRV loads due to sequential (i.e., all valves) actuation.

SRV_{ads} = SRV loads due to automatic depressurization system (ADS) (seven valves) actuation.

SRV_{asy} = SRV loads due to asymmetric (three valves) actuation.

SRV_{sngl} = SRV loads due to single valve actuation.

TABLE 3.8-8

STRESS

Stress Component	Location (el)	Governing Equation	Stress Actual	Stress Allowable
Concrete Stress				
Membrane and bending	175'	U = 1.0 D + 1.0 L + 1.0 Ta + 1.5 Pa + 1.0 Ra	2,800 psi	3,000 psi
Radial shear	180'	U = 1.0 D + 1.0 L + 1.0 Ta + 1.25 Pb + 1.0 Ra + 1.25 SRV	vu = 37 psi	vc = 37 psi
Reinforcement Stress				
Meridional				
Inside layer	175'	U = 1.0 D + 1.0 L + 1.0 Ta + 1.5 Pa + 1.0 Ra	44.9 ksi	45 ksi
Outside layer	201'	U = 1.0 D + 1.0 L + 1.0 Ta + 1.1 Pb + 1.0 Ra + 1.1 E + 1.1 SRV	44.5 ksi	45 ksi
Hoop				
Inside layer	220'	U = 1.0 D + 1.0 L + 1.0 Ta + 1.5 Pa + 1.0 Ra	43.8 ksi	45 ksi
Outside layer	220'	U = 1.0 D + 1.0 L + 1.0 Ta + 1.1 Pa + 1.0 Ra	43.7 ksi	45 ksi
Radial shear	260'	U = 1.0 D + 1.0 L + 1.0 Ta + 1.25 Pa + 1.0 Ra + 1.25 E + 1.0 (Rj + Rm + Rr)	35.7 ksi	42.5 ksi
Diagonal	201'	U = 1.0 D + 1.0 L + 1.0 Ta + 1.5 Pa + 1.0 Ra	45.0 ksi	45.0 ksi

KEY: vu = Nominal design shear stress resisted by concrete.
vc = Nominal permissible shear stress carried by concrete.

NOTE: Symbols are identified in Table 3.8-1.

TABLE 3.8-9

LOAD COMBINATIONS FOR THE BIOLOGICAL SHIELD WALL

Normal Operating Load

1. S = 1.0 D + 1.0 E

2. S = (1.0)D + (1.0)To + (1.0)Ro + E

Abnormal and/or Extreme Environmental Load

3. 1.6 S = (1.0)D + (1.0)To + (1.0)Ro + E′

4. 1.6 S = (1.0)D + (1.0)Td + (1.0)Ra′ + (1.0)Pd

5. 1.8 S = (1.0)D + (1.0)Td + (1.0)Ra′ + (1.0)Pd + 1.0(Rr+Rj+Rm) + E

6. 2.0 S = (1.0)D + (1.0)Td + (1.0)Ra′ + (1.0)Pd + 1.0(Rr+Rj+Rm) + E′

KEY:

D = Dead load of shield wall, equipment, and attached piping

E = Load due to operating basis earthquake (OBE)

E′ = Load due to safe shutdown earthquake (SSE)

Pd = Pressure differential across biological shield wall due to a pipe rupture

Ra′ = Pipe reactions under thermal conditions generated by the postulated accident

Rj = Equivalent static jet impingement load acting on a structure generated by a ruptured high-energy pipe during the postulated accident. The peak value of Rj is used unless a time-history analysis is performed to justify otherwise.

Rm = Equivalent static missile impact load on a structure generated by or during the postulated accident. The peak value of Rm is used unless a time-history analysis is performed to justify otherwise.

Ro = Pipe reaction during operating condition.

Rr = Equivalent static reaction load from the ruptured high-energy pipe during the postulated accident. The peak value of Rr is used unless a time-history analysis is performed to justify otherwise.

S = Required section strength is based on elastic design methods which are used in design as the allowable stresses defined in the AISC specification except that the 33 1/3 percent

TABLE 3.8-9 (Cont'd.)

		increase in allowable stresses for seismic loading is not used.
Td	=	Effect of temperature gradient across biological shield wall, including increase of temperature due to pipe rupture and pressure buildup.
To	=	Effect of temperature gradient across biological shield wall during operating condition.

TABLE 3.8-10

DESIGN STRENGTH FOR LOAD COMBINATIONS ON CATEGORY I STEEL STRUCTURES

PART I

Load Combinations Without SRV Discharge Loading[1,2,3]

Operating Conditions

Normal Loading:

1. S = 1.0[D+L]

Severe Environmental Loading:

2. S = 1.0[D+L+E]

3. S = 1.0[D+L+W]

Design Conditions

Extreme Environmental Loading:

4. 1.6 S = 1.0[D+L+To+Ro+E′]

5. 1.6 S = 1.0[D+L+To+Ro+Wt]

Abnormal Loading:

6. 1.6 S = 1.0[D+L+Ta+Ra+Pa]

7. 1.8 S = 1.0[D+L+Ta+Ra+Rr+Rj+Rm+Pa+E]

8. 2.0 S = 1.0[D+L+Ta+Ra+Rr+Rj+Rm+Pa+E′]

9. 1.6 S = 1.0[D+Ls]

PART II

Load Combinations With SRV Discharge Loading[1,2,3]

To account for the effect of SRV discharge loading and suppression pool hydrodynamic loading, the steel structures and steel framing within the reactor building are checked for the following combinations:

TABLE 3.8-10 (Cont'd.)

Design Conditions

Normal Without Temperature:

1. $S = 1.0\ D + 1.0\ L + 1.0\ Po + 1.0\ SRV_{seq}$

2. $S = 1.0\ D + 1.0\ L + 1.0\ Po + 1.0\ SRV_{asy}$

Normal With Temperature:

3. $S = 1.0\ D + 1.0\ L + 1.0\ Po + 1.0\ To + 1.0\ Ro + 1.0\ SRV_{seq}$

4. $S = 1.0\ D + 1.0\ L + 1.0\ Po + 1.0\ To + 1.0\ Ro + 1.0\ SRV_{asy}$

Normal/Severe Environmental:

5. $S = 1.0\ D + 1.0\ L + 1.0\ Po + 1.0\ Ro + 1.0\ To + 1.0\ E + 1.0\ SRV_{seq}$

6. $S = 1.0\ D + 1.0\ L + 1.0\ Po + 1.0\ Ro + 1.0\ To + 1.0\ E + 1.0\ SRV_{asy}$

Abnormal:

7. $1.6\ S = 1.0\ D + 1.0\ L + 1.0\ Pb + 1.0\ Ta + 1.0\ Ra + 1.0\ SRV_{ads}$

8. $1.6\ S = 1.0\ D + 1.0\ L + 1.0\ Pb + 1.0\ Ta + 1.0\ Ra + 1.0\ SRV_{asy}$

9. $1.6\ S = 1.0\ D + 1.0\ L + 1.0\ Pa + 1.0\ Ta + 1.0\ Ra + 1.0\ SRV_{sngl}$

Abnormal/Severe Environmental:

10. $1.8\ S = 1.0\ D + 1.0\ L + 1.0\ E + 1.0\ Pb + 1.0\ Ta + 1.0\ Ra + 1.0\ SRV_{ads}$

11. $1.8\ S = 1.0\ D + 1.0\ L + 1.0\ E + 1.0\ Pb + 1.0\ Ta + 1.0\ Ra + 1.0\ SRV_{asy}$

12. $1.8\ S = 1.0\ D + 1.0\ L + 1.0\ E + 1.0\ Pa + 1.0\ Ta + 1.0\ Ra + 1.0\ SRV_{sngl}$

TABLE 3.8-10 (Cont'd.)

Normal/Extreme Environmental:

13. $1.6\ S = 1.0\ D + 1.0\ L + 1.0\ Po + 1.0\ To + 1.0\ Ro + 1.0\ E' + 1.0\ SRV_{seq}$

14. $1.6\ S = 1.0\ D + 1.0\ L + 1.0\ Po + 1.0\ To + 1.0\ Ro + 1.0\ E' + 1.0\ SRV_{asy}$

Abnormal/Extreme Environmental:

15. $2.0\ S = 1.0\ D + 1.0\ L + 1.0\ E' + 1.0\ Pb + 1.0\ Ta + 1.0\ Ra + 1.0\ Rj + 1.0\ Rm + 1.0\ Rr + 1.0\ SRV_{ads}$

16. $2.0\ S = 1.0\ D + 1.0\ L + 1.0\ E' + 1.0\ Pb + 1.0\ Ta + 1.0\ Ra + 1.0\ Rj + 1.0\ Rm + 1.0\ Rr + 1.0\ SRV_{asy}$

17. $2.0\ S = 1.0\ D + 1.0\ L + 1.0\ E' + 1.0\ Pa + 1.0\ Ta + 1.0\ Ra + 1.0\ Rj + 1.0\ Rm + 1.0\ Rr + 1.0\ SRV_{sngl}$

---

(1) If thermal stresses due to To and Ro are present and are self-limiting in nature, a 50-percent increase in allowable stresses will be permitted.

(2) The peak values of Pa, Pb, Ta, Ra, Rr, Rj, and Rm are used unless a time-history analysis is performed to justify otherwise.

(3) Local stresses due to the concentrated load Rj, Rm, or Rr may exceed the allowables, but there will be no loss of function.

NOTES:

1. Key to notations follows Table 3.8-12.

2. Loads resulting from thermal stratification, if applicable, are included wherever temperature loads are considered.

TABLE 3.8-11

REQUIRED STRENGTH FOR LOAD COMBINATIONS ON CATEGORY I CONCRETE STRUCTURES

PART I

Load Combinations Without SRV Discharge Loading

Operating Conditions

Normal Loading:

1. U = 1.4 D + 1.7 L + 1.3[To+Ro]

Severe Environmental Loading:

2. U = 1.4 D + 1.7[L+W] + 1.3[To+Ro]

3. U = 1.4 D + 1.7 L + 1.9 E + 1.3[To+Ro]

Design Conditions

Extreme Environmental Loading:

4. U = 1.0[D+L+E'+To+Ro]

5. U = 1.0[D+L+Wt+To+Ro]

6. U = 1.0[D+L+F+To+Ro]

Abnormal Loading:

7. U = 1.0[D+L+Ta+Ra] + 1.5 Pa[1]

Abnormal/Severe Environmental Loading:

8. U = 1.0[D+L+Ta+Ra+Rj+Rr+Rm] + 1.25[Pa+E][1,2]

Abnormal/Extreme Environmental Loading:

9. U = 1.0[D+L+Ta+Ra+Rj+Rr+Rm+Pa+E'][1-3]

10. U = 1.0[D+Ls]

TABLE 3.8-11 (Cont'd.)

PART II

Load Combinations With SRV Discharge Loading

To account for the effect of SRV discharge loading and suppression pool hydrodynamic loading, the concrete structures within the reactor building are checked for the following combinations:

Design Conditions

Normal Without Temperature:

1. $U = 1.4\ D + 1.7\ L + 1.0\ Po + 1.5\ SRV_{seq}$

2. $U = 1.4\ D + 1.7\ L + 1.0\ Po + 1.5\ SRV_{asy}$

Normal With Temperature:

3. $U = 1.0\ D + 1.3\ L + 1.0\ Po + 1.0\ To + 1.0\ Ro + 1.3\ SRV_{seq}$

4. $U = 1.0\ D + 1.3\ L + 1.0\ Po + 1.0\ To + 1.0\ Ro + 1.3\ SRV_{asy}$

Normal Severe Environmental:

5. $U = 1.0\ D + 1.0\ L + 1.0\ Po + 1.0\ To + 1.0\ Ro + 1.25\ E + 1.25\ SRV_{seq}$

6. $U = 1.0\ D + 1.0\ L + 1.0\ Po + 1.0\ To + 1.0\ Ro + 1.25\ E + 1.25\ SRV_{asy}$

Abnormal:

7. $U = 1.0\ D + 1.0\ L + 1.25\ Pb + 1.0\ Ta + 1.0\ Ra + 1.25\ SRV_{ads}$

8. $U = 1.0\ D + 1.0\ L + 1.25\ Pb + 1.0\ Ta + 1.0\ Ra + 1.25\ SRV_{asy}$

9. $U = 1.0\ D + 1.0\ L + 1.25\ Pa + 1.0\ Ta + 1.0\ Ra + 1.0\ SRV_{sngl}$

Abnormal/Severe Environmental:

10. $U = 1.0\ D + 1.0\ L + 1.1\ E + 1.1\ Pb + 1.0\ Ta + 1.0\ Ra + 1.1\ SRV_{ads}$

TABLE 3.8-11 (Cont'd.)

11. $U = 1.0\ D + 1.0\ L + 1.1\ E + 1.1\ Pb + 1.0\ Ta + 1.0\ Ra + 1.1\ SRV_{asy}$

12. $U = 1.0\ D + 1.0\ L + 1.1\ E + 1.1\ Pa + 1.0\ Ta + 1.0\ Ra + 1.0\ SRV_{sngl}$

Normal/Extreme Environmental:

13. $U = 1.0\ D + 1.0\ L + 1.0\ Po + 1.0\ To + 1.0\ Ro + 1.0\ E' + 1.0\ SRV_{seq}$

14. $U = 1.0\ D + 1.0\ L + 1.0\ Po + 1.0\ To + 1.0\ Ro + 1.0\ E' + 1.0\ SRV_{asy}$

15. $U = 1.0\ D + 1.0\ L + 1.0\ Po + 1.0\ To + 1.0\ Ro + 1.0\ Wt + 1.0\ SRV_{seq}$

16. $U = 1.0\ D + 1.0\ L + 1.0\ Po + 1.0\ To + 1.0\ Ro + 1.0\ Wt + 1.0\ SRV_{asy}$

Abnormal/Extreme Environmental:

17. $U = 1.0\ D + 1.0\ L + 1.0\ E' + 1.0\ Pb + 1.0\ Ta + 1.0\ Ra + 1.0\ Rj + 1.0\ Rm + 1.0\ Rr + 1.0\ SRV_{ads}$

18. $U = 1.0\ D + 1.0\ L + 1.0\ E' + 1.0\ Pb + 1.0\ Ta + 1.0\ Ra + 1.0\ Rj + 1.0\ Rm + 1.0\ Rr + 1.0\ SRV_{asy}$

19. $U = 1.0\ D + 1.0\ L + 1.0\ E' + 1.0\ Pa + 1.0\ Ta + 1.0\ Ra + 1.0\ Rj + 1.0\ Rm + 1.0\ Rr + 1.0\ SRV_{sngl}$

---

(1) The peak values of Pa, Ta, Ra, Rr, Rj, and Rm will be used unless a time-history analysis is performed to justify otherwise.

(2) Local stresses due to the concentrated load Rm may exceed the allowables, but there will be no loss of function.

(3) Use capacity reduction factor = 1.0.

NOTE: Key to notations follows Table 3.8-12.

TABLE 3.8-12

LOAD COMBINATIONS FOR INTAKE TUNNELS*

Operating Conditions

Normal Loading:

1. U = 1.4 D + 1.7 L + 1.3 To + 1.4 Pd

Severe Environmental Loading:

2. U = 1.4 D + 1.7 L + 1.9 E + 1.3 To + 1.4 Pd

Design Conditions

Extreme Environmental Loading:

3. U = 1.0[D+L+E′+To] + 1.4 Pd

Abnormal Loading:

4. U = 1.0[D+L] + 1.4 Pd + 1.7 QL + 1.25 E

---

* These load combinations are used in designing the Category I reinforced concrete portions (e.g., concrete encasement) of intake tunnels.

In addition to these conditions, the concrete encasement is designed for the test (or preoperational) conditions considering the construction truck loading, the internal test pressure (Pi), or the external test pressure (Pe), as applicable.

KEY TO NOTATIONS FOR TABLES 3.8-10 THROUGH 3.8-12:

D = Dead loads and their related moments and forces, including any permanent loads and hydrostatic loads.
E = Loads generated by the operating basis earthquake.
E′ = Loads generated by the safe shutdown earthquake.
F = All forces and moments related to hydrostatic and saturated soil pressures due to the postulated maximum flood (PMF). For this loading, exterior walls and foundations are designed for a hydrostatic head to el 261 ft.
L = Live loads and their related moments and forces, including:
1. Movable equipment loads.
2. Lateral soil pressures.

TABLE 3.8-12 (Cont'd.)

3. Snow load during normal operating or severe environment conditions.
4. Pressure differences due to variation in heating and cooling and outside atmospheric changes.
5. Any other loads that vary with intensity and occurrence.

Pa	=	Pressure equivalent static load within or across a compartment and/or building, generated by the postulated accident (i.e., design basis pipe break accident with LOCA), and including an appropriate dynamic load factor applied to the peak of the pressure-time curve.
Pb	=	Pressure equivalent static load within or across a compartment and/or building, generated by the small pipe break accident (SBA) or the intermediate pipe break accident (IBA) events.
Pd	=	Pressure differential between the intake and discharge water (i.e., between inside and outside of the encasement) during the most critical of the operating or design conditions.
Po	=	Pressure loads on a structure during operating condition.
QL	=	Load caused by a postulated rock fall condition.
Ra	=	Pipe reactions under thermal conditions generated by the postulated accident.
Rj	=	Equivalent static jet impingement load acting on a structure generated by a ruptured high energy pipe during the postulated accident. Load includes an appropriate dynamic load factor applied to the peak of the jet pressure load time-history.
Rm	=	Equivalent static missile impact load acting on a structure generated by or during the postulated accident. Load includes an appropriate dynamic load factor applied to the peak of the missile impact-time curve.
Ro	=	Pipe reactions during normal operating or shutdown conditions, based on the most critical transient or steady-state condition.
Rr	=	Reaction equivalent static load on the ruptured high-energy pipe during the postulated accident, and including an appropriate dynamic load factor applied to the peak of the reaction-time curve.
S	=	Allowable design strength when using the allowable stresses defined in Part I of the AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, 1969 and 1978. The 33 1/3 percent increase in allowable

TABLE 3.8-12 (Cont'd.)

stresses for seismic or wind loadings is not permitted.

In computing the allowable design strength, actual yield stress of the material may be used in place of specified minimum yield stress of the material. The actual yield stress of the material shall be determined from the test reports of the material under consideration. This provision may be used to establish design adequacy in isolated cases where the specified minimum strength of the material is potentially exceeded.

SRV = SRV loads are defined as follows:

- SRV_{seq} = SRV loads due to sequential (i.e., all valves) actuation.
- SRV_{ads} = SRV loads due to automatic depressurization system (ADS) (seven valves) actuation.
- SRV_{asy} = SRV loads due to asymmetric (three valves) actuation.
- SRV_{sngl} = SRV loads due to single valve actuation.

Ta = Thermal effects and loads generated by the postulated accident.

To = Thermal effects and loads during normal operating or shutdown conditions, based on the most critical transient or steady-state condition.

U = Required strength to resist the design loads based on the strength design methods, as modified by the application of capacity reduction factors as described in Section 9.3, ACI-318-77.

In computing the required section strength, actual compressive strength of concrete may be used in place of specified minimum compressive strength of concrete. The actual compressive strength of concrete shall be determined from the compressive strength test reports of the concrete pour under consideration. This provision may be used to establish design adequacy in isolated cases where the specified minimum strength of the material is potentially exceeded.

W = Loads generated by the design wind specified for the unit (Section 3.3.1).

TABLE 3.8-12 (Cont'd.)

Wt	=	Loads generated by the design tornado specified for the unit. They include loads due to the tornado wind pressure, the tornado-created differential pressure, and the tornado-generated missiles (Sections 3.3.2 and 3.5).
Ls	=	Snow load or probable maximum precipitation (PMP) during abnormal/extreme environmental condition.

TABLE 3.8-14

FACTORS OF SAFETY FOR OVERTURNING, SLIDING AND FLOATATION OF MAJOR CATEGORY I STRUCTURES

Structure	Overturning				Sliding				Floatation
	Loading Conditions				Loading Conditions				Loading Condition
	1	2	3	4	1	2	3	4	5
Control and Diesel Generator Buildings	2.9	2.4	1.7	2.3	2.1	1.5	1.1	2.3	1.1
North and South Electrical Tunnels	1.5	[1]	[1]	[1]	2.1	[1]	[1]	[1]	5.1
Main Stack	[1]	4.2	[1]	[1]	[1]	19.8	[1]	[1]	11.6
Reactor Building	[1]	[1]	1.1	[1]	[1]	[1]	1.43	[1]	1.7
Screenwell Building	1.8	[1]	[1]	[1]	1.5	[1]	[1]	[1]	1.3
Standby Gas Treatment Building	3.6	[1]	1.2	[1]	1.5	[1]	1.1	[1]	1.2

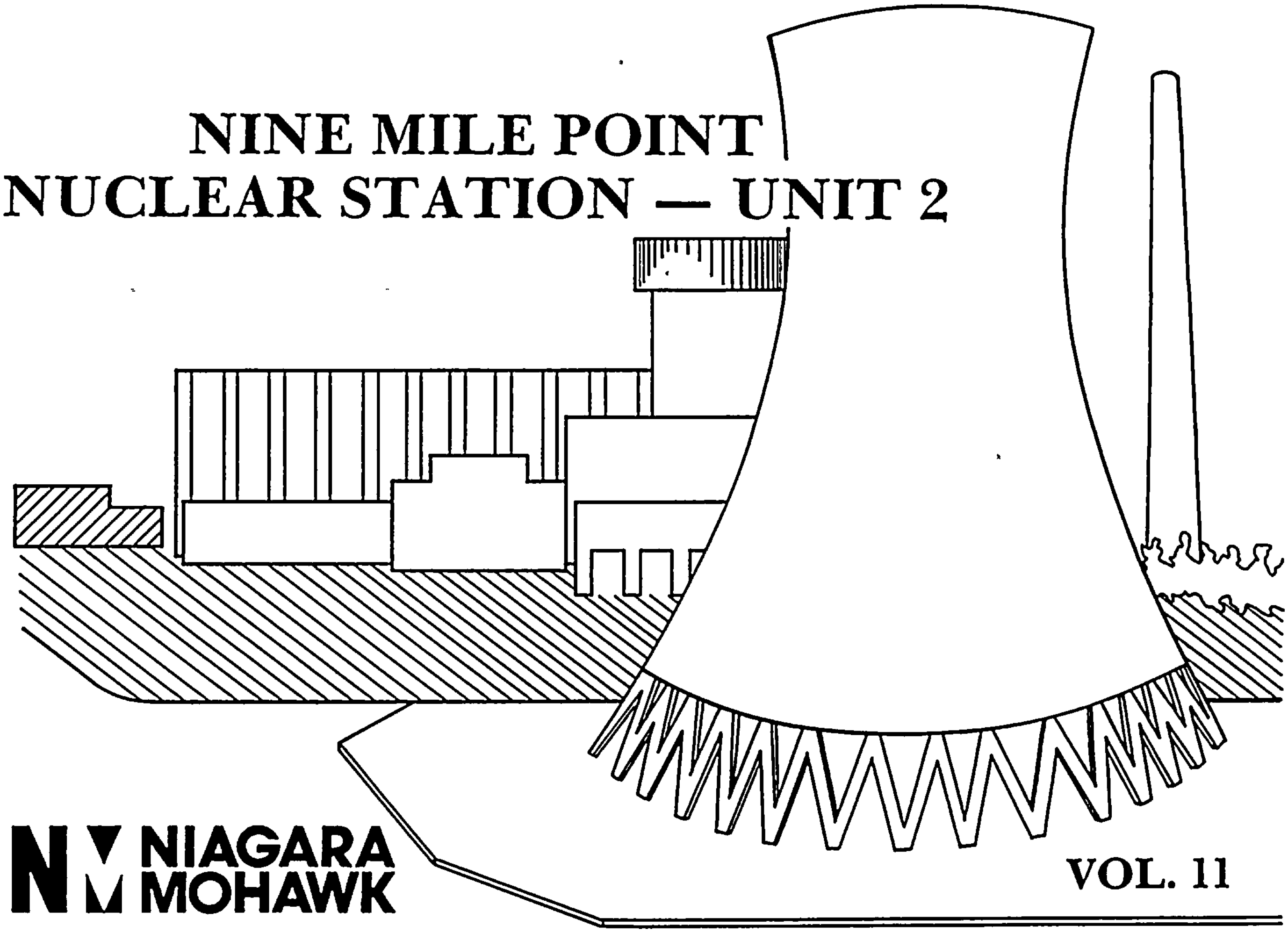
LEGEND:

1 D+H+E
2 D+H+W
3 D+H+E'
4 D+H+W+
5 D+F'

[1] Indicates that the loading condition yields a factor of safety higher than those listed herein.

UPDATED SAFETY ANALYSIS REPORT

NINE MILE POINT NUCLEAR STATION — UNIT 2



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3.9 MECHANICAL SYSTEMS AND COMPONENTS

This section is divided as follows: Section 3.9A applies to systems and components within SWEC scope of supply and Section 3.9B applies to systems and components within GE scope of supply.

3.9A MECHANICAL SYSTEMS AND COMPONENTS (SWEC SCOPE OF SUPPLY)

3.9A.1 Special Topics for Mechanical Components

3.9A.1.1 Design Transients

Table 3.9A-1 lists the plant events that were used for the design and analysis of ASME Section III Safety Class 1 components and supports. The table also shows the number of cycles per event and event classification. Application of these transients is discussed under load combinations in Section 3.9A.3.1.

3.9A.1.2 Computer Programs Used in Analyses

The computer programs used in analyses are described, and their applicability and validity are demonstrated in Appendix 3A.

3.9A.1.3 Experimental Stress Analysis

Experimental stress analysis for the design of balance-of-plant (BOP) equipment was not used.

3.9A.1.4 Consideration for the Evaluation of the Faulted Condition

3.9A.1.4.1 Equipment and Components

The elastic analysis techniques described in Section 3.7A.3 are utilized in the qualification of Category I ASME Code and non-Code equipment. Stress limits utilized for the faulted plant condition are outlined in Section 3.9A.3.1. Design conditions and stress limits defined are applicable for an elastic system (and equipment) analysis. Inelastic analyses have not been used.

3.9A.1.4.1.1 ASME III Compliance

Category I ASME Safety Class 1, 2, and 3 components are designed, analyzed, and certified in accordance with the appropriate ASME III Code edition and addenda as defined in their design specifications. However, if Code nameplates are removed from installed equipment, traceability is provided in accordance with ASME III 1980 Edition, Winter 1981 Addendum, Subsection NCA, Subarticle 8240(b).

3.9A.1.4.2 Piping Systems

Category I ASME Safety Class 1, 2, and 3 piping and pipe supports are analyzed and designed in accordance with requirements of ASME

Section III, Subsections NB, NC, ND, and NF, respectively. The analyses also comply with Appendix F of ASME Section III. The 1974 Edition is used with the following exceptions:

1. Building settlements, not applicable to the faulted condition, are analyzed according to the 1977 Edition.

2. The number of OBE load cycles is based on Appendix N of the 1977 Edition, Winter 1978 Addenda.

3. For pipe supports, the 1974 Edition is used with the additional requirements described in Section 3.9A.3.4.1.

4. The boundary of jurisdiction of ASME Code Section III, Class 1, 2, or 3 process piping extends to and includes the seat of the root valve to the instrument. The appropriate quality group extends from the root valve to the instrument and shall be designed to ASME Section III. Seismic Category I supports shall be installed with a 10CFR50 Appendix B program described in Section 3.9A.3.4.1.

5. Material upgraded by material manufacturer will meet the provisions of ASME III, 1977 Edition, Summer 1977 Addenda, Subsection NCA.

6. Installation of attachments to Category I ASME Safety Class 1, 2, and 3 piping systems after testing is to be accomplished in accordance with ASME Section III, 1980 Edition, Winter 1981 Addenda, Subarticle 4436, Subsections NB, NC, and ND.

7. The selection of the type and certification of penetrometers required for nondestructive examination is governed by ASME Section V as invoked by ASME Section III. The 1974 Edition, including Summer 1974 Addenda, is used.

8. The inside corner radius in Note 6d of Figures NC and ND 3673.2(b)-1 is as defined in ASME III, 1980 Edition, Winter 1980 Addendum. This radius is required on the inside wall of the run pipe at welded branch connections greater than 4 in. The radius is not required for nominal branch pipe size smaller than 4 in.

9. The requirements for the stamping of N-type nameplates are as defined in the subparagraphs NCA-8220 and NCA-8320 of the 1980 Edition of ASME III. The arrangement shall be substantially as shown on Figure NCA-8212-1 of the 1980 ASME III Code.

10. Nondestructive examination is governed by ASME Section V as invoked by ASME Section III, 1980 Edition of ASME Section V including Summer 1980 Addenda, is used for evaluation criteria of scattered radiation.

11. To establish acceptable criteria for venting during system fill operation, ASME Section III, 1981 Summer Addenda, Subarticles NB-6211, NC-6211, and ND-6211 are used.

12. The requirements for examination of socket weld components during in-process repair prior to reuse are defined in ASME III, 1980 Edition, NB-4121.3.

13. The requirements for the elimination of surface defects are as defined in Subparagraph NC-4452 of ASME Section II, 1983 Edition, Summer 1984 Addenda.

14. ASME III, Subsections NB-6211, NC-6211, and ND-6211, Summer 1980 Addenda, may be used for hydrostatic testing.

15. ASME III, paragraph NCA-1273, Summer 1980 Addenda is used when defining fluid conditioner and flow control devices other than valves.

16. ASME III Appendices Figure I-9.2 of the 1983 Edition is used for determination of acceptable stress limits for stainless steel piping during the preoperation and power ascension phases of the piping vibration test program.

17. Use of ASME Subparagraph NB-3630(d)(2) of ASME Section III, Division 1, Summer 1976 Addenda, is permitted for stress analysis of Class 1 piping in accordance with requirements of Subsection NC.

18. Residual heat removal (RHS) supply and discharge lines connected to the recirculation piping in primary containment are analyzed in accordance with the applicable ASME III Code governing the recirculation piping.

Loadings considered in the faulted condition include the following:

1. Loading associated with normal plant conditions, including hydrodynamic loads associated with suppression pool phenomena.

2. Loading associated with the postulated SSE.

3. Dynamic system loading associated with faulted plant. conditions, i.e., DBA, break of a MSL, or recirculation line.

4. Dynamic system loading associated with the intermediate break accident (IBA) and small break accident (SBA).

Procedures for developing the loading functions in Items 1 and 2 above are described in Sections 3.9A.1.5 and 3.7A.3.8. Loading functions in Items 3 and 4 are described in Section 3.6A.2. Loads associated with the suppression pool phenomena are described in the DAR (Appendix 6A).

3.9A.1.5 Analysis of Piping Systems

Category I piping systems (ASME Safety Class 1, 2, 3) are analyzed in accordance with ASME Section III, 1974 Edition, Subarticles NB-3600, NC-3600, and ND-3600 unless otherwise noted as an exception in Section 3.9A.1.4.2. ANSI B31.1 seismically supported and nonseismic piping systems are analyzed in accordance with ANSI B31.1 Code, 1973 Edition, including Addendum C, dated December 18, 1973. In addition, high-energy piping systems are analyzed for pipe rupture criteria.

All seismically supported and nonseismically supported ANSI B31.1 systems may be hydrostatically tested in accordance with a later Code edition than previously specified.

Later editions of ANSI B31.1 are considered when defining minimum welding dimensions required for socket welding components other than flanges.

Analytical modeling and seismic analysis are described in Section 3.7A.3.8. Static analysis and other dynamic analyses that contribute the remaining stresses in the Code stress criteria are described in the following sections.

Piping engineering and design specifications for Unit 2 allow the use of various types of branch connections, including pipe-to-pipe. Unless a specific branch connection is indicated in the specification or on the piping drawings, an unreinforced pipe-to-pipe connection is used in the pipe stress analysis. No further action is required if the allowable stresses are met. If the allowable stresses are not met, then the piping stress calculation identifies the reinforcement of the branch connection that is required.

For cases where the branch line is decoupled from the run piping, the proper stress intensification factor is used in the analysis of both the branch line and the main run piping. If reinforcement for mechanical loads is required, it is so identified in the piping stress calculation and drawings.

Reinforcement requirements for mechanical loads, identified by the pipe stress calculations, are incorporated on the piping drawings. Pressure reinforcement calculations required by ASME III, paragraph NB-3643, and ANSI B31.1, paragraph 104.3, are performed by the piping fabricator, and additional reinforcement, if required, is identified and added to the fabricated pipe.

3.9A.1.5.1 Static Analysis

The static equation of equilibrium for the idealized system may be written in matrix form, as follows:

$$KU = P-Q \qquad (3.9A\text{-}1)$$

Where:

K = Stiffness matrix for assembled system

U = Nodal displacement vector

P = External forces, weights, etc.

Q = Equivalent thermal forces = $\int_0^L AE\alpha\bar{T}d\ell$

A = Cross section area

E = Young's Modulus

α = Thermal expansion coefficient

$\bar{T}$ = Average wall temperature less 70°F installation temperature

ℓ = Coordinate along pipe axis

L = Length of pipe

The unknown nodal displacements are obtained from one of the piping analysis computer programs (Appendix 3A) by solving this equation using the Gaussian method. The nodal displacements are then applied to the individual members, and member stiffnesses are used to find internal forces. The nodal displacements at support locations can be used along with the support stiffness to determine support reactions.

<u>Dead Loads (Weight, Pressure) and Live Loads</u>

The effect of pressure, and the combined effects of weight, contents, and insulation, are calculated using one of the piping analysis computer programs (Appendix 3A). The analysis for deadweight assumes all flexible restraints, such as spring hangers, to be rigid. If a pipe has different contents (medium) and therefore different weights in various flow modes, this is taken into consideration. Other details are discussed in Section 3.7A.3.8.3. Live loads are considered if they are expected to constitute a significant component of the total mechanical load.

The filling of MSLs with water during vessel flooding and alternate shutdown events is indicated on the main steam thermal transients and considered in pipe stress analysis in accordance with NB-3600. Spring hangers are designed to carry the full water-filled piping load during hydrotest. Additional deadweight stress as a result of filling the piping with water is considered in the NB-3600 analysis of the system.

The main steam SRV discharge piping has been designed and qualified for the steam hammer load due to steam blowdown. The results of the BWR Owners' Group (BWROG) Safety Relief Valve Test Program, in which Unit 2 has been a participant, show that the measured spring and support response was significantly less for water than steam. The test report, as documented in NEDE-24988-P, stated that "the maximum pipe response due to liquid discharge was generally less than 30 percent of that due to steam discharge." The test program was established to measure the SRV discharge line (SRVDL) response for alternate shutdown cooling conditions and to compare these loads with steam loads.

Additional deadweight stress resulting from water-filled main steam safety relief is considered in the ND-3600 analysis of the main steam relief valve lines.

Initial Displacements (Anchor Movements)

The piping analysis computer programs (Appendix 3A) permit calculation of the thermal initial support displacements combined with the thermal response due to the average pipe wall temperature change.

Earthquake anchor movements are considered (Section 3.7A.3.8.3). In ASME Safety Class I analysis the loads due to OBE anchor movements are combined with the OBE inertia loads via absolute summation. In ASME Safety Class 2 and 3 analysis the Code permits their exclusion from occasional loads if they are included with the thermal expansion loads.

Thermal Loads

A piping system may experience various operating modes. All operating modes are modeled as follows: Portions of piping with flowing medium have the temperature of the medium, while inactive branches have ambient temperature. Nonuniform temperature distributions along the pipe near branch connections of active and inactive legs are considered.

In Safety Class I analysis, stresses due to temperature distribution across the thickness of the pipe wall and geometric and material discontinuities during thermal transients must be considered. These are represented in ASME Section III, Subarticle NB-3600, by:

$$E \propto \Delta T_1,\ E \propto \Delta T_2, \text{ and } E_{ab}\ (\alpha_a T_a - \alpha_b T_b) \qquad (3.9A\text{-}2)$$

Based on geometry, fluid type, insulation, thermal transients, environmental data:

$$\propto \Delta T_1,\ \propto \Delta T_2, \text{ and } E_{ab}\ (\alpha_a T_a - \alpha_b T_b) \qquad (3.9A\text{-}3)$$

are obtained from the HTLOAD program (Appendix 3A), or hand calculations.

3.9A.1.5.2 Occasional Dynamic Loads Excluding Seismic and Hydrodynamic Inertia Loads

Occasional loads are also analyzed using one of the piping analysis computer programs (Appendix 3A). In the matrix equation of motion:

$$M\ddot{U} + C\dot{U} + KU = F(t) \qquad (3.9A\text{-}4)$$

Where:

M = Mass matrix

C = Damping matrix

K = Stiffness matrix

U = Displacement vector

the forcing function F(t) is applied as a set of force time histories, one for each mass degree-of-freedom that experiences a dynamic load.

<u>Fluid Transients</u>

Fluid transients are considered in the following systems:

1. Main steam and main steam bypass systems.
2. Main steam SRV discharge system.
3. Moisture separator/reheater safety relief system.
4. Feedwater system.

5. ECCS, including ECCS pressure relief valve discharge piping.

6. SWP system.

7. RHR system.

8. RCIC system.

9. RWCU system.

10. SLCS system.

11. CRD system.

The computer programs (Appendix 3A) used to calculate these force time-histories due to water hammer, steam hammer, and pipe with air trapped in water lines, are WATHAM, STEHAM, and WATAIR, respectively.

Jet Impingement

The effects of direct jet impingement on piping are evaluated after all other piping analyses are completed and targets from all postulated breaks have been identified.

Relief Valve Reactions (Other Than Main Steam SRVs)

Valves that are subjected to jet reaction forces are either supported by static restraints adjacent to the valve body, in such a manner that the effects on the piping outside these restraints can be neglected, or the piping system is analyzed for relief valve discharge load case.

Suppression Pool Induced Dynamic Loads in the Reactor Building

These loads are described and assessed in the DAR (Appendix 6A).

3.9A.1.5.3 Field-Run Piping

There is no field-run ASME safety class piping in Unit 2.

3.9A.1.5.4 Load Combinations and Stress Criteria

In detailed analyses of ASME safety class piping systems, the individual load cases are combined as shown in Table 3.9A-2.

In the simplified analysis for small bore piping (Section 3.7A.3.8) the same principle is followed; however, the resulting seismic spans, thermal offsets, and support loads are bounding values determined from several fundamental configurations.

The classification for ASME Safety Class 1, 2, and 3 piping systems according to type of analysis is given in Table 3.9A-3.

3.9A.1.6 Safety-Related HVAC Ductwork and Supports

Safety-related duct systems are designed for internal pressure, deadweight, and dynamic loads which result from seismic events and plant operating conditions. Dynamic loads are applied statically as 'g' forces taken from building ARS curves. The 'g' values are taken as either maximum or the 'g' corresponding to the system natural frequency. Ductwork is qualified to the SMACNA Duct Construction Standards and the AISI Code; duct supports are qualified to the AISC Code.

3.9A.2 Dynamic Testing and Analysis

3.9A.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

A detailed preoperational test program was submitted 60 days before the start of the tests, as required by RG 1.68.

3.9A.2.1.1 Flow Modes

Tabulated flow modes for various systems are provided as part of the above test program.

3.9A.2.1.2 Preoperational Vibration Testing

Safety-related piping systems designated as Safety Class 1, 2, or 3 are designed in accordance with ASME Section III. Each system is designed to withstand dynamic loadings from operational transient conditions that are encountered during expected service as required by Paragraphs NB-3622, NC-3622, and ND-3622 of the ASME Code.

To verify that piping systems would withstand operational vibration conditions, a vibration monitoring program was implemented which included both safety-related and nonsafety-related process piping and instrument lines. A vibration monitoring test specification was prepared to categorize the requirements for the test program. Safety-related systems are categorized as follows:

a. Systems With Flow - Accessible lines (including attached instrument lines) were monitored visually or with hand-held instruments, and inaccessible lines and lines with transient vibrations were monitored by remote instrumentation.

b. Other Systems - No testing was required.

Instrument lines connected to inaccessible process lines were not individually monitored. Instrument lines were considered acceptable from a steady-state vibration point of view if the vibration of the process pipe to which the instrument lines are connected was within the acceptance test limits. If the vibration levels in the process pipe were above the acceptable test limits, consideration was given to the connected instrument lines.

During the vibration monitoring program, vibration testing was performed either during the preoperation or power ascension testing phases on the systems identified below.

System	Preoperation Phase	Power Ascension Phase*
Low-Pressure Core Spray (CSL)	X	
High-Pressure Core Spray (CSH)	X	
Reactor Water Cleanup (WCS)	X	
Feedwater (FWS)	X	X
Spent Fuel Pool Cooling (SFC)	X	
Service Water (SWP)	X	
Residual Heat Removal (RHS)	X	X
Main Steam (MSS)		X
Main Steam Safety Relief (SVV)		X
Air Startup Standby Diesel Generator (EGA)	X	
Service Air (SAS)	X	
Reactor Core Isolation Cooling (ICS)		X
Condensate (CNM)		X
Standby Liquid Control (SLS)	X	
Control Building Chilled Water (HVK)	X	
Instrument Air (IAS)	X	
Reactor Building Closed Loop Cooling Water (CCP)	X	
Nitrogen System (GSN)	X	
Standby Gas Treatment (GTS)	X	
Containment Purge System (CPS)	X	
Reactor Coolant Recirculation (RCS)		X
Control Rod Drive (RDS)	X	
Nuclear Boiler Instrumentation (ISC)		X

See Section 3.9B.2.1 for vibration testing of GE-supplied systems.

Vibration measurements were conducted for steady-state and transient conditions such as pump starts and valve operation. Also, visual inspections to determine vibration response were performed, with emphasis placed on vents, drains, and branch piping.

* Testing on these systems is accomplished during the startup test phase as described in Table 14.2-303.

3.9A.2.1.3 Preoperational Thermal Expansion Testing

Preoperational tests for BWRs are conducted at near ambient conditions; therefore, thermal expansion testing during the preoperational test phase is very limited. For the systems delineated in Section 3.9A.2.1.2 that are operated at other than ambient conditions during the preoperational test phase, pipe deflections are observed or measured at selected locations. The startup expansion testing program is discussed in further detail in Section 3.9B.2.1.2.

3.9A.2.1.4 Measurement Locations

The exact locations of measuring devices and identification of visual inspection points are supplied in the test program. Measurements taken at points with dynamic instrumentation show whether the stress and fatigue limits are within acceptable levels, and measurements taken at points with expansion instrumentation in an expansion test, excluding dynamic effects, are checked against displacement criteria.

3.9A.2.1.5 Acceptance Criteria

Acceptance criteria for vibrations were dependent upon whether steady-state or transient vibration was measured.

For steady-state vibrations, acceptance criteria were based on ANSI/ASME OM3-1982 rules. The majority of the piping was tested by a two-phase process. Phase 1 consisted of visually observing the pipe to determine if a vibration was perceived. If vibration was not observed, that portion of the pipe was acceptable. If vibration was observed, Phase 2 was implemented. This consisted of taking local measurements using hand-held instruments at points where steady-state vibration was observed. Vibration velocity was measured and, if it was less than 0.5 in/sec, the piping was acceptable. At velocities equal to or greater than 0.5 in/sec, displacement measurements were taken and forwarded to engineering for resolution.

For the remaining piping, where significant steady-state vibration was anticipated, or where inaccessible for normal viewing, vibration was monitored by fixed displacement transducers (lanyard potentiometers) with remote readouts. The recorded displacements were compared to the acceptance criteria as determined by ANSI/ASME OM3-1982. If the acceptance criteria were exceeded, the recorded displacements were evaluated by engineering to determine a resolution.

For all steady-state vibration, OM3-1982 guidelines were used; however, displacements for carbon steel were based on 80 percent of stress endurance limits divided by a factor of 1.3. Displacements for stainless steel piping were based on stress allowables for 10E11 cycles, as shown on Figure I-9-2 of ASME III of the 1983 Code. Curve C of the figure was used for initial screening. If detailed analysis was required, Curve B was used in accordance with Code requirements.

For transient vibration testing, vibration also was measured by fixed displacement transducers (lanyard potentiometers) with remote readouts, and two levels of acceptance criteria were used.

1. Level 1 criteria establish the maximum limits for the level of pipe motion which, if exceeded, mandates a test hold or termination. Level 1 criteria ensure that the pipe stress level will not exceed $1.2S_h$, the applicable Code allowable.

   The displacement limits for Level 1 criteria were determined from those predicted for loading conditions that were used to evaluate the applicable Code equation for an occasional load. If any Level 1 criteria were exceeded, an engineering evaluation was performed to develop corrective action or show that the measured results were acceptable.

2. Level 2 criteria are based on pipe stresses as analyzed and predicted for the fluid transient for the particular event. If any Level 2 limits were exceeded, a detailed engineering evaluation was performed to develop corrective action or show that the measured results were acceptable.

Acceptance criteria for vibration on systems listed in Section 3.9A.2.1.2 are specified in the vibration test program. The stress calculated based on measured displacements represents the combined stress of pressure, deadweight, and fluid transient loads, and was combined with the analytical stress of the load cases not simulated, such as the OBE, and then compared with the combined analytical result. The allowable stresses are listed in Table 3.9A-2.

The limits for thermal displacements depend on the equipment design parameters. Under all plant conditions the piping is not permitted to touch another object that may interfere with the operation of the piping system or equipment.

3.9A.2.1.6 Corrective Actions

If during the vibration test it should be noted that the vibrations are beyond the acceptable design level, additional supports and restraints may be provided. The possibility of piping rerouting would also be considered, and a reanalysis or retest would be performed to assure that the design meets the acceptance criteria.

Similarly, if the design tolerances for thermal displacements are not satisfied at a point along the piping, the equipment affected can usually be realigned. Otherwise, supports and restraints would be rearranged, and pipe rerouting would also be considered.

3.9A.2.2 Seismic Qualification of Safety-Related Mechanical Equipment

This section provides the qualification criteria and methods for equipment affected by seismic loads. The methods for the qualification of equipment affected by hydrodynamic loads associated with SRV discharge and the postulated LOCA are provided in the DAR, Appendix 6A, Subsection 6A.9.

3.9A.2.2.1 Seismic Qualification Criteria

The purpose of qualifying Category I mechanical equipment is to demonstrate its ability to perform a safety-related function during and after a postulated seismic occurrence of a magnitude up to and including the SSE. Equipment that does not perform any safety-related function, but whose failure could jeopardize the function of Category I equipment, is required only to maintain its structural integrity.

Seismic qualification of equipment is accomplished by one of the four methods discussed in Section 3.7A.3.1. Analysis is used to demonstrate structural integrity of the equipment. When mechanical equipment is qualified by analysis, the calculated stresses are maintained within the specified allowables that contain the required margins of safety described in Section 3.9A.2.2.2. Where the equipment is classified as active, additional deflection analysis and/or testing is performed. Details of qualification methods for specific equipment are contained in Table 3.9A-4.

These methods are applied to mechanical equipment as follows.

<u>Analysis</u>

The listing below is for equipment where the maintenance of structural integrity only is required to assure performance of the design-intended function. This equipment is qualified by analysis:

1. Piping.
2. Ductwork.
3. Tanks and vessels.
4. Heat exchangers.
5. HVAC - passive components.
6. Pump and valve pressure boundary parts that are not required to operate and perform a safety function.

Analysis is also used to qualify rotating machinery items where verification must be obtained to demonstrate that deformations resulting from seismic loadings do not cause binding of the rotating element, to the extent that the component cannot perform its design-intended function. Components in this category include:

1. Active pumps and valves.

2. Fans and dampers.

The large size and weight of some of these components, together with the difficulties encountered in applying operating loads during dynamic testing, serve to make analysis the most viable qualification method for the rotating machine elements.

Dynamic Testing

The following equipment whose functional capability cannot be adequately demonstrated by analysis is qualified by dynamic testing:

1. Standby diesel generator components.

2. Hydrogen recombiner control panels.

3. Electric motor valve actuators, including limit switches.

4. Pneumatic and hydraulic valve limit switches and solenoid valves.

5. Electrical control panels, relay boards, switchgear and MCCs, and radiation monitoring equipment.

6. Control instrumentation such as flow switches, thermocouples, and transmitters.

7. Batteries, battery chargers, and inverters.

8. Electrical penetrations.

Combination of Analysis with Testing

A combination of analysis with static or dynamic testing is used for seismic qualification of active valves, as follows:

1. The natural frequencies of the valve assembly are determined by analysis or test.

2. A static deflection test is performed to verify that deformation due to seismic loadings does not cause binding of internal valve parts, which prevents valve operations within specified time limits.

3. The electric motor-driven, pneumatic, and hydraulic valve actuator and other electrical appurtenances are qualified by dynamic testing.

For those active valves that are simple in design or do not have significant extended structures or electrical appurtenances, seismic qualification is achieved by analysis alone to ensure that the valve can perform its design-intended function.

Equipment that is qualified by testing is mounted and operated in a manner similar to that of the actual system. For testing procedures refer to Section 3.7A.3.

3.9A.2.2.2 Acceptance Criteria

The acceptance criteria used are as follows:

1. Tests, when used, demonstrate that the component performs its required safety function during and after the test. The TRS envelope the applicable frequency range of the RRS with the required 10-percent margin in accordance with IEEE-323-1974. Where the TRS does not envelope the RRS with the suggested margins of IEEE-323-1974, a justification is provided.

2. Analysis, when used, verifies that stresses do not exceed the specified allowable stress limits for the loading conditions shown in Tables 3.9A-5 and 3.9A-6 and that deformations do not exceed those which will not permit the component to perform its design-intended function.

For ASME components, the specified allowable stress limits are those shown in Tables 3.9A-7 and 3.9A-8.

For non-ASME components, the Design Condition I loading has allowable stresses limited to 75 percent of the minimum yield strength at the design temperature of the material, in accordance with applicable ASTM specification. For the Design Condition II loading the stresses do not exceed the smaller of:

1. 100 percent of the minimum yield strength, or

2. 70 percent of the minimum ultimate tensile strength of the material (at temperature), in accordance with the ASTM or equivalent specification for the material.

For definitions of Design Conditions I and II, see Section 3.9A.3.1.2.

3.9A.2.2.3 Seismic Qualification of Specific Non-NSSS Mechanical Equipment

Piping

All safety-related piping, including piping in pipe tunnels, is seismically analyzed in accordance with Section 3.7A.3.8.

Tanks

The safety-related tanks have been seismically qualified as follows.

The seismic analysis on the buried standby diesel generator fuel oil storage tank consisted of the following:

1. Selection of the applicable seismic acceleration factors at the elevation in the diesel generator building at which the tank is installed.

2. Calculation of the lowest natural frequency of the filled tank in its buried environment taking into account both the mass and spring rate of this environment. This frequency occurs in the rigid range.

3. Choice of the correct seismic factors by combining analysis parameters 1 and 2.

4. Determination of loads on both the tank and support rings by static analysis with seismic g-factors applied to all tank and sand masses.

5. ASME Code methods for the design of the tank shell, heads, stiffening, and support rings were used. Local stress analysis, by BIJLAARD or other methods, as appropriate, was used in determining stresses at nozzles and support rings.

6. Analyses were performed for both normal and upset conditions (including live and dead loads, thermal and pressure stresses, and OBE seismic factors) and faulted conditions composed of live and dead loads plus full SSE inertial loads.

7. Adequacy of the tank at design pressure was determined. The tank was hydrotested at 1.5 times design pressure in compliance with ASME Code.

The seismic analysis for the air damper/accumulators, the chilled water expansion tanks, the skimmer surge tanks, and the standby diesel generator fuel oil day tanks, consisted of the following:

1. An analysis of the vessel was performed to prove that it has rigid characteristics, i.e., the natural

frequency of vibration of the predominant mode of the supported vessel is in the flat portion of the applicable response spectrum curves. The applicable seismic acceleration coefficients were chosen according to the location of each vessel.

2. The seismic acceleration coefficients were applied statically, and a static analysis was performed on the equipment and supports. The vertical and horizontal seismic effects were applied simultaneously to the subject vessel at its gravitational center for the seismic load calculation and design.

3. Determination of loads for both the tanks and supports by static analysis with seismic coefficients applied to all tank masses.

4. The remainder of the analysis was performed according to preceding steps 5 through 7 for safety-related tanks.

Since the ADS and main steam SRV accumulators are located inside the reactor building, seismic as well as hydrodynamic effects were considered in their analysis. The preceding Steps 1 and 2 were therefore performed with the applicable acceleration coefficients.

Pumps

Qualification of pumps is shown in Table 3.9A-9 and further discussed in Section 3.9A.3.2. The results of tests and analyses are described in Table 3.9A-4 for pumps listed in Table 3.9A-9.

Valves

The qualification of active valves is discussed in Section 3.9A.3.2. The results of tests and analyses are described in Table 3.9A-4 for the valves listed in Table 3.9A-12.

There are no manually operated valves which must change position for any safety system to perform its function in the short term, following any event. The operation of certain manual valves may be required in the long term. These valves include those necessary to replenish fuel oil to the diesel generator fuel oil storage tanks and nitrogen to the ADS valve accumulator receiving tanks, and to accomplish boron replenishment in the SLCS following an anticipated transient without scram (ATWS) event. The only other valves which may be required to change position to accomplish a safety function are those RHR valves located in the SFC/RHR interties. As discussed in Section 9.1.3.3, these interties may be used to provide additional fuel pool cooling following a full core offload.

Other Mechanical Equipment

The qualification method for mechanical equipment other than the above is discussed in Section 3.7A.3. The qualification results are described in Table 3.9A-4.

Electrical Equipment and Instrumentation

The seismic qualification criteria and methods of qualification of Category I electrical equipment and instrumentation, other than those items discussed in this section, are described in Section 3.10.

Cranes

Cranes are seismically qualified in accordance with the following criteria:

1. The possibility of the crane being dislodged by a seismic disturbance is precluded.

2. No part of the crane becomes detached and falls during an earthquake.

3. The crane load will not lower in an uncontrolled manner during, or as the result of, an earthquake.

3.9A.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9A.3.1 Loading Combinations, Design Transients, and Stress Limits

The design basis for all safety-related piping, components, equipment, and supports considers all applied loads such as pressure temperature, deadweight, external mechanical, thermal, fluid transient, seismic, and hydrodynamic loads.

Hydrodynamic loads are unique to the Mark II containment of Unit 2 and other similar suppression pool-type containments. The design basis for all safety-related piping, components, and equipment subjected to hydrodynamic loads meets the requirements of the following NRC documents:

1. NUREG-0487, Supplements 1 and 2, Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria.

2. NUREG-0808, Mark II Containment Program Load Evaluation and Acceptance Criteria.

3. NUREG-0802, Safety/Relief Valve - Quencher Loads Evaluation Reports - BWR Mark II and III Containments.

4. NUREG-0783, Suppression Pool Temperature Limits for BWR Containments.

5. NUREG-0763, Guidelines for Confirmatory Inplant Tests of Safety-Relief Valve Discharges for BWR Plants.

All safety-related equipment, piping, and components and their supports located in the reactor building are evaluated using hydrodynamic loads. All other structures are not affected by hydrodynamic loads.

See Appendix 6A, Design Assessment Report, for further details.

3.9A.3.1.1 ASME Section III, Class 1 Components

Equipment

ASME III, Class 1 mechanical equipment, i.e., valves, pumps, and cooling coils, is designed in accordance with ASME Section III, Subsection NB. Loading combinations and service conditions are outlined in Table 3.9A-5. Corresponding stress limits in accordance with Article NB-3000 are listed in Table 3.9A-7. This equipment is listed in Table 3.9A-4.

These load descriptions include Dynamic Load 1, Dynamic Load 2, and Dynamic Load 3 notations, which are load combinations for equipment in the reactor building only, resulting from consideration of hydrodynamic loading conditions; for equipment outside the reactor building, these reduce to OBE, SSE, and OBE, respectively.

For the conditions specified, the allowable stress limits defined in Table 3.9A-7 are applicable to stress results obtained by elastic analysis techniques. The analysis methods described in Section 3.7A.3 are used in implementing this criterion. Computer programs used in these analyses are discussed in Appendix 3A.

Piping

The pipe stress analysis load combinations and stress limits for ASME Class 1 piping are given in Table 3.9A-2. The design transients and number of associated stress cycles for the various plant conditions are given in Table 3.9A-1, which includes the dynamic load events OBE, SSE, LOCA-related load cases, and SRV discharge cases. The suppression pool events are discussed in Appendix 6A. There are several SRV cases. In Table 3.9A-2 SRV refers to the envelope of the response of all SRV cases applicable to a particular load combination. The number of load cycles used for different SRV cases is given in Table 3.9A-1. Under emergency and faulted conditions no fatigue analysis need be performed.

Figures 3.9A-6 through 3.9A-67 are typical examples of response spectra for the load conditions of:

1. Seismic OBE.

2. Seismic SSE.

3. SRV loads.

4. LOCA-related loads

    a. Chugging.
    b. Basic CO.
    c. ADS CO.

5. Seismic OBE 2-5 percent damping in accordance with Code Case N-411.

6. Seismic SSE 2-5 percent damping in accordance with Code Case N-411.

These response spectras are provided at the following locations:

1. Top of RPV.

2. Top of BSW.

3. Primary containment at the suppression pool water level.

4. Reactor building mat (only hydrodynamic loads are provided).

The provided response spectra have been broadened in accordance with RG 1.122. Both vertical and horizontal spectra are provided for each location.

In order to ensure their continued operation during emergency and faulted events, ECCS and other essential systems are required to meet the functional capability criteria of NEDO-21985, Functional Capability Criteria of Essential Mark II Piping, September 1978.

ASME Class 1 piping meets the criteria of ASME Section III, 1974 Edition, and 10CFR50.55a, Section (d).

Analysis of the individual load cases is described or referred to in Section 3.9A.1.5.

ASME Code Class 1 piping fatigue evaluation was performed for the SRV piping in the suppression pool area. All of the thermal and dynamic loads and respective operating cycle data were used for evaluation of the SRV line. The CUF obtained from the analysis is less than 1.0; hence, no fatigue crack is anticipated due to all of the prescribed loads.

ASME Code Class 1 piping fatigue evaluation was performed for the downcomers. The CUF obtained from the analysis is less than 1.0;

hence, no fatigue crack is anticipated due to all of the prescribed loads.

3.9A.3.1.2 ASME Class 2 and 3 Components

Equipment

Tables 3.9A-6 and 3.9A-8 list loading conditions and stress limits for ASME Section III, Class 2 and 3 components of the Category I fluid systems constructed in accordance with ASME Section III, Subsections NC and ND. These conditions are:

1. Design Condition I Includes the specified design loads (temperature, pressure, etc.), plus Dynamic Load 1 loads.

2. Design Condition II Includes the specified design loads (as above), plus Dynamic Load 2 loads, plus pipe rupture loads (if applicable).

The design load combinations are analogous to either the Code Class 1 normal or upset conditions for Design Condition I and to the faulted condition for Design Condition II. See Table 3.9A-5 for the definitions of Dynamic Load 1 and Dynamic Load 2.

These requirements, which supplement the present scope of ASME Section III, Subsections NC and ND, are consistent with the present Code format and philosophy. Further extension of terminology (normal, upset, etc.) is not required, since Code Class 2 and 3 systems are not generally evaluated for such varieties of operating conditions and transients, but rather to design conditions which conservatively envelop all operating conditions.

Generally, only design conditions of pressure and temperature are necessary to satisfy ASME Code requirements. These conditions envelop all service level conditions for the component such as normal, upset, emergency, and faulted plant conditions. Use of design conditions plus seismic loading is therefore a conservative criterion.

The stress limits and design conditions presented in Table 3.9A-8 are intended to ensure that no gross deformation of the component occurs. These limits are applicable for an elastic system (and component) analysis. No inelastic analysis has been performed for any ASME Class 2 or 3 component.

Piping Systems

The load combinations and stress limits for ASME Class 2 and 3 piping are given in Table 3.9A-2. They conform to the criteria of ASME Section III, which imply elastic analysis. Under faulted condition, with primary stress limit 2.4 S_H, gross inelastic deformations that may occur are permitted by the Code.

Analysis of the individual load cases is described or referred to in Section 3.9A.1.5. The application of detailed or simplified analysis depends on criteria stated in Table 3.9A-3.

Typical examples of ARS used in the design of piping systems are described in Section 3.9A.3.1.1.

3.9A.3.1.3 Compliance with Regulatory Guide 1.48

Unit 2 compliance to the regulatory guide is documented in Table 1.8-1.

3.9A.3.2 Pump and Valve Operability Assurance

This section provides the operability assurance programs for pumps and valves affected by seismic loads. The operability assurance programs for pumps and valves affected by hydrodynamic loads associated with SRV discharge and the postulated LOCA are provided in the DAR, Appendix 6A, Subsection 6A.9.

Active pumps and valves are those whose operability is relied upon to perform a safety function such as safe shutdown of the reactor or mitigation of the consequences of a postulated accident. Pumps and valves installed in seismic Category I piping systems are designed in accordance with the requirement of ASME Section III, Subsections NB, NC, and ND. Active pumps and valves are listed in Tables 3.9A-9 and 3.9A-12, respectively.

Active valves are qualified by testing and analysis, and active pumps by testing and analysis with appropriate stress limits and nozzle loads. The content of these programs is detailed in the following sections.

3.9A.3.2.1 Pump Operability Program

All active pumps are qualified for operability by being subjected to tests both prior to installation in the plant and after installation in the plant. The in-shop tests include:

1. Hydrostatic tests to ASME Section III requirements.

2. Performance tests while the pump is operated with flow to determine total developed head, minimum and maximum head, net positive suction head (NPSH) requirements and other pump/motor parameters. As a result of these tests, a certified pump curve is developed for each pump that may be used to verify continued satisfactory operation subsequent to pump installation. Proper seal function is verified during the performance test.

Also monitored during these operational tests are bearing temperatures and vibration levels that are shown to be below appropriate limits specified to the manufacturer for design of each active pump.

After the pump is installed in the plant, it undergoes cold hydro tests, preoperational tests, and the required periodic ISI and operational tests as applicable.

These tests demonstrate reliability of the pump for the design life of the plant.

In addition to these tests, active pumps are qualified for operability during a SSE condition to assure that 1) the pump is not damaged during the seismic event, and 2) the pump continues operating when subjected to the SSE loads.

The pump manufacturer is required to show that the pump operates normally when subjected to the maximum applicable amplified seismic (floor) accelerations, attached piping nozzle loads, and dynamic system loads associated with the faulted plant operating condition. Analysis procedures are utilized in accordance with those outlined in Section 3.7A.3. Natural frequencies are determined in order to obtain maximum seismic accelerations based on applicable amplified (floor) response spectra.

In order to avoid damage during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE, and dynamic system loads are limited to the values indicated in Table 3.9A-8. The maximum seismic nozzle loads are also considered in an analysis of the pump supports to assure that a system misalignment cannot occur. A static shaft deflection analysis of the rotor is performed with horizontal and vertical accelerations based on floor response levels. The deflections determined from the static shaft analysis are compared to allowable rotor clearances. The results of the pump stress/deflection analyses are summarized in Table 3.9A-10.

Performing these analyses with the conservative loads stated, and with the restrictive stress limits of Table 3.9A-8 as allowables, assures that critical parts of the pump are not damaged during the short duration of the faulted condition; therefore, the reliability of the pump for postfaulted condition operation is not impaired by the seismic event.

In addition to the postfaulted condition operation, it is necessary to assure that the pump functions throughout the SSE. The pump/motor combination is designed to rotate at a constant speed under all conditions unless the rotor becomes completely seized, i.e., no rotation. Typically, the rotor can be seized 5 full seconds before a circuit breaker trip shuts down the pump to prevent damage to the motor. However, the high rotary inertia in the operating pump rotor, and the random nature and short duration loading characteristics of the seismic event prevent the rotor from becoming seized. In actuality, the seismic loadings cause only a slight increase, if any, in the torque (i.e., motor current) necessary to drive the pump at the constant design speed. Therefore, the pump does not shut down during the SSE and operates at the design speed despite the SSE loads.

When seismic testing of the pump assembly is impractical, a seismic analysis is performed on the pump assembly to ensure operability. The analysis considers the pump, motor, and supporting structures together. In addition, the pump motor is independently qualified for operation during the maximum seismic event. Any auxiliary equipment that is identified to be vital to the operation of the pump or pump motor, and that is not qualified for operation during the pump analysis or motor qualifications, is separately qualified for operation at the accelerations that it would experience at its mounting. The pump motor and vital auxiliary equipment are qualified by meeting the requirements of IEEE-344-1975.

The functional ability of active pumps after a faulted condition is assured since only normal operating loads and steady-state nozzle loads exist. Since it is demonstrated that the pumps would not be damaged during the faulted condition, the postfaulted condition operating loads are identical to the normal plant operating loads. This is assured by requiring that the imposed nozzle loads (steady-state loads) for normal conditions and postfaulted conditions are limited by the magnitudes of the normal condition nozzle loads. The postfaulted condition ability of the pumps to function under these applied loads is proven during the normal operating plant conditions for active pumps. The active pump motors are qualified to operate satisfactorily when subjected to their surrounding environmental conditions for both normal operation and postaccident operation by meeting the requirements of IEEE-323-1974 (Section 3.11).

3.9A.3.2.2 Valve Operability Program

Safety-related active valves are required to perform their mechanical function during and/or after the course of a postulated accident. Assurance must be supplied that these valves can operate during and/or after a seismic event. Qualification tests accompanied by analyses are conducted for all active valves.

Valves without significant extended structures are considered seismically adequate as a result of piping seismic adequacy.

For valves with operators having significantly extended structures, an analysis is performed for static equivalent seismic SSE loads applied at the center of gravity of the extended structure. The maximum stress limits allowed in these analyses ensure the maintenance of structural integrity. The limits used for valves are shown in Tables 3.9A-7 and 3.9A-8, depending upon the class.

The safety-related valves are also subjected to a series of tests prior to service and during the plant life. Prior to installation, the following tests are performed: shell hydrostatic test to ASME Section III requirements; back seat and main seat leakage tests; disc hydrostatic test; and functional tests to verify that the valve opens and closes within the specified time limits, when subjected to the design differential pressure and operability qualification of motor operators for the environmental conditions over the installed life (i.e., aging, radiation, accident environment simulation, etc.) according to IEEE-323-1974. Cold hydro qualification tests, preoperational tests, periodic ISIs, and periodic in-service operation are performed to verify and assure the functional ability of the valve.

In addition to these tests and analyses, representative active valves of each design type, pressure, and size group are tested for verification of operability during a simulated seismic event, by demonstrating operational capabilities within the specified limits. The basic criteria used in selecting the representative value for qualification testing is based on an evaluation of the following parameters:

1. Assembly weight

2. Size, type, and pressure ratings

3. Actuator type and performance characteristics

4. Mounting arrangement and appurtenances

The methodology utilized in assessing the degree of similarity of evaluating the differences follows generally the guidelines of ANSI Standard B16.41-1983, Functional Qualification Requirements for Power Operated Active Valve Assemblies for Nuclear Power Plants.

The proposed testing procedures are as follows. The valve is mounted in a manner that conservatively represents the actual valve installation. The valve assembly includes the operator and all appurtenances normally attached to the valve in service. The operability of the valve during a SSE is demonstrated by satisfying the following criteria:

1. All the active valves are required to have a fundamental natural frequency that is generally greater than 33 Hz. This is shown by suitable test or analysis.

2. The valve is operated in the normal unloaded position for baseline data. The actuator and yoke of the valve system are then statically loaded by an amount equal to that determined from an analysis as representing SSE accelerations applied at the center of gravity of the operator about the weaker axis of the yoke. The design differential pressure of the valve is simultaneously applied to the valve during the static deflection tests.

3. The valve is then operated while in the deflected position, i.e., from the normal operating mode to the faulted operating mode. The valve is again operated in the normal position after the static load is removed. The valve is required to perform its safety-related function within the specified operating time limits in both the deflected and the normal position.

4. Electric motor operators and other electrical appurtenances necessary for operation are qualified in accordance with IEEE-323-1974 and IEEE-344-1975.

5. Environmental qualification of nonmetallic components in accordance with EQD Section 4.1.3.

The accelerations used for the valve qualification are generally 3.0 g horizontal and 3.0 g vertical. The piping design maintains the motor operator accelerations to these levels with an adequate margin of safety.

Selection of parent valve operators for testing generally follows the methodology outlined in Appendix A of IEEE-382-1980, IEEE Standard for Qualification of Safety-Related Valve Actuators. This standard provides a comprehensive method of analyzing all the relevant parameters of valve operators such as: type, size, weight, electrical characteristics (ac/dc, voltage, current) performance characteristics (speed, motor torque), and materials for the selection of a valve operator for qualification testing.

Consideration is also given to the effects of thermal and vibration aging in conjunction with applicable margins by utilizing the worst-case parameters in qualification.

Testing is conducted on a representative number of valves from each of the primary safety-related design types. Selected valve sizes are qualified by the tests and the results used to qualify that group of valves which the tested valve represents. Stress and deformation analyses are used to support the interpolation.

An assessment of the stresses at the pipe/valve interface generally indicates that distortion, if any, due to seismic loads will not cause binding of internal components. Therefore, additions of piping and loads during the operability tests is unnecessary.

For valves where stresses in the valve body could be significant, the piping and loads were imposed during the operability tests. Examples include solenoid valves and air-operated control valves.

For selected "active" valve categories specific qualification programs are conducted to demonstrate operability. The method of qualification for these valves is detailed as follows:

1. Butterfly Valves

The containment and drywell vent/pipe isolation valves are evaluated for operability during and after a postulated accident by both analyses and testing methods.

a. The valve assembly is analytically evaluated and shown to perform its safety-related function (i.e., to close within the required response time). Valve analysis considers seismic, hydrodynamic, operating, air flow, and LOCA loads.

b. The valve assembly is statically loaded by an amount equal in magnitude to the dynamic force and applied at the actuator C.G. The design pressure of the valve is simultaneously applied and the valve is operated while in the deflected position.

c. Electrical appurtenances (limit switches and solenoid-operated valves [SOVs]) are qualified according to the requirements of IEEE-323-1974 and IEEE-344-1975.

d. In addition, assurance of operability is demonstrated by the following tests:

   (1) In-shop shell hydrostatic tests

   (2) Cold cyclic tests

   (3) Seat leakage tests

   (4) Pre/postinstallation functional tests

2. Check Valves

Check valves are characteristically simple in design, and their operation is not affected by seismic accelerations or the applied piping end loads. Check valve design is compact, and there are no extended structures or masses whose motion could cause distortions or restrict operation of the valve. The piping end loads due to maximum seismic excitation do not affect the functional ability of the valve since clearance is provided between the valve disc and the casing wall. This clearance around the disc prevents the disc from becoming bound or restricted due to any casing distortions caused by piping end loads. Therefore, the design of these valves is such that when the structural integrity of the valve is assured, using standard design or analysis methods, the ability of the valve to operate is assured by the design features. In addition to these design considerations, the valves are also subjected to the following tests and analysis:

a. Stress analysis, including the SSE loads.

b. In-shop hydrostatic test.

c. In-shop seat leakage test.

d. Periodic in situ valve exercising and inspection to assure the functional ability of the valve.

For the feedwater check valve, the operability following a postulated feedwater line break is also demonstrated. The maximum disc impact velocity and the pressure differential across the disc are determined. A stress analysis of the valve, which considers the impact and the seismic inertia loads, demonstrates valve design adequacy.

3. <u>Safety Relief Valves</u>

SRVs are evaluated for operability during and after a postulated accident of both analyses and testing methods.

a. The valve is analytically evaluated for seismic/hydrodynamic and operating loads and shown to perform its safety-related functions.

b. The valve is statically loaded by an amount equal in magnitude to the dynamic force. A pressure, representative of the design pressure, is simultaneously applied and the valve is operated while in the deflected positions.

c. In addition, assurance of operability is demonstrated by the following tests:

(1) In-shop hydrostatic seat leakage tests.

(2) In-shop hydrostatic body leakage tests.

(3) Performance tests.

(4) Periodic in situ valve inspections and an applicable periodic valve removal, refurbishment, and performance testing.

Using the methods described, all the safety-related valves in the system are qualified for operability during a seismic event. These methods conservatively simulate the seismic event and ensure that the active valves can perform their safety-related function when necessary.

3.9A.3.3 Design and Installation Details for Mounting of Pressure-Relief Devices

Pressure-relieving devices for ASME Safety Class 1 and 2 system components are:

1. Main steam SRVs.

2. SRVs for protecting RHR system heat exchangers.

The design and installation of main steam SRVs is described in Section 3.9B.3.3.

The design and installation of SRVs for protecting the RHR system heat exchangers (Section 5.4.7.2.3) is in accordance with ASME Section III, Article NC-7000, and RG 1.67.

Piping to and from SRVs is designed in accordance with ASME Section III, Paragraph NC-3677.

The STEHAM computer program (Appendix 3A) is used to calculate fluid transient forces in each piping segment (straight pipe piece between two elbows, an elbow and a tee, or an elbow and a terminal end) downstream of the SRVs. A conservatively low value of valve opening time is used in this calculation. Water slugs in pipe segments ending in the suppression pool are taken into account.

Dynamic stresses in the piping are computed by time-history integration or the equivalent static methods using one of the piping analysis computer programs described in Appendix 3A. These stresses are combined with those due to other mechanical loads, in accordance with load combinations described in Section 3.9A.3.1. Both SRVs protecting a heat exchanger are assumed to discharge concurrently. These loads meet design allowables provided by the vendor.

3.9A.3.4 Component Supports

Expansion anchor bolts, used in supporting the mechanical components from concrete structures, are drilled-in wedge-type uniform hole anchors. Drilled-in, bearing-type, flared hole anchors are also used in supporting the mechanical components from concrete structures.

Drilled-in, wedge-type, uniform hole expansion anchors are designed for a minimum safety factor of four, as determined by the ultimate load tests performed by the manufacturer. The setting torque is determined from in situ tests.

Drilled-in, bearing-type, flared hole anchors are designed for a minimum safety factor of three, as determined by field testing. The loads are transferred into concrete by direct bearing against concrete. The bolt material is capable of reaching full ductility prior to failure. Due to these reasons, the anchors afford greater reliability, and a lower safety factor is justified.

The design, procurement, and installation of building steel comply with requirements of the AISC specification for the design, fabrication, and erection of structural steel for buildings, as described in Sections 3.8.4.2 and 3.8.4.6.3. The examination and inspection of building steel comply with the requirements of NRC RG 1.94, as described in Table 1.8-1.

3.9A.3.4.1 Pipe Supports

The pipe support designs, using base plates and concrete expansion anchor bolts, are performed using the flexibility criteria of NRC IE Bulletin 79-02 before they are released for fabrication. Verification of as-built conditions in accordance with NRC IE Bulletin 79-14 is described in Section 3.7A.3.8.1.

The bases for design and construction of ASME and non-ASME piping supports are given in Table 3.9A-16.

Nonnuclear Piping

Nonnuclear piping supports satisfy the requirements of the American National Standard Code for Pressure Piping, ANSI B31.1-1973, up to and including the Winter 1973 Addenda, paragraph 120 and 121. An exception is taken to paragraph 120.2.4 of this edition by invoking the same paragraph of ANSI B31.1-1980, permitting the use of the 8th Edition - 1980 Edition of the AISC Manual for Steel Construction for the design of partial penetration groove welds in accordance with Table 1.17.5.

Nuclear Piping

Pipe supports for nuclear piping are designed and fabricated in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NF, 1974 Edition, no Addenda dated July 1, 1974, subject to the exceptions and additions listed below. Code stamping of pipe supports is not a requirement of the 1974 Edition. All pipe supports applicable to the RCPB piping or other piping are considered either linear or component standard, with portions of component standard supports designed to plate and shell rules.

The pipe support jurisdictional boundaries are in accordance with NF-1000 (see examples on Figure 3.9A-5). Portions of supports that are integrally attached to piping are designed, including local pipe stresses, in accordance with ASME III, Subsection NB, NC, or ND as applicable. The applicable dimensional standards of Table NB-3691-1 apply.

See Appendix 3E for a discussion of the criteria in the 1974 Edition of ASME Subsection NF regarding stresses in supports due to thermal growth in piping and seismic anchor motion as it compares to similar criteria in the 1983 Edition.

As permitted by NA-1140, the portions of the 1974 Edition of the ASME Code for which specific provisions of later ASME Code addenda or editions are substituted are listed below.

NA-3256 Filing of Design Specifications

The Summer 1978 Addenda dated June 30, 1978, is invoked for the new subparagraph NCA-3256(b) to permit design specifications for component standard supports to be provided by the manufacturer and to permit/facilitate the implementation of ASME III Code Case N-247. See Table 5.2-1.

NA-3352 Stress Reports

The Summer 1978 Addenda dated June 30, 1978, is invoked for paragraph NCA-3351 to permit the use of the term design report in lieu of stress report and to permit/facilitate the implementation of ASME III Code Case N-247. See Table 5.2-1.

NF-1214 Component Standard Supports

The Summer 1976 Addenda dated June 20, 1976, is invoked to delete the specific reference to hydraulic snubbers.

NF-2121 Permitted Material Specifications

The Summer 1974 Addenda dated June 30, 1974, is invoked to permit the use of SA672 material.

NF-2121 Permitted Material Specifications

The Winter 1974 Addenda dated December 31, 1974, is invoked to permit the use of increased allowable stress for SA515 G65.

NF-2121 Permitted Material Specifications

The Summer 1976 Addenda dated June 30, 1976, is invoked to include the new subparagraph NF-2121(c) to permit the exclusion of certain shim stock from the requirements of Article NF-2120.

NF-2121 Permitted Material Specifications

The 1977 Edition dated July 1, 1977, is invoked to permit the use of SA36 material.

NF-2121 Permitted Material Specifications

The 1980 Edition dated July 1, 1980, is invoked to permit the use of SA564, Type 630 material.

NF-2121 Permitted Material Specifications

The Winter 1981 Addenda dated December 31, 1981, is invoked to permit the use of SA-194-2H nuts.

NF-2130 Certification by Material Manufacturer

The Summer 1982 Addenda dated June 30, 1982, is invoked for material certification.

NF-2610 Documentation and Maintenance of Quality Systems Programs

The 1977 Edition dated July 1, 1977, is invoked to revise the material manufacturers and material suppliers responsibilities for materials defined as small products or materials permitted to be supplied with Certificates of Compliance.

NF-3274 Snubbers

The Summer 1976 Addenda dated June 30, 1976, is invoked for NF-3134.6 to permit the use of mechanical snubbers.

NF-3226.5 Special Stress Limits
NF-3321.1 Design Conditions
XVII-2211 Stress in Tension

The Winter 1978 Addenda dated December 31, 1978, is invoked for these paragraph sections which in effect delete the code methods for consideration of through thickness stresses in plates and elements of rolled shapes.

NF-3391.1 Allowable Stress Limits
NF-3392.1 Allowable Stress Limits

The Winter 1979 Addenda dated December 31, 1979, is invoked for these paragraph sections which in effect delete the code methods for consideration of through thickness stresses in plates and elements of rolled shapes.

XVII-2454 Butt and Groove Welds

The 1980 Edition dated July 1, 1980, is invoked to redefine the throat thickness of partial penetration groove welds in accordance with Table XVII-2452.1-1.

In the case that material cannot be purchased to meet the specified ASME III Code, then material that meets subsequent ASME

III Code Editions/Addenda up to and including the 1980 Edition/Summer 1982 Addenda may be substituted after a review and reconciliation of related requirements of the ASME III Code are performed and documented.

Table 3.9A-14 lists the load conditions, load combinations, and allowable stresses. Loads are applied in whatever manner is necessary to attain the worst possible stress levels for all support elements. Component standard supports are qualified either by analysis or by a combination of analysis and load rating. All other supports are qualified by analysis.

No specific deformation limits are required; however, pipe support deformations are consistent with pipe stress analysis.

The pipe support buckling criteria are consistent with the requirements of ASME III, Appendix XVII.

The design criteria and dynamic testing requirements for component and pipe supports listed in the following paragraphs are applicable under all plant operating conditions.

Instrument Lines The requirements for instrument lines are listed in Table 3.9A-15.

Component Supports All component supports are designed, fabricated, and assembled so they cannot become disengaged by the movement of the supported pipe or equipment during operation. All component supports are designed in accordance with the rules of ASME Section III, Subsection NF.

Spring Hangers (Variable and Constant Support) The design load on spring hangers is the load caused by deadweight alone. Variable spring hangers are calibrated to ensure that they support the deadweight at both their hot and cold load settings. For constant support spring hangers, the deadweight is always supported as a constant load, not subject to separate hot and cold loads. Spring hangers also allow for a down-travel and up-travel in excess of the specified thermal movement to account for dynamic movement.

Rod Hangers Rod hangers are only used as a rigid restraint when there is no possibility of compression.

Struts The design loads on struts include those loads caused by deadweight, thermal expansion, primary seismic (OBE and SSE), system anchor displacements, and reaction forces caused by relief valve discharge and turbine stop valve closure, etc. Struts are designed in accordance with Article NF-3000.

Snubbers (Mechanical) The design loads on snubbers include all dynamic loads such as seismic forces (OBE and SSE), system dynamic anchor movements, and reaction forces caused by short duration relief valve discharge and turbine stop valve closure, and dynamic loads produced by suppression pool phenomena. The snubbers are designed and load-rated in accordance with Article NF-3000 to be capable of carrying the design load for all dynamic operating conditions. Faulted condition design uses the criteria outlined in Appendix F of the ASME Code. The prototype snubbers have been tested dynamically to ensure that they can perform as required in the following manner:

1. The snubber was subjected to a force that varied approximately as the sine wave.

2. The frequency (Hz) of the input force was varied by small increments within the specified range.

3. The resulting relative displacements and corresponding loads across the working components, including end attachments, were recorded.

4. The test was conducted with the snubber at various temperatures.

5. The peak load in both static tension and compression tests was higher than the rated load.

6. The duration of the tests at each frequency was specified.

7. Snubbers were tested for various abnormal environmental conditions, including salt-fog, sand and dust, and humidity, followed by operational tests. The environmental test results are filed at the snubber manufacturer's location. The other test results are forwarded with the shipment of each snubber and are incorporated into the permanent plant file.

<u>Anchors</u> Anchors are designed to restrain all rotations and translations of piping. Terminal anchors are those which are common to two independently analyzed piping subsystems, one on each side of the anchor. For each load type, loads from both sides of the anchor are combined to form a total anchor load. For vibratory loads the total anchor load is ± (the SRSS of two loads from both sides of the anchor). For static loads the total anchor load is the algebraic sum of loads from both sides of the anchor. Design transient cyclic data are not applicable to piping supports, since no fatigue evaluation is necessary to meet the code requirements, unless the design specification identifies more than 20,000 load cycles. Design of anchors separating seismically designed and nonseismic piping is discussed in Section 3.7A.3.1.3.1.

3.9A.3.4.2 Pump Supports

The pump pedestal and pedestal bolt analysis includes consideration of loads from operating and seismic events, connecting pipes, temperature, and deadweight. The stress limits of ASME Section III Subsection NF are met. The analysis includes deflection of the pedestal.

3.9A.3.4.3 Other Components Supports

Equipment supports and their connections to building structures that are governed by ASME are in accordance with ASME Section III, Subsection NF. ASME classifies these supports as either plate and shell- or linear-type supports.

Plate and Shell-Type Supports

These supports, e.g., vessel skirts and saddles, are fabricated from plate and shell elements and have the same ASME Code classification as the vessel.

Jurisdictional Boundaries

Figures 3.9A-2 and 3.9A-3 show the boundaries for different subsections of the ASME Code and building structures. As shown, the NF jurisdiction typically includes the connection between the component support and the building, with the exception of concrete anchorages.

Basis for Design and Construction

These supports are designed, fabricated, and installed in accordance with ASME Section III, Subsection NF.

Loads, Load Combinations, and Stress Limits

The combination of design loadings for these supports is categorized with respect to plant operating conditions. These conditions are identified as service levels A through D (Table 3.9A-13). Stress limits for the corresponding service levels also are given in Table 3.9A-13.

Deformation Limits

Deformations are considered so there is no interference with adjacent equipment, piping, or structures. If support deformations are determined to be critical, they become an integral part of the design and are held within the required limits; otherwise, deformations are consistent with support stress analysis.

Buckling Criteria

Analysis is performed to determine critical buckling strength, including local instabilities. Actual loads are compared to critical buckling loads in accordance with ASME Section III, Appendix F.

Linear-Type Supports

These supports, e.g., structural elements such as beams, columns, and frames, have the same ASME Code classification as the component.

Jurisdictional Boundaries

A typical linear equipment support for a HVR unit cooler is illustrated on Figure 3.9A-4. The jurisdictional boundary on the typical support is the connection between the supporting beams and the framing structure. The bolted or welded connection is NF-designed.

Basics for Design and Construction

These supports are designed, fabricated, and installed in accordance with ASME Section III, Subsection NF.

Loads, Load Combinations, and Stress Limits

The combination of design loading for these supports is categorized with respect to plant operating conditions. These conditions are identified as service levels A through D (Table 3.9A-13). Stress limits are also in accordance with ASME Section III, Subsection NF.

Deformation Limits

Deformations are considered so there are no interferences with adjacent equipment, piping, or structures. If support deformations are determined to be critical, they become an integral part of the design and are held within the required limits; otherwise, deformations are consistent with support stress analysis.

Buckling Criteria

Support buckling criteria is consistent with the requirements of ASME Section III, Appendix XVII.

Bolting

The allowable stress limits used for bolts in equipment anchorage, component supports, and flanged connections are given by the following:

1. Anchor Bolts Used in Equipment Anchorage - Appendix B of ACI 349, Code Requirements for Nuclear Safety-Related Concrete Structures.

2. Bolts Used in Component Supports - ASME Section III, Division I, Subsection NF, and Appendix XVII, paragraph 2460. For service levels C and D,

XVII-2460 with factors indicated under XVII-2110 is applicable to the design requirements of bolting. The calculated stresses under these categories do not exceed the specified minimum yield stresses at temperature.

3. Bolts Used in Flanged Connections - ASME III.

Equipment mounted with high-strength bolts include vessels, unit coolers, and heat exchangers. The material for high-strength bolts used for the mounting of component supports to building structures conforms to the assigned jurisdictional boundary. Concrete high-strength anchor bolts used at component supports include A193, A325, and A490 steel. ASME Section III, NF high-strength bolts include SA-193 and SA-325 material.

High-strength bolts and low-strength bolts are used in pipe and duct support designs.

3.9A.4 Control Rod Drive Systems

See Section 3.9B.4.

3.9A.5 Reactor Pressure Vessel Internals

See Section 3.9B.5.

3.9A.6 In-service Testing of Pumps and Valves

An IST program was prepared in conformance with the applicable portions of GDC 37, 40, 43, and 46. This program, submitted in November 1985, included baseline preservice testing and a periodic IST program for pumps and valves and is based on the ASME Boiler and Pressure Vessel Code, Section XI, 1980 Edition, through the Winter 1980 Addenda. The purpose of this program is to ensure that certain Safety Class 1, 2, and 3 pumps provided with an emergency power source, and Safety Class 1, 2, and 3 valves required to perform a specific function in bringing the reactor to a cold shutdown condition or in mitigating the consequences of an accident, are in a state of operational readiness throughout the life of the plant. This test program is based on the ASME Boiler and Pressure Vessel Code, Section XI, and will be periodically updated in accordance with 10CFR50.55a(g).

3.9A.6.1 In-service Testing of Pumps

The IST program for certain Safety Class 1, 2, and 3 pumps that have an emergency power source is in accordance with Subsection IWP of ASME Section XI. The basis of the test program is to detect changes in the hydraulic and mechanical condition of the pump relative to a reference set of parameters. Reference values will be established in accordance with IWP-3000 of ASME Section

XI and listed in the IWP pump testing section of the Unit 2 ISI program. Pumps are tested periodically during plant operation and during shutdown periods, if practical, in accordance with the requirements of ASME Section XI, IWP-3100 and IWP-3400(a). If a pump is normally operated more frequently than once a month, it need not be specially tested provided that the requirements of ASME Section XI, IWP-3400(b) are met.

The pumps are to be run at least 5 min under conditions as stable as the system permits. The methods of measurement are in accordance with IWP-4100, -4200, -4300, -4400, -4500, and -4600 of ASME Section XI with regard to instruments, pressure measurements, temperature measurements, rotational speed, vibration measurements, and flow measurement. The specific methods for measuring pump test parameters will be described in the test procedures for the IWP pump plan. The pump test plan and schedule are provided in the IWP pump testing section of the Unit 2 ISI program.

3.9A.6.2 In-service Testing of Valves

The IST plan for all Safety Class 1, 2, and 3 valves that are required to perform a safety function will be in accordance with the requirements of Subsection IWV of ASME Section XI. In accordance with the exemptions listed in Subarticle IWV-1200 of ASME Section XI, valves used only for maintenance or for operating convenience, such as manual vent, drain, instrument, and test valves, will not be included in the plan. All valves requiring in-service testing will be listed in the IWV valve testing section of the Unit 2 ISI plan. Each valve in the plan is categorized in accordance with the requirements of Subarticle IWV-2100 of ASME Section XI. Valve tests with regard to preservice tests, valve replacement, valve repair and maintenance, indication of valve position, and in-service tests will be in accordance with the requirements of Article IWV-3000 of ASME Section XI. Test methods for each valve tested under the valve IST plan will be described in the valve test procedures.

The valves which separate the RCPB, identified in Table 3.4.3.2-1 of the Technical Specifications, from interfacing low-pressure systems shall be leak tested in accordance with Technical Specification 3.4.3.2.

These valves are included in the Unit 2 Pump and Valve IST Program which was developed in accordance with the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWV. The Unit 2 leak testing program for these valves is described as follows:

1. The leak rate test will be performed, as a minimum, at least every refueling outage (IWV-3422).

2. The leak test medium will be water for the valves which are not Type C per 10CFR50 Appendix J, with a test acceptance criteria of 0.5 gpm leakage per nominal in

of valve size up to a maximum of 5 gpm at a RCS pressure of 1,020 ±20 psig.

For those check and globe valves which require a 10CFR50 Appendix J Type C test, the air leak rate data may be converted to a water leakage rate at 1,020 ±20 psig and compared with the above acceptance criteria for compliance to the Technical Specification requirement.

3. The periodic leak test will be done during refueling outages.

4. After maintenance which can affect leak-tightness of the valve, leak testing will be performed in accordance with ASME XI, prior to returning the valve to service.

P&IDs have been supplied with the FSAR and Preservice and In-Service Inspection Plan which describe the RCS pressure isolation valves. Procedures to support Technical Specification testing shall be generated in accordance with the startup schedule during 1985 and 1986.

3.9A.6.3 Relief Requests

Requests for relief from the requirements of Article IWP of ASME Section XI for IST of pumps will be documented in the IWP pump testing section of the Unit 2 ISI program (Technical Specifications). Requests for relief from the requirements of Article IWV of ASME Section XI for IST of valves will be documented in the IWV valve testing section of the Unit 2 ISI program. If future updating of the Unit 2 ISI program reveals additional cases where testing of pumps or valves is impractical, with respect to the current or future revisions of Articles IWP or IWV, additional requests for relief from testing requirements will be documented and submitted in a timely manner.

3.9A.6.4 Pipe Welds Within Break Exclusion Area

A 100-percent volumetric preservice and in-service inspection of high-energy fluid system piping welds within the break exclusion area will be conducted during each inspection interval, as defined in IWA-2400, ASME Section XI.

[illegible]

TABLE 3.9A-1

TRANSIENTS AND THE NUMBER OF ASSOCIATED CYCLES CONSIDERED IN THE DESIGN AND FATIGUE ANALYSES OF CLASS 1 PIPING

	Transients	No. of Cycles
	Normal, Upset, and Testing Conditions	
	Dynamic loads caused by SRV discharge events[3]	
	a. Piping	5,200
	b. Equipment	6,000
	OBE at rated operating conditions[4]	50
1.	Boltup	123
2.	Design hydrotest	130
3.	Startup (100°F/hr heatup rate)[1]	120
4.	Turbine roll and increase to rated power	120
5.	Daily power reduction to 75%	10,000
6.	Weekly power reduction to 50%	2,000
7.	Rod pattern change	400
8,10.	Scram - turbine generator trip, feedwater on, isolation valves stay open	50
9.	Partial feedwater heater bypass	70
11.	Other scrams	140
12.	Rated power normal operation - inadvertent actuation	10
13.	Reduction to 0% power	111
14.	Hot standby	111
15.	Shutdown prior to vessel flooding	111
16.	Vessel flooding	111
17.	Shutdown	111
18.	Vessel unbolt	123
19.	Refueling	0
20.	Loss of feedwater pumps - isolation valves closed	10
21.	Single relief or safety valve blowdown	8
29.	Inadvertent/accidental vessel overfilling[7]	4
	Emergency	
22.	Reactor overpressure with delayed scram[2]	1
23.	Automatic blowdown[2]	1
24.	Improper start of cold recirc loop[2]	1
25.	Sudden start of pump in cold recirc loop[2]	1
26.	Hot standby-drain shutoff pump restart[2]	1
27.	Dynamic loads caused by suppression pool events during SBA, IBA[2]	[6]

TABLE 3.9A-1 (Cont'd.)

Transients		No. of Cycles
Faulted		
28.	Pipe rupture and blowdown[2]	1
	SSE at rated operating conditions[2,5]	10
	Dynamic loads caused by suppression pool events during DBA[2]	1,500

(1) Bulk average vessel coolant temperature change in any 1-hr period.

(2) The probability of event to occur in 40-yr plant life, P_{40}, is:

Emergency conditions: $10^{-1} > P_{40} \geq 10^{-3}$

Faulted conditions: $10^{-3} > P_{40} \geq 10^{-6}$

(3) The SRV discharge events used for analysis are given in Table 3.9A-2.

(4) In some cases, considered as an emergency event.

(5) Includes 10 maximum load cycles per event.

(6) Fatigue analysis is not required for emergency and faulted conditions.

(7) Number of cycles is based on maximum temperature differential (ΔT) of 141°F between the main steam pipe wall temperature and the incoming fluid temperature, and it includes one cycle for vessel overfilling which occurred in January 1988 during power ascension testing.

TABLE 3.9A-3

PIPE STRESS ANALYSIS CLASSIFICATIONS
FOR ASME CODE CLASSES 1, 2, 3

Piping Class	Analysis Classification		
Class 1	Nominal pipe size	D>1"	D≤1"
	Type of analysis	Class 1	Class 2[1]
Class 2,3	Nominal pipe size or tubing OD	D>6"	D≤6"
	Type of analysis	Computer analysis	Noncomputer analysis[2]

(1) See Section 3.7.3.8.2A for acceptance criteria.
(2) Piping or instrumentation tubing is qualified by placing the supports in accordance with a generic procedure or by hand calculations of stress and support loads.

TABLE 3.9A-4

BOP SEISMIC/DYNAMIC QUALIFICATION RESULTS - MECHANICAL EQUIPMENT

Equipment	Methods	Results
Motor-operated rotary gates	A static analysis and test were performed. The applicable standards and guidelines are IEEE-323-1974 and IEEE-344-1975 and RG 1.61, 1.89, 1.92, and 1.100.	The motor-operated rotary gates are affected by seismic loads only. The analysis indicated that the stress levels are within the allowable limits of 3.9A.2.2.2. Refer to Table 3.10A-1 for the qualification of the Limitorque actuators.
Special air filter assemblies	A static analysis and test were performed. The applicable standards, codes, and guidelines are IEEE-344-1975, ASME Section III, and RG 1.61, 1.92, 1.84, 1.85, 1.89, and 1.100.	The filter assemblies are affected by seismic loads only. The results of the analysis of structural and functional elements of the equipment indicate that the stresses are within the allowable limits of 3.9A.2.2.2. Refer to Table 3.10A-1 for the qualification of the heater/flow switches.
Spent fuel pool cooling water heat exchangers	A static analysis was performed. The applicable standards, codes, and guidelines are IEEE-344-1975, ASME Section III, and RG 1.61, 1.92, 1.89, and 1.100.	The heat exchangers are affected by seismic loads. The analysis indicates that the stress levels are within the allowable limits of 3.9A.2.2.2.
Self-cleaning strainers	The strainers are qualified by static analysis and test. The applicable standards, codes, and guidelines are IEEE-323-1974, IEEE-334-1974, IEEE-344-1975, ASME Section III, and RG 1.61, 1.84, 1.85, 1.89, 1.92, and 1.100.	The strainers are subjected to seismic loads. The analysis indicates that the stress intensity for the strainer components is within the allowable stress limits of 3.9A.2.2.2. The deflections do not affect the operability of the strainer. Refer to Table 3.10A-1 for the qualification of the motors.
Simplex strainers	The strainers are qualified by static analysis. The applicable standards are ASME Code Section III and RG 1.61 and 1.92.	The strainers are subjected to seismic loads. The structural integrity of the strainers is demonstrated since the analysis indicates the stress levels are within the allowable limits of 3.9A.2.2.2.
ECCS and RCIC suppression pool strainers	The strainers are qualified by static analysis. The applicable standards are ASME Code Section III and RG 1.61 and 1.92.	The strainers are subjected to seismic, hydrodynamic, and suppression pool drag loads. The analysis indicates that the stress intensity for the strainer components is within the allowable stress limits of 3.9A.2.2.2.
Active pumps horizontal centrifugal Diesel generator fuel oil transfer	Pumps are qualified by dynamic and static analysis and operability tests. The applicable standards and guidelines are IEEE-344-1975, IEEE-334-1974, RG 1.48, 1.61, 1.89, 1.92, and 1.100, and ASME Code Section III.	The pumps are affected by seismic loads only. The structural integrity and functional capability of the pumps have been demonstrated by static and dynamic analysis. The centrifugal pumps have been determined to be rigid. The fundamental frequency of the fuel oil transfer pumps was 6 Hz. Stresses are maintained within the allowable limits of Table 3.9A-8. All of the deflections are within the normal clearances. The lowest margin of safety with respect to both the stresses and deflections is approximately

TABLE 3.9A-4 (Cont'd.)

Equipment	Methods	Results
		1 percent. In addition to the seismic analysis, the operability of the pumps is ensured through the program described in Paragraph 3.9A.3.2.1. The qualification of the motors conforms to IEEE-334-1974. See Table 3.10A-1 for the qualification results of the pump motors and for the qualification summary of the electric motors.
Local instrument racks	The instrument racks are qualified by analysis. The applicable standards and guidelines are IEEE-323-1974, IEEE-344-1975, and RG 1.61 and 1.100.	The racks are affected by seismic loads only. The analysis was performed using a finite element model. The equipment was determined to be rigid and, therefore, static analysis was utilized. Results of the analysis indicate that the stresses are within the allowable stresses of 3.9A.2.2.2.
Flexible metal hoses	Flexible metal hoses are qualified by analysis. Applicable standards and guidelines are EJMA, ASME Code Section III, ASME Code Case N-192-2, IEEE-344-1975, and RG 1.61, 1.84, 1.85, 1.92, and 1.100.	Flexible hoses are affected by both seismic and hydrodynamic loads. Design adequacy was verified by analysis in accordance with the EJMA Standard. This analysis takes into account design temperature, pressure, dynamic loads, differential displacements, and the number of cycles of displacement. In addition, representative hoses are qualified by two separate dynamic test programs. In the first dynamic test, hoses are subjected to a total of 1 million cycles of vibrations in the frequency range of 5 to 100 Hz, at accelerations ranging from 3 g to 51 g. In the second test, the hoses are subjected to six biaxial, random, multifrequency input motions of 30-sec durations each. The six tests are repeated in the other horizontal orientations. The TRS envelops the applicable portion of RRS with at least a 10-percent margin. Hoses were pressurized at the start of each test series to at least the design pressure. During and following the dynamic tests the hoses maintained their pressure integrity.
Miscellaneous HVAC Axial fans Centrifugal fans Air conditioning units Backdraft dampers Bubbletight dampers Butterfly damper Tornado damper Fire damper Multileaf damper Single-bladed dampers	The HVAC equipment listed was qualified by analysis and test. Applicable codes, standards, and guidelines include RG 1.60, 1.61, 1.89, 1.92, 1.84, 1.85, and 1.100, IEEE-323-1974, IEEE-334-1974, IEEE-334-1975, and ASME Section III.	This equipment is affected by seismic loads only. The results of the analysis of structural and functional elements of the equipment indicate that the stresses are within the allowable limits of 3.9A.2.2.2. The deflection of rotating members was determined to be within the clearances. Refer to Table 3.10A-1 for the qualification results of the fan motors and the pneumatic actuators used on the dampers.

TABLE 3.9A-4 (Cont'd.)

Equipment	Methods	Results
Centrifugal liquid chillers	Seismic qualification is by static analysis and testing. Applicable standards and guidelines include ASME Code Section III, RG 1.61, 1.89, 1.92, and 1.100, IEEE-323-1974, IEEE-334-1974, and IEEE-334-1975.	This equipment is affected by seismic loads only. The analysis of the chiller assembly and structural components is performed using a detailed finite element model. Results of the analysis indicate that the stresses are within the allowable stresses of 3.9A.2.2.2 and deflections of critical components are within limits required to maintain functional capability. The electronic control panel and the System Class 1E components were qualified by dynamic testing. See Table 3.10A-1 for test results.
Unit space cooler	The unit space coolers and air conditioning units were qualified by both test and analysis. The applicable standards and guidelines include IEEE-344-1975, IEEE-323-1974, and IEEE-334-1974, ASME Section III, and RG 1.61, 1.89, 1.92, and 1.100.	The unit space coolers and air conditioning units are affected by seismic loads only. The unit coolers are composed of three parts: the fan-motor section, coil section, and filter section. The units have either a propeller-type fan (Industrial Air) or a vaneaxial fan (Joy Manufacturing). They all have electric motors (Reliance) and motor control panels. (The air conditioning units are essentially the same as the unit coolers, except that they have no motor control panels.) All but two units have air filters (American Air Filters) and each has one or two ASME Section III cooling coils. A propeller fan space cooler and a vaneaxial fan space cooler were chosen for dynamic testing, since they were representative of the dimensions and characteristics of the other coolers. The dynamic testing is performed as follows: The units were mounted on the vibration test table so that the in-service condition is simulated. The units were instrumented to record accelerations. A resonance search was performed from 1 to 35 Hz for each of the 3 orthogonal axes. The seismic simulation vibration testing consisted of biaxial random multifrequency tests, 5 OBEs and 1 SSE in each of 2 test orientations, 90 deg apart. The units were pressurized and operational during the tests. The TRS enveloped the RRS. The cooling coils for the unit coolers were qualified by analysis. The results of the analysis indicate that the stress levels are within the allowable limits of 3.9A.2.2.2. The fan and coil sections of the air conditioning unit were qualified by dynamic testing. The units were mounted on the vibration test table to simulate plant installation. They were instrumented to record accelerations. A resonance search was performed from 1 to 33 Hz for each of the 3 orthogonal axes. The seismic simulation vibration testing consisted of biaxial random multifrequency test of 4 SSEs in each of the two test orientations, 90 deg apart. The equipment remained operational during the tests and the TRS enveloped the RRS.

TABLE 3.9A-4 (Cont'd.)

Equipment	Methods	Results
Nonactive valves Motor-operated Air-operated Manual Solenoid	Nonactive valves are qualified by analysis. Applicable standards and guidelines are ASME Section III and RG 1.61 and 1.92.	Valves affected by seismic loads only are qualified for 3 g horizontal and 3 g vertical loadings. Those valves affected by seismic and hydrodynamic loadings have been qualified for up to 20 g horizontal and 20 g vertical loadings. Piping design acceptance criteria ensure actual loadings to be within the qualified levels for each valve. All valves were determined to have a natural frequency that is generally greater than 33 Hz. Structural and pressure integrity of the valve assemblies has been demonstrated by static analysis or through their similarity to active valves. Stresses are maintained within the limits of Tables 3.9A-7 and 3.9A-8. For ASME Class 1 valves, design reports are also prepared in accordance with ASME Section III, Subsection NB-3500. For valves affected by hydrodynamic loads, fatigue analyses of the critical components were also performed, and the CUFs are maintained below one.
Active valves Motor-operated Air-operated Solenoid Relief valves Electrohydraulic	Active valves are qualified by analysis and test. Applicable standards and guidelines are ASME Section III, RG 1.48, 1.61, 1.89, 1.92, and 1.100, and IEEE-323-1974, IEEE-344-1975 and IEEE-382-1972.	Valves affected by seismic loads only are generally qualified for 3 g horizontal and 3 g vertical loadings. The valves affected by seismic and hydrodynamic loads were qualified for up to 11 g horizontal and 11 g vertical loadings. Piping design acceptance criteria ensures actual loadings to be within the qualified levels for each valve. All valves were determined to have natural frequencies that are generally greater than 33 Hz. For valves with a fundamental natural frequency below 33 Hz, the appropriate valve mass and stiffness properties were included in piping models, and the valve acceleration responses obtained from the piping analyses were maintained below the qualification levels. Structural and pressure integrity of the valve assemblies are demonstrated by analysis. Stresses are maintained within the limits of Tables 3.9A-7 and 3.9A-8. Deflection of critical components is well within the allowable limits. Design stress analyses were performed for ASME Class 1 valves in accordance with ASME Section III, Subsection NB-3500. For valves affected by hydrodynamic loads, fatigue analyses of the critical components were also performed, and CUFs are maintained below one. Actuators for AOVs are qualified by dynamic testing. Each tested actuator was mounted on the shake table as it normally would be in service, and biaxial, random multifrequency tests of 30-sec duration were performed for each of the five OBE and one SSE conditions. The tests were repeated in the second horizontal and vertical orientation. The actuator was operated through one complete cycle for each OBE and SSE test. The actuator performed its safety function and successfully completed the test. Test spectra ZPA has a minimum margin of 200 percent over the peak accelerations obtained from piping analyses for valves affected by seismic loads only. For valves affected by combined seismic and hydrodynamic

TABLE 3.9A-4 (Cont'd.)

Equipment	Methods	Results
		loads, the OBE and SSE test spectra enveloped the upset and faulted RRS, respectively, by a minimum margin of 10 percent, except for one direction, where the margin at one frequency is below 10 percent. However, the adjacent 1/6 octave points are above the 10-percent margin, and the adjacent 1/3 octave points are above a 20-percent margin. Additionally, test margins adequately envelope the stress cycles and durations required for the hydrodynamic loads. For the SOVs (Target Rock) and the electric components of the valves, such as solenoid valves, electrohydraulic operators, motor operators, limit switches, etc., qualification is achieved through comprehensive environmental and dynamic test programs. Detailed results are provided in Table 3.10A-1. Operability of the valve assemblies is demonstrated by both dynamic and static load tests. Selected valves were subjected to dynamic tests to simulate the seismic and hydrodynamic loads. Other valves were qualified through static deflection tests of parent valve assemblies. The test programs conform to Paragraph 3.9A.3.2.2. Functional adequacy was verified during and after these tests.
Feedwater check valves	The feedwater check valves are qualified by analysis. The Class 1E components of the air-operated check valves are qualified by testing. Applicable standards and guidelines are ASME Section III, NRC RG 1.48, 1.61, 1.89, 1.92, and 1.100, and IEEE-323-1974 and IEEE-344-1975.	The feedwater check valves are affected by both seismic and hydrodynamic loads. The valves are qualified by dynamic analysis for the worst transient condition following a pipe break, together with the seismic/hydrodynamic loads. Stresses are maintained within the limits of Table 3.9A-7. Design analysis of the valves is also prepared in accordance with ASME Section III, Subsection NB-3500. The electrical appurtenances of the air operators (limit switch, solenoid valves) are qualified by testing. Detailed results are provided in Table 3.10A-1.
Vacuum relief valves	The vacuum relief valves are qualified by analysis. The applicable standards and guidelines are ASME Section III and NRC RG 1.48, 1.61, and 1.92.	The vacuum relief valves are affected by both seismic and hydrodynamic loads. The valves are rigid (natural frequency >100 Hz) and are analyzed for up to 16 g horizontal and 14 g vertical loadings together with the operating loads (opening and/or closing pressure transients). The design analysis met the ASME III, Subsection NC-3500, requirements. The stresses in all the critical valve components are maintained within the limits of Table 3.9A-8, and the calculated deflections do not affect the operability of the valves.

TABLE 3.9A-4 (Cont'd.)

Equipment	Methods	Results
Polar crane	Dynamic analysis is performed utilizing the time-history technique. Applicable standards and guidelines are CMAA-70, the AISC Code, and NRC RG 1.61 and 1.92.	The polar crane is affected by seismic loads. A finite element lumped mass mathematical model is developed to simulate the mass and stiffness characteristics of the crane, including the trucks, trolleys, and hoist rope. Dynamic responses due to seismic loadings are evaluated for several trolley positions and load lift heights, as appropriate, in order to determine the maximum stress levels in all critical members and connections. The analysis indicates that the stresses are within the allowable limits.

TABLE 3.9A-5

LOAD COMBINATIONS FOR ASME SECTION III CLASS 1 VALVES[1]

Classification	Combination
Design	Design pressure Design temperature[2] Deadweight Piping reactions OBE
Normal	Normal condition pressure Normal condition metal temperature Deadweight Piping reactions
Upset	Upset condition pressure Upset condition metal temperature Deadweight Dynamic load 1[3] Piping reactions
Emergency	Emergency condition pressure Emergency condition metal temperature Deadweight Dynamic load 3[3] Piping reactions
Faulted	Faulted condition pressure Faulted condition metal temperature Deadweight Dynamic load 2[3] Piping reactions

(1) The only ASME Class 1 components within the BOP scope are valves.
(2) Temperature is used to determine allowable stress only.
(3) The definitions of these loads are given on page 2 of this table.

TABLE 3.9A-5 (Cont'd.)

Load Definitions

Dynamic Load 1

Location a = $[(OBE)^2+(SRV_{ALL})^2]^{1/2}$

Location b = OBE

Dynamic Load 2

Location a; the envelope of

$$(i) = [(SSE)^2 + (AP)^2]^{1/2}$$
$$(ii) = [(SSE)^2 + (SRV_{ONE})^2 + (CO1)^2]^{1/2}$$
$$(iii) = [(SSE)^2 + (SRV_{ADS})^2 + (CO2)^2]^{1/2}$$
$$\text{and } (iv) = [(SSE)^2 + (SRV_{ALL})^2 + (CHUG)^2]^{1/2}$$

Location b = SSE

Dynamic Load 3

Location a; the envelope of

$$(i) = [(OBE)^2 + (AP)^2]^{1/2}$$
$$(ii) = [(OBE)^2 + (SRV_{ONE})^2 + (CO1)^2]^{1/2}$$
$$(iii) = [(OBE)^2 + (SRV_{ADS})^2 + (CO2)^2]^{1/2}$$
$$\text{and } (iv) = [(OBE)^2 + (SRV_{ALL})^2 + (CHUG)^2]^{1/2}$$

Location b = OBE

Where:

Location a = Equipment inside the reactor building
Location b = Equipment outside the reactor building
OBE = Operating basis earthquake
SSE = Safe shutdown earthquake
SRV_{ALL} = Envelope of all safety relief valve actuation cases, including symmetric, asymmetric, ADS, and single subsequent actuations
SRV_{ONE} = One stuck open safety relief valve actuation case
SRV_{ADS} = ADS safety relief valves actuation case
CO1 = Basic condensation oscillation phase of LOCA
CO2 = Condensation oscillation phase of LOCA, concurrent with actuation of ADS valves
AP = Annulus pressurization due to LOCA
CHUG = Envelope of symmetric and asymmetric chugging phases of LOCA

TABLE 3.9A-5 (Cont'd.)

LOCA = Envelope of CO chugging and annulus pressurization due to LOCA

NOTE: For a detailed discussion of these loads, see the Design Assessment Report (Appendix 6A).

TABLE 3.9A-6

LOAD COMBINATIONS FOR ASME SECTION III
CLASS 2 AND 3 AND NON-ASME COMPONENTS

Design Conditions I and II are defined as:

Design Condition I = Specified design loads (temperature, pressure, etc.) + Dynamic Load 1

Design Condition II = Specified design loads (as above) + Dynamic Load 2 pipe rupture loads (if applicable)

Dynamic Load 1*

Location a = $[(OBE)^2 + (SRV_{ALL})^2]^{1/2}$

Location b = OBE

Dynamic Load 2*

Location a; the envelope of

$$\begin{aligned}
(i) &= [(SSE)^2 + (AP)^2]^{1/2} \\
(ii) &= [(SSE)^2 + (SRV_{ONE})^2 + (CO1)^2]^{1/2} \\
(iii) &= [(SSE)^2 + (SRV_{ADS})^2 + (CO2)^2]^{1/2} \\
\text{and } (iv) &= [(SSE)^2 + (SRV_{ALL})^2 \; (CHUG)^2]^{1/2}
\end{aligned}$$

Location b = SSE

* See Table 3.9A-5 for hydrodynamic load nomenclature.

TABLE 3.9A-7

STRESS LIMITS FOR ASME SECTION III CLASS 1 (NB)
SEISMIC CATEGORY I COMPONENTS (ELASTIC ANALYSIS)

PRESSURE BOUNDARY-DESIGNED BY ANALYSIS

Condition of Design[1]	Reference Paragraph ASME III	Primary Stress Limits Pm	PL	PL+Pb	Expansion Stress Limits Pe	Primary plus Secondary Stress Limits PL+Pb+Pe+Q	Peak Stress Limits PL+Pb+Pe+Q+F
Upset[2]	NB-3223	[4]	[4]	[4]	3 S_m	3 S_m	S_a
Emergency[2]	NB-3224	Greater of 1.2S_m or 1.0S_y	Greater of 1.8S_m or 1.5S_y	Greater of 1.8S_m or 1.5S_y		Not required	Not required
Faulted[2,3]	NB-3225, NB-3221, App. F F 1323.1	Lesser of 2.4S_m or 0.7S_u	Lesser of 3.6S_m or 1.05S_u	Lesser of 3.6S_m or 1.05S_u		Not required	Not required

[1] Since design loads are used in the actual analysis, only the conditions shown require evaluation.
[2] Use design loads.
[3] Use above limits for materials of Table I-1.2 (ASME Section III). Use 0.7S for materials of Table I-1.1 (ASME Section III).
[4] Primary stresses are evaluated and combined with secondary effects as appropriate.

NOTE: The nomenclature, conditions, and applications of the above allowables are in accordance with ASME Section III. Stress limits apply to design by elastic analysis. Limit and plastic analysis is allowed in accordance with ASME Section III criteria. Special stress limits of Paragraph NB-3227 apply as applicable.

[illegible]

[illegible]

[illegible]

TABLE 3.9A-8

STRESS LIMITS FOR ASME SECTION III CLASS 2 AND 3 COMPONENTS (ELASTIC ANALYSIS)

Design Condition[1]	ASME III Code Class	Primary Stress Limits: Membrane (P_m)	Primary Stress Limits: Membrane + Bending ($P_m + P_b$)
Pressure Vessels			
I	2(NC-3300) or	1.1 S	1.65 S
II	3(ND-3300)	2.0 S	2.40 S
I[2]	2(NC-3200)	1.1 S_m	1.65 S_m
II[3]		2.0 S_m	2.40 S_m
Pumps Nonactive[4,5]			
I	2(NC-3400) or	1.1 S	1.65 S
II	3(ND-3400)	2.0 S	2.40 S
Pumps Active[4,5]			
I	2(NC-3400) or	1.0 S	1.50 S
II	3(ND-3400)	1.2 S	1.80 S
Valves Active and Nonactive[5,6]			
I	2(NC-3500) or	1.1 S	1.65 S
II	3(ND-3500)	2.0 S	2.40 S
Tanks (Steel)[5]			
I	2(NC38-3900) or	1.1 S	1.65 S
II	3(ND38-3900)	2.0 S	2.40 S

KEY: S = Allowable stress values at design temperature from ASME Section III, Appendix I, as allowed by class

S_m = Design stress intensity values at design temperature from ASME Section III, Appendix I, as allowed by class

(1) Refer to Table 3.9A-7 for the definitions of Design Conditions I and II.

(2) Fatigue analysis may be required with operating conditions; see Paragraph NC-3219 and Appendix XIV of ASME Section III.

TABLE 3.9A-8 (Cont'd.)

(3) When a complete analysis is performed in accordance with Subparagraph NC-3211.1(c), the faulted stress limits of Appendix F apply.

(4) In accordance with Subarticles NC-3400 and ND-3400, any design method that has been demonstrated to be satisfactory for the specified design conditions may be used.

(5) Stress limits of ASME Section III, Subsection NF, are used for the design of supports as applicable (Table 3.9A-14).

(6) The standard or alternative design rules of Subarticles NC-3500 and ND-3500 may be used in conjunction with the stress limits specified.

Valve nozzle (piping load) stress analysis is not required when both the following conditions are satisfied by calculation:

a. Section modulus and area at the plane normal to the flow passage through the region at the valve body crotch is at least 110 percent of that for the piping connected (or joined) to the valve body inlet and outlet nozzles; and,

b. Code allowable stress, S, for valve body material, is equal to or greater than code allowable stress, S, of connected piping material. If valve body material allowable stress is less than that of the connected piping, the valve section modulus and area as calculated in Item a is multiplied by the ratio of the allowable stress for the pipe divided by the allowable stress of the valve.

The design by analysis procedure of Subparagraph NB-3545.2 is an acceptable alternative method if these requirements cannot be met. A casting quality factor of 1.0 is used.

Design requirements listed in this table are not applicable to stems, cast rings, or other nonpressure-retaining parts of valves which are contained within the confines of the body and bonnet.

TABLE 3.9A-11

THIS TABLE HAS BEEN DELETED

TABLE 3.9A-12

ACTIVE VALVES (BOP)

System Name	Mark Number	Size	Valve Type	Pressure Rating(*)	ASME Class	Mfg.	Valve Operator Model (Mfg.)	Active Function
Reactor Plant Component Cooling (CCP)	2CCP*MOV14A,B	12	Gate	150	3	1	SMB-0-25(1)	63
	2CCP*MOV15A,B	4	Gate	150	2	1	SMB-000-5(1)	9
	2CCP*MOV16A,B	4	Gate	150	2	1	SMB-000-5(1)	9
	2CCP*MOV17A,B	4	Gate	150	2	1	SMB-000-5(1)	9
	2CCP*MOV18A,B	12	Gate	150	3	1	SMB-0-25(1)	63
	2CCP*MOV122	8	Gate	150	2	1	SMB-00-15(1)	9
	2CCP*MOV124	8	Gate	150	2	1	SMB-00-15(1)	9
	2CCP*MOV265	8	Gate	150	2	1	SMB-00-15(1)	9
	2CCP*MOV273	8	Gate	150	2	1	SMB-00-15(1)	9
	2CCP*RV60A,B,C	3/4 x 1	SRV	150/150	3	8	None	4
	2CCP*RV64A,B	2 x 3	SRV	150/150	3	8	None	4
	2CCP*RV170	3/4 x 1	SRV	150/150	2	8	None	4
	2CCP*RV171	3/4 x 1	SRV	300/150	2	8	None	4
	2CCP*MOV94A,B	4	Gate	150	2	1	SMB-000-5(1)	9
	2CCP*AOV37A	1 1/2	Plug	150	3	4	NCB520-SR80(2)	64
	2CCP*AOV37B	2	Plug	150	3	4	NCB725-SR80(2)	64
	2CCP*AOV38A	1 1/2	Plug	150	3	4	NCB520-SR80(2)	64
	2CCP*AOV38B	2	Plug	150	3	4	NCB725-SR80(2)	64
Containment Atmosphere Monitoring (CMS)	2CMS*SOV23A-F	3/4	Globe	1500	2	6	76P-001(7)	18
	2CMS*SOV24A-D	3/4	Globe	1500	2	6	76P-001(7)	16,83
	2CMS*SOV26A-D	3/4	Globe	1500	2	6	76P-001(7)	16,83
	2CMS*SOV32A,B	3/4	Globe	1500	2	6	76P-002(7)	16,83
	2CMS*SOV33A,B	3/4	Globe	1500	2	6	76P-002(7)	16,83
	2CMS*SOV34A,B	3/4	Globe	1500	2	6	76P-002(7)	16,83
	2CMS*SOV35A,B	3/4	Globe	1500	2	6	76P-002(7)	16,83
	2CMS*SOV60A,B	3/4	Globe	1500	2	6	76P-001(7)	16
	2CMS*SOV61A,B	3/4	Globe	1500	2	6	76P-002(7)	16
	2CMS*SOV62A,B	3/4	Globe	1500	2	6	76P-002(7)	16
	2CMS*SOV63A,B	3/4	Globe	1500	2	6	76P-002(7)	16
	2CMS*SOV64A,B	3/4	Globe	1500	2	6	76P-001(7)	62
	2CMS*SOV65A,B	3/4	Globe	1500	2	6	76P-001(7)	62
	2CMS*EFV1A,B	3/4	Check	45	2	13	None	16
	2CMS*EFV3A,B	3/4	Check	45	2	13	None	16
	2CMS*EFV5A,B	3/4	Check	45	2	13	None	16
	2CMS*EFV6	3/4	Check	45	2	13	None	16
	2CMS*EFV8A,B	3/4	Check	45	2	13	None	16
	2CMS*EFV9A,B	3/4	Check	45	2	13	None	16
	2CMS*EFV10	3/4	Check	45	2	13	None	16

TABLE 3.9A-12 (Cont'd.)

System Name	Mark Number	Size	Valve Type	Pressure Rating(*)	ASME Class	Mfg.	Valve Operator Model (Mfg.)	Active Function
Primary Containment Purge (CPS)	2CPS*AOV104	14	Butterfly	150	2	2	N721C-SR80-M3HW(2)	9
	2CPS*AOV105	12	Butterfly	150	2	2	N721C-SR80-M3HW(2)	9
	2CPS*AOV106	14	Butterfly	150	2	2	N721C-SR80-M3HW(2)	9
	2CPS*AOV107	12	Butterfly	150	2	2	N721C-SR80-M3HW(2)	9
	2CPS*AOV108	14	Butterfly	150	2	2	N721C-SR80-M3HW(2)	9
	2CPS*AOV109	12	Butterfly	150	2	2	N721C-SR80-M3HW(2)	9
	2CPS*AOV110	14	Butterfly	150	2	2	N721C-SR80-M3HW(2)	9
	2CPS*AOV111	12	Butterfly	150	2	2	N721C-SR80-M3HW(2)	9
	2CPS*SOV119	2	Globe	1500	2	6	76P-020(7)	9
	2CPS*SOV120	2	Globe	1500	2	6	76P-027(7)	9
	2CPS*SOV121	2	Globe	1500	2	6	76P-020(7)	9
	2CPS*SOV122	2	Globe	1500	2	6	76P-027(7)	9
	2CPS*SOV132	1	Globe	1500	2	6	76P-035(7)	
	2CPS*SOV133	1	Globe	1500	2	6	76P-035(7)	
High Pressure Core Spray (CSH)	2CSH*AOV108	12	Check	900	1	3	Series 2A(6)	9,16
	2CSH*RV113	3/4 x 1	SRV	150/150	2	8	None	4,9,16,56
	2CSH*RV114	3/4 x 1	SRV	150/150	2	8	None	4,9,16,56
	2CSH*V9	16	Check	900	2	3	None	23
	2CSH*RV160	3/4 x 1	SRV	150/150	2	8	None	4,56
	2CSH*V17	3	Check	900	2	1	None	30
	2CSH*V55	3	Check	900	2	1	None	30
	2CSH*EFV1	2	Check	100	2	13	None	9,16
	2CSH*EFV2	2	Check	100	2	13	None	9,16
	2CSH*EFV3	3/4	Check	1575	2	13	None	9,16
Low Pressure Core Spray (CSL)	2CSL*AOV101	12	Check	900	1	3	Series 2A(6)	9,16
	2CSL*FV114	10	Globe	300	3	5	SMB-00-5(1)	27
	2CSL*MOV104	12	Gate	600	1	1	SB-2-60(1)	28
	2CSL*MOV107	4	Gate	300	2	1	SMB-00S-15(1)	31
	2CSL*MOV112	20	Butterfly	150	2	9	SMB-0-10/H4BC(1)	32
	2CSL*RV105	1 1/2 x	SRV	300/150	2	8	None	4,9,56
	2CSL*RV123	2	SRV	150/150	2	8	None	4,9,56
	2CSL*V4	3/4 x 1	Check	300	2	1	None	74
	2CSL*V14	16	Check	600	2	1	None	30
	2CSL*V17	2	Gate	600	2	1	None	30
	2CSL*EFV1	2	Check	1250	2	13	None	16
	2CSL*V9	3/4	Check	150	2	1	None	22,31
	2CSL*V21	12	Check	600	2	1	None	22
	2CSL*EFV31	2	Check	1250	2	13	None	16
		3/4						

TABLE 3.9A-12 (Cont'd.)

System Name	Mark Number	Size	Valve Type	Pressure Rating(*)	ASME Class	Mfg.	Valve Operator Model (Mfg.)	Active Function
Reactor Building Equipment Drains (DER)	2DER*MOV119	4	Gate	150	2	1	SMB-000-5(1)	9
	2DER*MOV120	4	Gate	150	2	1	SMB-000-5(1)	9
	2DER*MOV130	2	Globe	1500	2	1	SMB-000-5(1)	9
	2DER*MOV131	2	Globe	1500	2	1	SMB-000-5(1)	9
	2DER*EFV31	3/4	Check	1250	2	13	None	16
Reactor Building Floor Drains (DFR)	2DFR*MOV120	6	Gate	150	2	1	SMB-00-10(1)	17
	2DFR*MOV121	6	Gate	150	2	1	SMB-00-10(1)	17
	2DFR*MOV139	3	Gate	150	2	1	SMB-000-5(1)	17
	2DFR*MOV140	3	Gate	150	2	1	SMB-000-5(1)	17
Standby Diesel Generator Air Startup (EGA)	2EGA*RV125	30	SRV	150	3	10	None	56
	2EGA*RV126	30	SRV	150	3	10	None	56
	2EGA*RV127	22	SRV	150	3	10	None	56
Feedwater (FWS)	2FWS*AOV23A,B	24	Check	900	1	3	Series 2A(6)	9
	2FWS*MOV21A,B	24	Gate	900	1	1	SMB-4-200(1)	17
	2FWS*V12A,B	24	Check	900	1	3	None	9
Standby Diesel Generator Fuel (EGF)	2EGF*V12	1	Check	600	3	1	None	22
	2EGF*V13	1	Check	600	3	1	None	22
	2EGF*V32	1	Check	600	3	1	None	22
	2EGF*V33	1	Check	600	3	1	None	22
	2EGF*V52	1	Check	600	3	1	None	22
	2EGF*V53	1	Check	600	3	1	None	22
Nitrogen Tanks (GSN)	2GSN*SOV166	1	Globe	1500	2	6	76P-035(7)	9
	2GSN*V70A,B	1	Check	600	3	1	None	22,75
	2GSN*V75A,B	1	Check	600	3	1	None	76
Standby Gas Treatment (GTS)	2GTS*MOV1A,B	20	Butterfly	150	2	9	SMB-00-10/H3BC(1)	10
	2GTS*MOV2A,B	20	Butterfly	150	2	9	85430(3)	11
	2GTS*MOV3A,B	20	Butterfly	150	2	9	85430(3)	11
	2GTS*MOV4A,B	8	Gate	150	2	1	SMB-00-15(1)	11
	2GTS*MOV28A,B	8	Butterfly	150	2	9	86040(3)	12
	2GTS*PV5A,B	14	Butterfly	150	2	9	86060(3)	13
Hydrogen Recombiner (HCS)	2HCS*MOV1A,B	3	Gate	150	2	15	SMB-00-10(1)	3
	2HCS*MOV2A,B	3	Globe	150	2	1	SMB-000-5(1)	3
	2HCS*MOV3A,B	3	Gate	150	2	15	SMB-00-10(1)	3
	2HCS*MOV4A,B	3	Gate	150	2	15	SMB-000-5(1)	3
	2HCS*MOV5A,B	3	Globe	150	2	1	SMB-000-5(1)	3
	2HCS*MOV6A,B	3	Gate	150	2	15	SMB-000-5(1)	3
	2HCS*SOV10A,B	1	Globe	1500	2	6	76P-024(7)	3
	2HCS*SOV11A,B	1	Globe	1500	2	6	76P-024(7)	3

TABLE 3.9A-12 (Cont'd.)

System Name	Mark Number	Size	Valve Type	Pressure Rating(*)	ASME Class	Mfg.	Valve Operator Model (Mfg.)	Active Function
Control Building Air Conditioner (HVC)	2HVC*MOV1A,B	18	Butterfly	150	3	2	SMB-000-2/H1BC(1)	8
Control Building Chilled Water (HVK)	2HVK*RV14A,B	3/4 x 1	SRV	150/150	3	8	None	4
	2HVK*RV35A,B	3/4 x 1	SRV	150/150	2	8	None	4
	2HVK*RV37A,B	3/4 x 1	SRV	150/150	2	8	None	4
	2HVK*RV43A,B	3/4 x 1	SRV	150/150	2	8	None	4
	2HVK*SOV36A,B	3	Globe	1500	3	6	76P-034(7)	5
	2HVK*TV21A,B	4	Globe	150	3	5	(3)	6
	2HVK*TV22A,B	4	Globe	150	3	5	(3)	7
Instrument Air (IAS)	2IAS*V448	1 1/2	Check	600	3	1	None	16,22,59
	2IAS*V449	1 1/2	Check	600	3	1	None	16,22,59
	2IAS*V372	1	Check	600	3	1	None	22
	2IAS*V371	1	Check	600	3	1	None	22
	2IAS*SOV164	1 1/2	Globe	1500	2	6	76P-019(7)	16,59
	2IAS*SOV165	1 1/2	Globe	1500	2	6	76P-019(7)	16,59
	2IAS*SOV166	1 1/2	Globe	1500	2	6	76P-019(7)	16
	2IAS*SOV167	1 1/2	Globe	1500	2	6	76P-019(7)	16
	2IAS*SOV168	1 1/2	Globe	1500	2	6	76P-019(7)	16
	2IAS*SOV180	1 1/2	Globe	1500	2	6	76P-019(7)	16
	2IAS*SOV184	1 1/2	Globe	1500	2	6	76P-019(7)	16
	2IAS*SOV185	1 1/2	Globe	1500	2	6	76P-019(7)	16
	2IAS*SV19A,B	3/4 x 1	SRV	600/150	3	8	None	56,59
	2IAS*SV20A,B	3/4 x 1	SRV	150/150	3	8	None	56,59
	2IAS*SOVY181	3/4	Globe	1500	3	6	76P-036	59
	2IAS*SOVY186	3/4	Globe	1500	3	6	76P-036	59
	2IAS*SOVX181	1 1/2	Globe	1500	3	6	76P-037	59
	2IAS*SOVX186	1 1/2	Globe	1500	3	6	76P-037	59
	2IAS*V571	1 1/4	Check	600	3	1	None	22,56
	2IAS*V471	1 1/4	Check	600	3	1	None	22,56
	2IAS*V421	1 1/4	Check	600	3	1	None	22,56
	2IAS*V431	1 1/4	Check	600	3	1	None	22,56
	2IAS*V526	1 1/4	Check	600	3	1	None	22,56
	2IAS*V546	1 1/4	Check	600	3	1	None	22,56
	2IAS*V581	1 1/4	Check	600	3	1	None	22,56
	2IAS*EFV200	3/4	Check	350	2	13	None	59,60
	2IAS*EFV201	3/4	Check	350	2	13	None	59,60
	2IAS*EFV202	3/4	Check	350	2	13	None	59,60
	2IAS*EFV203	3/4	Check	350	2	13	None	59,60
	2IAS*EFV204	3/4	Check	350	2	13	None	59,60
	2IAS*EFV205	3/4	Check	350	2	13	None	59,60
	2IAS*EFV206	3/4	Check	350	2	13	None	59,60

TABLE 3.9A-12 (Cont'd.)

System Name	Mark Number	Size	Valve Type	Pressure Rating(*)	ASME Class	Mfg.	Valve Operator Model (Mfg.)	Active Function
Reactor Core Isolation Cooling (ICS)	2ICS*AOV109	2	Globe	150	2	5	None	38
	2ICS*AOV110	2	Globe	150	2	5	None	38
	2ICS*AOV130	2	Globe	900	2	5	None	38
	2ICS*AOV131	2	Globe	900	2	5	None	38
	2ICS*AOV156	6	Check	900	1	3	Series 2A(6)	40
	2ICS*AOV157	6	Check	900	1	3	Series 2A(6)	40
	2ICS*MOV116	2	Globe	1500	2	1	SMB-00-5(1)	42
	2ICS*MOV120	4	Globe	900	2	1	SMB-00-10(1)	43
	2ICS*MOV121	10	Gate	900	1	1	SB-2-60(1)	9
	2ICS*MOV122	12	Gate	150	2	1	SMB-0-25(1)	44
	2ICS*MOV124	4	Gate	900	2	1	SMB-00-10(1)	41
	2ICS*MOV126	6	Gate	900	1	1	SMB-0-40(1)	9
	2ICS*MOV128	10	Gate	900	1	1	SB-2-60(1)	9
	2ICS*MOV129	6	Gate	150	2	1	SMB-00-10(1)	46
	2ICS*MOV136	6	Gate	150	2	1	SMB-00-10(1)	47
	2ICS*MOV143	2	Globe	1500	2	1	SMB-00-5(1)	31
	2ICS*MOV148	1 1/2	Globe	1500	2	1	SMB-000-5(1)	48
	2ICS*MOV159	1	Globe	1500	2	1	SMB-000-2(1)	43
	2ICS*MOV164	1 1/2	Globe	1500	2	1	SMB-000-5(1)	48
	2ICS*MOV170	1	Globe	1500	2	1	SMB-000-2(1)	9
	2ICS*RV112	3/4 x 1	SRV	150/150	2	8	None	4
	2ICS*RV114	3/4 x 1	SRV	150	2	8	None	4
	2ICS*V27	6	Check	150	2	1	None	22
	2ICS*V28	6	Check	150	2	1	None	22
	2ICS*V29	12	Check	150	2	1	None	22
	2ICS*V38	2	Check	1500	2	1	None	22
	2ICS*V39	1 1/2	Check	600	2	1	None	22
	2ICS*V40	1 1/2	Check	600	2	1	None	22
	2ICS*PCV115	2	E/H	900	2	5	(3)	42
	2ICS*EFV1	3/4	Check	1250	2	13	None	9
	2ICS*EFV2	3/4	Check	1250	2	13	None	9
	2ICS*EFV3	3/4	Check	1250	2	13	None	9
	2ICS*EFV4	3/4	Check	1250	2	13	None	9
Reactor Vessel Instrumentation (ISC)	2ISC*RV33A,B	24	Check	150	2	10	(4)	1
	2ISC*RV34A,B	24	Check	150	2	10	(4)	1
	2ISC*RV35A,B	24	Check	150	2	10	(4)	1
	2ISC*RV36A,B	24	Check	150	2	10	(4)	1
	2ISC*EFV1	3/4	Check	1250	2	13	None	60
	2ISC*EFV2	3/4	Check	1250	2	13	None	60
	2ISC*EFV3	3/4	Check	1250	2	13	None	60
	2ISC*EFV4	3/4	Check	1250	2	13	None	60
	2ISC*EFV5	3/4	Check	1250	2	13	None	60
	2ISC*EFV6	3/4	Check	1250	2	13	None	60
	2ISC*EFV7	3/4	Check	1250	2	13	None	60

TABLE 3.9A-12 (Cont'd.)

System Name	Mark Number	Size	Valve Type	Pressure Rating(*)	ASME Class	Mfg.	Valve Operator Model (Mfg.)	Active Function
Reactor Vessel Instrumentation (ISC) (cont'd.)	2ISC*EFV8	3/4	Check	1250	2	13	None	60
	2ISC*EFV9	3/4	Check	1250	2	13	None	60
	2ISC*EFV10	3/4	Check	1250	2	13	None	60
	2ISC*EFV11	3/4	Check	1250	2	13	None	60
	2ISC*EFV12	3/4	Check	1250	2	13	None	60
	2ISC*EFV13	3/4	Check	1250	2	13	None	60
	2ISC*EFV14	3/4	Check	1250	2	13	None	60
	2ISC*EFV15	3/4	Check	1250	2	13	None	60
	2ISC*EFV16	3/4	Check	1250	2	13	None	60
	2ISC*EFV17	3/4	Check	1250	2	13	None	60
	2ISC*EFV18	3/4	Check	1250	2	13	None	60
	2ISC*EFV19	3/4	Check	1250	2	13	None	60
	2ISC*EFV20	3/4	Check	1250	2	13	None	60
	2ISC*EFV21	3/4	Check	1250	2	13	None	60
	2ISC*EFV22	3/4	Check	1250	2	13	None	60
	2ISC*EFV23	3/4	Check	1250	2	13	None	60
	2ISC*EFV24	3/4	Check	1250	2	13	None	60
	2ISC*EFV25	3/4	Check	1250	2	13	None	60
	2ISC*EFV26	3/4	Check	1250	2	13	None	60
	2ISC*EFV27	3/4	Check	1250	2	13	None	60
	2ISC*EFV28	3/4	Check	1250	2	13	None	60
	2ISC*EFV29	3/4	Check	1250	2	13	None	60
	2ISC*EFV30	3/4	Check	1250	2	13	None	60
	2ISC*EFV31	3/4	Check	1250	2	13	None	60
	2ISC*EFV32	3/4	Check	1250	2	13	None	60
	2ISC*EFV33	3/4	Check	1250	2	13	None	60
	2ISC*EFV34	3/4	Check	1250	2	13	None	60
	2ISC*EFV35	3/4	Check	1250	2	13	None	60
	2ISC*EFV36	3/4	Check	1250	2	13	None	60
	2ISC*EFV37	3/4	Check	1250	2	13	None	60
	2ISC*EFV38	3/4	Check	1250	2	13	None	60
	2ISC*EFV39	3/4	Check	1250	2	13	None	60
	2ISC*EFV40	3/4	Check	1250	2	13	None	60
	2ISC*EFV41	3/4	Check	1250	2	13	None	60
	2ISC*EFV42	3/4	Check	1250	2	13	None	60
Containment Leakage Monitoring (LMS)	2LMS*SOV152	3/4	Globe	1500	2	6	76P-001(7)	16
	2LMS*SOV153	3/4	Globe	1500	2	6	76P-001(7)	16
	2LMS*SOV157	3/4	Globe	1500	2	6	76P-001(7)	16
	2LMS*SOV156	3/4	Globe	1500	2	6	76P-001(7)	16

TABLE 3.9A-12 (Cont'd.)

System Name	Mark Number	Size	Valve Type	Pressure Rating(*)	ASME Class	Mfg.	Valve Operator Model (Mfg.)	Active Function
Main Steam System (MSS)	2MSS*MOV111	6	Globe	600	1	1	SMB-2-25(1)	8,9,16
	2MSS*MOV112	6	Globe	600	1	1	SMB-2-25(1)	8,9,16,45
	2MSS*MOV208	2	Globe	1500	1	1	SMB-000-5(1)	16,8,9
	2MSS*PSV120	8 x 10	SRV	1500/300	1	14	None	56
	2MSS*PSV121	8 x 10	SRV	1500/300	1	14	None	56
	2MSS*PSV122	8 x 10	SRV	1500/300	1	14	None	56,59
	2MSS*PSV123	8 x 10	SRV	1500/300	1	14	None	56
	2MSS*PSV124	8 x 10	SRV	1500/300	1	14	None	56
	2MSS*PSV125	8 x 10	SRV	1500/300	1	14	None	56
	2MSS*PSV126	8 x 10	SRV	1500/300	1	14	None	56,59
	2MSS*PSV127	8 x 10	SRV	1500/300	1	14	None	56,59
	2MSS*PSV128	8 x 10	SRV	1500/300	1	14	None	56
	2MSS*PSV129	8 x 10	SRV	1500/300	1	14	None	56,59
	2MSS*PSV130	8 x 10	SRV	1500/300	1	14	None	56,59
	2MSS*PSV131	8 x 10	SRV	1500/300	1	14	None	56
	2MSS*PSV132	8 x 10	SRV	1500/300	1	14	None	56
	2MSS*PSV133	8 x 10	SRV	1500/300	1	14	None	56
	2MSS*PSV134	8 x 10	SRV	1500/300	1	14	None	56,59
	2MSS*PSV135	8 x 10	SRV	1500/300	1	14	None	56
	2MSS*PSV136	8 x 10	SRV	1500/300	1	14	None	56
	2MSS*PSV137	8 x 10	SRV	1500/300	1	14	None	56,59
	2MSS*EFV1A-D	3/4	Check	1250	2	13	None	60,16
	2MSS*EFV2A-D	3/4	Check	1250	2	13	None	60,16
	2MSS*EFV3A-D	3/4	Check	1250	2	13	None	60,16
	2MSS*EFV4A-D	3/4	Check	1250	2	13	None	60,16
Reactor Coolant Recirculation (RCS)	2RCS*SOV68A,B	3/4	Globe	2500	2	6	76P-040(7)	9
	2RCS*SOV65A,B	2	Globe	1500	2	6	76P-038(7)	9
	2RCS*SOV66A,B	1	Globe	2500	2	6	76P-039(7)	9
	2RCS*SOV67A,B	2	Globe	1500	2	6	76P-038(7)	9
	2RCS*SOV79A,B	2	Globe	1500	2	6	76P-038(7)	9
	2RCS*SOV80A,B	1	Globe	2500	2	6	76P-039(7)	9
	2RCS*SOV81A,B	2	Globe	1500	2	6	76P-038(7)	9
	2RCS*SOV82A,B	3/4	Globe	2500	2	6	76P-040(7)	9
	2RCS*SOV104	3/4	Globe	1500	2	6	76P-049(7)	9
	2RSC*SOV105	3/4	Globe	1500	2	6	76P-049(7)	9
	2RCS*V59A,B	3/4	Check	1500	2	1	None	9
	2RCS*V60A,B	3/4	Check	1500	2	1	None	9
	2RCS*V90A,B	3/4	Check	1500	2	1	None	9
	2RCS*EFV43A,B	3/4	Check	1250	2	13	None	16
	2RCS*EFV44A,B	3/4	Check	1250	2	13	None	16
	2RCS*EFV45A,B	3/4	Check	1250	2	13	None	16
	2RCS*EFV46A,B	3/4	Check	1250	2	13	None	16
	2RCS*EFV47A,B	3/4	Check	1250	2	13	None	16
	2RCS*EFV48A,B	3/4	Check	1250	2	13	None	16

TABLE 3.9A-12 (Cont'd.)

System Name	Mark Number	Size	Valve Type	Pressure Rating(*)	ASME Class	Mfg.	Valve Operator Model (Mfg.)	Active Function
Reactor Coolant Recirculation (RCS) (cont'd.)	2RCS*EFV52A,B	3/4	Check	1250	2	13	None	16
	2RCS*EFV53A,B	3/4	Check	1250	2	13	None	16
	2RCS*EFV62A,B	3/4	Check	1250	2	13	None	16
	2RCS*EFV63A,B	3/4	Check	1250	2	13	None	16
Residual Heat Removal (RHS)	2RHS*MOV1A,B	24	Butterfly	300	2	9	SMB-2-60(1)	49,52,53
	2RHS*MOV1C	24	Butterfly	300	2	9	SMB-0-25(1)	55,16
	2RHS*MOV2A,B	18	Butterfly	300	2	9	SMB-0-25(1)	50,52
	2RHS*MOV9A,B	18	Butterfly	300	2	2	SMB-00-10(1)	50,53,55
	2RHS*V3	18	Check	300	2	1	None	49,50,53, 55
	2RHS*AOV16A-C	12	Check	900	1	3	Series 2A(6)	40
	2RHS*AOV39A,B	12	Check	900	1	3	Series 2A(6)	50,16
	2RHS*AOV150	16	Check	300	2	3	4-A-FFX-8-3/4-Y(5)	51
	2RHS*FV38A-C	18	Globe	300	2	5	SMB-00-5(1)	52,53
	2RHS*MOV4A-C	6	Gate	300	2	1	SMB-00S-15(1)	54,52
	2RHS*MOV15A,B	16	Gate	300	2	1	SMB-2-80(1)	52,55,16
	2RHS*MOV22A,B	8	Globe	900	2	1	SMB-1-25(1)	52
	2RHS*MOV23A,B	8	Globe	900	2	1	SMB-1-25(1)	52
	2RHS*MOV24A,B,C	12	Gate	900	1	1	SMB-3-100(1)	49,52,16
	2RHS*MOV25A,B	16	Gate	300	2	1	SMB-2-80(1)	52,55,16
	2RHS*MOV26A,B	1	Globe	1500	2	1	SMB-000-2(1)	52,16
	2RHS*MOV27A,B	1	Globe	1500	2	1	SMB-000-2(1)	52,16
	2RHS*MOV30A,B	18	Butterfly	300	2	9	SMBO-25/H4BC(1)	53,54,16
	2RHS*MOV8A,B	18	Butterfly	300	2	9	SMB-1-25/H5BC(1)	49,50,52
	2RHS*V60	2	Check	600	2		None	22
	2RHS*V143	6	Check	900	1	1	None	50
	2RHS*RV117	3/4 x 1	SRV	185/180	2	8	None	56
	2RHS*MOV32A,B	4	Gate	300	2	1	SMB-00-10(1)	52
	2RHS*MOV33A,B	4	Globe	300	2	1	SMB-000-5(1)	52,55,16
	2RHS*MOV37A,B	4	Globe	300	2	1	SMB-000-5(1)	52
	2RHS*MOV40A,B	12	Globe	900	1	1	SMB-3-80(1)	50,52,16
	2RHS*MOV67A,B	2	Globe	1500	1	1	SMB-000-5(1)	16
	2RHS*MOV80A,B	1	Globe	1500	2	1	SMB-000-2(1)	52
	2RHS*MOV104	6	Globe	900	1	1	SMB-0-10(1)	50,52,16
	2RHS*MOV112	20	Gate	900	1	1	SB-3-150(1)	50,16
	2RHS*MOV113	20	Gate	900	1	1	SB-3-150(1)	50,16
	2RHS*MOV115	16	Gate	300	2	1	SMB-0-25(1)	51
	2RHS*MOV116	16	Gate	150	3	1	SMB-0-25(1)	51
	2RHS*MOV142	3	Globe	300	2	1	SMB-000-5(1)	52
	2RHS*MOV149	3	Gate	300	2	1	SMB-000-5(1)	52
	2RHS*RV20A-C	3/4 x 1	SRV	300/150	2	8	None	56
	2RHS*RV61A-C	3/4 x 1	SRV	300/150	2	8	None	56
	2RHS*RV108	3 x 4	SRV	150/150	2	8	None	56
	2RHS*RV110	3/4 x 1	SRV	300/150	2	8	None	56

TABLE 3.9A-12 (Cont'd.)

System Name	Mark Number	Size	Valve Type	Pressure Rating(*)	ASME Class	Mfg.	Valve Operator Model (Mfg.)	Active Function
Residual Heat Removal (RHS) (cont'd.)	2RHS*RV139	3/4 x 1	SRV	300/150	2	8	None	56
	2RHS*RV152	3/4 x 1	SRV	900/150	1	8	None	56
	2RHS*SOV35A,B	3/4	Globe	1500	2	6	76P-032(7)	52,34
	2RHS*SOV36A,B	3/4	Globe	1500	2	6	76P-030(7)	52,34
	2RHS*SOV70A,B	1	Globe	1500	2	6	76P-025(7)	52
	2RHS*SOV71A,B	1	Globe	1500	2	6	76P-025(7)	52
	2RHS*SOV126	3/4	Globe	1500	2	6	76P-023(7)	51,52
	2RHS*SV34A,B	4 x 6	SRV	600/300	2	8	None	56
	2RHS*SV62A,B	6 x 8	SRV	600/300	2	8	None	56
	2RHS*RVV35A,B	10	Vac Brkr	150	2	10	None	57
	2RHS*RVV36A,B	10	Vac Brkr	150	2	10	None	57
	2RHS*MOV12A,B	18	Butterfly	300	2	2	SMB-00-10(1)	50,53,55
	2RHS*V17	2	Stop Chk	600	2	1	None	22
	2RHS*V47	2	Stop Chk	600	2	1	None	22
	2RHS*V61	2	Stop Chk	600	2	1	None	22
	2RHS*V18	2	Check	600	2	1	None	22
	2RHS*V48	2	Check	600	2	1	None	22
	2RHS*V1	18	Check	300	2	1	None	49,50,53, 55
	2RHS*V2	18	Check	300	2	1	None	49,50,53, 55
	2RHS*EFV5	3/4	Check	1250	2	13	None	16
	2RHS*EFV6	3/4	Check	1250	2	13	None	16
	2RHS*EFV7	3/4	Check	1250	2	13	None	16
	2RHS*EFV8	3/4	Check	1250	2	13	None	16
	2RHS*EFV9	3/4	Check	1250	2	13	None	16
	2RHS*EFV10	3/4	Check	1250	2	13	None	16
	2RHS*V19	3/4	Check	600	2	1	None	57
	2RHS*V20	3/4	Check	600	2	1	None	57
	2RHS*V117	3/4	Check	600	2	1	None	57
	2RHS*V118	3/4	Check	600	2	1	None	57
Spent Fuel Pool Cooling and Cleanup (SFC)	2SFC*AOV153	8	Butterfly	300	3	2	N721C-SR80-M3HW(2)	45
	2SFC*AOV154	8	Butterfly	300	3	2	N721C-SR80-M3HW(2)	45
	2SFC*AOV19A,B	8	Butterfly	300	3	2	N721C-SR80-M3HW(2)	45
	2SFC*HV6A,B	10	Butterfly	150	3	2	N721C-SR80-M3HW(2)	35
	2SFC*HV17A,B	8	Butterfly	300	3	2	N721C-SR80-M3HW(2)	35
	2SFC*HV18A,B	8	Butterfly	300	3	2	N721C-SR80-M3HW(2)	35
	2SFC*HV37A,B	8	Butterfly	300	3	2	N721C-SR80-M3HW(2)	35
	2SFC*V11	8	Check	150	3	1	None	78
	2SFC*V20A,B	8	Check	300	3	1	None	22
	2SFC*V9	8	Check	150	3	1	None	78

TABLE 3.9A-12 (Cont'd.)

System Name	Mark Number	Size	Valve Type	Pressure Rating(*)	ASME Class	Mfg.	Valve Operator Model (Mfg.)	Active Function
Standby Liquid Control System (SLS)	2SLS*MOV1A,B	3	Globe	150	2	1	SB-00-5(1)	66
	2SLS*MOV5A,B	2	Stop Chk	1500	1	1	SMB-00-10(1)	66,16
	2SLS*RV2A,B	3/4 x 1	SRV	1500/150	2	8	None	56
Main Steam Safety/ Relief Valves, Vents and Drains (SVV)	2SVV*RVV101	10	Check	600	3	10	None	21
	2SVV*RVV102	10	Check	600	3	10	None	21
	2SVV*RVV103	10	Check	600	3	10	None	21
	2SVV*RVV104	10	Check	600	3	10	None	21
	2SVV*RVV105	10	Check	600	3	10	None	21
	2SVV*RVV106	10	Check	600	3	10	None	21
	2SVV*RVV107	10	Check	600	3	10	None	21
	2SVV*RVV108	10	Check	600	3	10	None	21
	2SVV*RVV109	10	Check	600	3	10	None	21
	2SVV*RVV110	10	Check	600	3	10	None	21
	2SVV*RVV111	10	Check	600	3	10	None	21
	2SVV*RVV112	10	Check	600	3	10	None	21
	2SVV*RVV113	10	Check	600	3	10	None	21
	2SVV*RVV114	10	Check	600	3	10	None	21
	2SVV*RVV115	10	Check	600	3	10	None	21
	2SVV*RVV116	10	Check	600	3	10	None	21
	2SVV*RVV117	10	Check	600	3	10	None	21
	2SVV*RVV118	10	Check	600	3	10	None	21
	2SVV*RVV201	10	Check	600	3	10	None	21
	2SVV*RVV202	10	Check	600	3	10	None	21
	2SVV*RVV203	10	Check	600	3	10	None	21
	2SVV*RVV204	10	Check	600	3	10	None	21
	2SVV*RVV205	10	Check	600	3	10	None	21
	2SVV*RVV206	10	Check	600	3	10	None	21
	2SVV*RVV207	10	Check	600	3	10	None	21
	2SVV*RVV208	10	Check	600	3	10	None	21
	2SVV*RVV209	10	Check	600	3	10	None	21
	2SVV*RVV210	10	Check	600	3	10	None	21
	2SVV*RVV211	10	Check	600	3	10	None	21
	2SVV*RVV212	10	Check	600	3	10	None	21
	2SVV*RVV213	10	Check	600	3	10	None	21
	2SVV*RVV214	10	Check	600	3	10	None	21
	2SVV*RVV215	10	Check	600	3	10	None	21
	2SVV*RVV216	10	Check	600	3	10	None	21
	2SVV*RVV217	10	Check	600	3	10	None	21
	2SVV*RVV218	10	Check	600	3	10	None	21
	2SVV*RVV301	2 1/2	Check	150	3	10	None	21
	2SVV*RVV302	2 1/2	Check	150	3	10	None	21
	2SVV*RVV303	2 1/2	Check	150	3	10	None	21
	2SVV*RVV304	2 1/2	Check	150	3	10	None	21
	2SVV*RVV305	2 1/2	Check	150	3	10	None	21

TABLE 3.9A-12 (Cont'd.)

System Name	Mark Number	Size	Valve Type	Pressure Rating(*)	ASME Class	Mfg.	Valve Operator Model (Mfg.)	Active Function
Main Steam Safety/ Relief Valves, Vents and Drains (SVV) (cont'd.)	2SVV*RVV306	2 1/2	Check	150	3	10	None	21
	2SVV*RVV307	2 1/2	Check	150	3	10	None	21
	2SVV*RVV308	2 1/2	Check	150	3	10	None	21
	2SVV*RVV309	2 1/2	Check	150	3	10	None	21
	2SVV*RVV310	2 1/2	Check	150	3	10	None	21
	2SVV*RVV311	2 1/2	Check	150	3	10	None	21
	2SVV*RVV312	2 1/2	Check	150	3	10	None	21
	2SVV*RVV313	2 1/2	Check	150	3	10	None	21
	2SVV*RVV314	2 1/2	Check	150	3	10	None	21
	2SVV*RVV315	2 1/2	Check	150	3	10	None	21
	2SVV*RVV316	2 1/2	Check	150	3	10	None	21
	2SVV*RVV317	2 1/2	Check	150	3	10	None	21
	2SVV*RVV318	2 1/2	Check	150	3	10	None	21
Service Water (SWP)	2SWP*AOV20A,22A	1 1/2	Plug	150	3	4	NCB520-SR80(2)	64
	2SWP*AOV20B,22B	2	Plug	150	3	4	NCB725-SR80(2)	64
	2SWP*AOV97A,B	6	Plug	150	3	4	NTB12-SR3-M3HW(2)	68
	2SWP*AOV572	2 1/2	Plug	150	3	4	NCB725-SR80(2)	68
	2SWP*AOV78A,B	2	Plug	150	3	4	NCB725-SR80(2)	68
	2SWP*FV54A,B	30	Butterfly	150	3	9	PD87265-500(7)	70
	2SWP*MOV1A-F	4	Ball	150	3	11	SMB-000-2/H1BC(1)	2
	2SWP*MOV3A,B	30	Butterfly	150	3	9	SMB-2-40/H6BC(1)	45
	2SWP*MOV19A,B	20	Butterfly	150	3	9	SMB-1-15/H4BC(1)	45
	2SWP*MOV33A,B	18	Butterfly	150	3	9	SMB-0-25/H4BC(1)	67
	2SWP*FV47A,B	30	Butterfly	150	3	9	PD89265-500-003&4(9)	70
	2SWP*MOV50A,B	36	Butterfly	150	3	9	SMB-3-60/H6BC(1)	71
	2SWP*MOV74A-F	18	Butterfly	150	3	9	SMB-0-40/H4BC(1)	69
	2SWP*MOV90A,B	18	Butterfly	150	3	9	SMB-0-25/H4BC(1)	67
	2SWP*MOV15A,B	2 1/2	Gate	150	3	1	SMB-000-5(1)	68
	2SWP*MOV17A,B	12	Gate	150	3	1	SMB-0-25(1)	63
	2SWP*MOV18A,B	12	Gate	150	3	1	SMB-0-25(1)	63
	2SWP*MOV21A,B	3	Gate	150	3	1	SMB-000-5(1)	36
	2SWP*MOV66A,B	8	Gate	150	3	1	SMB-00-15(1)	65
	2SWP*MOV67A,B	4	Gate	150	3	1	SMB-000-5(1)	68
	2SWP*MOV94A,B	8	Gate	150	3	1	SMB-00-15(1)	65
	2SWP*MOV95A,B	8	Gate	150	3	1	SMB-00-15(1)	65
	2SWP*MOV599	30	Butterfly	150	3	9	SMB-1-25/H5BC(1)	45
	2SWP*MOV93A,B	24	Butterfly	150	3	9	SMB-1-15/H4BC(1)	45
	2SWP*MOV30A,B	48 x 72	Butterfly	150	3	12	SMB-0-15/H4BC(1)	72
	2SWP*MOV77A,B	48 x 48	Butterfly	150	3	12	SMB-00-10/H3BC(1)	73
	2SWP*RV9A,B	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RV10A,B	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RV11A,B	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RV27A,B	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RV34A,B	4 x 6	SRV	300/150	3	8	None	56

TABLE 3.9A-12 (Cont'd.)

System Name	Mark Number	Size	Valve Type	Pressure Rating(*)	ASME Class	Mfg.	Valve Operator Model (Mfg.)	Active Function
Service Water (SWP) (cont'd.)	2SWP*RV53A,B	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RV58A,B	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RV68A,B	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RV72A,B	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RV80A-F	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RV82A-D	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RV83A-E	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RV84A-C	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RV85A-C	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RV87A,B	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RV89A,B	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RV155A,B	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RV81A,B	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RV202A,B	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RV203	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RV515	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RV518	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RV556	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RV558	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RV564	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RV575	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RV576	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RVX46A,B	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RVX157A,B	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RVY46A,B	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*RVY157A,B	3/4 x 1	SRV	150/150	3	8	None	4
	2SWP*TV35A,B	4	Globe	150	3	5	(3)	68
	2SWP*V1A-F	18	Check	150	3	9	None	22
	2SWP*V201A,B	1 1/4	Check	600	3	16	None	22
	2SWP*V202A,B	30	Check	150	3	9	None	22,45
	2SWP*V203A,B	2	Check	600	3	1	None	22
	2SWP*V219A,B	4	Check	150	3	1	None	22
	2SWP*V240A	4	Check	150	3	16	None	22
	2SWP*V240B	4	Check	150	3	16	None	22
	2SWP*V259	8	Check	150	3	1	None	22
	2SWP*V260	8	Check	150	3	1	None	22
	2SWP*V76A,B	8	Check	150	3	1	None	22
	2SWP*V720A,B	1	Check	600	3	1	None	22
	2SWP*V1002A,B	3	Check	150	3	1	None	22
	2SWP*V1027	30	Check	150	3	9	None	22
	2SWP*V1028	30	Check	150	3	9	None	22
	2SWP*V1024	6	Check	150	3	1	None	22
	2SWP*V1025	6	Check	150	3	1	None	22
	2SWP*V1029	30	Check	150	3	9	None	22
	2SWP*RV566	3/4 x 1	SRV	150/150	3	8	None	4

TABLE 3.9A-12 (Cont'd.)

System Name	Mark Number	Size	Valve Type	Pressure Rating(*)	ASME Class	Mfg.	Valve Operator Model (Mfg.)	Active Function
Service Water (SWP) (cont'd.)	2SWP*AOV154A,B	1 1/2	Plug	150	3	4	NCB520-SR80(2)	68
	2SWP*AOV571	1 1/2	Plug	150	3	4	NCB520-SR80(2)	68
	2SWP*AOV581	1 1/2	Plug	150	3	4	NCB520-SR80(2)	68
	2SWP*AOV573	2	Plug	150	3	4	NCB725-SR80(2)	68
	2SWP*AOV574	2	Plug	150	3	4	NCB725-SR80(2)	68
Reactor Water Cleanup (WCS)	2WCS*MOV102	8	Globe	600	1	1	SB-2-80(1)	16
	2WCS*MOV112	8	Globe	600	1	1	SB-2-80(1)	16
	2WCS*MOV200	8	Globe	900	1	1	SMB-1-25(1)	16
	2WCS*EFV221	3/4	Check	1250	2	13	None	16
	2WCS*EFV222	3/4	Check	1250	2	13	None	16
	2WCS*EFV223	3/4	Check	1250	2	13	None	16
	2WCS*EFV224	3/4	Check	1250	2	13	None	16
	2WCS*EFV300	3/4	Check	1250	2	13	None	16

Key to Manufacturer

1 = Velan Corp.
2 = Posi-Seal
3 = Anchor-Darling
4 = Atwood & Morill
5 = Copes-Vulcan
6 = Target Rock
7 = Gulf & Western
8 = Crosby
9 = Clow
10 = GPE Controls
11 = Contromatics
12 = Henry Pratt Co.
13 = Dragon
14 = Dikkers
15 = Westinghouse
16 = Enertech

Key to Valve Operator/Manufacturer

1 = Motor/Limitorque
2 = Air/Bettis
3 = Electrohydraulic/Borg-Warner
4 = Pneumatic/Parker-Hannifin (cylinder)
5 = Air/Anchor-Darling
6 = Air/Parker-Hannifin (piston)
7 = Target Rock
8 = (Deleted)
9 = Electrohydraulic/Paul Monroe

(*) Pressure Rating - Inlet/Outlet

Key to Active Functions

1 = Relieve pressure from suppression chamber to drywell.
2 = Pressure control for self-cleaning strainers for service water pumps.
3 = Hydrogen recombiner isolation (active function required only after approximately 2 days following a LOCA).
4 = Pressure relief.
5 = Computer room isolation (safety class change).
6 = Temperature control of control room air-conditioning unit.
7 = Temperature control of relay room air-conditioning unit.
8 = High radiation isolation valve.
9 = Containment isolation during LOCA.

TABLE 3.9A-12 (Cont'd.)

10	=	Redundant supply flow path to GTS from reactor building ventilation system.
11	=	GTS filter train operation.
12	=	GTS filter train isolation upon high temperature of one of the GTS trains.
13	=	Modulate to maintain the required reactor building pressure differential.
14	=	Isolation between GTS and CPS.
15	=	(Deleted)
16	=	Primary containment isolation.
17	=	Isolation valves for penetrations through primary containment wall.
18	=	Drywell sample selector valves.
19	=	(Deleted)
20	=	Containment atmosphere monitoring and postaccident sampling system isolation valves.
21	=	Vacuum breaker, water hammer mitigation for main steam SRV discharge piping.
22	=	Prevent reverse flow.
23	=	HPCS system injection valve.
24	=	HPCS suction from 2CNS-TK1B, isolates on low level in tank.
25	=	HPCS suction from suppression pool, opens on low level in 2CNS-TK1B.
26	=	HPCS system test valve to be closed to ensure maximum flow is injected into the vessel. Valve is opened to allow flow to return to suppression pool during full flow.
27	=	LPCS system test valve to be closed to ensure maximum flow is injected into the vessel. Valve is opened to allow flow to return to suppression pool during full flow.
28	=	LPCS system injection valve and containment isolation when LPCS service is terminated.
29	=	LPCS system test valve to be closed to ensure maximum flow is injected into condensate tank.
30	=	Prevent reverse flow and provide pressure boundary.
31	=	Minimum flow bypass to protect pump.
32	=	Containment isolation on low suppression pool level.
33	=	Reactor vessel drain line isolation and class break (ASME III to ANSI B31.1).
34	=	Sample isolation and class break (ASME III to ANSI B31.1).
35	=	SFC loop isolation.
36	=	SFC makeup fill isolation.
37	=	Cask handling area isolation.
38	=	RCIC turbine drain pot drain isolation.
39	=	Cooling loop shutoff valve.
40	=	Containment isolation, open for injection.
41	=	Isolates RCIC test return.
42	=	Open to allow lube oil cooling.
43	=	RCIC turbine steam supply valve.
44	=	RCIC turbine exhaust - containment isolation.
45	=	Safety class change.
46	=	Isolates on low CST level.
47	=	Opens on low CST level.
48	=	Turbine exhaust vacuum relief.
49	=	LPCI injection.
50	=	Shutdown cooling.
51	=	RHR/SWP cross-connect for post-LOCA containment flooding.
52	=	System boundary isolation.
53	=	Suppression pool cooling.
54	=	RHR pump miniflow.
55	=	Containment spray.

TABLE 3.9A-12 (Cont'd.)

56	=	Overpressure protection.
57	=	Vacuum breaker/water hammer mitigation.
58	=	(Deleted)
59	=	Automatic depressurization system.
60	=	Prevent excess flow.
61	=	(Deleted)
62	=	Hydrogen/oxygen analyzer isolation valves to open on loss of offsite power.
63	=	SWP/CCP crosstie for SFC heat exchanger.
64	=	SWP/CCP crosstie for RHR pump seal coolers.
65	=	SWP supply to/from diesel generators.
66	=	Standby liquid control injection.
67	=	SWP supply to/from RHR heat exchangers.
68	=	SWP supply to/from safety-related unit coolers/chillers.
69	=	SWP pump discharge valves.
70	=	SWP makeup to CWS.
71	=	SWP divisional cross-connect isolation.
72	=	Intake bay cross-connect isolation.
73	=	Intake bay traveling screen bypass.
74	=	LPCS injection.
75	=	Allow flow to ADS accumulator tanks.
76	=	Allow emergency nitrogen flow.
78	=	Allow flow in the forward direction.
83	=	CMS atmosphere sample valves.

TABLE 3.9A-13

LOAD COMBINATIONS FOR COMPONENT SUPPORTS AND STRESS LIMITS FOR PLATE AND SHELL-TYPE SUPPORTS

A. Load Combinations for Component Supports

Plant Condition	Loading Combination	Service Level
Normal	DL+S+D+R+A+E	A
Upset	DL+S+D_1+R+A+E+D	B
Emergency	Not Applicable	C
Faulted	DL+S+D_2+E+D	D

KEY:

To Loading Combinations

DL = Dead load
S = Superimposed loads
D_1,D_2 = Dynamic Loads 1 and 2, respectively (for definitions, see Table 3.9A-6)
R = Restrained thermal expansion
A = Anchor and support movement
E = Environmental loads
D = Other external dynamic loads

NOTES: For each operating condition, the loadings as given in the table are to be considered simultaneously. Symbols used are defined in the key.

The specific loads of each type which are applied during the applicable operating condition are dependent on the particular system conditions. The following listing identifies some of these specific loads which are used as a general checklist when determining loading conditions as related to plant operating conditions.

Dead Load

Component maximum operating weight (with appurtenances)
Hydrostatic test weight
Operational test weight
Component support weight

TABLE 3.9A-13 (Cont'd.)

Superimposed
- Pressure
- Temperature
- Piping system reactions
- LOCA building deflections

Dynamic
- OBE
- SSE
- Pipe rupture
- Hydrodynamic loads
- Jet impingement
- Missile impact
- Vibrations
- Handling loads (construction, installation, servicing)
- Thermal transients
- Water hammer
- Steam hammer
- Valve trips

Anchor and Support Movement
- OBE/SSE effects
- Thermal growth
- LOCA

Environmental
- Radiation
- Moisture
- Chemicals

TABLE 3.9A-13 (Cont'd.)

B. Stress Limits for Plate and Shell-Type Supports

Service Level	1	(1 + 2)	3
A	1.0S	1.5S	0.5S
B	1.0S	1.5S	0.5S
C	1.2S	1.8S	0.5S
D	lesser of 1.5S or 0.4Su	lesser of 2.25S or 0.6Su	0.5S

KEY:

To Stress Limits

1 = Membrane stress
2 = Bending stress
3 = Maximum tensile stress at the contact surface of a weld producing a tensile load in a direction through the thickness of a plate
S = ASME Section III allowable stress
Su = ASME Section III minimum ultimate tensile strength

TABLE 3.9A-15

REQUIREMENTS FOR SAFETY CLASSES 2 AND 3
INSTRUMENT AND PNEUMATIC TUBING AND SUPPORTS

(TUBING SIZES UP TO AND INCLUDING 1/2 IN O.D.)*

1. Design loads and limits are calculated in accordance with ASME III.

2. Procurement of material is in accordance with ASME III except that alternate QA Category I materials may be used for supports. See Figure 3.9A-1.

3. Fabrication and installation control utilizes QA Category I material marking of exclusive purchase of QA Category I materials with control to point of use.

4. Automatic Class 2 welding follows the requirements of ASME III Code Case N-127 except that Category I documentation is used in lieu of N-127 and the Authorized Nuclear Inspector (ANI) involvement as noted in 5.a below.

5. Fabrication, installation, NDE, and hydrostatic inspections are as follows:

    a. Pressure boundary welds

        (1) Welders and welding procedures are qualified to ASME Section IX.

        (2) 100% documentation of completion of all construction steps by Contractor's construction prior to release to Contractor's QA Program.

        (3) 100% liquid penetrant check of Class 2 field welds (automatic and manual) and visual inspection of all Class 3 field welds by Contractor FQC documented by the Contractor's QA Program.

        (4) Contractor FQC in-process inspection documented by the Contractor's QA Program.

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* This program also applies to safety-related instrument tubing for radiation protection process monitoring up to 1 in when the process piping, equipment, or component is not ASME Section III construction (e.g., HVAC duct).

TABLE 3.9A-15 (Cont'd.)

(5) Pressure test to be performed in accordance with ASME Section III pressure test requirements and document via pressure test reports.

(6) Surveillance inspections by the ANI of approximately 10 percent of in-process activities (welding and hydro) documented by SIS report.

(7) Compression fittings are an acceptable substitute for welded fittings.

b. Compression Fittings

(1) Compression fittings which meet ASME material requirements and the following installation requirements shall be used. The specific design criteria used in the application of compression fittings at Unit 2 are ASME Section III, Subsections NC/ND, Paragraphs 3671.4 and 3673.2.

(2) 100-percent visual inspection of compression fitting makeup is performed by construction prior to release to Contractor's QA Program.

(3) Documentation of the completion of all construction is required.

(4) 100-percent inspection of fitting makeup is performed by FQC using vendor-supplied tools and procedures.

(5) Documentation of the FQC inspection is required.

(6) Pressure test to be performed in accordance with ASME Section III pressure test requirements and document via pressure test reports.

c. Supports

(1) Welders and welding procedures are qualified to Category I specification requirements invoking ASME Section IX or AWS standards as appropriate to the support type.

(2) 100-percent documentation of completion of all construction steps by Contractors's construction prior to release to Contractor's QA Program.

TABLE 3.9A-15 (Cont'd.)

(3) 100-percent Contractor FQC visual inspection of all field welds using ASME, ANSI, or AWS acceptance criteria as required in accordance with the installation specification (except undercut not exceeding 1/32-in deep is acceptable in lieu of AWS D1.1 requirements). These inspections shall be documented by the Contractor's QA Program. For alternate weld inspection, refer to Section 3.8.4.6.

(4) ASTM A515, Gr 65, may be considered an AWS D1.1 prequalified group no. 1 material.

6. Visual examination acceptance for pressure-retaining field welds.

All weld surfaces are sufficiently free from coarse ripples, grooves, overlaps, abrupt ridges, and valleys to allow examination. The following indications are unacceptable:

a. Cracks, external surface.

b. Fillet weld dimension not meeting Figure NC/ND 4427-1 or butt weld reinforcement greater than specified in Figure NC/ND 4427-1.

c. Lack of fusion on the surface.

7. Unsatisfactory conditions noted by the SWEC FQC on SIS reports are to be addressed and resolved via existing Engineering and QA procedures.

3.9B MECHANICAL SYSTEMS AND COMPONENTS (GE SCOPE OF SUPPLY)

3.9B.1 Special Topics for Mechanical Components

3.9B.1.1 Design Transients

This section describes the transients that are used in the design of major NSSS ASME Section III, Safety Class 1 core support, reactor internals, and CRD components. The number of cycles or events for each transient is included. These transients are included in the design specifications and/or stress reports for components. Transients or combinations of transients are classified with respect to the component operating condition categories identified as Normal, Upset, Emergency, Faulted, or Testing in ASME Section III as applicable. (The first four conditions correspond to Service Levels A, B, C, and D, respectively.)

3.9B.1.1.1 Control Rod Drive Transients

The normal and test service load cycles used for the design and fatigue analysis for the 40-yr life of the CRD are as follows:

	Transient	Category	Cycles
1.	Reactor startup/shutdown	Normal/upset	120
2.	Vessel pressure tests	Normal/upset	130
3.	Vessel overpressure	Normal/upset	10
4.	Scram test plus startup scrams	Normal/upset	300
5.	Operational scrams	Normal/upset	300
6.	Jog cycles	Normal/upset	30,000
7.	Shim/drive cycles	Normal/upset	1,000

In addition to the above cycles, the following have been considered in the design of the CRD.

	Transient	Category	Cycles
8.	Scram with inoperative buffer	Normal/upset	10
9.	Scram with stuck control blade	Normal/upset	1

	Transient	Category	Cycles
10.	Operating basis earthquake (OBE)*	Normal/upset	10
11.	Safe shutdown earthquake (SSE)**	Faulted	1
12.	Control rod ejection accident	Faulted	1

All ASME Section III, Class 1 components of the CRD have been evaluated according to the requirements of the Code. The capacity of the CRD system to withstand emergency and faulted conditions is verified by tests rather than analysis.

3.9B.1.1.2 Control Rod Drive Housing and In-core Housing Transients

The number of transients, their cycles, and classification as considered in the design and fatigue analysis of the CRD housing and in-core housing are as follows:

	Transient	Category	Cycles
1.	Normal startup and shutdown	Normal/upset	120
2.	Vessel pressure tests	Normal/upset	130
3.	Vessel overpressure tests	Normal/upset	10
4.	Interruption of feedwater flow	Normal/upset	80
5.	Scrams	Normal/upset	200
6.	OBE	Normal/upset	10
7.	SSE	Faulted	1
8.	Stuck rod scram	Normal/upset	1
9.	Scram with inoperative buffer	Normal/upset	10

* The frequency of occurrence of this transient would indicate emergency category. However, for conservatism the OBE condition is analyzed as an upset condition. Ten peak OBE cycles are postulated.

** SSE is a faulted condition; however, in the stress analysis it was treated as emergency with lower stress limits.

3.9B.1.1.3 Hydraulic Control Unit Transients

The transients used in the design and analysis of the HCU and its components are:

	Transient	Category	Cycles
1.	Reactor startup/shutdown	Normal/upset	120
2.	Scram tests	Normal/upset	300
3.	Operational scrams	Normal/upset	300
4.	Jog cycles	Normal/upset	30,000
5.	Scram with stuck scram discharge valve	Normal/upset	1
6.	OBE	Normal/upset	10
7.	SSE	Faulted	1

3.9B.1.1.4 Core Support and Reactor Internals Transients

The cycles listed in Table 3.9B-1 were considered in the design and fatigue analysis for the reactor internals.

3.9B.1.1.5 Main Steam System Transients

See Section 3.9A.1.

3.9B.1.1.6 Recirculation System Transients

The following transients are considered in the stress analysis of the recirculation piping:

	Transient	Category	Cycles
1.	Startup	Normal	120
2.	Turbine roll and increase to power	Normal	120
3.	Loss of feedwater heater	Upset	10
4.	Partial feedwater heater bypass	Upset	70
5.	Scrams	Upset	180
6.	Shutdown	Normal	111
7.	Loss of feedwater pumps isolation valves closed	Upset	10

	Transient	Category	Cycles
8.	Single SRV blowdown	Upset	8
9.	Hydrotest	Test	130
10.	OBE	Upset	50

3.9B.1.1.7 Reactor Assembly Transients

The reactor assembly includes the RPV, support skirt, shroud support, and shroud plate. The cycles listed in Table 3.9B-1 were specified in the reactor assembly design and fatigue analysis.

3.9B.1.1.8 Main Steam Isolation Valve Transients

The transients considered in the analysis of the MSIVs are as follows:

	Transient	Category	Cycles
1.	Heatup from 70°F to 552°F (100°F/hr)	Normal/upset	300
2.	Cooldown from 552°F to 70°F (100°F/hr)	Normal/upset	300
3.	Small temperature changes of 29°F (either increase or decrease) at any temperature between 70°F and 552°F	Normal/upset	600
4.	Temperature changes of 50°F (either increase or decrease) at any temperature between 70°F and 552°F	Normal/upset	200
5.	Loss of feedwater pumps in which the temperature jumps from 552°F to 573°F in 3 sec, drops down to 525°F in 9 min, rises to 573°F in 6 min, drops down to 485°F in 7 min, rises to 573°F again in 8 min, and drops down to 485°F in 7 min	-	10
6.	Turbine bypass, single relief or safety valve blowdown in which the	-	8

	Transient	Category	Cycles
	temperature drops from 552°F to 375°F in 10 min		
7.	Reactor overpressure with delay scram in which the temperature rises from 552°F to 586°F in 2 sec, and the pressure rises from 1050 to 1375 psig immediately followed by cooling transient in which the temperature drops from 586°F to 561°F in 30 sec. The pressure drops down to 1125 psig.	Emergency	1
8.	Automatic blowdown in which the temperature changes from 552°F to 375°F in 3.3 min immediately followed by a change from 375°F to 259°F in 19 min (300°F/hr)	Emergency	1
9.	Pipe rupture and blowdown in which the temperature changes from 552°F to 259°F in 15 sec	Faulted	1
10.	Installed hydrotests at 100°F		
	a. 1250 psig	Testing	130
	b. 1575 psig	Testing	3

3.9B.1.1.9 Safety/Relief Valve Transients

The transients considered in the analysis of the SRVs are as follows:

	Transient	Category	Cycles
1.	Preoperational and in-service testing (100°F/hr)	Normal/upset	150
2.	Startup (100°F/hr) and pressure increase (0 psig to 1,000 psig)	Normal/upset	120
3.	Shutdown (100°F/hr, pressure decrease to 0 psig)	Normal/upset	120

	Transient	Category	Cycles
4.	Scram	Normal/upset	180
5.	System pressure and temperature decay from 1,000 psig and 546°F to 35 psig and 281°F within 15 sec	Emergency/ faulted	1
6.	System temperature change from 546° to 375°F within 3.3 min and from 375° to 281°F at 300°F/hr. Pressure change from 1,000 to 35 psig.	Emergency/ faulted	1
7.	System temperature change from 546° to 375°F within 10 min and from 375° to 281°F at 100°F/hr. Pressure change from 1,000 to 35 psig.	Emergency/ faulted	8
8.	System temperature change from 546° to 583°F within 2 sec, from 583° to 538°F within 30 sec, and from 538° to 400°F and return to 546°F at 100°F/hr. Pressure change from 1,000 to 1,350 psig, then to 240 psig and return to 1,000 psig.	Emergency/ faulted	1
9.	System temperature changes, greater than 30°F, from 561° to 500°F within 7 min and from 500° to 400°F and return to normal operating temperature of 546°F at 100°F/hr. Pressure change from 1,000 to 1,180 to 240 psig and return to normal operating of 1,000 psig.	Emergency/ faulted	10

Paragraph NB-3552 of ASME Section III excludes various transients and provides a means for combining those that are not excluded. Review and approval of the equipment supplier's certified calculations provides assurance of proper accounting of the specified transients.

The SRVs used for Unit 2 are those normally supplied for BWR 6 projects. The stress (including fatigue) analysis of this SRV model is performed on the basis of BWR 6 plant conditions. These include transients that are anticipated to be more numerous and more severe than the transients shown above for Unit 2. The SRV is, therefore, qualified for the above transients.

3.9B.1.1.10 Recirculation Flow Control Valve Transients

The following pressure and temperature transients were considered in the design of the recirculation system flow control valve (FCV):

	Transient	Category	Cycles
1.	Startup (100°F/hr heatup rate 70°F to design temperature)	Normal/upset	300
2.	Small temperature step changes (29°F step)	Normal/upset	600
3.	50°F step changes	Normal/upset	200
4.	SRV blowdowns (single valve) 522° to 375°F in 10 min)	Normal/upset	8
5.	Safety valve transient (110% of design pressure)	Normal/upset	1
6.	Installed hydrostatic tests		
	a. 1,300 psig	Testing	130
	b. 1,670 psig	Testing	3
7.	Automatic blowdown 552° to 375°F in 3.3 min, followed by a change from 375° to 281°F in 19 min	Emergency	1
8.	Improper start of pump in cold loop over a period of 15 sec	Emergency	1

3.9B.1.1.11 Recirculation Pump Transients

The following pressure transients were considered in the design of the recirculation pumps:

	Transient	Category	Cycles
1.	Startup (100°F/hr heatup rate 70°F to design temperature)	Normal/upset	300
2.	Small temperature changes (29°F step)	Normal/upset	600
3.	50°F step changes	Normal/upset	200
4.	SRV blowdowns (single valve) (552° to 375°F in 10 min)	Normal/upset	8
5.	Safety valve transient (110% of design pressure)	Normal/upset	1
6.	Installed hydrotests		
	a. 1,300 psig	Testing	130
	b. 1,670 psig	Testing	3
7.	Automatic blowdown (552° to 375°F in 3.3 min and 375° to 281°F in 19 min)	Emergency	1
8.	Improper start of pump in cold loop (100° to 552°F over a period of 15 sec)	Emergency	1
9.	Cooling transient, 552° to 281°F in 15 sec	Faulted	1

3.9B.1.1.12 Recirculation Gate Valve Transients

The following transients are considered in the design of the recirculation gate valves:

	Transient	Cycles
1.	50° to 575° to 50°F at a rate of 100°F/hr	300
2.	±29°F between limits of 50° and 575°F, instantaneous	600
3.	±50°F between limits of 50° and 546°F, instantaneous	200
4.	552° to 375°F, instantaneous	9
5.	546° to 281°F, instantaneous	1
6.	130° to 546°F, instantaneous	1

	Transient	Cycles
7.	110% of design pressure at 575°F	1
8.	1,300 psi at 100°F installed hydrostatic test	130
9.	1,670 psi at 100°F installed hydrostatic test	3

3.9B.1.2 Computer Programs Used in Analysis

The following sections discuss computer programs used in the analysis of specific components. (Computer programs were not used in the analysis of all components, thus, not all components are listed.) The NSSS programs can be divided into two categories, GE programs and vendor programs.

GE Programs

Verification of the following GE programs has been performed in accordance with the requirements of 10CFR50 Appendix B. Evidence of the verification of input, output, and methodology is documented in GE Design Record Files.

SEISM		FTFLG01
MASS		ANSYS
SNAP (MULTISHELL)		POSUM
HEATER	PDA	BILRD
ANSI7	EZPYP	DYSEA
PISYS	SAP4G	SPECA
		COSMOS/M

Vendor Programs

Verification of the following vendor (CB&I) programs is assured by contractual requirements between GE and the vendor. In accordance with the requirements, the QA procedure of these proprietary programs used in the design of N-stamped equipment is in full compliance with 10CFR50 Appendix B.

711 GENOZZ	9-28 TGRV	953
948 NAPALM	962 E0962A	1666
1027	984	1684
846	992 GASP	E1702A
781 KALNINS	1037 DUNHAM'S	955 MESHPLOT
979 ASFAST	1335	1028
7-66 TEMAPR	1606 & 1657 HAP	1038
7-67 PRINCESS	1635	

3.9B.1.2.1 Reactor Pressure Vessel and Internals

Reactor Pressure Vessel

CB&I Program 7-11 - GENOZZ The GENOZZ computer program is used to proportion barrel and double taper-type nozzles to comply with the specifications of ASME Section III and contract documents. The program will either design such a configuration or analyze the configuration input into it. If the input configuration does not comply with the specifications, the program modifies the design and redesigns it to yield an acceptable result.

CB&I Program 9-48 - NAPALM The basis for the program NAPALM (Nozzle Analysis Program - All Loads Mechanical) is to analyze nozzles for mechanical loads and find the maximum stress intensity and location. The program analyzes at specified locations from the point of application of the mechanical loads. At each location, the program calculates the maximum stress intensity for both the inside and outside surfaces of the nozzle, and its angular location around the circumference of the nozzle from the reference location. The principal stresses are also printed. The stresses resulting from each component of loading (bending, axial, shear, and torsion) are printed, as well as the loads that caused these stresses.

CB&I Program 1027 This program is a computerized version of the analysis method contained in the Welding Research Council Bulletin No. 107, August 1965. Part of the program provides for the determination of the shell stress intensities (S) at each of four cardinal points at both the upper and lower shell plate surfaces (ordinarily considered outside and inside surfaces) around the perimeter of a loaded attachment on a cylindrical or spherical vessel. With the determination of each S, the components of that S (two normal stresses, σ_x and σ_z , and one shear stress τ) are also determined. This program provides the same information as the manual calculation, and the input data are essentially the geometry of the vessel and attachment.

CB&I Program 846 This program computes the required thickness of a hemispherical head with a large number of circular parallel penetrations by means of the area replacement method in accordance with the ASME Section III. In cases where the penetration has a counterbore, the thickness is determined so that the counterbore does not penetrate the outside surface of the head.

CB&I Program 781 - KALNINS This program is a thin elastic shell program for shells of revolution. The basic method of analysis was developed and published by Dr. A. Kalnins of Lehigh University[1]. Extensive revisions and improvements have been made by Dr. J. Endicott to yield the CB&I version of this program. The program is used to establish the shell influence coefficient and to perform detail stress analysis of the vessel.

The stresses and the deformations of the vessel can be computed for any combination of the following axisymmetric loading:

1. Preload condition.

2. Internal pressure.

3. Thermal load.

CB&I Program 979 - ASFAST The ASFAST program performs stress analysis of axisymmetric, bolted closure flanges between head and cylindrical shell.

CB&I Program 7-66 - TEMAPR This program reduces any arbitrary temperature gradient through the wall thickness to an equivalent linear gradient. The resulting equivalent gradient has the same average temperature and the same temperature-moment as the given temperature distribution. Input consists of plate thickness and actual temperature distribution. Output contains average temperature and total gradient through the wall thickness. The program is written in FORTRAN IV language.

CB&I Program 7-67 - PRINCESS The PRINCESS program calculates the maximum alternating stress amplitudes from a series of stress values by the method in ASME Section III.

CB&I Program 9-28 - TGRV The TGRV program is used to calculate temperature distributions in structures or vessels. Although it is primarily a program for solving the heat conduction equations, some provisions have been made for including radiation and convection effects at the surfaces of the vessel.

The TGRV program is a greatly modified version of the TIGER heat transfer program written about 1958 at Knolls Atomic Power Laboratory by A. P. Bray. There have been many versions of TIGER in existence including TIGER II, TIGER II B, TIGER IV, and TIGER V, in addition to TGRV.

The program uses an electrical network analogy to obtain the temperature distribution of any given system as a function of time. The finite difference representation of the three-dimensional equations of heat transfer are repeatedly solved for small time increments and continually summed. Linear mathematics are used to solve the mesh network for every time interval. Included in the analysis are the three basic forms of heat transfer, i.e., conduction, radiation, and convection, as well as internal heat generation.

Given any odd-shaped structure, which is represented by a three-dimensional field, its geometry and physical properties, boundary conditions, and internal heat generation rates, TGRV calculates and gives as output the steady-state or transient temperature distributions in the structure as a function of time.

CB&I Program 962 - E0962A Program E0962A is one of a group of programs (E0953A, E1606A, E0962A, E0992N, E1037N, and E0984N) used together to determine the temperature distribution and stresses in pressure vessel components by the finite element method.

Program E0962A is primarily a plotting program. Using the nodal temperatures calculated by program E1606A or program E0928A, and the node and element cards for the finite element model, it calculates and plots lines of constant temperature (isotherms). These isotherm plots are used as part of the stress report to present the results of the thermal analysis. They are also very useful in determining at which points in time the thermal stresses should be determined.

In addition to its plotting capability, the program can determine the temperatures of some of the nodal points by interpolation. This feature of the program is intended primarily for use with the compatible TGRV and finite element models generated by program E0953A.

CB&I Program 984 Program 984 is used to calculate the stress intensity of the stress differences, on a component level, between two different stress conditions. The calculation of the stress intensity of stress component differences (the range of stress intensity) is required by ASME Section III.

CB&I Program 992 - GASP The GASP program, originated by Prof. E. L. Wilson of the University of California at Berkeley, uses the finite element method to determine the stresses and displacements of plane or axisymmetric structures of arbitrary geometry and is written in FORTRAN IV[2]. The structures may have arbitrary geometry and linear or nonlinear material properties. The loadings may be thermal, mechanical, accelerational, or a combination of these.

The structure to be analyzed is broken up into a finite number of discrete elements or finite elements which are interconnected at a finite number of nodal points or nodes. The actual loads on the structure are simulated by statically equivalent loads acting at the appropriate nodes. The basic input to the program consists of the geometry of the stress model and the boundary conditions. The program then gives the stress components at the center of each element and the displacements at the nodes, consistent with the prescribed boundary conditions.

CB&I Program 1037 - DUNHAM'S DUNHAM'S program is a finite ring element stress analysis program. It determines the stresses and displacements of axisymmetric structures of arbitrary geometry subjected to either axisymmetric loads or nonaxisymmetric loads represented by Fourier series. This program is similar to the GASP program (CB&I 992). The major differences are that DUNHAM'S can handle nonaxisymmetric loads (which requires that each node have 3 degrees of freedom) and the material properties for

DUNHAM'S must be constant. As in GASP, the loadings may be thermal, mechanical, and accelerational.

<u>CB&I Program 1335</u> To obtain stresses in the shroud support, the baffle plate must be considered as a continuous circular plate. This program makes this modification and allows the baffle plate to be included in CB&I Program 781 as two isotropic parts and an orthotropic portion at the middle (where the diffuser holes are located).

<u>CB&I Programs 1606 and 1657 - HAP</u> The HAP program is an axisymmetric nonlinear heat analysis program. It is a finite element program and is used to determine nodal temperatures in a two-dimensional or axisymmetric body subjected to transient disturbances. Programs 1606 and 1657 are identical except that 1606 has a larger storage area allocated and can thus be used to solve larger problems. The model for Program 1606 is compatible with CB&I stress Programs 992 and 1037.

<u>CB&I Program 1635</u> Program 1635 offers three features to aid the stress analyst in preparing a stress report:

1. Generates punched card input for Program 7-67 (PRINCESS) from the stress output of Program 781 (KALNINS).

2. Writes a stress table in a format that can be incorporated into a final stress report.

3. Has the option to remove through-wall thermal bending stress and report these results in a stress table similar to the one mentioned in Item 2.

<u>CB&I Program 953</u> The program is a general purpose program which does the following:

1. Prepares input cards for the thermal model.

2. Prepares the node and element cards for the finite element model.

3. Sets up the model in such a way that the nodal points in the TGRV model correspond to points in the finite element model. They have the same number so that there is no possibility of confusion in transferring temperature data from one program to the other.

<u>CB&I Program 1666</u> This program is primarily written to calculate the temperature differences at selected critical sections of the nuclear reactor vessel components at different time points of thermal transients during its life of operation and to list them all in a tabular form. Since there is no involved calculation applicable particularly to nuclear components, this program can be used with any other kind of model that is subjected to thermal

transients over a period of time. This program helps ascertain the time points in thermal transients when the thermal stresses may be critical.

CB&I Program 1684 This program, an expansion of Program 984, is written to expedite the fatigue analysis of nuclear reactor components as required by ASME Section III. The features of this program allow the user to easily perform the complete secondary stress and fatigue evaluation including partial fatigue usage calculation of a component in one run. An additional option allows the user to completely document the input stress values in a format suitable for a stress analysis report. The program is written to allow for a minimum amount of data handling by the user once the initial deck is established.

CB&I Program E1702A This program evaluates the stress-intensity factor K_I due to pressure, temperature, and mechanical load stresses for a number of different stress conditions (times) and at a number of different locations (elements). It then calculates the maximum reference temperature nil ductility transition (RT_{NDT}) the actual material can have based on a 1/4T flaw size and compares it with the ordered RT_{NDT}. If the ordered RT_{NDT} is larger than the maximum RT_{NDT}, the maximum allowable flaw size is calculated. The rules of ASME Code Appendix G are used except that Welding Research Council (WRC) 175 can be used to calculate K_I due to pressure in a nozzle-to-shell junction.

For a more thorough description of the fracture problem, see WRC Bulletin No. 175[3].

CB&I Program 955 - MESHPLOT This program plots input data used for finite element analysis. The program plots the finite element mesh in one of three ways: without labels, with node labels, or with element labels. The output consists of a listing and a plot. The listing gives all node points with their coordinates and all elements with their node points. The plot is a finite element model with the requested labels.

CB&I Program 1028 This program calculates the necessary form factors for the nodes of the model that simulates heat transfer by radiation. Inputs are shape and dimensions of the head-to-skirt knuckle junction. The program is limited to junctions with a toroidal knuckle part.

CB&I Program 1038 This program calculates the loads required to satisfy the compatibility between the shroud baffle plate and the jet pump adaptors for a GE BWR vessel.

Vessel Internals

Fuel Support Loads Program - SEISM SEISM computes the vertical fuel support loads using the component element methods in dynamics[4].

Other Programs The following programs are also used in the analysis of core support structures and other safety-related reactor internals: MASS, SNAP (MULTISHELL), and HEATER. These programs are described in detail in Section 4.1.

3.9B.1.2.2 Piping

Piping Analysis Program - PISYS PISYS is a specialized computer code for piping load calculations. It utilizes selected stiffness matrices representing standard piping components, which are assembled to form a finite element model of a piping system. The technique relies on dividing the pipe model into several discrete substructures, called pipe elements, which are connected to each other via nodes called pipe joints. It is through these joints that the model interacts with the environment, and loading of the structure becomes possible. PISYS is based on linear classical elasticity in which the resultant deformation and stresses are proportional to the loading, and the superposition of loading is valid.

PISYS has a full range of static and dynamic analysis options that include distributed weight, thermal expansion, differential support motion modal extraction, response spectra, and time-history analysis by modal or direct integration. The PISYS program has been benchmarked against five NRC piping models for the option-of-response-spectrum analysis, and the results are documented in a report to the NRC, NEDO-24210[5].

Component Analysis - ANSI 7 The ANSI 7 program determines stress and accumulative usage factors in accordance with Subarticle NB-3600 of ASME Section III. The program was written to perform stress analysis in accordance with the ASME sample problem, and has been verified by reproducing the results of the sample problem analysis.

SUPERPIPE Computer Program The SUPERPIPE computer program is described in Appendix 3B.

Piping Dynamic Analysis Program - PDA The pipe whip analysis was performed using the PDA program to determine the response of a pipe subjected to the thrust force occurring after a pipe break. The program treats the situation in terms of a generic pipe break configuration, which involves a straight, uniform pipe fixed at one end and subjected to a time-dependent thrust force at the other end. A typical restraint used to reduce the resulting deformation is also included at a location between the two ends. Nonlinear and time-dependent stress-strain relations are used to model the pipe and the restraint. Similar to the popular elastic hinge concept, bending of the pipe is assumed to occur only at the fixed end and at the location supported by the restraint.

Shear deformation is neglected. The pipe bending moment-deflection (or rotation) relation used for these locations is obtained from a static nonlinear cantilever beam analysis.

Using moment-rotation relations, nonlinear equations of motion are formulated using energy considerations and the equations are numerically integrated in small time steps to yield the time-history of the pipe motion.

Piping Analysis Program - EZPYP EZPYP links the ANSI-7 and SAP programs. The EZPYP program can be used to run several SAP cases by making user-specified changes to a basic SAP pipe model. By controlling files and SAP runs, the EZPYP program gives the analyst the capability to perform a complete piping analysis in one computer run.

3.9B.1.2.3 Recirculation Pump

The ANSYS code is used in the analysis of the recirculation pump casing for various thermal and mechanical loads during plant operating and postulated conditions.

In general, the finite element techniques are used to solve temperature distribution in heat transfer transient problems, and to perform stress analysis for various thermal and mechanical loadings by using the same finite element model representing the pump body. The output of these programs is in the form of temperature profiles, deflections, and stresses at the nodal points of the finite element idealization of the pump structure.

3.9B.1.2.4 Emergency Core Cooling System Pumps and Motors

Structural Analysis Program - SAP4G SAP4G is used to analyze the structural and functional integrity of the ECCS pump and motor systems. This is a general structural analysis program for static and dynamic analysis of linear elastic complex structures. The finite element displacement method is used to solve the displacements and stresses of each element of the structure. The structure can be composed of an unlimited number of three-dimensional truss, beam, plate, shell, solid, plate strain-plane stress and spring elements that are axisymmetric. The program can treat thermal and various forms of mechanical loading. The dynamic analysis includes mode superposition, time-history, and response spectrum analysis. Seismic loading and time-dependent pressure can be treated. The program is versatile and efficient in analyzing large and complex structural systems. The output contains displacements of each nodal point as well as stresses at the surface of each element.

Effects of Flange Joint Connections - FTFLG01 The flange joints connecting the pump bowl castings are analyzed using FTFLG01. This program uses the local forces and moments determined by SAP4G to perform flat flange calculations in accordance with the rules set forth in Appendix II and ASME Boiler and Pressure Vessel Code Section III.

Structural Analysis of Discharge Head - ANSYS ANSYS is used to analyze the pump discharge head flange and bolting taking into account the prying action developed by the flat face contact surface. The program is described in detail in Section 4.1.

Beam Element Data Processing - POSUM POSUM is a computer code designed to process SAP-generated beam element data for pump or heat exchanger models. The purpose is to determine the load combination that would produce the maximum stress in a selected beam element. It is intended to be used on RHR heat exchangers with four nozzles or ECCS pumps with two nozzles.

3.9B.1.2.5 RHR Heat Exchangers

Structural Analysis Program - SAP4G SAP4G is used to evaluate the structural and functional integrity of the RHR heat exchangers. A description of this program is provided in Section 3.9B.1.2.4.

Calculation of Shell Attachment Parameters and Coefficients/BILRD BILRD is used to calculate the shell attachment parameters and coefficients used in the stress analysis of the support-to-shell junction. The method, in accordance with WRC Bulletin No. 107, is implemented in BILRD to calculate local membrane stress due to the support reaction loads on the heat exchanger shell.

Beam Element Data Processing/POSUM POSUM is used to process SAP-generated beam element data. The description of this program is provided in Section 3.9B.1.2.4.

3.9B.1.2.6 Dynamic Load Analysis

Dynamic Analysis Program - DYSEA DYSEA simulates a beam model in the annulus pressurization dynamic analysis. A description of DYSEA is provided in Section 4.1. DYSEA employs a preprocessor program named GEAPL. GEAPL converts pressure time-histories into time-varying loads and forcing functions for DYSEA. The overall resultant forces and moments time-histories at specified points of resolution can also be obtained from GEAPL.

Acceleration Response Spectrum Program - SPECA SPECA generates acceleration response spectra for an arbitrary input time-history of piece-wise linear accelerations, i.e., to compute maximum acceleration responses for a series of single degree-of-freedom systems subjected to the same input. It can accept acceleration time-histories from a random file. It also can generate the broadened/enveloped spectra when the spectral points are generated equally spaced on a logarithmic scale axis of period/frequency. This program is also used in seismic and SRV transient analyses.

3.9B.1.3 Experimental Stress Analysis

The following sections list those NSSS components for which experimental stress analysis was used and provide a discussion of the analysis.

3.9B.1.3.1 Experimental Stress Analysis of Piping Components

The following components have been tested to verify their design adequacy:

1. Snubbers

2. Pipe whip restraints

Descriptions of the snubbers and pipe whip restraint tests are contained in Sections 3.9B.3.4 and 3.6B.2.2.2, respectively.

3.9B.1.3.2 Orificed Fuel Support, Vertical and Horizontal Load Tests

A series of horizontal and vertical load tests were performed on the orificed fuel support (OFS) in order to verify the design. Results from these tests indicate that the seismic and hydrodynamic loading of the OFS are below allowable load limits, with a safety margin of at least 2.0 for normal, upset and faulted conditions. (The allowable load limits were arrived at by applying a 0.75 quality factor to the ASME Code allowables of 0.44 x test load for upset and 0.80 x test load for faulted condition.)

3.9B.1.4 Considerations for the Evaluation of Faulted Conditions

Each item of Category I equipment is evaluated for faulted loading conditions. In all cases, calculated stresses are within the allowable limits. This section provides examples of the treatment of faulted conditions for the major components on a component-by-component basis. Additional discussion of faulted analysis is found in Sections 3.9B.3 and 3.9B.5, and Table 3.9B-2.

Sections 3.9B.2.2 and 3.7B discuss the treatment of dynamic loads resulting from the postulated faulted condition. Section 3.9B.2.5 discusses the dynamic analysis of loads on reactor internals resulting from blowdown. Deformations under faulted conditions have been evaluated in critical areas, and no cases have been identified where design limits, such as clearance limits, are exceeded.

3.9B.1.4.1 Control Rod Drive System Components

<u>Control Rod Drives</u>

The major CRD components that have been analyzed for the faulted conditions are the ring flange, main flange, and indicator tube. The maximum stresses for these components and for various plant operating conditions including the faulted condition are given in Table 3.9B-2a.

The ASME Section III Code components of the CRD have been analyzed for conditions in Section 3.9B.1.1.1. The loads and stresses are within the elastic limits of the material.

The design adequacy of noncode components of the CRD has been verified by extensive testing programs on both (Code and non-Code) component parts, specially instrumented prototype drives, and production drives. The testing has included postulated abnormal events as well as the service life cycle listed in Section 3.9B.1.1.1.

Hydraulic Control Unit

The seismic and hydrodynamic loads adequacy of the HCU is demonstrated by tests. Section 3.9B.2.2.2 discusses the dynamic qualification of the HCU.

3.9B.1.4.2 Standard Reactor Internal Components

Control Rod Guide Tube

The maximum calculated stress on the control rod guide tube occurs in its base during the faulted condition. The faulted limit is the lesser of 2.4 S or 0.7 S at the design temperature in accordance with ASME Section III, Table F 1322-1; according to ASME Section III, Table I-1.2, S_u = 57,500 psi and S_m = 16,000 psi at 575°F. The analysis and limiting stresses for various plant operating conditions are given in Table 3.9B-2b.

In-core Housing

The maximum calculated stress on the in-core housing occurs at the outer surface of the vessel penetration during the faulted condition. The allowable stress for the elastic analysis used is 2.4 S_m = 40,000 psi. The analysis for various operating conditions is summarized in Table 3.9B-2c which shows that the calculated stresses are within the allowables.

Jet Pump

The elastic analysis for the jet pump faulted conditions shows that the maximum stress is due to impulse loading of the diffuser during a pipe rupture and blowdown. The maximum allowable for this condition, in accordance with ASME Section III Appendix F is 3.6 S_m or 60,000 psi. Table 3.9B-2d summarizes the results of the stress analysis.

LPCI Coupling The calculated stress at the highest stressed location is bounded by the allowable stress which is 3.6 S_m. Table 3.9B-2e summarizes the criteria, loading conditions, the calculated and allowable stresses.

Orificed Fuel Support

OFS is analyzed for the faulted condition. The analysis and testing are described in Section 3.9B.1.3.2. Results of the analysis are provided in Table 3.9B-2f.

CRD Housing

The CRD housing is analyzed for the faulted condition, considering SSE and hydrodynamic loads.

Table 3.9B-2g shows that the calculated stress values for the highly stressed areas of the CRD housing are within the allowable limits.

3.9B.1.4.3 Reactor Pressure Vessel Assembly

For the faulted condition, the RPV and the shroud support were evaluated using elastic analysis methods. For the support skirt and shroud support, an elastic analysis was performed, and buckling was evaluated for the compressive load.

The support skirt is designed in compliance with ASME III requirements for the Class 1 pressure-retaining portion of the vessel.

A generic BWR 4/5 study was conducted on the Limerick 1 and 2 cylindrical support skirt, which has the smallest ratio of thickness to radius. The study examined the skirt buckling under axial compression, hoop stress, and transverse shear (Section 3.9.6-13) and showed that, in each case, the critical buckling stress was much greater than the yield stress.

Since this study showed that inelastic stability limits the skirt integrity, the permissible compressive load is limited to 90 percent of the load which produces yield stress, divided by a safety factor of 1.125 for faulted conditions to account for the effects of fabrication or any eccentricity.

For faulted conditions, the RPV support skirt design can meet the allowable limit of two-thirds of the critical buckling stress in accordance with ASME Boiler and Pressure Vessel Code, Section III, paragraph F-1370(c). An analysis of the RPV support skirt shows that the design has the capability to meet this allowable stress at temperature.

Table 3.9B-2h lists the calculated and allowable stresses for the various load combinations.

3.9B.1.4.4 Core Support Structure

The core support structure is evaluated for the faulted condition. The bases for determining the faulted loads due to seismic and hydrodynamic events are discussed in Sections 3.7B and 3.9B.5, respectively. The calculated stresses and allowables are summarized in Table 3.9B-2i.

3.9B.1.4.5 Recirculation Gate, Safety/Relief Valves, and Main Steam Isolation Valves

Tables 3.9B-2j, 3.9B-2k, and 3.9B-2z provide a summary of the analysis of the SRV, recirculation gate valve, and MSIV, respectively.

Standard design rules, as defined in ASME Section III, are used in the analysis of pressure boundary components of Category I valves. Conventional, elastic stress analysis is used to evaluate components not defined in the Code. The Code allowable stresses are applied to determine acceptability of the structure under applicable loading conditions including the faulted condition.

3.9B.1.4.6 Recirculation System Flow Control Valve

The recirculation system FCV is analyzed for faulted conditions using the elastic analysis methods from ASME Section III. The analysis is summarized in Table 3.9B-2l.

3.9B.1.4.7 Recirculation Piping

For recirculation system piping, elastic analysis methods are used for evaluating faulted loading conditions. The equivalent allowable stresses using elastic techniques are obtained from the ASME Section III, Appendix F, and these are above elastic limits. Additional information on the recirculation piping is in Table 3.9B-2m.

3.9B.1.4.8 Nuclear Steam Supply System Pumps, Heat Exchanger, and Turbine

The recirculation, ECCS, RCIC, and SLC pumps, RHR heat exchangers, and RCIC turbine have been analyzed for the faulted loading conditions identified in Section 3.9B.1.1. In all cases, stresses were within the elastic limits. The analytical methods, stress limits, and allowable stresses are summarized in Table 3.9B-2 in the respective equipment table.

3.9B.1.4.9 Control Rod Drive Housing Supports

The calculated stresses and the allowable stress limits for faulted conditions for the CRD housing supports are shown in Table 3.9B-2y.

3.9B.1.4.10 Fuel Storage Racks

Examples of the calculated stresses and stress limits for the faulted conditions for the new fuel storage racks are shown in Table 3.9B-2n.

3.9B.1.4.11 Fuel Assembly (Including Channels)

GE BWR fuel assembly design bases, analytical methods, and evaluation results, including those applicable to the faulted conditions, are contained in NEDE-24011[6], NEDE-24011-US[7], and NEDE-21175-3-P[13]. The acceleration profiles and fuel lift gap are summarized in Table 3.9B-2o.

3.9B.1.4.12 Refueling Equipment

Refueling equipment and servicing equipment that are important to safety are classified as essential components in accordance with the requirements of 10CFR50 Appendix A. This equipment and other equipment whose failure would degrade an essential component are defined in Section 3.9B.1 and are classified as Category I. These components are subjected to an elastic dynamic finite element analysis to generate loadings. This analysis utilizes appropriate seismic floor response spectra and combines loads at frequencies up to 33 Hz for seismic and up to 80 Hz for hydrodynamic loads in three directions. Imposed stresses are generated and combined for normal, upset, and faulted conditions. Stresses are compared, depending on the specific safety class of the equipment, to those allowed by Industrial Codes, ASME, ANSI, AISC, or Industrial Standards allowables.

The calculated stresses and allowable limits for the faulted condition for the fuel storage rack, fuel preparation machine, and refueling platform are documented in Table 3.9B-2n.

3.9B.2 Dynamic Testing and Analysis

3.9B.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

The test program is divided into three phases: piping vibration, thermal expansion, and dynamic effects.

3.9B.2.1.1 Piping Vibration

Preoperational and Startup Vibration Testing of Recirculation Piping

The purpose of the preoperational vibration test phase is to verify that operating vibrations in the recirculation piping are within acceptable limits. This phase of the test uses visual observation to supplement remote measurements. If, during steady-state operation, visual observation indicates that vibration is significant, measurements are made with a hand-held vibrograph. Visual observations and manual and remote measurements are made during the following steady-state conditions:

1. Recirculation pumps at minimum flow.

2. Recirculation pumps at 50 percent of rated flow.

3. Recirculation pumps at 75 percent of rated flow.

4. Recirculation pumps at 100 percent of rated flow.

5. RHR suction piping at 100 percent of rated flow in the shutdown cooling mode.

Preoperational Vibration Testing of Small Attached Piping

During visual observation of test conditions 1 through 5, special attention will be given to small attached piping and instrument connections to ensure that they are not in resonance with the recirculation pump motors or flow-induced vibrations. If the operating vibrations acceptance criteria are not met, corrective action such as modification of supports will be undertaken.

Operating Transient Loads on Recirculation Piping

The purpose of the operating transient test phase is to verify that pipe stresses are within code limits. The amplitude of displacements and number of cycles per transient of the recirculation piping are measured and the displacements compared with acceptance criteria. The deflections are correlated with stresses to verify that the pipe stresses remain within Code limits. Remote vibration and deflection measurements are taken during the following transients:

1. Recirculation pump starts.

2. Recirculation pump trip at 100 percent of rated flow.

3. Turbine stop valve closure at 100 percent power.

4. Manual discharge of each SRV at 1,000 psig and at planned transient tests that result in SRV discharge.

3.9B.2.1.2 Thermal Expansion Testing of Recirculation Piping

A thermal expansion, preoperational and startup testing program, performed through the use of potentiometer sensors, has been established to verify that normal thermal movement occurs in the piping systems. The main purpose of this program is to ensure the following:

1. The piping system during system heatup and cooldown is free to expand, contract, and move without unplanned obstruction or restraint in the x, y, and z directions.

2. The piping system is working in a manner consistent with the assumption of the NSSS stress analysis.

3. There is adequate agreement between calculated values of displacements and measured value of displacement.

4. There is consistency and repeatability in thermal displacements during heatup and cooldown of the NSSS systems.

Limits of thermal expansion displacements are established prior to the start of piping testing to which the actual measured displacements are compared, to determine acceptability of the actual motion. If the measured displacement does not vary from the acceptance limits values by more than the specified tolerances, the piping system is responding in a manner consistent with predictions and is, therefore, acceptable. Two levels of displacement limits are established to check the systems as explained in Section 3.9B.2.1.4.

3.9B.2.1.3 Dynamic Effects Testing of Recirculation Piping

To verify that snubbers are adequately performing their intended function during plant operation, a program for dynamic testing, as a part of the normal startup operation testing, is planned. The main purpose of this program is to ensure the following:

1. The vibration levels from the various dynamic loadings during transient and steady-state conditions are below the predetermined acceptable limits.

2. Long-term fatigue failure does not occur due to underestimating the dynamic effects caused by cyclic loading during plant transient operations.

This dynamic testing is to account for the acoustic wave due to the SRV lifts (RV1), SRV load resulting from air clearing (RV2), and turbine stop valve closure (TSVC) load. The maximum stresses developed in the piping by the RV1, RV2, and TSVC transient analyses are used as a basis for establishing criteria that assure proper functioning of the snubbers. If field measurements exceed criteria limits, the snubbers are not operating properly. Sample production snubbers of each size (i.e., 10, 20, 50 kips) are qualified and tested for design and faulted condition loadings prior to shipment to the field. Snubbers are tested to allow free piping movements at low velocity. During plant startup, the snubbers are checked for proper settings.

The criteria for vibration displacements are based on assumed linear relationship between displacements, snubber loads, and the magnitude of applied loads for any function and response of system. Thus, the magnitude of limits of displacements, snubber loads, and nozzle loads are all proportional. Maximum displacements (Level 1 limits) are established to prevent the maximum stress in the piping systems from exceeding the normal and upset primary stress limits and/or the maximum snubber load from exceeding the maximum load to which the snubber has been tested.

Based on the above criteria, Level 1 displacement limits are established for all instrumented points in the piping system. These limits are compared with the field measured piping displacements. The method of acceptance is explained in the following section.

3.9B.2.1.4 Test Evaluation and Acceptance Criteria for Recirculation Piping

The piping response to test conditions is considered acceptable if the test results verify that the piping responded in a manner consistent with the predictions of the stress report and/or that piping stresses are within code limits (ASME Section III, Subarticle NB-3600). Acceptable deflection and acceleration limits are determined after the completion of piping system stress analysis and are provided in the startup test specifications. To ensure test data integrity and test safety, criteria have been established to facilitate assessment of the test while it is in progress. These criteria, designated Levels 1 and 2, are described in the following sections.

3.9B.2.1.4.1 Level 1 Criterion

Level 1 establishes the maximum limits for the level of pipe motion which, if exceeded, makes a test hold or termination mandatory. If the Level 1 limit is exceeded, the plant will be placed in a satisfactory hold condition, and the responsible piping design engineer will be advised. Following resolution, applicable tests must be repeated to verify that the requirements of the Level 1 limits are satisfied.

3.9B.2.1.4.2 Level 2 Criteria

If the Level 2 criteria are satisfied for both steady-state and operating transient vibrations, there will be no fatigue damage to the piping system due to steady-state vibration, and all operating transient vibrations are bounded by the values in the stress report.

Exceeding the Level 2 specified pipe motion requires that the responsible piping design engineer be advised. Plant operating and startup testing plans would not necessarily be altered. Investigations of the measurements, criteria, and calculations used to generate the pipe motion limits would be initiated. An acceptable resolution must be reached by all appropriate and involved parties, including the responsible piping design engineer. Detailed evaluation is needed to develop corrective action or show that the measurements are acceptable. Depending upon the nature of such resolution, the applicable tests may or may not have to be repeated.

3.9B.2.1.4.3 Acceptance Limits

For steady-state vibration, the piping peak stress due to vibration only (neglecting pressure) will not exceed 10,000 psi for Level 1 criteria and 5,000 psi for Level 2 criteria. These limits are below the piping material fatigue endurance limits defined in Design Fatigue Curves in Appendix I of the ASME Code for 10^6 cycles.

For operating transient vibration, the piping bending stress (zero to peak) will not exceed 1.2 S_m or pipe support loads will not exceed service Level D ratings for Level 1 criteria. The 1.2 S_m limit ensures that the total primary stress, including pressure and deadweight, will not exceed 1.8 S_m, the Code service Level B limit. Level 2 criteria are based on pipe stresses and support loads not exceeding design basis predictions. Design basis criteria require that operating transient stresses and loads not exceed any service Level B limits, including primary stress limits, fatigue usage factor limits, and allowable loads on snubbers.

3.9B.2.1.5 Corrective Actions for Recirculation Piping

During the course of the tests, the remote measurements are regularly checked to determine compliance with Level 1 criteria. If trends indicate that Level 1 criteria may be violated, the measurements are monitored at more frequent intervals. The test is interrupted as soon as Level 1 criteria are violated. As soon as possible after the test hold or termination, the following corrective actions are taken:

1. Installation Inspection A walkdown of the piping and suspension is made to identify any obstruction or improperly operating suspension components. If vibration exceeds criteria, the source of the excitation must be identified to determine if it is related to equipment failure. Action is taken to correct any discrepancies before repeating the test.

2. Instrumentation Inspection The instrumentation installation and calibration are checked, and any discrepancies are corrected. Additional instrumentation is added, if necessary.

3. Repeat Test If actions 1 and 2 identify discrepancies that could account for failure to meet the Level 1 criteria, the test will be repeated.

4. Resolution of Findings If the Level 1 criteria are violated on the repeat test, or no relevant discrepancies are identified as described in actions 1 and 2, the test results and criteria are reviewed to ensure that the test can be safely continued.

If the test measurements indicate failure to meet the Level 2 criteria, the following corrective actions are taken after completion of the test:

1. Installation Inspection A walkdown of the piping and suspension is made to identify any obstruction or improperly operating suspension components.

   Snubbers are installed at about the midpoint of the total range at operating temperature. Hangers are installed in their operating range between the hot and cold settings. If vibration exceeds limits, the source of the vibration is identified. Action is taken to correct any discrepancies.

2. Instrumentation Inspection The instrumentation installation and calibration are checked and any discrepancies are corrected.

3. Repeat Test If inspections described in actions 1 and 2 above identify a malfunction or discrepancy that could account for failure to comply with Level 2 criteria and appropriate corrective action has been taken, the test is repeated.

4. Documentation of Discrepancies If the test is not repeated, the discrepancies found under actions 1 and 2 are documented in the test evaluation report and correlated with the test condition. The test is not complete until the test results are reconciled with the acceptance criteria.

3.9B.2.1.6 Measurement Locations for Recirculation Piping

Remote shock and vibration measurements are made in the three orthogonal directions on the piping between the recirculation pump discharge and the first downstream valve. During preoperational testing prior to fuel load, visual inspection of the piping is made, and any visible vibration is measured with a hand-held instrument.

For each of the selected remote measurement locations, Level 1 and 2 deflection and acceleration limits are prescribed in the startup test specification.

3.9B.2.2 Seismic and Hydrodynamic Qualification of Safety-Related Mechanical Equipment

This section describes the criteria for dynamic qualification of safety-related mechanical equipment and the qualification testing and/or analysis applicable to this plant for all the major components on a component-by-component basis. In some cases, a module or assembly consisting of mechanical and electrical equipment is qualified as a unit (e.g., ECCS pumps). These modules are generally discussed in this section rather than in Sections 3.10B and 3.11. Dynamic load qualification testing for pumps and valves is also discussed in Section 3.9B.3.2. Electrical supporting equipment such as control consoles, cabinets, and panels that are part of the NSSS are discussed in Section 3.10B.

3.9B.2.2.1 Tests and Analysis Criteria and Methods

The ability of equipment to perform its safety-related function during and after the application of dynamic loads (earthquake) is demonstrated by tests and/or analysis. Selection of testing, analysis, or a combination of the two is determined by the type, size, shape, and complexity of the equipment being considered. When practical, equipment operability is demonstrated by testing. Otherwise, operability is demonstrated by mathematical analysis.

Equipment that is large, simple, and/or consumes large amounts of power is usually qualified by analysis or test to show that the loads, stresses, and deflections are less than the allowable maximum. Analysis testing is also used to show that there are no natural frequencies below 33 Hz for seismic loads and 60 Hz for hydrodynamic loads. If a lower natural frequency is discovered, dynamic tests may be conducted and, in conjunction with mathematical analysis, used to verify operability and structural integrity at the required dynamic input conditions. A similar dynamic test and/or analysis is performed for hydrodynamic loads over a frequency range to include contributions from all significant modes in the total response. When the equipment is qualified by dynamic test, the response spectrum or time-history of the attachment point is used in determining input motion.

Natural frequency may be determined by running a continuous sweep frequency search using a sinusoidal steady-state input of low magnitude. Seismic conditions are simulated by testing using random vibration input or single frequency input (within equipment capability) over the frequency range of interest. Whichever method is used, the input amplitude during testing envelops the actual input amplitude expected during hydrodynamic load conditions.

Equipment being dynamically tested is mounted on a fixture that simulates the intended service mounting and causes no dynamic coupling to the equipment.

Equipment having an extended structure, such as a valve operator, is analyzed by applying static equivalent dynamic loads at the center of gravity of the extended structure. In cases where the equipment structural complexity makes mathematical analysis impractical, a static bend test is used to determine operational capability at maximum equivalent dynamic load conditions. Pipe-mounted equipment is analyzed in the piping system dynamic analysis.

Random Vibration Input

When random vibration input is used, the actual input motion envelops the appropriate floor input motion at the individual modes. However, single frequency input such as sine waves can be used provided one of the following conditions is met:

1. The characteristics of the required input motion are dominated by one frequency.

2. The anticipated response of the equipment is adequately represented by one mode.

3. The input has sufficient intensity and duration to excite all modes to the required magnitude, in such a way that the testing response spectra envelop the corresponding response spectra of the individual modes.

Application of Input Motion

When dynamic tests are performed, the input motion is applied to one vertical and one horizontal axis simultaneously. However, if the equipment response along the vertical direction is not sensitive to the vibratory motion along the horizontal direction, and vice versa, then the input motion is applied to one direction at a time. In the case of single frequency input, the time phasing of the inputs in the vertical and horizontal directions are such that a purely rectilinear resultant input is avoided.

Fixture Design

The fixture design simulates the actual service mounting and causes no dynamic coupling to the equipment.

Prototype Testing

Equipment testing is conducted on prototypes of the equipment installed in this plant.

3.9B.2.2.2 Seismic and Hydrodynamic Load Qualification of Specific NSSS Mechanical Components

The following sections discuss the testing or analytical qualification of NSSS equipment. Seismic qualification is also described in Sections 3.9B.1.4, 3.9B.3.1, and 3.9B.3.2.

Jet Pumps

A dynamic analysis of the jet pumps is performed and stresses from the analysis are below the design allowables.

CRD and CRD Housing

The dynamic qualification of the CRD housing (with enclosed CRD) is done analytically, and the stress results of their analysis established the structural integrity of these components. Preliminary dynamic tests have been conducted to verify the operability of the CRD during a dynamic event. A simulated test, imposing a static bow in the fuel channels, is performed with the CRD functioning satisfactorily.

Core Support (Fuel Support and Control Rod Guide Tube)

A detailed analysis imposing dynamic effects due to seismic and hydrodynamic events showed that the maximum stresses developed during these events are much lower than the maximum allowed for the component material.

Hydraulic Control Unit

The seismic and hydrodynamic load adequacy of the HCU has been demonstrated by tests. A complete HCU assembly was qualified by multiaxis/multifrequency testing in the frequency range from 1 to 100 Hz. The required safety function of initiating reactor scram was demonstrated successfully.

Fuel Assembly (Including Channels)

GE BWR fuel channel design bases, analytical methods, and evaluation results, including seismic and hydrodynamic considerations, are contained in NEDE-24011[6], NEDE-24011-US[7], and NEDE-21175-3-P[13]. Section 3.9B.1.4.11.

Recirculation Pump and Motor Assembly

Calculations were made to assure that the recirculation pump and motor assembly is designed to withstand the specific static equivalent seismic and hydrodynamic loads. The flooded assembly was analyzed as a free body supported by constant support hangers from the brackets on the motor mounting member with mechanical snubbers attached to brackets located on the pump case and the top of the motor frame.

Primary stresses due to horizontal and vertical seismic (including hydrodynamic) forces are considered to act simultaneously and are conservatively added directly. Horizontal and vertical dynamic forces are applied to mass centers, and equilibrium reactions are determined for motor and pump brackets.

ECCS Pump and Motor Assembly

The qualification of the ECCS pump and motor assemblies as a unit while operating under faulted conditions was provided in the form of a static earthquake-acceleration analysis. The maximum specified vertical and horizontal accelerations were constantly applied simultaneously in the worst-case combination and the results of the analysis indicate the pump is capable of sustaining these loadings without overstressing the pump components.

Analysis is used for qualification when motors are similar in design features and insulation to previously qualified motors. Differences in design features and insulation are identified in a comparison study or similarity analysis. Also included in this comparison study are data showing that the differences have no impact on qualification. In addition to the comparison study, motor unique seismic analysis is required to assure that the motors can handle the design loads.

A motor of similar design has been seismically qualified via a combination of static analysis and dynamic testing. The complete motor assembly has been seismically qualified via dynamic testing, in accordance with IEEE-344-1975. The qualification test program included demonstration of startup and shutdown capabilities, as well as no-load operability during seismic and hydrodynamic loading conditions.

RCIC Pump Assembly

The RCIC pump construction is a barrel-type on a large cross-section pedestal. The RCIC pump assembly is analytically qualified by static analysis for seismic and hydrodynamic loading as well as the design operating loads of pressure, temperature, and external piping loads. The results of this analysis confirm that the stresses are substantially less than the allowables.

Because of their large size and weight, pumps are not included in the test list. Analysis is the most viable qualification method.

RCIC Turbine Assembly

The RCIC turbine is qualified for seismic and hydrodynamic loads via a combination of static analysis and dynamic testing. The turbine assembly consists of rigid masses, wherein static analysis has been utilized, interconnected with control levers and electronic control systems, necessitating final qualification by dynamic testing. Static loading analysis has been employed to verify the structural integrity of the turbine assembly and the adequacy of bolting under operating and seismic loading conditions. The RCIC electrohydraulic system integrated with the turbine governing valve is of a safety-grade design. The entire turbine assembly has been tested for seismic qualification in accordance with IEEE-344-1975. The electrohydraulic system was in its operational modes during the test program. The qualification test program included demonstration of startup and shutdown capabilities, as well as no-load operability during seismic loading conditions.

The specification for seismic qualification of the RCIC turbine and its accessories states that they shall be capable of withstanding the specified seismic accelerations at all frequencies within the range of 0.25 to 33 Hz. Proper performance may be demonstrated by tests, analysis, or a combination of both. If all natural frequencies of the turbine, the component parts, and the accessories are greater than 33 Hz (as defined by test and/or analysis), a static load analysis may be performed. The seismic forces of each component or assembly are obtained by concentrating its mass at the center of mass and multiplying by the seismic acceleration (earthquake coefficient). The magnitude of the earthquake coefficients is 1.5 g for both horizontal and vertical. If component parts and/or accessories have natural frequencies below 33 Hz, these parts must be dynamically analyzed or tested, demonstrating satisfaction of the floor response spectra.

Standby Liquid Control Pump and Motor Assembly

Each of the two SLC pumps is a positive displacement pump and motor mounted on a common base plate that is qualified by static analysis.

The SLC pump and motor assembly is analytically qualified by static analysis for seismic and hydrodynamic loads as well as the design operating loads of pressure, temperature, and external piping loads. The results of this analysis confirm that the stresses are substantially less than 90 percent of allowable.

RHR Heat Exchangers

A dynamic analysis is performed to verify that the RHR heat exchanger can withstand seismic and hydrodynamic loads. Seismic testing is an impractical method to verify the seismic adequacy of passive equipment.

Standby Liquid Control Tank

The SLC storage tank is a cylindrical tank, 9 ft in diameter and 12 ft high, bolted to the concrete floor. The SLC tank is qualified for seismic and hydrodynamic loads by analysis for:

1. Stresses in the tank bearing plate.

2. Bolt stresses.

3. Sloshing loads imposed at natural frequency of sloshing = 0.58 Hz.

4. Minimum wall thickness.

5. Buckling.

The results of the analysis confirm that stresses are less than the allowables.

Main Steam Isolation Valves

The MSIVs are qualified for seismic and hydrodynamic loads by analysis and test.

The fundamental requirement of the MSIV following a SSE or other faulted hydrodynamic loading is to close and remain closed after the event. This is demonstrated by the test and analysis as outlined in Section 3.9B.3.2.3.

Main Steam Safety/Relief Valves

Due to the complexity of the structure and the performance requirements of the valve, the total assembly of the SRV (including electrical, pneumatic devices) is dynamically tested at seismic acceleration equal to or greater than the combined SSE and hydrodynamic loading determined for this plant. Satisfactory operation of the valves was demonstrated during and after the test.

3.9B.2.3 Dynamic Response of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

The major reactor internal components are subjected to extensive testing coupled with dynamic system analyses to properly describe the resulting flow-induced vibration phenomena incurred from normal reactor operation and anticipated operational transients.

In general, the vibration forcing functions for operational flow transients and steady-state conditions are not predetermined by detailed analysis. Special analysis of the response signals measured for reactor internals of many similar designs are performed to obtain the parameters that determine the amplitudes and modal contributions in the vibration responses. These studies provide useful predictive information for extrapolating the results from tests of components with similar designs to components of different designs. This vibration prediction method is appropriate where standard hydrodynamic theory cannot be applied due to the complexity of the structure and flow conditions. Elements of the vibration prediction method are outlined as follows:

1. Dynamic analysis of major components and subassemblies is performed to identify vibration modes and frequencies. The analysis models used for Category I structures are similar to those outlined in Section 3.7B.2.

2. Data from previous plant vibration measurements are assembled and examined to identify predominant vibration response modes of major components. In

general, response modes are similar, but response amplitudes vary among BWRs of differing size and design.

3. Parameters are identified that are expected to influence vibration response amplitudes among the several reference plants. These include hydraulic parameters such as velocity and steam flow rates and structural parameters such as natural frequency and significant dimensions.

4. Correlation functions of the variable parameters are developed which, multiplied by response amplitudes, tend to minimize the statistical variability between plants. A correlation function is obtained for each major component and response mode.

5. Predicted vibration amplitudes for components of the prototype plant are obtained from these correlation functions, based on applicable values of the parameters for the prototype plant. The predicted amplitude for each dominant response mode is stated in terms of a range, taking into account the degree of statistical variability in each of the correlations. The predicted mode and frequency are obtained from the dynamic analyses of Item 1 above.

The dynamic modal analysis also forms the basis for interpretation of the preoperational and initial startup test results (Section 3.9B.2.4). Modal stresses are calculated, and relationships are obtained between sensor response amplitudes and peak component stresses for each of the lower normal modes. The allowable amplitude in each mode is that which produces a peak stress amplitude of ±10,000 psi.

3.9B.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

Vibration testing of reactor internals is performed on all GE BWR plants. At the time of the original issue of AEC RG 1.20, test programs for compliance were instituted. The first BWR 5 plant of each size is considered a prototype and is instrumented and subjected to preoperational and startup flow testing to demonstrate that flow-induced vibrations similar to those expected during operation cause no damage. Subsequent plants that have internals similar to those of the prototypes are also tested in compliance with the requirements of RG 1.20.

Unit 2 reactor internals will be tested in accordance with RG 1.20, Revision 2, for nonprototype, Category IV plants using Tokai-2 as the limited valid prototype. The test procedure will require vibration measurements to determine the vibration characteristics of vessel internals during the initial power ascension. Vibratory responses at various power levels and recirculation flow rates are recorded using accelerometers on the shroud head assembly and strain gauges on two selected jet pump riser pipe braces.

Reactor internals for Unit 2 are substantially the same as the internals design configurations that have been tested in prototype BWR 4 plants. Exceptions are the jet pumps, which are of the BWR 5 design. A vibration measurement and inspection program has been conducted at Tokai-2 to verify the design of the jet pumps with respect to vibration. Results of the prototype tests are presented in NEDE-24057-P (Class III) and NEDO-24057 (Class I)[8].

3.9B.2.5 Dynamic System Analysis of Reactor Internals Under Faulted Conditions

To ensure that no significant dynamic amplification of load occurs as a result of the oscillatory nature of the blowdown forces, a comparison is made of the periods of the applied forces and the natural periods of the core support structures being acted upon by the applied forces. These periods are determined from a 12-node vertical dynamic model of the RPV and internals. In addition to the real masses of the RPV and core support structures, hydrodynamic masses including fluid-structure interaction effects are accounted for.

Time-varying pressures are applied to the dynamic model of the reactor internals described above. Except for the nature and locations of the forcing functions and the dynamic model, the dynamic analysis method is identical to that described for seismic analysis and is detailed in Section 3.7B.2.1. The dynamic components of forces from these loads are combined with dynamic force components from other dynamic loads (including seismic and hydrodynamic), all acting in the same direction, by the SRSS method. This resultant force is then combined with other steady-state and static loads on an absolute sum basis to determine the design load in a given direction. Results of the dynamic analysis are summarized in Table 3.9B-2i.

3.9B.2.6 Correlations of Reactor Internals Vibration Tests With Analytical Results

Prior to initiation of the instrumented vibration test program for the prototype plant, extensive dynamic analyses of the reactor and internals were performed. The results of these analyses were used to generate the allowable vibration levels during the vibration test. The vibration data obtained during the test were analyzed in detail. The results of the data analysis, vibration amplitudes, natural frequencies, and mode shapes were then compared to those obtained from the theoretical analysis.

Such comparisons provided insight into the dynamic behavior of the reactor internals. The additional knowledge gained was utilized in the generation of the dynamic models for seismic and LOCA analyses for this plant. The models used for this plant are similar to those used for the vibration analysis of the prototype plant, Tokai-2.

3.9B.3 ASME Section III, Safety Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9B.3.1 Load Combinations, Design Transients, and Stress Limits

This section delineates criteria for selection and definition of design limits and load combinations associated with normal operation, postulated accidents, and specified seismic and hydrodynamic events for the design of safety-related ASME Code NSSS components.

This section also lists the major ASME Section III, Safety Class 1, 2, and 3, NSSS pressure parts and associated equipment on a component-by-component basis and identifies the applicable loadings, calculation methods, calculated stresses, and allowable stresses. Design transients for ASME Section III, Safety Class 1 equipment are addressed in Section 3.9B.1.1. Seismic-related loads are discussed in Section 3.7B. The hydrodynamic loads are described in the Mark II Containment Dynamic Forcing Functions Information Report (DFFR)[9].

Table 3.9B-2 is the major part of this section; it presents the load combinations, analytical methods (by reference or example), and calculated stress or other design values for the most critical areas in the design of each component. These values are also compared to applicable Code allowables. Table 3.9B-2 presents the generic load combinations required to be considered for the design and analysis of a plant, and is applicable to all ASME Safety Class 1, 2, and 3 component supports and core support structures.

3.9B.3.1.1 Plant Conditions

All events that the plant might credibly experience during a reactor year are evaluated to establish a design basis for plant equipment. These events are divided into four plant conditions. The plant conditions described in the following sections are based on event probability (i.e., frequency of occurrence) and correlated design conditions defined in the ASME Section III.

<u>Normal Condition</u>

Normal conditions are any conditions in the course of system startup, operation in the design power range, normal hot standby (with condenser available), and system shutdown other than upset, emergency, faulted, or testing.

Upset Condition

Upset conditions are any deviations from normal conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The upset conditions include transients that result from any single Operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system, and transients due to loss of load or power. Vibrations due to OBE are conservatively treated as upset. Hot standby with the main condenser isolated is an upset condition.

Emergency Condition

Emergency conditions are deviations from normal conditions that require shutdown for correction of the conditions or repair of damage in the RCPB. These conditions have a low probability of occurrence, but are included to provide assurance that no gross loss of structural integrity results as a concomitant effect of any damage developed in the system. Emergency condition events include, but are not limited to, transients caused by one of the following: a multiple valve safety/relief blowdown of the reactor vessel; loss of reactor coolant from a small break or crack that does not depressurize the reactor system or result in leakage beyond normal makeup system capacity, but which requires the safety functions of isolation of containment and reactor shutdown; improper assembly of the core during refueling; and vibration of an OBE in combination with associated system transients.

Faulted Condition

These are combinations of conditions associated with extremely low probability, postulated events whose consequences are such that the integrity and operability of the system may be impaired to the extent that considerations of public health and safety are involved. Faulted conditions encompass events that are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These postulated events are the most drastic that must be designed against and thus represent limiting design bases. Faulted condition events include, but are not limited to, one of the following: a control rod drop accident (CRDA), a fuel handling accident, a MSL break, a recirculation loop break, the combination of any pipe break plus the seismic motion associated with a SSE and hydrodynamic loads plus a LOOP, or the SSE.

Correlation of Plant Conditions with Event Probability

The probability of an event occurring per reactor year associated with the plant conditions is listed below. This correlation can be used to identify the appropriate plant condition for any hypothesized event or sequence of events.

Plant Condition	Event Encounter Probability Per Reactor Year
Normal (planned)	1.0
Upset (moderate probability)	$1.0 > P > 10^{-2}$
Emergency (low probability)	$10^{-2} > P > 10^{-4}$
Faulted (extremely low probability)	$10^{-4} > P > 10^{-6}$

Safety Class Functional Criteria

For any normal or upset design condition event, Safety Class 1, 2, and 3 equipment is capable of accomplishing its safety functions as required by the event and incurs no permanent changes that adversely affect its ability to accomplish its safety functions as required by any subsequent design condition event.

For any emergency or faulted design condition event, Safety Class 1, 2, and 3 equipment is capable of accomplishing its safety functions as required by the event, but repairs could be required to ensure its ability to accomplish its safety functions as required by any subsequent design condition event.

Compliance with Regulatory Guide 1.48 RG 1.48 was issued after the design of this plant was established and was therefore not used as a design basis requirement. However, GE design basis was representative of good industry practices at the time of design, procurement, and manufacture and is shown to be in general agreement with the requirement of RG 1.48 through the use of the alternate approach cited in Table 3.9B-3.

RG 1.48 delineates acceptable design limits and appropriate combinations of loadings associated with normal operation, postulated accidents, and specified seismic events for the design of the Category I fluid system components.

For a comparison of NSSS compliance with RG 1.48 refer to Table 3.9B-3. This comparison reflects general GE practice on BWR 5 plants and therefore is applicable to this plant.

3.9B.3.1.2 Reactor Pressure Vessel Assembly

The RPV assembly consists of the RPV support structure and shroud support. The RPV support structure and shroud support are constructed in accordance with ASME Section III. The shroud support consists of the shroud support plate and the shroud support cylinder and its legs. The RPV assembly components are classified as ASME Safety Class 1. Complete stress reports on these components have been prepared in accordance with ASME requirements. Table 3.9B-2h summarizes the loading conditions, calculated stresses, and allowables. The stress analyses performed for the reactor vessel assembly, including the faulted conditions, were completed using elastic methods. Except as noted in Section 3.9B.1.4.3, the load combinations and stress analyses for the core support structure and other reactor internals are discussed in Section 3.9B.5.

3.9B.3.1.3 Main Steam Piping

The main steam piping is discussed in Section 3.9A.

3.9B.3.1.4 Recirculation Loop Piping

The recirculation system piping bounded by the RPV nozzles is designed in accordance with ASME Section III, Subarticle NB-3600. The load conditions, stress criteria, calculated stresses, and allowables are shown in Table 3.9B-2m. The rules contained in Appendix F of ASME Section III are used in evaluating faulted loading conditions independently of all other design and operating conditions. Stresses calculated on an elastic basis are evaluated in accordance with Appendix F.

3.9B.3.1.5 Recirculation System Valves

The recirculation system flow control and suction and discharge gate valves are designed in accordance with ASME Section III, Safety Class 1, Subarticle NB-3500. These valves are not required to operate under the SSE. Load combinations and other stress analysis information are presented in Tables 3.9B-2l (FCVs) and 3.9B-2k (gate valves).

3.9B.3.1.6 Recirculation Pump

In the design of the recirculation pumps, the ASME Boiler and Pressure Vessel Code, Section VIII, Division 1, 1971 Edition with latest addenda was used as a guide in calculations made for determining the thickness of pressure-retaining parts and in sizing the pressure-retaining bolting.

The pump vendor made calculations for the design of the pressure-containing components to include the determination of minimum wall thickness, allowable stress, and pressures. The loading conditions and other stress analysis information are presented in Table 3.9B-2p.

Load, shear, and moment diagrams were constructed to scale, using live loads, dead loads, and calculated snubber reactions. Combined bending, tension, and shear stresses were determined for each major component of the assembly, including the pump driver mount, motor flange bolting, and pump case. The maximum combined tensile stress in the cover bolting was calculated using tensile stress from design pressure. Combined primary stresses did not exceed 150 percent of the Code allowable stress shown in Section

VIII of the ASME Boiler and Pressure Vessel Code, 1971 Edition. These methods and calculations demonstrate that the pump will maintain pressure integrity at all times.

3.9B.3.1.7 Standby Liquid Control Tank

The SLC tank is designed in accordance with ASME Boiler and Pressure Vessel Code, Section III. The loading conditions, stress criteria, calculated stresses, and allowables are summarized in Table 3.9B-2q.

3.9B.3.1.8 Residual Heat Removal Heat Exchangers

The RHR heat exchangers are designed in accordance with the ASME Boiler and Pressure Vessel Code Section III. The calculated stresses and allowables are shown in Table 3.9B-2r.

3.9B.3.1.9 RCIC Turbine

Although not under the jurisdiction of the ASME Code, the RCIC turbine is designed and fabricated following the basic guidelines for an ASME Section III, Safety Class 2 component.

Design operating conditions for the RCIC turbine include:

1. Surveillance Testing Monthly operation with reactor pressure at 1,000 psia, nominal, and saturated temperature, turbine exhaust pressure at 25 psia, peak, and saturated temperature.

2. Automatic Startup 30 cycles/yr with reactor pressure at 1,150 psia, nominal, and saturated temperature, turbine exhaust pressure at 25 psia, peak, and saturated temperature.

Design conditions for the RCIC turbine include:

1. Turbine inlet - 1,250 psig at saturated temperature.

2. Turbine exhaust - 165 psig at saturated temperature.

Table 3.9B-2s summarizes the criteria, calculated stresses, and allowables for the RCIC turbine components.

3.9B.3.1.10 RCIC Pump

The RCIC pump is designed and fabricated to the requirements for an ASME Section III Safety Class 2 component. Operating conditions for the RCIC pump are tested under surveillance together with the RCIC turbine. A monthly operation test is performed where the RCIC pump takes condensate from the CST and at design flow discharges condensate back to the CST via a closed test loop.

Design conditions for the RCIC pump include:

1.	Available NPSH minimum	21 ft
2.	Total head	
	High speed	2,895 ft
	Low speed	610 ft at 165 psia reactor pressure
3.	Constant flow rate	625 gpm
4.	Normal ambient operating temperature	60° to 100°F

5. Normal plus upset conditions which control the pump design include:

Design pressure	1,525 psig
Design temperature	40° min - 140°F max
OBE	2/3 of SSE

Table 3.9B-2t contains a summary of the design calculations for the RCIC pump components.

3.9B.3.1.11 ECCS Pumps

The RHR, LPCS, and HPCS pumps are designed and fabricated to the requirements of ASME Section III.

Table 3.9B-2u summarizes the design calculations for the ECCS pumps.

3.9B.3.1.12 Standby Liquid Control Pump

The SLC pump is designed and fabricated following the requirements for an ASME Section III, Safety Class 2 component. Operating conditions for the SLC pump and motor are functionally tested by pumping demineralized water through a closed test loop. The SLC pump is capable of injecting the net contents of the storage tank into the reactor in 50 to 125 min. The pump is capable of injecting flow into the reactor against zero psig up to the initial setpoint of the reactor relief valves.

Design conditions for the SLC pump include:

1.	Flow rate	43 gpm
2.	Available NPSH, maximum	12.9 psi
3.	Maximum operating discharge pressure	1,220 psig

4. Ambient conditions:

Temperature	70° - 104°F
Relative humidity	20 - 95%

5. Normal plus upset conditions that control the pump design include:

Design pressure	1,400 psig
Design temperature	150°F
OBE	2/3 of SSE

A summary of the design calculations for the SLC pump components is provided in Table 3.9B-2v.

3.9B.3.1.13 Main Steam Isolation and Safety/Relief Valves

The MSIVs and SRVs are designed in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section III, Subarticle NB-3500, Safety Class 1 components. Load combination, analytical methods, calculated stresses, and allowable limits for the SRVs and MSIVs are shown in Tables 3.9B-2j and 3.9B-2z.

3.9B.3.1.14 Reactor Water Cleanup System Pump and Heat Exchangers

The RWCU pump and regenerative and nonregenerative heat exchangers are not part of a safety system and are not designed to Category I requirements.

The requirements of ASME Boiler and Pressure Vessel Code, Section III, Safety Class 3 components are used as guidelines in evaluating the RWCU system pump and heat exchanger components. The loading conditions, stress criteria, and calculated and allowable stresses are summarized in Tables 3.9B-2w and 3.9B-2x.

3.9B.3.2 Pump and Valve Operability Assurance

The active pumps and valves are listed in Table 3.9B-4. Active mechanical equipment classified as Category I is designed to perform its functions during the life of the plant under postulated plant conditions. Equipment with faulted condition functional requirements includes active pumps and valves in fluid systems such as the ECCS and MSS system. (Active equipment must perform a mechanical motion during the course of accomplishing a safety function.)

Safety-related valves are qualified by testing and analysis, and satisfy stress and deformation criteria at critical locations. Operability is assured by satisfying the requirements of the programs detailed in the following sections.

3.9B.3.2.1 ECCS Pumps and Motors

All active pumps and motors are qualified for operability by first being subjected to rigorous tests before and after installation in the plant. The in-shop tests include 1) hydrostatic tests of pressure-retaining parts to 125 percent of the design pressure (multiplied by the ratio of material allowable stress at room temperature to the allowable stress value at the design temperature), 2) seal leakage tests, and 3) performance tests, while the pump is operated with flow, to determine total developed head, minimum and maximum head, and NPSH requirements. Also monitored during these operating tests are bearing temperatures (except water-cooled bearings) and vibration levels. Both are shown to be below specified limits. After the pump is installed in the plant, it undergoes cold hydro tests, functional tests, and the required periodic ISI and operation. These tests demonstrate reliability of the pump for the design life of the plant.

In addition to these tests, the safety-related active pumps are analyzed for operability during a faulted condition by imposing the following criteria: 1) the pump is not damaged during the faulted event, and 2) the pump continues operating despite the faulted loads.

<u>Analysis of Loading, Stress, and Acceleration Conditions</u>

To avoid damage during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE, hydrodynamic, and dynamic system loads are limited to the material elastic limit, as indicated in Section 3.9B.3.1 and Table 3.9B-2.

A three-dimensional finite element model of the pump/motor and its support was developed and dynamically analyzed using the response spectrum analysis method. The same model was analyzed for static nozzle loads, pump thrust loads, and deadweight. Critical location stresses were evaluated and compared with the allowable stress criteria. Critical location deflection and acceleration were evaluated to assure operability. The maximum seismic nozzle loads from the attached piping system are also considered in an analysis of the pump support to assure that there will be no geometric/dimensional deformation of the pump components.

Since the pumps and motors are structurally coupled, the dynamic acceleration values at the motor were obtained by performing a pump/motor response spectrum dynamic analysis to transfer the floor RRS to the motor and determine the peak vibration acceleration amplitude at the point of highest acceleration in the motor. This analysis showed that the maximum acceleration was less than the valves used in the detailed motor analyses.

Pump Operation During and Following SSE Loading

Active pump/motor rotor combinations are designed to rotate at a constant speed under all conditions. Motors are designed to withstand short periods of severe overload. The high rotary inertia in the operating pump rotor and the nature of the random, short duration loading characteristics of the seismic event prevent the rotor from becoming seized. In actuality, the seismic and hydrodynamic loadings cause only a slight increase, if any, in the torque (i.e., motor current) necessary to drive the pump at the constant design speed. Therefore, the pump does not shut down during the faulted load and continues to operate at the design speed.

The functional ability of the active pumps after a faulted condition is assured since only normal operating loads and steady-state nozzle loads exist. For the active pumps, the faulted condition is greater than the normal condition only due to seismic SSE and hydrodynamic loads on the equipment. These events are infrequent and of relatively short duration compared to the design life of the equipment.

Since it is demonstrated that the pumps are not damaged during the faulted event, the postfaulted condition operating loads are no worse than the normal plant operating limits. This is ensured by requiring that the imposed nozzle loads (steady-state loads) for normal conditions and postfaulted conditions are limited by the magnitudes of the normal condition nozzle loads. The postfaulted condition ability of the pumps to function under these applied loads is proven during the normal operating plant conditions for active pumps.

ECCS Motors

The analysis of the ECCS motors is performed by a computer program that consists of the mechanical analysis of the motor rotor assembly when acted upon by external forces including magnetic and centrifugal forces at any point along the shaft. The calculation for the seismic and hydrodynamic condition assumes that the motor is operating and the seismic, hydrodynamic, magnetic, and centrifugal forces all act simultaneously and in phase on the rotor shaft assembly. Other components of the motor, such as stator frame, lower-end shield, stator supports, base fasteners, top cap, and conduit box, are checked for the combined effects of seismic, self-weight, hydrodynamic, and operational loads, including consideration of bending, shear, torsion, and direct bearing loads.

The analysis and tests that are used for qualifications of ECCS pump motors were performed on an ECCS test motor of very similar mechanical construction.

The type test has been performed on a 1,250-hp vertical motor in accordance with IEEE-323-1974, by first simulating normal operation during the design life, then with the motor being subjected to a number of seismic and hydrodynamic events, and to the abnormal environmental conditions possible during and after a LOCA.

The test plan for the type test was as follows:

1. Thermal aging of the motor electrical insulation system (which is a part of the stator only) was based on extrapolation in accordance with the temperature life characteristic curve to satisfy the requirements of IEEE-275-1966 and from test data for the insulation type used on the ECCS motors. The amount of aging was equivalent to the total estimated days at maximum insulation temperature.

2. Radiation aging of the motor electrical insulation equals the maximum estimated integrated dose of gamma during normal and abnormal conditions.

3. The dynamic deflection analysis on the rotor shaft, performed to ensure adequate rotation clearance, has been verified by static loading and deflection of the rotor for the type test motor.

4. Dynamic load aging and testing has been performed on a biaxial test table in accordance with IEEE-344-1975. During this type test the shake table input simulated the maximum design limit of the SSE and hydrodynamic loads, combined with motor starts and operational combinations that may possibly occur during plant life. The acceleration values of the motor in the type test were significantly higher than those found in the structurally-coupled motor and pump dynamic analyses.

5. An environmental test simulating a LOCA condition with 100-days duration time has been performed with the test motor fully loaded, simulating pump operation. The test consisted of startup and 6-hr operation at 212°F ambient temperature and 100-percent steam environment. Another startup and operation of the test motor after 1-hr standstill in the same environment was followed by sufficient operation at high humidity and temperature, based on extrapolation in accordance with the temperature life characteristic curve to satisfy the requirements of IEEE-275-1966 for the insulation type used on the ECCS motors.

3.9B.3.2.2 SLC Pump and Motor Assembly and RCIC Pump Assembly

These equipment assemblies are small, compact, rigid assemblies, with natural frequencies well above 33 Hz. With this fact verified, each equipment assembly has been seismically qualified by static analysis. This static qualification verifies operability under seismic conditions, and assures structural loading stresses within Code limitations.

3.9B.3.2.3 NSSS Valves

3.9B.3.2.3.1 Safety Class 1 Active Valves

The Safety Class 1 active valves are the MSIVs, SRVs, SLC valves, and HPCS injection valves. Each of these valves is designed to perform its mechanical function in conjunction with a DBA including hydrodynamic loads. Qualification for operability is unique for each valve type. The method of qualification is described below.

Main Steam Isolation Valves

The MSIVs are evaluated for operability during seismic and hydrodynamic load events by both analysis and test.

1. The valve body is designed in accordance with ASME Code Section III, Subsection NB (Table 3.9B-2z), which limits deformation to within the elastic limit of the material by limiting pressure and pipe reaction input loads (including seismic and hydrodynamic loads). This ensures that only small deformations are allowed in the operating area of the valve body, hence, no interference with valve operability.

2. The entire topworks assembly was dynamically qualified by a bidirectional, random-frequency shake test. The loadings include SRV aging, OBE and SSE motions, and chugging motions. The SRV aging lasted 15 min for each pair of vertical axes and one of the two major horizontal axes. The motion simulation involved 5 intervals of 30 sec each for the 2 bidirectional combinations. The SSE simulation involved 1 interval of 30 sec each for the two bidirectional combinations. The chugging motion involved 15 min of bidirectional loadings lasting 15 min for each pair of major orthogonal axes. The testing covered seismic and hydrodynamic loads. The TRS exceeded the RRS by 10 percent. During each test interval, the MSIV topworks was cycled from full open to full closed to demonstrate operability. After the complete dynamic test program, the MSIV topworks was again cycled to ensure operability.

   Pipe anchors and restraints are provided in such a way as to limit the dynamic response and amplified accelerations to within design limits for the MSIVs. The mathematical modeling of the assembly accounts for the natural frequencies of the assembly as determined by the analysis and confirmed by a generic test.

3. MSIV operability following a downstream line break was demonstrated by the "state line test," as defined in the report APED-5750 (March 1969)[14]. The test specimen was a 20-in valve of a design representative of the MSIVs.

Main Steam Safety/Relief Valves

SRVs are qualified by test for operability during a seismic and hydrodynamic loading event. Each valve is designed for maximum moments that may be imposed when installed in service. These moments are resultants due to deadweight plus seismic and hydrodynamic loading of both the valve and the connecting pipe, the thermal expansion of the connecting pipe, and the reaction forces from valve discharge.

The SRVs were qualified by testing for seismic and hydrodynamic loads. The natural frequencies were determined to be greater than 33 Hz for seismic and 60 Hz for hydrodynamic loading.

The SRV design has been upgraded to NUREG-0588 Category 1 requirements. The SRV qualification program consists of:

1. Radiation aging of electropneumatic actuator assembly for a 5-yr (minimum) period.

2. Thermal aging of the SRV assembly for a 5-yr (minimum) period at 300°F.

3. Thermal cycling of the SRV from 135°F to 220°F back to 135°F 80 times and simultaneously actuating the SRV assembly approximately 130 times during the environmental transient condition.

4. Mechanical cycling of the SRV assembly 1,250 times in a 150°F ambient environment before subjecting the actuator assembly to a series of external pressurization tests.

The dynamic tests for SRV assembly envelope the Unit 2 RRS and hydrodynamic loading condition. The dynamic testing consists of vibration aging in accordance with IEEE-382-1980, 40-yr equivalent hydrodynamic aging, and upset and faulted loading conditions. SRV operability was demonstrated by periodically actuating (opening and closing) the valve successfully without malfunction.

Standby Liquid Control (Explosive) Valve

The SLC valve has been qualified by test in compliance with NUREG-0588, Category 1 requirements. The qualification includes compliance with IEEE-323-1974, IEEE-344-1975, and IEEE-382-1980. Prior to seismic and hydrodynamic testing, the explosive valve was subjected to radiation and thermal aging. No mechanical cycling was performed since this valve is designed for one-time use and, therefore, not subjected to operational cycles. Fatigue testing due to pipe-induced vibration, however, was performed by simulating SRV, OBE, and SSE loads to demonstrate functional operability.

HPCS Gate Valve

There is one Class 1 HPCS valve. This valve is a motor-operated gate valve. The valve body design, analysis, and testing is in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class 1 requirements. The environmental testing (radiation, thermal, and mechanical aging) in accordance with IEEE-382-1980 and dynamic testing per IEEE-344-1975 of a specimen motor actuator will be performed for the equivalent of 40-yr normal environment and 100-day post-LOCA environment to demonstrate functional operability.

3.9B.3.2.3.2 Safety Class 2 and 3 Active Valves

There are six HPCS gate valves and four CRD globe valves in this category. There is no Class 3 active valve in the NSSS scope of supply.

HPCS Gate Valves

These MOVs are qualified by testing valves that are generally typical of the valves supplied by GE. Operability is ensured by testing at the static design basis load. The actuators are qualified to IEEE-382-1980 to levels that exceed the design loadings.

CRD Globe Valves

These four CRD SDV vent and drain valves are air-operated globe valves. They were dynamically qualified by test, in accordance with IEEE-344-1975, to demonstrate operational and structural integrity under seismic and hydrodynamic load conditions.

3.9B.3.3 Design and Installation of Pressure Relief Devices

3.9B.3.3.1 Main Steam Safety/Relief Valves

SRV valve opening results in a transient that produces momentary unbalanced forces acting on the discharge piping system for the period from opening of the SRV until a steady discharge flow from the RPV to the suppression pool is established. This period includes clearing of the water slug from the end of the discharge piping submerged in the suppression pool. Pressure waves traveling through the discharge piping following the relatively rapid opening of the SRV cause the SRV discharge piping to vibrate. This in turn produces forces that act on the main steam piping.

The analysis of the relief valve discharge transient consists of a stepwise time-history solution of the fluid flow equation to generate a time-history of the fluid properties at numerous locations along the pipe. The fluid transient properties are calculated based on the maximum SRV set pressure specified in the steam system specification, and the value of the ASME flow rating increased by a factor to account for the conservative method of establishing the rating. Simultaneous discharge of all valves is assumed in the analysis as this is considered to induce maximum stress in the piping. Reaction loads on the pipe are determined at each elbow location. These loads are composed of pressure times area, momentum change, and fluid friction terms. The method of analysis to determine piping system response to relief valve operation is time-history integration. The forces are applied at locations on the piping system where fluid flow changes direction, thus causing momentary reactions. The resulting loads on the SRV, the MSL, and the discharge piping are combined with loads due to other effects as specified in Section 3.9B.3.1. The Code stress limits corresponding to load combination classifications of normal, upset, emergency, and faulted are applied to the steam and discharge pipe.

3.9B.3.4 Component Supports

3.9B.3.4.1 Piping

Piping supports are designed in accordance with Subsection NF of ASME Section III. Supports are either designed by load rating in accordance with Subsubarticle NF-3260 or to the stress limits for linear supports in accordance with Subsubarticle NF-3231. To avoid buckling in the component supports, Appendixes F and XVII of ASME Section III require that the allowable loads be limited to two-thirds of the critical buckling loads. The critical buckling loads for ASME Safety Class 1 component supports in the NSSS scope subjected to faulted loads that are more severe than normal, upset, and emergency loads, are determined by the vendor using the methods discussed in Appendix F of the ASME Code. In general, the load combinations for the conditions correspond to those used to design the supported pipe. Design transient cyclic data are not applicable to piping supports as no fatigue evaluation is necessary to meet the Code requirements. See Appendix 3E for a discussion of stresses in supports due to thermal growth of piping and seismic anchor motion.

The design criteria and dynamic testing requirements for component supports are as follows:

<u>Stiff Pipe Clamps</u>

Stiff pipe clamps are used on the recirculation piping system. There are 3 E-system pipe clamps on each recirculation loop. This is the only use of stiff pipe clamps.

The clamps were not used to meet stiffness criteria; they were designed to meet the requirements for strength and load distribution using a minimum of space.

The clamp design utilizes a double nut arrangement to prevent the nuts from backing off. The low temperature (<600°) and stresses in the bolt from preloads will not cause a relaxation of the material; consequently, no lift-off from the piping will occur.

Although bolt preloads are not addressable under ASME III rules for piping, preload could result in damage to the pipe if a clamp is poorly designed. Calculations have been made to ensure that bolt preload will not result in plastic deformation of recirculation pipe walls.

Equation 9 (of ASME III, Subsection NB) is aimed at preventing collapse of the piping system due to loads that produce primary stresses. Collapse is prevented by keeping the stresses due to pressure, deadweight, and inertia effects of dynamic loads less than prescribed values. The existence of clamps on piping systems does not adversely affect the moment carrying capability or reduce the ability of the piping system to resist collapse under combined loadings that produce primary stresses.

The only concern is the loading transmitted from the snubber through the clamp pads to the pipe. This bearing load will result in local stress in the pipe wall. These stresses are conservatively calculated using the indices method and added to the membrane and overall bending stresses computed by Equation 9 of the Code.

Clamp-induced stresses caused by the constraint of pipe expansion due to internal pressure have been added to other operating secondary and peak stresses by calculating effective increases in local bending stresses.

Clamp-induced stresses due to differential temperatures and material expansion coefficients have been accounted for by computing effective increases in local bending stresses. These stresses have been added to other operating secondary and peak stresses.

The fatigue usage at each clamp location has been conservatively computed, taking into consideration clamp-induced stresses from internal pressure, differential thermal expansion, and snubber loads.

The clamp-induced stresses were added to the stresses computed for each load set using Equations 10 and 11 of NB-3650. Cumulative fatigue usage was computed by the rules of the Code.

The stresses induced at each clamp location were calculated and compared to Code acceptance criteria. The primary stresses computed by Equation 9 were shown to be nongoverning. The thermal expansion stresses computed by Equation 12 were also shown to be nongoverning. The stress ratchet criteria of Equation 13 and the fatigue usage criteria of Equation 14 meet Code criteria, with significant margins.

Component Supports

All component supports, which include piping clamps, hangers, snubbers, struts, and attachments (e.g., clevis) to the building structure are designed, fabricated, and assembled so they cannot become disengaged by the movement of the supported pipe or equipment after they have been installed. All component supports are designed in accordance with the rules of Subsection NF of the Code. For the NSSS scope of supply, valve operators mounted on Safety Class 1 piping are not used as component supports.

Table 3.9B-2 includes loads and load combinations which are used also for the NSSS piping supports. The stress limits are in accordance with ASME III, Subsection NF.

No specific deformation limit is required. Deformation is limited by requiring proper stress limits.

Hangers

The design load on hangers is the load caused by deadweight. The hangers are calibrated to ensure that they support the design load at both their hot and cold load settings. Hangers provide a specified down travel and up travel in excess of the specified thermal movement.

Snubbers

Required Load Capacity and Snubber Location The entire piping system, including valves and suspension system between anchor points, is mathematically modeled for complete structural analysis. In the mathematical model, the snubbers are modeled as springs with a given spring stiffness depending on the snubber size. The analysis determines the forces and moments acting on each component and the forces acting on the snubbers due to all dynamic loading conditions defined in the piping design specification. The design load on snubbers includes those loads caused by seismic forces (OBE and SSE), hydrodynamic forces, system anchor movements, and reaction forces caused by relief valve discharge, turbine stop valve closure, and others.

The snubber location and loading direction are first decided by estimation to confine the stresses in the piping system to acceptable values. The snubber locations and direction are refined by performing the computer analysis on the piping system as described above.

The spring constant required by the suspension design specification for a given load capacity snubber is compared against the spring constant used in the piping system model. If the spring constants are the same, then the snubber location and load direction have been confirmed. If the spring constants are not in agreement, they are brought into agreement, and the system analysis is redone to confirm the snubber loads. This iteration is continued until all snubber load capacities and spring constants are compatible.

Design Specification Requirements To assure that the required structural and mechanical performance characteristics and product quality are achieved, the following requirements for design and testing are imposed on the manufacturer:

1. The snubbers are required by the suspension design specification to be designed in accordance with all rules and regulations of ASME Section III, Subsection NF. This design requirement includes analysis wherein the stresses in the snubber component parts are calculated under normal, upset, emergency, and faulted loads. These calculated stresses are then compared against the allowable stresses of the material as given in ASME Section III to ensure that they are below the allowable limit.

2. The snubbers are tested to ensure that they can perform as required during the OBE, the SSE, and hydrodynamic events under anticipated operational transient loads or other mechanical loads associated with the design requirements for the plant. The test requirements include:

   a. Snubbers are subjected to force or displacement versus time loading at frequencies within the range of significant modes of the piping system.

   b. Displacements are measured to determine the performance characteristics specified.

   c. Tests are conducted at various temperatures to ensure operability over the specified range.

   d. Peak test loads in both tension and compression are equal to or higher than the rated load requirements.

   e. Tests are conducted for various abnormal environmental conditions. Upon completion of the above abnormal environmental transient test, the snubber is tested dynamically at a frequency with a specified frequency range. The snubber must operate as designed during the dynamic test.

Snubber Installation Requirements An installation instruction manual is required by the suspension design specification. This manual is required to contain instructions for storage, handling, erection, and adjustments (if necessary) of snubbers. Each snubber has an installation location drawing, which contains the installation location of the snubber on the pipe and structure, the hot and cold settings, and additional information needed to install the particular snubber.

The suspension design specification requires that snubbers be provided with position indicators to identify the rod position. This indicator facilitates the checking of hot and cold settings of the snubber, as specified in the installation manual, during plant preoperational and startup testing.

Inspection, Testing, Repair, and/or Replacement of Snubbers The suspension design specification requires that the snubber supplier prepare an installation instruction manual. This manual is required to contain complete instructions for the testing, maintenance, and repair of the snubber. It also contains inspection points and the period for inspection.

Struts

The design load on struts includes those loads caused by deadweight, thermal expansion, primary seismic forces (OBE and SSE), hydrodynamic loads, system anchor displacements, and reaction forces caused by relief valve discharge, turbine stop valve closure, etc. Struts are designed in accordance with ASME Section III, Article NF-3000 to be capable of carrying the design load for all operating conditions.

3.9B.3.4.2 Reactor Pressure Vessel Stabilizer

The RPV stabilizer, which is massive and well supported, is designed as a Safety Class 1 linear-type component support in accordance with the requirements of ASME Boiler and Pressure Vessel Code Section III, Subsection NF. The RPV stabilizers attach to the ring girder/star truss structure. The ring girder/star truss structure, which is the top extension of the shield wall, is considered building steel and is designed to AISC criteria (see Section 3.8.3.2). The stabilizer provides a reaction point near the upper end of the RPV to resist horizontal loads due to effects such as earthquake and pipe rupture. The design loads and load combinations, stress criteria, calculated stresses, and allowable stresses in the critical areas are summarized in Table 3.9B-2f.

Deformation is limited by requiring proper stress limits.

3.9B.3.4.3 NSSS Floor-Mounted Equipment (Pumps, Heat Exchanger, and RCIC Turbine)

The NSSS floor-mounted equipment is analyzed to verify the adequacy of its support structures under various plant operating conditions. In all cases the stress loads in the critical support areas are within the ASME Code allowables. The loading conditions, stress criteria, and the allowable stresses in the critical support areas are given in Table 3.9B-2 in the respective equipment table.

3.9B.3.4.4 Supports for ASME Safety Class 1, 2, and 3 Active Components

ASME Safety Class 1, 2, and 3 active components are either pumps or valves. Since valves are supported by piping and not tied to building structures, pipe design criteria govern.

Category I active pump supports are qualified for seismic and hydrodynamic loads by testing when the pump supports and the pumps fulfill the following conditions:

1. Simulate actual mounting conditions.

2. Simulate all static and dynamic loadings on the pump.

3. Monitor pump operability during testing.

4. Normal operation of the pump during and after the test indicates that the supports are adequate. Any deflection or deformation of the pump supports that precludes the operability of the pump is not accepted.

5. Supports are inspected for structural integrity after the test. Any cracking or permanent deformation is not accepted.

Seismic and hydrodynamic qualification of component supports by analysis is generally accomplished as follows:

1. Stresses at all support elements and parts such as pump holddowns, and baseplate holddown bolts, pump support pads, pump pedestal, and foundation are checked to be within the allowable limits as specified in ASME Subsection NF.

2. For normal and upset plant conditions, the deflections and deformations of the support are assured to be within the elastic limits and must not exceed the values permitted by the designer based on his design verification tests to ensure the operability of the pumps.

3. For emergency and faulted plant conditions, the deformations must not exceed the values permitted by the designer to ensure the operability of the pumps.

3.9B.3.4.5 Bolting Support

Component Support Bolting

The support bolting of the RWCU pump is designed for the effects of pipe and SSE loads to the requirements of ASME Section III, Appendix XVII. The stress limits of 0.25Sy for tension and 0.20Sy for shear are used.

The equipment-to-base plate bolting of RCIC/SLC pumps and RCIC turbine satisfies the following design criteria: For normal and upset conditions, 1.0S is used for primary membrane and 1.5S for primary membrane plus bending, where S is the allowable stress limit from ASME Section III, Appendix I, Table I-7.3. For emergency and faulted conditions, stresses shall be less than 1.2 times the allowable limits for normal and upset conditions.

Piping Supports and Pipe-Mounted Equipment Supports

The allowable stresses for bolts meet the criteria of ASME Section III, Subsection NF. For service levels A and B, the bolts meet the criteria of NF-3280. For service levels C and D, XVII-2460 with factors indicated under XVII-2110 is applicable to the design requirements of bolting. The calculated stresses under these categories do not exceed the specified minimum yield stresses at temperature.

High-Strength Bolts

Bolts SA-193, Gr B7, and ASTM A-490-76a are used on Bergen-Patterson riser clamps for hanger attachments and E-systems clamps for snubber attachments, respectively.

The RWCU pump and driver motor holddown bolts are SA-193, Gr B7, and SAE Gr 8, respectively.

The RCIC pump and holddown bolts are SA-449.

The SLC pump and driver motor holddown bolts are SA-193, Gr B7, and SAE Gr 8, respectively.

3.9B.4 Control Rod Drive System

Unit 2 is equipped with a hydraulic CRD system that includes the CRD mechanism, the HCU, the condensate supply system, and the SDV, and extends to the coupling interface with the control rods.

3.9B.4.1 Descriptive Information on CRD System

Descriptive information on the CRDs and the entire drive and control system is contained in Section 4.6.

3.9B.4.2 Applicable CRD System Design Specifications

The CRD system is designed to meet the functional design criteria outlined in Section 4.6 and consists of the following:

1. Locking piston CRD.

2. Hydraulic control unit.

3. Hydraulic power supply (pumps).

4. Interconnecting piping.

5. Flow, and pressure and isolation valves.

6. Instrumentation and electrical controls.

Quality group classification is not applicable to the CRD.

Those components of the CRD forming part of the primary pressure boundary are designed according to ASME Section III.

The quality group classification of the CRD hydraulic system is outlined in Table 3.2-1, and the components are designed according to the codes and standards governing the individual quality groups.

Pertinent aspects of the design and qualification of the CRD system components are discussed in the following sections: transients in Section 3.9B.1.1, faulted conditions in Section 3.9B.1.4, seismic testing in Section 3.9B.2.2, load combinations and stress limits in Table 3.9B-2a.

Tables 3.9B-2g and 3.9B-2s show the load combinations, analytical methods, and allowable and calculated stress values for the highly stressed areas of the CRD housing and supports.

3.9B.4.3 Design Loads, Stress Limits, and Allowable Deformation

The ASME Code components of the CRD system have been evaluated analytically, and the design load combinations and stress limits are listed in Table 3.9B-2a. For the non-Code components, experimental testing was used to determine the CRD performance under all possible conditions as described in Section 3.9B.4.4. Deformation has been compared with the allowables and is not a limiting factor in the analysis of the CRD components.

3.9B.4.4 CRD Performance Assurance Program

The CRD test program consists of the following tests:

1. Development tests.

2. Factory quality control tests.

3. 5-yr maintenance life tests.

4. 1.5 x design life tests.

5. Operational tests.

6. Acceptance tests.

7. Surveillance tests.

All the above tests except 3 and 4 are discussed in Sections 4.6.3 through 4.6.3.1.1.5. Tests 3 and 4 are discussed as follows:

Test 3 - 5-Yr Maintenance Life Tests Four CRDs are normally picked at random from the production stock each year and subjected to various tests under simulated reactor conditions and more than one-eighth of the cycles specified in Section 3.9B.1.1. Upon completion of the test program, CRDs must meet or surpass the minimum specified performance requirements. This sample size is based on the large production volume during the manufacturing of CRDs through and including Model 7RDB144C.

The practice of testing the CRDs continues for Model 7RDB144EG001. However, due to the lower production volume expected for this model, fewer drives per year are tested.

Test 4 - 1.5 x Design Life Tests When a significant design change is made to the components of the drive, the drive is subjected to a series of tests equivalent to 1.5 times the life test cycles specified in Section 3.9B.1.1. Two CRDs underwent such testing in 1976. Upon completion of the test program, these CRDs met or exceeded the minimum specified performance requirements.

3.9B.5 Reactor Core Support Structures and Pressure Vessel Internals

3.9B.5.1 Design Arrangements

The core support structures and reactor vessel internals (exclusive of fuel, control rods, CRDs, and in-core nuclear instrumentation) are identified below:

1. Core support structures:

    a. Shroud.

    b. Shroud support.

    c. Core plate and holddown bolts.

    d. Top guide.

e. Fuel supports.

f. CRD housing.

g. Control rod guide tubes.

2. Reactor internals:

a. Jet pump assemblies and instrumentation penetration seal, instrumentation lines.*

b. Feedwater spargers.*

c. Vessel head spray nozzle.

d. Differential pressure and liquid control lines.

e. In-core flux monitor guide tubes.

f. Initial startup neutron sources.*

g. Surveillance sample holders.*

h. Core spray lines and spargers.

i. In-core instrument housings.

j. LPCI coupling.

k. Steam dryer.*

l. Shroud head and steam separator assembly.*

m. Guide rods.*

n. CRD thermal sleeves.*

A general assembly drawing of the important reactor components is shown on Figure 5.3-4.

The floodable inner volume of the RPV inside the core shroud up to the level of the jet pump suction inlet and internal flow path following a postulated recirculation line break are depicted in Figure 3.9B-2.

The design arrangement of the reactor internals such as the jet pumps, steam separators, and guide tube, is such that one end is unrestricted and thus free to expand.

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* Nonsafety-class equipment.

3.9B.5.1.1 Core Support Structures

These structures form partitions within the reactor vessel to sustain pressure differentials across the partitions, direct the flow of the coolant water, and laterally locate and support the fuel assemblies.

Shroud

The shroud support, shroud, and top guide make up a stainless steel cylindrical assembly. The first two structures provide a partition to separate the upward flow of coolant through the core from the downward recirculation flow. This partition separates the core region from the downcomer annulus, providing a floodable region following a recirculation line break. The volume enclosed by this assembly is characterized by three regions. The upper portion surrounds the core discharge plenum, which is bounded by the shroud head on top and the top guide grid plate below. The central portion of the shroud surrounds the active fuel and forms the longest section of the assembly. This section is bounded at the top by the grid plate and at the bottom by the core plate. The lower portion, surrounding part of the lower plenum, is welded to the RPV shroud support.

Shroud Support

The shroud support is designed to support the shroud and to support and locate the jet pumps. The shroud support provides an annular baffle between the RPV and the shroud. The jet pump discharge diffusers penetrate the shroud support to introduce the coolant to the inlet plenum below the core.

Shroud Head and Steam Separator Assembly

This component is not a core support structure or safety class component. It is discussed here to describe the coolant flow paths in the RPV. The shroud head and steam separator assembly is bolted to the top of the top guide to form the top of the core discharge plenum. This plenum provides a mixing chamber for the steam-water mixture before it enters the steam separators. Individual stainless steel axial flow steam separators are attached to the top of standpipes that are welded into the shroud head. The steam separators have no moving parts. In each separator, the steam-water mixture rising through the standpipe passes vanes that impart a spin to establish a vortex, separating the water from the steam. The separated water flows from the lower portion of the steam separator into the downcomer annulus.

Core Plate

The core plate consists of a circular stainless steel plate with bored holes stiffened with a rim-and-beam structure. The plate provides lateral support and guidance for the control rod guide tubes, in-core flux monitor guide tubes, peripheral fuel supports, and startup neutron sources. The last two items are also supported vertically by the core support plate. The entire assembly is bolted to a support ledge on the lower portions of the shroud.

Top Guide

The top guide is formed by a series of stainless steel beams joined at right angles to form square openings and fastened to a peripheral rim. Each opening provides lateral support and guidance for four fuel assemblies or, in the case of peripheral fuel, less than four fuel assemblies. Sockets are provided in the bottom of the beam intersections to anchor the in-core flux monitors and startup neutron sources. The rim of the top guide rests on a ledge between the upper and central portions of the shroud. The top guide has alignment pins that engage and bear against slots in the shroud that are used to position the assembly correctly before it is secured. Lateral restraint is provided by wedge blocks between the top guide and the shroud wall.

Fuel Support

The fuel supports shown on Figure 3.9B-3 are of two basic types: peripheral supports and four-lobed orificed fuel supports. The peripheral fuel support is located at the outer edge of the active core and is not adjacent to control rods. Each peripheral fuel support supports one fuel assembly and contains a single orifice assembly designed to ensure proper coolant flow to the peripheral fuel assembly. Each four-lobed orificed fuel support supports four fuel assemblies and has four orifice plates to ensure proper coolant flow distribution to each rod-controlled fuel assembly. The four-lobed orificed fuel supports rest in the top of the control rod guide tubes which are supported laterally by the core plate. The control rods pass through slots in the center of the four-lobed orificed fuel support. A control rod and the four adjacent fuel assemblies represent a core cell described in Section 4.4.2.

Control Rod Guide Tubes

The control rod guide tubes, located inside the vessel, extend from the top of the CRD housings up through holes in the core plate. Each tube is designed as the guide for a control rod and as the vertical support for a four-lobed orificed fuel support piece and the four fuel assemblies surrounding the control rod. The bottom of the guide tube is supported by the CRD housing, which in turn transmits the weight of the guide tube, fuel support, and fuel assemblies to the reactor vessel bottom head. A thermal sleeve is inserted into the CRD housing from below and is rotated to lock the control rod guide tube in place. A key is inserted into a locking slot in the bottom of the CRD housing to hold the thermal sleeve in position.

3.9B.5.1.2 Reactor Vessel Internals

Jet Pump Assemblies

The jet pump assemblies are not core support structures but are discussed here to describe coolant flow paths in the RPV. The jet pump assemblies are located in two semicircular groups in the downcomer annulus between the core shroud and the RPV wall. The design and performance of the jet pump are covered in detail in APED-5460[10] and NEDO-10602[11]. Each stainless steel jet pump consists of driving nozzles, suction inlet, throat or mixing section, and diffuser (Figure 3.9B-4). The driving nozzle, suction inlet, and throat are joined together as a removable unit, and the diffuser is permanently installed. High-pressure water from the recirculation pumps is supplied to each pair of jet pumps through a riser pipe welded to the recirculation inlet nozzle thermal sleeve. A riser brace consists of cantilever beams welded to a riser pipe and to pads on the RPV wall.

The nozzle entry section is connected to the riser by a metal-to-metal, spherical-to-conical seal joint. Firm contact is maintained by a holddown clamp. The throat section is supported laterally by a bracket attached to the riser. There is a slip-fit joint between the throat and diffuser. The diffuser is a gradual conical section changing to a straight cylindrical section at the lower end.

NMPC will reduce the preload on the beams from 30 to 25 kips in accordance with GE's recommendations. The expected life of these beams without cracking is 19 to 40 yr. ISI of the jet pump holddown beam will be performed to detect cracking. Inspection frequencies will be based on a lead plant experience and GE testing.

Steam Dryers

The steam dryer assembly is neither a core support structure nor a safety class component. It is discussed here to describe coolant flow paths in the vessel. The steam dryers remove moisture from the wet steam leaving the steam separators. The extracted moisture flows down the dryer vanes to the collecting troughs, then flows through tubes into the downcomer annulus. A skirt extends from the bottom of the dryer vane housing to the steam separator standpipe below the water level. This skirt forms a seal between the wet steam plenum and the dry steam flowing from the top of the dryers to the steam outlet nozzles.

The steam dryer and shroud head are positioned in the vessel during installation with the aid of vertical guide rods. The dryer assembly rests on steam dryer support brackets attached to the RPV wall. Upward movement of the dryer assembly, which may occur under accident conditions, is restricted by steam dryer holddown brackets attached to the RPV top head.

Feedwater Spargers

These components are not core support structures nor safety class components. They are discussed here to describe flow paths in the vessel. The feedwater spargers are stainless steel headers located in the mixing plenum above the downcomer annulus. A separate sparger is fitted to each feedwater nozzle and is shaped to conform to the curve of the vessel wall. Sparger end brackets are pinned to vessel brackets to support the spargers. Feedwater flow enters the center of the spargers and is discharged radially inward to mix the cooler feedwater with the downcomer flow from the steam separators and steam dryer before it contacts the vessel wall. The feedwater also serves to condense the steam in the region above the downcomer annulus and to subcool the water flowing to the jet pumps and recirculation pumps.

Core Spray Lines

This component is not a core support structure. It is discussed here to describe a safety class feature inside the RPV. The core spray lines are the means for directing flow to the core spray nozzles, which distribute coolant during accident conditions.

Two core spray lines enter the RPV through the two core spray nozzles (Section 5.4). The lines divide immediately inside the RPV. The two halves are routed to opposite sides of the RPV and are supported by clamps attached to the vessel wall. The lines are then routed downward into the downcomer annulus and pass through the top guide cylinder immediately below the flange. The flow divides again as it enters the center of the semicircular sparger, which is routed halfway around the inside of the top guide cylinder. The two spargers are supported by brackets designed to accommodate thermal expansion. The line routing and supports are designed to accommodate differential movement between the top guide and vessel. The other core spray line is identical except that it enters the opposite side of the vessel and the spargers are at a slightly different elevation inside the top guide cylinder. The correct spray distribution pattern is provided by a combination of distribution nozzles pointed radially inward and downward from the spargers (Section 6.3).

Vessel Head Spray Nozzle

This component is not a core support structure. It is included here to describe a safety class feature in the RPV. When reactor coolant is returned to the RPV, part of the flow can be diverted to a spray nozzle in the reactor head. This spray maintains saturated conditions in the RPV head volume by condensing steam being generated by the hot RPV walls and internals. The spray also decreases thermal stratification in the RPV coolant. This ensures that the water level in the RPV can rise. The higher water level provides conduction cooling to more of the mass of metal of the RPV and, therefore, helps to maintain the cooldown rate. The vessel head spray nozzle is mounted to a short length of pipe and a flange, which is bolted to a mating flange on the RPV head nozzle (Section 5.4.7).

Differential Pressure and Liquid Control Line

This component is not a core support structure. It is included here to describe a safety class component in the RPV. The differential pressure and liquid control lines enter the vessel through two bottom head penetrations and serve a dual function within the reactor vessel: to sense the differential pressure across the core support plate (Section 5.4), and to provide a path for the injection of the SLC solution into the coolant stream. One line terminates near the lower shroud with a perforated length below the core support plate. It is used to sense the pressure below the core support plate during normal operation and to inject liquid control solution if required. This location facilitates good mixing and dispersion. The other line terminates immediately above the core support plate and senses the pressure in the region outside the fuel assemblies.

In-core Flux Monitor Guide Tubes

These components are not core support structures. They are a safety class feature and provide a means of positioning fixed detectors in the core as well as provide a path for calibration monitors (TIP system).

The in-core flux monitor guide tubes extend from the top of the in-core flux monitor housing (Section 5.4) in the lower plenum to the top of the core support plate. The power range detectors for the power range monitoring units and the dry tubes for the SRM and IRM detectors are inserted through the guide tubes. A latticework of clamps, tie bars, and spacers give lateral support and rigidity to the guide tubes. The bolts and clamps are welded, after assembly, to prevent loosening during reactor operation.

Surveillance Sample Holders

This component is not a core support structure or a safety class component. The surveillance sample holders are welded baskets containing impact and tensile specimen capsules (Section 5.4). The baskets hang from the brackets that are attached to the inside wall of the RPV and extend to mid-height of the active core. The radial positions are chosen to expose the specimens to the same environment and maximum neutron fluxes experienced by the RPV itself while avoiding jet pump removal interference or damage.

Low-Pressure Coolant Injection Lines Penetrations

This component is a safety class feature, not a core support structure, but is discussed here to describe the coolant flow paths in the RPV. Three LPCI lines penetrate the core shroud through separate LPCI nozzles. Coolant is discharged inside the core shroud immediately below the top guide to restore and maintain the water level in the vessel required after a LOCA.

3.9B.5.2 Design Loading Conditions

3.9B.5.2.1 Events to Be Evaluated

Examination of the spectrum of conditions for which the safety design basis must be satisfied by core support structures and ESF components reveals the following significant faulted events:

1. Recirculation Line Break A break in a recirculation line between the RPV and the recirculation pump suction.

2. Steam Line Break Accident A break in one MSL between the RPV and the flow restrictor. The accident results in significant pressure differentials across some of the structures within the reactor.

3. Earthquake Subjects the core support structures and reactor internals to significant forces as a result of ground motion.

4. SRV discharge in combination with a SSE.

Analysis of other conditions existing during normal operation, abnormal operational transients, and accidents shows that the loads affecting the core support structures and other ESF reactor internals are less severe than these four postulated events. The faulted conditions for the RPV internals are discussed in Section 3.9B.1.4. Load combinations and analysis for the RPV internals are discussed in Section 3.9B.3.1 and Tables 3.9B-2h and 3.9B-2i. The stress deformation and fatigue limits are discussed in Sections 3.9B.5.3.5 and 3.9B.5.3.6.

3.9B.5.2.2 Pressure Differential During Rapid Depressurization

A digital computer code is used to analyze the transient conditions within the RPV following the recirculation line break accident and the steam line break accident. The analytical model of the vessel consists of nine nodes that are connected to the necessary adjoining nodes by flow paths having the required resistance and inertial characteristics. The program solves the energy and mass conservation equations for each node to give the depressurization rates and pressure in the various regions of the reactor. Figure 3.9B-5 shows the nine reactor nodes. The computer code used is the GE Short-Term Thermal-Hydraulic Model[12]. This model has been approved for use in ECCS conformance evaluation under 10CFR50 Appendix K. In order to adequately describe the blowdown pressure effect on the individual assembly components, three features are included in the model that are not applicable to the ECCS analysis and are therefore not described in NEDE-20566[12].

These additional features are discussed as follows:

1. The liquid level in the steam separator region and in the annulus between the dryer skirt and the pressure vessel is tracked to more accurately determine the flow and mixture quality in the steam dryer and in the steam line.

2. The flow path between the bypass region and the shroud head is more accurately modeled since the fuel assembly pressure differential P is influenced by flashing in the guide tubes and bypass region for a steam line break. In the ECCS analysis, the momentum equation is solved in this flow path, but its irreversible loss coefficient is conservatively set at an arbitrary low value.

3. The enthalpies in the guide tubes and the bypass are calculated separately, since the fuel assembly P is influenced by flashing in these regions. In the ECCS analysis, these regions are lumped.

3.9B.5.2.3 Recirculation Line and Steam Line Break

Accident Definition

Both a recirculation line break (the largest liquid break) and a steam line break inside containment (the largest steam break) are considered in determining the DBA for the ESF reactor internals. The recirculation line break is the same as the design basis LOCA (Section 6.3). A sudden, complete circumferential break is assumed to occur in one recirculation loop. The pressure differentials on the reactor internals and core support structures are in all cases lower than those for the MSL break.

The analysis of the steam line break assumes a sudden, complete circumferential break of one MSL between the RPV and the MSL restrictor. A steam line break upstream of the flow restrictors produces a larger blowdown area and thus a faster depressurization rate than a break downstream of the restrictors. The larger blowdown area results in greater pressure differentials across the reactor internal structures.

The steam line break accident produces significantly higher pressure differentials across the reactor internal structures than does the recirculation line break. This results from the higher reactor depressurization rate associated with the steam line break. Therefore, the steam line break is the DBA for internal pressure differentials.

Effects of Initial Reactor Power and Core Flow

The maximum internal pressure loads can be considered to be composed of two parts: steady-state and transient pressure differentials. For a given plant, the core flow and power are the two major factors that influence the reactor internal pressure differentials. The core flow essentially affects only the steady-state differential pressure. For a fixed power, the greater the core flow, the larger will be the steady-state pressure differentials. The core power affects both the steady-state and the transient differential pressure. As the power is decreased, there is less voiding in the core and, consequently, the steady-state core pressure differential is less. However, less voiding in the core also means that less steam is generated in the RPV and thus the depressurization rate and the transient part of the maximum pressure load is increased. As a result, the total loads on some components are higher at low power.

To ensure that the calculated pressure differences bound those that could be expected if a steam line break should occur, an analysis is conducted at a low power, high recirculation flow condition in addition to the standard safety analysis condition at high power, rated recirculation flow. The power chosen for analysis is the minimum value permitted by the recirculation system controls at rated recirculation drive flow (rated recirculation drive flow is the drive flow used to achieve rated core flow. This condition maximizes those loads that are inversely proportional to power.

Seismic and Hydrodynamic Events

The seismic and hydrodynamic loads acting on the structures within the RPV are based on a dynamic analysis as described in Sections 3.9B and 3.9B.2.5. Dynamic analysis is performed by coupling the lumped mass model of the RPV and internals with the building model to determine the forces, acceleration, and moment time-history in the reactor vessel and internals. This is done using the modal superposition method. ARS are also generated for subsystem analyses of selected components.

3.9B.5.3 Design Bases

3.9B.5.3.1 Safety Design Bases

The reactor core support structures and internals meet the following safety design bases:

1. They are arranged to provide a floodable volume in which the core can be adequately cooled in the event of a breach in the nuclear system process barrier external to the RPV.

2. Deformation is limited to assure that the control rods and core standby cooling systems can perform their safety functions.

3. Mechanical design of applicable structures assures that safety design bases 1 and 2 are satisfied so that the safe shutdown of the plant and removal of decay heat are not impaired.

3.9B.5.3.2 Power Generation Design Bases

The reactor core support structures and internals are designed to the following power generation design bases:

1. They provide the proper coolant distribution during all anticipated normal operating conditions up to full power operation of the core without fuel damage.

2. They are arranged to facilitate refueling operations.

3. They are designed to facilitate inspection.

3.9B.5.3.3 Design Loading Categories

The basis for determining faulted loads on the reactor internals is shown for seismic and hydrodynamic loads in Sections 3.7B, 3.8B, and 3.9B.2.5, and for pipe rupture loads in Sections 3.9B.5.2.3 and 3.9B.5.3.4. Loading conditions for shroud support, core support structures, CRD housing, jet pumps, LPCI coupling, and control rod guide tubes are given in Table 3.9B-2 under the respective equipment table.

Core support structure and safety class internals stress limits are consistent with ASME Section III, Paragraph NA-2140, and associated stress limits contained in Addenda dated through Summer 1976.. Level A, B, C, and D service limits defined in Winter 1976 Addenda which replace normal, upset, emergency, and faulted condition limits are not reflected in design documents for core support structures and other safety class internals for this reactor. However, for these components, Level A, B, C, and D service limits are judged to be equivalent to the normal, upset, emergency, and faulted loading condition limits and, therefore, for clarity, both sets of nomenclature are retained herein.

Stress intensity and other design limits are discussed in Section 3.9B.5.3.5. The core support structures that are fabricated as part of the RPV assembly are discussed in Section 3.9B.1.3.

The design requirements for equipment classified as "other internals" (e.g., steam dryers and shroud heads) were specified by the designer with appropriate consideration of the intended service of the equipment and expected plant and environmental conditions under which it is to operate. Where possible, design requirements are based on applicable industry codes and standards. If these are not available, the designer relies on accepted industry or engineering practices.

3.9B.5.3.4 Response of Internals Due to Inside Steam Break Accident

The maximum pressure loads acting on the reactor internal components result from an inside steam line break, and on some components the loads are maximum when operating at the minimum power associated with the maximum core flow. This has been substantiated by the analytical comparison of liquid versus steam breaks and by the investigation of the effects of core power and core flow.

It has also been pointed out that it is possible but not probable that the reactor would be operating at the rather abnormal condition of minimum power and maximum core flow. More realistically, the reactor would be at or near a full power condition and thus the maximum pressure loads would act on the internal components.

3.9B.5.3.5 Stress, Deformation, and Fatigue Limits for Engineered Safety Feature Reactor Internals (Except Core Support Structure)

The stress deformation and fatigue criteria listed in Tables 3.9B-5 through 3.9B-8 are used, or the criteria are based on the criteria established in applicable codes and standards for similar equipment, by manufacturers' standards, or by empirical methods based on field experience and testing. For the quantity SF_{min} (minimum safety factor) appearing in those tables, the following values were used.

Service Level	Design Condition	SF_{min}
A	Normal	2.25
B	Upset	2.25
C	Emergency	1.5
D	Faulted	1.125

Components inside the RPV, such as control rods that must move during accident conditions, have been examined for adequate clearances during emergency and faulted conditions. No mechanical clearance problems have been identified. The forcing functions applicable to the reactor internals are discussed in Section 3.9B.2.5.

3.9B.5.3.6 Stress, Deformation, and Fatigue Limits for Core Support Structures

The stress, deformation, and fatigue criteria presented in Tables 3.9B-9 through 3.9B-11 are used. These criteria are supplemented, where applicable, by the criteria for the reactor internals in the previous section.

3.9B.6 References

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3. PVRC Recommendations on Toughness Requirements for Ferritic Materials. Welding Research Council Bulletin No. 175, (date later).

4. Levy, S. and Wilkinson, J. D. P. The Component Element Methods in Dynamics, McGraw Hill Co., New York, NY, 1976.

5. PISYS Analysis of NRC Problem, NEDO-24210, August 1979.

6. General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A (latest approved revision).

7. General Electric Standard Application for Reactor Fuel-United States Supplement, NEDE-24011-P-A-US, (latest approved revision).

8. NEDE-24057-P (Class III) and NEDO-24057 (Class I). Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants, November 1977. Also NEDO-24075-1-P (Amendment No. 1 dated December 1978) and NEDE-2-P-24075 (Amendment No. 2 dated June 1979).

9. Mark II Containment Dynamic Forcing Functions Information Report, NEDO-21061, Revision 3, June 1978.

10. Design and Performance of G.E. BWR Jet Pumps. General Electric Company, Atomic Power Equipment Department, APED-5460, July 1968.

11. Moen, H. H. Testing of Improved Jet Pumps for the BWR/6 Nuclear System. General Electric Company, Atomic Power Equipment Department, NEDO-10602, June 1972.

12. Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K. Proprietary Document, General Electric Company, NEDE-20566.

13. BWR Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings, NEDE-21175-3-P, General Electric Company, July 1982.

14. Design and Performance of General Electric Boiling Water Reactor Main Steam Isolation Valves, APED-5750, General Electric Company, March 1969.

TABLE 3.9B-1

PLANT EVENTS

Normal, Upset, and Testing Condition	No. of Cycles
1. Boltup[1]	123
2. Design hydrostatic test	130
3. Startup (100°F/hr heatup rate)[2]	120
4. Daily reduction to 75% power[1]	10,000
5. Weekly reduction 50% power[1]	2,000
6. Control rod pattern change[1]	400
7. Loss of feedwater heaters (80 cycles total)	80
8. 50% SSE event at rated operating conditions (OBE)	10/50[3]
9. Scram:	
a. Turbine generator trip, feedwater on, isolation valves stay open	40
b. Other scrams	140
c. Loss of feedwater pumps, isolation valves closed	10
d. Single safety or relief valve blowdown	8
10. Reduction to 0% power, hot standby, shutdown (100°F/hr cooldown rate)[2]	111
11. Unbolt	123
Emergency Condition	
12. Scram:	
a. Reactor overpressure with delayed scram, feedwater stays on, isolation valves stay open	1[4]

TABLE 3.9B-1 (Cont'd.)

Emergency Condition	No. of Cycles
b. Automatic blowdown	1[4]
13. Improper start of cold recirculation loop	1[4]
14. Sudden start of pump in cold recirculation loop	1[4]
15. Hot standby, RPV drain shutoff, recirculation pumps restart	1[4]
Faulted Condition	
16. Pipe rupture and blowdown	1[4]
17. Safe shutdown earthquake at rated operating conditions	1[4]

(1) Applies to reactor pressure vessel only.
(2) Bulk average vessel coolant temperature change in any 1-hr period.
(3) 50 peak OBE cycles for NSSS piping; 10 peak OBE cycles for other NSSS equipment and components.
(4) Annual encounter probability of the one-cycle events is $<10^{-2}$ for emergency and $<10^{-4}$ for faulted events.

TABLE 3.9B-2

LOAD COMBINATIONS, STRESS LIMITS,
AND ALLOWABLE STRESSES

Load Combinations for ASME
Safety Class 1, 2, and 3 NSSS Components

INTRODUCTION

This table lists the major safety-related components in the plant, and both calculated and allowable stresses. Various parts of the table are referenced in Section 3.9B. The formats in various parts of the table are not consistent since variation in analytical method and depth of detail, necessary to demonstrate the safety aspects of various components, differs.

INDEX

a. Control Rod Drive

b. Control Rod Guide Tube

c. In-core Housing

d. Jet Pumps

e. Highest Stressed Region on the LPCI Coupling (Attachment Ring)

f. Reactor Vessel Support Equipment

g. Control Rod Drive Housing

h. Reactor Pressure Vessel and Shroud Support Assembly

i. Reactor Vessel Internals and Associated Equipment

j. Acting Type Safety/Relief Valves Spring-Loaded Direct

k. Reactor Recirculation System Gate Valves

l. Recirculation Flow Control Valve

m. ASME Safety Class 1 Recirculation Loop Piping and Pipe-Mounted Equipment (Class 1)

n. Reactor Refueling and Servicing Equipment

o. Fuel Assembly (including Channel)

TABLE 3.9B-2 (Cont'd.)

p.	Recirculation Pump
q.	Standby Liquid Control Tank
r.	Residual Heat Removal Heat Exchanger
s.	RCIC Turbine
t.	RCIC Pump
u.	ECCS Pumps
v.	Standby Liquid Control Pump
w.	Reactor Water Cleanup System Pump
x.	Reactor Water Cleanup Heat Exchangers
y.	Control Rod Drive Housing Supports
z.	Main Steam Isolation Valve

TABLE 3.9B-2 (Cont'd.)

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA
FOR ASME SAFETY CLASS 1, 2, AND 3
NSSS PIPING AND EQUIPMENT

Load Combination	Design Basis	Evaluation Basis	Service Level
N + SRV_{ALL}	Upset	Upset	B
N + OBE	Upset	Upset	B
N + OBE + SRV_{ALL}	Emergency	Upset	B
N + SSE + SRV_{ALL}	Faulted	Faulted(1)	D
N + SBA + SRV	Emergency	Emergency(1)	C
N + IBA + SRV	Faulted	Faulted(1)	D
N + SBA + SRV_{ADS}	Emergency	Emergency(1)	C
N + SBA + OBE + SRV_{ADS}	Faulted	Faulted(1)	D
N + IBA + OBE + SRV_{ADS}	Faulted	Faulted(1)	D
N + SBA/IBA + SSE + SRV_{ADS}	Faulted	Faulted(1)	D
N + LOCA(2) + SSE	Faulted	Faulted(1)	D

LOAD DEFINITION KEY:

N = Normal and/or abnormal loads depending on acceptance criteria

OBE = Operating basis earthquake loads

SSE = Safe shutdown earthquake loads

SRV = Safety/relief valve discharge-induced loads from two adjacent valves (one valve actuated when adjacent valve is cycling)

SRV_{ALL} = Loads induced by actuation of all safety/relief valves that activate within milliseconds of each other (e.g., turbine trip operational transient)

SRV_{ADS} = Loads induced by actuation of safety/relief valves associated with the automatic depressurization system that actuate within milliseconds of each other during the postulated small or intermediate size pipe rupture

TABLE 3.9B-2 (Cont'd.)

LOCA	=	Loss-of-coolant accident associated with the postulated pipe rupture of large pipes (e.g., main steam, feedwater, recirculation piping)
$LOCA_1$	=	Pool swell drag/fallout loads on piping and components located between the main vent discharge outlet and the suppression pool water upper surface
$LOCA_2$	=	Pool swell impact loads on piping and components located above the suppression pool water upper surface
$LOCA_3$	=	Oscillating pressure-induced loads on submerged piping and components during condensation oscillations
$LOCA_4$	=	Building motion-induced loads from chugging
$LOCA_5$	=	Building motion-induced loads from main vent air clearing
$LOCA_6$	=	Vertical and horizontal loads on main vent piping
$LOCA_7$	=	Annulus pressurization loads
SBA	=	Abnormal transients associated with small break accident
IBA	=	Abnormal transients associated with intermediate break accident

(1) All ASME Safety Class 1, 2, and 3 piping systems that are required to function for safe shutdown under the postulated events are designed to meet the functional capability criteria in accordance with NEDO-21985.

(2) The most limiting case of load combination among $LOCA_1$ through $LOCA_7$.

TABLE 3.9B-2d

JET PUMPS

Criteria	Load Combinations	Stress Type	Allowable Stress (psi)	Calculated* Stress (psi)
Primary membrane plus bending stress based on ASME Section III				
For Service Levels A and B (normal and upset) condition: For Type 304 @ 550°F S_m = 16,900 psi S_{limit} = 1.5 S_m psi	1. Dead weight 2. Pressure 3. OBE 4. SRV	Primary membrane plus bending	25,350	19,346
For Service Level C (emergency) condition: For Type 304 @ 550°F S_m = 16,900 psi S_{limit} = 1.8 S_m psi	1. Dead weight 2. Pressure 3. OBE 4. SRV	Primary membrane plus bending	30,420	19,346
For Service Level D (faulted) condition: For Type 304 @ 550°F S_m = 16,900 psi S_{limit} = 3.6 S_m psi	1. Deadweight 2. Pressure 3. Chugging 4. SRV 5. SSE	Primary membrane plus bending	60,840	34,417

* The New Loads Adequacy Evaluation has concluded that the listed limiting design basis loads envelope new loads.

TABLE 3.9B-2e

HIGHEST STRESSED REGION ON THE LPCI COUPLING (ATTACHMENT RING)

(Bellow-Type Design)

Criteria	Load Combinations	Stress Type	Allowable Stress (psi)	Calculated* Stress (psi)
Primary membrane plus bending stress based on ASME Section III for Type 304L				
For Service Levels A & B (normal and upset) condition: S_{limit} = 1.5 S_m = 20,925 psi	1. Normal loads 2. Pressure 3. OBE 4. SRV	Primary membrane plus bending	20,925	18,900
For Service Level C (emergency) condition: S_{limit} = 2.25 S_m = 31,400 psi	1. Normal loads 2. Pressure 3. OBE 4. SRV	Primary membrane plus bending	31,400	18,900
For Service Level D (faulted) condition: S_{limit} = 3.6 S_m = 50,220 psi	1. Normal Loads 2. Pressure 3. Annulus pressurization 4. SSE	Primary membrane plus bending	50,220	35,700

* The New Loads Adequacy Evaluation has concluded that the listed design basis loads envelope new loads.

TABLE 3.9B-2f

REACTOR PRESSURE VESSEL SUPPORT EQUIPMENT

RPV Support (Bearing Plate)

Criteria	Loading	Location	Allowable Stress (psi)	Calculated[4] Stress (psi)
Primary Stress Limit: AISC specification for the design, fabrication, and erection of structural steel for buildings				
For normal and upset condition: AISC allowable stresses, but without the usual increases for earthquake loads	Normal and upset condition: 1. Deadweight 2. OBE 3. Scram	Bearing plate	F_b (bearing) 22,000	f_b = 3,680
For emergency condition: 1.5 x AISC allowable stresses	Emergency condition: 1. Deadweight 2. OBE 3. Scram	Bearing plate	F_b (bearing) 33,000	f_b = 7,360
For faulted condition: 1.67 x AISC allowable stresses for structural steel members	Faulted condition: 1. Deadweight 2. SSE 3. Jet reaction load	Bearing plate	F_b (bearing) 36,000	f_b = 9,080

TABLE 3.9B-2f (Cont'd.)

RPV Stabilizer

Criteria	Loading	Location[1]	Allowable Stress (psi)	Calculated[4] Stress (psi)
Primary Stress Limit: ASME Section III, Subsection NF-Linear type support Material: Bracket and yoke: SA-516, Gr. 70 Rod: SA-540, B24, C1.2				
For normal and upset conditions: Subsection NF allowables	Upset condition: 1. Spring preload 2. OBE	Bracket Yoke Rod	[2]	[2]
For emergency conditions: 1.33 x normal/upset allowables	Emergency condition: 1. Spring preload 2. SSE	Bracket Bracket Yoke Yoke	F_b = 29,000 F_v = 19,300 F_b = 34,600 F_t = 27,610	f_b = 16,430 f_v = 4,390 f_b = 26,550 f_t = 19,000
For faulted conditions: (0.7 S/ft) x normal/upset allowables	Faulted condition: 1. Spring preload 2. SSE 3. Jet reaction load	Rod	F_t = 104,490	f_t = 84,220

TABLE 3.9B-2f (Cont'd.)

Stabilizer Bracket-Adjacent Shell

Criteria	Loading	Location	Allowable Stress (psi)	Calculated[4] Stress (psi)
ASME Section III primary local membrane plus primary bending limit for SA 533 Grade B, Class I:				
For normal and upset condition: $S_{limit} = 1.5 \times S_m$	Normal and upset condition loads: 1. OBE 2. Pressure	Local membrane plus bending	40,050	38,895
For emergency condition: $S_{limit} = 1.5 \times S_y$	Emergency condition loads: 1. SSE 2. Pressure	Local membrane plus bending	[3]	[3]
For faulted condition: $S_{limit} = 1.5 \times S_y$	Faulted condition loads: 1. SSE 2. Jet reaction 3. Pressure	Local membrane plus bending	63,450	56,604

TABLE 3.9B-2f (Cont'd.)

Orificed Fuel Supports

Criteria	Loading	Direction	Allowable Loads (LBF)	Calculated Loads (LBF)
Based on ASME Subsection NG-3228.4				
For normal, upset, and emergency conditions:	Normal, upset, and emergency loads:			
$L_{limit} = 0.44\ L_u$ [5]	Horizontal: Normal operating loads: OBE SRV	Horizontal	4,495	2,109
	Vertical: Normal operating loads: SRV SCRAM	Vertical	49,632	4,028
For faulted condition:	Faulted loads:			
$L_{limit} = 0.80\ L_u$ [5]	Horizontal: Normal operating loads Jet Reaction AP SSE	Horizontal	8,172	3,075
	Vertical: Normal operating loads: SRV SSE SCRAM	Vertical	90,240	8,044
Cumulative usage factor Limit = 0.047				

(1) Bracket and yoke have least stress margin in emergency condition, and rod in faulted condition.
(2) For the three locations, normal and upset condition has higher stress margins as compared to other conditions.
(3) Faulted category loads are evaluated with emergency allowable stresses; hence, emergency condition is not evaluated.
(4) The New Loads Adequacy Evaluation has concluded that the listed design basis loads envelope the new loads.
(5) Ultimate test load.

TABLE 3.9B-2h

REACTOR PRESSURE VESSEL AND SHROUD SUPPORT ASSEMBLY

I. Vessel Support Skirt

ASME Section III Primary Stress Limit Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress* (psi)
Material: SA-533 Gr. B Class I				
A. Normal and Upset Conditions:				
$P_m \leq S_m$ $S_m = 26{,}700$ @ 575°F	1. Deadweight 2. Pressure loads	Primary membrane	26,700	21,900
$P_L + P_b \leq 1.5\ S_m$ $S_m = 26{,}700$ @ 575°F	3. OBE 4. SRV	Primary membrane plus bending	40,050	30,910
B. Emergency Condition:				
$P_m \leq 1.2\ S_m$ $S_m = 26{,}700$ @ 575°F	1. Deadweight 2. Pressure loads	Primary membrane	32,040	26,930
$P_L + P_b \leq 1.8\ S_m$ $S_m = 26{,}700$ @ 575°F	3. OBE 4. SRV	Primary membrane plus bending	48,060	47,960
C. Faulted Condition:				
$P_m \leq 0.7\ S_m$ $S_y = 80{,}000$ @ 575°F	1. Deadweight 2. Pressure	Primary membrane	56,000	26,930
$P_L + P_b \leq 1.05\ S_y$ $S_y = 80{,}000$ @ 575°F	3. Jet reaction 4. Annulus pressurization 5. SSE	Primary membrane plus bending	84,000	47,960
D. Maximum Cumulative Usage Factor:	0.248 at knuckle			

TABLE 3.9B-2h (Cont'd.)

II. Shroud Support

ASME Section III Primary Stress Limit Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress* (psi)
Material: SB-168				
A. Normal and Upset Conditions:				
$P_m \leq S_m$ $S_m = 23{,}300$ psi @ 575°F	1. Deadweight 2. Pressure	Primary membrane	23,300	11,820
$P_L + P_b \leq S_y$ $S_y = 28{,}125$ psi @ 575°F	3. OBE 4. SRV	Primary membrane plus bending	28,125	28,030
B. Emergency Condition:				
$P_m \leq S_y$ $S_y = 28{,}548$ @ 528°F	1. Deadweight 2. Pressure	Primary membrane	28,548	17,990
$P_L + P_b \leq 1.5\ S_y$ $S_y = 28{,}548$ psi @ 528°F	3. Chugging 4. SRV	Primary membrane plus bending	42,822	40,040
C. Faulted Condition:				
$P_m \leq S_y$ $S_y = 28{,}548$ psi @ 528°F	1. Deadweight 2. Pressure	Primary membrane	28,548	21,420
$P_L + P_b \leq 1.5\ S_y$ $S_y = 28{,}548$ @ 528°F	3. Chugging 4. SRV 5. SSE	Primary membrane plus bending	42,822	40,040
D. Maximum Cumulative Usage Factor:	0.047 at Inconel section 0.053 at low alloy steel section			

TABLE 3.9B-2h (Cont'd.)

III. RPV Feedwater Nozzle

ASME Section III Primary Stress Limit Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress* (psi)
Material: SA-508 Class I safe end SRSS values only				
A. Normal and Upset Conditions:				
$P_m \leq 17,700$ psi $S_m = 17,700$ @ 575°F	1. Deadweight 2. Pressure loads	Primary membrane	17,700	16,220
$P_L + P_b \leq 26,550$ psi $1.5\ S_m = 26,550$ @ 575°F	3. OBE 4. SRV	Primary membrane plus bending	26,550	22,930
B. Emergency Condition:				
$P_m \leq 25,900$ psi $S_y = 25,900$ @ 594°F	1. Deadweight 2. Pressure loads	Primary membrane	25,900	21,420
$P_L + P_b \leq 38,850$ psi $1.5\ S_y = 38,850$ @ 594°F	3. SRV 4. SBA	Primary membrane plus bending	38,850	22,400
C. Faulted Condition:				
$P_m \leq 53,100$ psi $3\ S_m = 53,100$ @ 575°F	1. Deadweight 2. Pressure loads	Primary membrane	53,100	28,300
$P_L + P_b \leq 38,850$ psi $1.5\ S_m = 38,850$ @ 594°F	3. SSE 4. SRV 5. IBA	Primary membrane plus bending	38,850	33,740
D. Maximum Cumulative Usage Factor:	0.965 at safe end			

TABLE 3.9B-2h (Cont'd.)

IV. CRD Penetration

ASME Section III Primary Stress Limit Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress* (psi)
Material: SB 167 (Inconel stub tube)				
A. Normal and Upset Conditions:				
$P_m \leq S_m$ $S_m = 20{,}000$ @ 575°F	1. Normal loads 2. Pressure	Primary membrane	20,000	8,250
$P_L + P_b \leq 1.5\ S_m$ $S_m = 20{,}000$ @ 575°F	3. OBE 4. SRV	Primary membrane plus bending	30,000	27,051
B. Emergency Condition:				
$P_m \leq S_y$ $S_y = 24{,}100$ @ 575°F	1. Normal loads 2. Pressure	Primary membrane	24,100	8,250
$P_L + P_b \leq 1.5\ S_y$ $S_y = 24{,}100$ @ 575°F	3. OBE 4. SRV	Primary membrane plus bending	36,150	27,051
C. Faulted Condition:				
$P_m \leq 2.4\ S_m$ $S_m = 20{,}000$ @ 575°F	1. Normal loads 2. Pressure	Primary membrane	48,000	9,760
$P_L + P_b \leq 3.6\ S_m$ $S_m = 20{,}000$ @ 575°F	3. Jet reaction 4. Vent clearing 5. SSE 6. Scram	Primary membrane plus bending	72,000	31,900
D. Maximum Cumulative Usage Factor:	0.645 at stub tube			

* The New Loads Adequacy Evaluation has concluded that the listed design basis loads envelope new loads.

TABLE 3.9B-2j

SAFETY/RELIEF VALVES SPRING-LOADED DIRECT-ACTING TYPE[1]

Topic	Method of Analysis	Dikkers Analysis	Allowable Value	Calculated Value
1. Body inlet and outlet flange stresses[2]	$S_H = \frac{fMo}{Lg_1^2B} + \frac{PB}{4g_o} < 1.5S_m$ $S_R = \frac{(4te/3+1)Mo}{Lt^2B} < 1.5S_m$ $S_T = \frac{YMo}{t^2B} - ZS_R < 1.5S_m$ Where: S_H = longitudinal "hub" wall stress, psi S_R = Radial "flange" (body base, inlet) stress, psi S_T = Tangential "flange" stress, psi	[(Uses same notation as codes)] Body material: ASME SA352 LCB Inlet: S_m at 585°F = 17,540 psi Outlet: S_m at 500°F = 18,900 psi	1.5 S_m = 26,310 psi (inlet) and = 28,350 psi (outlet)	Inlet: S_H = 1.15 S_m =0.77 (allowable) S_R = 0.23 S_m =0.16 (allowable) S_T = 0.98 S_m =0.66 (allowable) Outlet: S_H = 1.21 S_m =0.81 (allowable) S_R = 0.79 S_m =0.53 (allowable) S_T = 0.49 S_m =0.33 (allowable)
2. Inlet and outlet stud area requirements[2]	Total cross-sectional area shall exceed the greater of: $Am_1 = \frac{Wm_1}{Sb}$, or $Am_2 = \frac{Wm_2}{Sa}$	[(Uses same notation as codes)]	Inlet: $Am_1 (>Am_2)$=12.45 in² Outlet: $Am_1 (>Am_2)$=4.65 in²	Inlet: Ab(actual area)=1.52 Am (required min) Outlet: Am(actual area)=1.84 Am (required min)

TABLE 3.9B-2j (Cont'd.)

Topic	Method of Analysis	Dikkers Analysis	Allowable Value	Calculated Value
	Where: Am_1 = Total required bolt (stud) area for condition Am_2 = Total required bolt (stud) area for gasket seating	Bolting material: ASME SA193 Gr. B7		
3. Nozzle wall thickness	1. Minimum wall thickness criterion: $t_{min} < t$ (1) Where: t_{min} = Minimum calculated thickness requirement, including corrosion allowance t_A = Actual nozzle wall thickness	Section near nozzle base: $t_{m_e} < t_{m_e}$ *(actual)* Nozzle midsection: $t_{m_c} < t_{m_c}$ *(actual)* Thin section near valve seat: $t_{m_b} < t_{m_b}$ *(actual)* Thinnest section at nozzle tip - just below valve seat: $t_{m_a} < t_{m_a}$ *(actual)* Nozzle Material: ASME SA350 LF2	 t_{m_e}=0.84 *inch* t_{m_c}=0.81 *inch* t_{m_b}=0.79 *inch* t_{m_a}=0.206 *inch* Actual thickness greater than t_m at the section under consideration	 t_{m_e}*(actual)* = 1.58 t_{m_e} t_{m_c}*(actual)* = 1.54 t_{m_c} t_{m_b}*(actual)* = 1.012 t_{m_b} t_{m_a}*(actual)* = 1.68 t_{m_a}

Table 3.9B-2j (Cont'd.)

Topic	Method of Analysis	Dikkers Analysis	Allowable Value	Calculated Value
(Refer to Section 3.9B.1.1.9 for thermal transients information.)	2. Cyclic Rating: Thermal: $It = \Sigma \frac{Nri}{Ni}$ Fatigue: $N_a \geq 2{,}000$ cycles, as based on Sa, where Sa is defined as the larger of: $Sp_1 = (2/3)Qp + \frac{P_{eb}}{2} + Q_{T_3} + 1.3\ Q_{T_1}$ or $Sp_2 = 0.4Q_p + \frac{K}{2}(P_{eb} + 2Q_{T_3})$ Where: Sp_1 = Fatigue stress intensity at inside surface of crotch, psi Sp_2 = Fatigue stress intensity at inside surface of crotch, psi	$It = \Sigma \frac{Nri}{Ni}$ $(i = 1,2,3,4,5)$ $N_a \geq 2{,}000$ cycles, as based on Sa, where $Sa = Sp_1 (> Sp_2)$ [(Uses same notation as codes)]	It (max) ≤ 1.0 $N_a \geq 2{,}000$ cycles	It=0.0014 (=0.0014 x It [max]) N_a (based on $Sa = Sp_2$) = 400,000 cycles, therefore satisfies criterion

Table 3.9B-2j (Cont'd.)

Topic	Method of Analysis	Dikkers Analysis	Allowable Value	Calculated Value
4. Bonnet flange strength	Flange treated as a loose type flange without hub: $S_R = \pm \frac{6M_p}{t^2(3.14C-nD)}$ $S_T = \pm \frac{5.46M_p}{BT_2}[0.318 \left[\frac{C-B}{C+B} + \frac{2h_c}{C+A}\right] + r_B] - \frac{E\theta_A t}{B}$ Where: S_R = Radial "flange" stress, psi S_T = Tangential "flange" stress, psi	[(Uses same notation as codes)] Bonnet Material: ASME SA-352LCB S_m at 500°F = 18,900 psi	1.5 S_m (for max S_R, S_R, and S_T), = 28,350 psi	S_R = 1.35 S_m = 0.9 (allowable) S_T = 0.53 S_m = 0.35 (allowable) (max S_T at back face of flange)
5. Bonnet bolting area requirements	Total cross-sectional area shall exceed the greater of: $AM_1 = \frac{Wm_1}{Sb}$ or $Am_2 = \frac{Wm_2}{Sa}$ Where: Am_1 = Total required bolt (stud) area for operating condition Am_2 = Total required bolt (stud) area for gasket seating	$Am_1 = \frac{Wm_1}{Sb}$ $Am_2 = \frac{Wm_2}{Sa}$ Where: Am (required minimum) is the greater of Am_1 and AM_2; and Ab (actual bolt area) must exceed Am. Body to bonnet bolting material: ASME SA-193 Gr. B7	$Am_1(>Am_2)$ = 7.399 in^2	A_b (actual area) = 1.34 Am (required min)

Table 3.9B-2j (Cont'd.)

Topic	Method of Analysis	Dikkers Analysis	Allowable Value	Calculated Value
6. Disc	The disc stress is calculated by treating the disc as a flat circular plate, edges supported, uniform load over area with radius r_o; reference Bach's Formulas, Machinery's Handbook, 15th Ed., p 414. From the reference: $t=1.2\frac{\sqrt{W\left[1-\frac{2r_o}{3R}\right]}}{S}$ W is based on p=1,375 psi under the disc	w = 27,430 lb r_o = 0.785 in R = 2.48 in Disc material: ASME SA351 CF3A Temperature: 585°F S_m (585°F) = 18,235 psi. Allowable stress is 1.5 S_m. This is the value S in the above formula. (1.5 S_m = 27,353 psi)	t (min allowable) = 1.067 in	Actual t_{min} = 1.068 in = 1.0009 (required min)
7. Seismic capability	Stress analysis uses $F_{vertical}$ = (mass of valve) x (4.5 g), and $F_{horizontal}$ = (mass of valve) x (6.5 g), with 800,000 in-lb and 300,000 in-lb applied at the inlet and outlet, respectively. Actual capability verifiable by test (with the moments concurrently applied) and exceeds these values.			

(1) ASME Section III, July 1974, including addenda through summer 1976.

(2) Design pressures:
P_b = 1,375 psig (inlet)
P_b = 625 psig (outlet)
These are the maximum anticipated pressures under all operating conditions. Analyses include applied moments of: M=800,000 in-lb (inlet) and M=300,000 in-lb (outlet). The analyses also include consideration of seismic, operational, and flow reaction forces. Since these SRVs are pipe-mounted equipment, refer to the piping analysis for verification that the moments are not exceeded.

(3) This t_{min} is t_m in accordance with notation of the codes.

TABLE 3.9B-2k

REACTOR RECIRCULATION SYSTEM GATE VALVES

Suction Valves

Item No.	Component/Load/Stress Type	Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowed
1.0	Body and Bonnet				
1.1	Loads: Design pressure Design temperature	 System requirement System requirement	 1,250 psi 575°F	 1,250 psi 575°F	 N/A N/A
1.2	Pressure rating	ASME Section III(1), Figure NB-3545.1-2	P_r = 734.96 psi	P_r = 734.96 psi	N/A
1.3	Minimum wall thickness	ASME Section III(1), Paragraph NB-3542	t_{min} = 1.931 in	t_{min} = 1.931 min	N/A
1.4	Primary membrane stress	ASME Section III(1), Paragraph NB-3545.1	$P_m \leq S$ (500°F) = 19,600 psi	P_m = 8,087 psi	0.54
1.5	Secondary stress due to pipe reaction	ASME Section III(1), Paragraph NB-3545.2(b)(i)	P_e = greatest value of p	P_{ed} = 6,300 psi P_{ed} = 13,894 psi	0.21 0.47
			P_{eb} and $P_{et} \leq$ 1.5S (500°F) (1.5) (19,600) = 29,400 psi	P_{et} = 13,894 psi	0.47
1.6	Primary plus secondary stress due to internal pressure	ASME Section III(1), Paragraph NB-3545.2(a)(1)	See Paragraph 1.8	Q_p = 20,632 psi	
1.7	Thermal secondary stress	ASME Section III(1), Paragraph NB-3545.2(c)	See Paragraph 1.8	Q_{T_2} = 998 psi	
1.8	Range of primary plus secondary stress at crotch region	ASME Section III(1), Paragraph NB-3545.2	$S_n \leq 3\ S_m$ (500°F) = 58,800 psi	$S_n = Q_p + P_e + 2Q_t$ = 28,928 psi	0.49
1.9	Cycle requirements for fatigue analysis	ASME Section III(1), Paragraph NB-3545.3	$N_a \geq$ 2,000 cycles	$N\delta = 1 \times 10^6$ cycles	N/A
1.10	Usage factor requirements for fatigue analysis	ASME Section III(1), Paragraph NB-3550	$I_t \leq 1.0$	I_t = 0.0025	

TABLE 3.9B-2k (Cont'd.)

Suction Valves (Cont'd.)

Item No.	Component/Load/Stress Type	Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowed
2.0	Body to Bonnet Bolting				
2.1	Loads: design pressure and temperature, gasket loads, stem operational load, seismic load (SSE)				
2.2	Bolt area	ASME Section III[1], Paragraph NB-3647.1	$A_b \geq 28.96$ in^2 $S_b = 28.675$ psi	$A_b = 29.70$ in^2 $S_b = 28.675$ psi	N/A N/A
2.3	Body Flange Stresses	ASME Section III[1], Paragraph NB-3647.1	--	--	--
2.3.1	Operating condition	ASME Section III[1], Paragraph NB-3647.1	$S_h \leq 1.5$ S (575°F) = 28,832 psi $S_r \leq 1.5$ S_m (575°F) = 28,837 psi $S_t \leq 1.5$ S_m (575°F) = 28,837 psi	S_h = 14,293 psi S_r = 11,568 psi S_t = 3,914 psi	0.50 0.40 0.14
2.3.2	Gasket seating condition	ASME Section III[1], Paragraph NB-3647.1	$S_h \leq 1.5$ S_m (100°F) = 30,000 psi $S_r \leq 1.5$ S_m (100°F) = 30,000 psi $S_t \leq 1.5$ S_m (100°F) = 30,000 psi	S_h = 18,437 psi S_r = 16,242 psi S_t = 5,498 psi	0.61 0.54 0.18
2.4	Bonnet Flange Stresses	ASME Section III[1], Paragraph NB-3647.1	--	--	--
2.4.1	Operating condition	ASME Section III[1], Paragraph NB-3647.1	$S_h \leq 1.5$ S_m (575°F) = 28,837 psi $S_r \leq 1.5$ S_m (575°F) = 28,837 psi $S_t \leq 1.5$ S_m (575°F) = 28,837 psi	S_h = 18,783 psi S_r = 14,968 psi S_t = 4,691 psi	0.65 0.52 0.16

TABLE 3.9B-2k (Cont'd.)

Suction Valves (Cont'd.)

Item No.	Component/Load/Stress Type	Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowed
2.4.2	Gasket seating condition	ASME Section III[1], Paragraph NB-3647.1	$S_h \leq 1.5\ S_m$ (100°F) = 30,000 psi $S_r \leq 1.5\ S_m$ (100°F) = 30,000 psi $S_t \leq 1.5\ S_m$ (100°F) = 30,000 psi	S_h = 18,214 psi S_r = 14,391 psi S_t = 4,507 psi	0.61 0.48 0.15
3.0	<u>Stresses in Stem</u>				
3.1	Loads: operator thrust and torque	--	--	--	--
3.2	Stem thrust stress	Calculate stress due to operator thrust in critical cross section	$S_t \leq S_m$ = 42,275 psi	S_t = 3,347 psi	0.08
3.3	Stem shear stress	Calculate shear stress due to operator torque in critical cross section	$S_s \leq 0.6\ S_m$ = 25,365 psi	S_s = 2,288 psi	0.09
3.4	Buckling on stem	Calculate slenderness ratio. If greater than 30 calculate allowable load from Rankine's formula using safety factor of 4	Max. allowable load = 34,284 lb	Slenderness ratio = 115 Actual load on stem = 15,115 lb (therefore, no buckling)	N/A 0.44
4.0	<u>Disc Analysis</u>				
4.1	Loads: maximum differential pressure[2]	--	--	--	--
4.2	Maximum stress in disc	ASME Section III[1], Paragraphs NB-3215 and NB-3221.3	$S_{max} \leq 1.5\ S_m$ (575°F) = 27,487 psi	Max stress = 20,225 psi	0.74

TABLE 3.9B-2k (Cont'd.)

Suction Valves (Cont'd.)

Item No.	Component/Load/Stress Type	Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowed
5.0	Yoke and Yoke Connections				
5.1	Loads: stem operational loads	Calculate stresses in the yoke and yoke connections to acceptable structural analysis methods	--	--	--
5.2	Tensile stress in yoke leg bolts	--	$S_{max} \leq S_m$ (100°F) = 35,000 psi	S_{max} = 8,718 psi	0.25
5.3	Bending stress of yoke legs	--	$S_b \leq 1.5\ S_m$ (185°F) = 33,165 psi	S_b = 14,011 psi	0.42

TABLE 3.9B-2k (Cont'd.)

Discharge Valves

Item No.	Component/Load/Stress Type	Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowed
1.0	Body and Bonnet				
1.1	Loads:				
	Design pressure	System requirement	1,650 psi	N/A	N/A
	Design temperature	System requirement	575°F	N/A	N/A
	Pipe reaction	Not specified	N/A	N/A	N/A
	Thermal effects	Not specified	N/A	N/A	N/A
1.2	Pressure rating	ASME Section III(1), Figure NB-3545.1-2	P_r = 969.68 psi	P_r = 969.68 psi	N/A
1.3	Minimum wall thickness	ASME Section III(1), Paragraph NB-3542	t (nominal) = 2.432 in	t_m = 2.432 min, in	N/A
1.4	Primary membrane stress	ASME Section III(1), Paragraph NB-3545.1	$P_m \leq S_m$ (500°F) = 19,600 psi	P_m = 9,831 psi	0.50
1.5	Secondary stress due to pipe reaction	ASME Section III(1), Paragraph NB-3545.2	P_e = greatest value of P_{ed}	P_{ed} = 5,917 psi P_{ed} = 13,834 psi	0.20 0.47
			P_{eb} and $P_{et} \leq$ 1.5 S_m (500°F) 1.5 (19,600) = 29,400 psi	P_{et} = 13,834 psi $P_e = P_{et}$ = 13,834 psi	0.47
1.6	Primary plus secondary stress due to internal pressure	ASME Section III(1), Paragraph NB-3545.2(a)(1)	$S_n \leq$ 3 S_m (500°F) = 58,800 psi	Q_p = 23,264 psi	--
1.7	Thermal secondary stress	ASME Section III(1), Paragraph NB-3545.2(c)	$S_n \leq$ 3 S_m (500°F) = 58,000 psi	Q_t = 1,020 psi	--
1.8	Sum of primary plus secondary stress	ASME Section III(1), Paragraph NB-3545.2	$S_n \leq$ 3 S_m (500°F) = 58,500 psi	$S_n = Q + P_e + 2Q_t$ = 31,221 psi	0.53
1.9	Fatigue requirements	ASME Section III(1), Paragraph NB-3545.3	$N_a \geq$ 2,000 cycles	$N\delta = 4 \times 10^5$ cycles	N/A
1.10	Cyclic rating	ASME Section III(1), Paragraph NB-3550	$I_t \leq 1.0$	I_t = 0.0032	N/A

TABLE 3.9B-2k (Cont'd.)

Discharge Valves (Cont'd.)

Item No.	Component/Load/Stress Type	Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowed
2.0	<u>Body to Bonnet Bolting</u>				
2.1	Loads: design pressure and temperature, gasket loads, stem operational load, seismic load (design basis earthquake)	ASME Section III[1], Paragraph NB-3647.1	--	--	--
2.2	Bolt area	ASME Section III[1], Paragraph NB-3647.1	$A_b \leq 41.23$ in^2 $S_b \leq 28{,}675$ psi	$A_b \geq 47.52$ in^2 $S_b = 28{,}675$ psi	N/A N/A
2.3	<u>Body Flange Stresses</u>	ASME Section III[1], Paragraph NB-3647.1	--	--	--
2.3.1	Operating conditions	ASME Section III[1], Paragraph NB-3647.1	$S_h \leq 1.5\ S_m$ (575°F) = 28,837 psi $S_r \leq 1.5\ S_m$ (575°F) = 28,837 psi $S_t \leq 1.5\ S_m$ (575°F) = 28,837 psi	$S_h = 14{,}682$ psi $S_r = 18{,}889$ psi $S_t = 4{,}815$ psi	0.51 0.65 0.17
2.3.2	Gasket seating condition	ASME Section III[1], Paragraph NB-3647.1	$S_h \leq 1.5\ S_m$ (100°F) = 30,000 psi $S_r \leq 1.5\ S_m$ (100°F) = 30,000 psi $S_t \leq 1.5\ S_m$ (100°F) = 30,000 psi	$S_h = 19{,}884$ psi $S_r = 28{,}378$ psi $S_t = 7{,}235$ psi	0.66 0.94 0.24
2.4	<u>Bonnet Flange Stresses</u>				
2.4.1	Operating condition	ASME Section III[1], Paragraph NB-3647.1	$S_h \leq 1.5\ S_m$ (575°F) = 28,837 psi $S_r \leq 1.5\ S_m$ (575°F) = 28,837 psi $S_t \leq 1.5\ S_m$ (575°F) = 28,837 psi	$S_h = 18{,}352$ psi $S_r = 24{,}546$ psi $S_t = 6{,}400$ psi	0.64 0.85 0.22

TABLE 3.9B-2k (Cont'd.)

Discharge Valves (Cont'd.)

Item No.	Component/Load/Stress Type	Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowed
2.4.2	Gasket seating condition	ASME Section III[1], Paragraph NB-3647.1	$S_h \leq 1.5\ S_m$ (100°F) = 30,000 psi $S_r \leq 1.5\ S_m$ (100°F) = 30,000 psi $S_t \leq 1.5\ S_m$ (100°F) = 30,000 psi	S_h = 18,540 psi S_r = 24,875 psi S_t = 6,485 psi	0.62 0.83 0.22
3.0	<u>Stresses in Stem</u>				
3.1	Load: operator thrust and torque	--	--	--	--
3.2	Stem thrust stress	Calculate stress due to operator thrust in critical cross section	$S_t \leq S_m$ = 42,275 psi	S_t = 7,295 psi	0.17
3.3	Stem torque stress	Calculate shear stress due to operator torque in critical cross section	$S_s \leq 0.6\ S_m$ = 25,365 psi	S_s = 4,986 psi	0.20
3.4	Buckling on stem	Calculate slenderness ratio. If greater than 30, calculate allowable load from Rankine's formula using safety factor of 4.	Max allowable load = 44,322 lb	Slenderness ratio = 96.5 Actual load on stem = 32,944 lb (therefore, no buckling)	N/A 0.74
4.0	<u>Disc Analysis</u>				
4.1	Loads: maximum differential pressure[3]	--	--	--	--
4.2	Maximum stress in disc	ASME Section III[1], Paragraphs NB-3215 and NB-3221.3	$S_{max} \leq 1.5\ S_m$ (575°F) = 27,487 psi	Max stress = 26,179 psi	0.95

TABLE 3.9B-2k (Cont'd.)

Discharge Valves (Cont'd.)

Item No.	Component/Load/Stress Type	Design Procedure	Allowable Limit	Design/ Calculated Value	Ratio Calculated/ Allowed
5.0	Yoke and Yoke Connections				
5.1	Loads: stem operational load	Calculate stresses in yoke and yoke connections to acceptable structural analysis methods	--	--	--
5.2	Tensile stress in yoke leg bolts	--	$S_{max} \leq S_m$ (100°F) = 35,000 psi	S_{max} = 14,654 psi	0.42
5.3	Bending stress of yoke legs	--	$S_b \leq$ 1.5 S (185°F) = 33,165 psi	S_b = 19,988 psi	0.60

(1) ASME Section III, 1971 Edition.
(2) Valve differential pressure is 50 psig.
(3) Valve differential pressure is 450 psig.

TABLE 3.9B-2m

ASME SAFETY CLASS 1 RECIRCULATION PIPING AND PIPE MOUNTED EQUIPMENT

HIGHEST STRESS SUMMARY

Acceptance Criteria	Limiting Stress Type	Calculated Stress[1] or Usage Factor	Allowable Limits	Ratio Actual/ Allowable	Loading	Identification[2] of Locations of Highest Stress Points
ASME Section III, NB-3600 Design Condition: Eq. 9 $\leq$ 1.5 S_m	Primary	15,846 psi	25,875 psi	0.61	1. Pressure 2. Weight	Hanger lug (Loop B)
Service Levels A and B (normal & upset) condition: Eq. 12 $\leq$ 3.0 S_m	Secondary	30,015 psi	51,750 psi	0.58	1. Thermal	Header sweepolet (Loop B)
Service Levels A and B (normal & upset) condition: Eq. 13 $\leq$ 3.0 S_m	Primary plus secondary (except thermal expansion)	40,978 psi	51,750 psi	0.79	1. Pressure 2. Weight 3. OBE 4. Operating transients 5. SRV	RHS return sweepolet (Loop B)
Service Levels A and B (normal and upset) condition: Cumulative usage factor	N/A	0.56	1.0	0.56		Header sweepolet (Loop B)
Service Level B (upset) condition: Eq. 9 $\leq$ 1.8 S_m & 1.5 S_y	Primary	29,239 psi	29,388 psi	0.99	1. Pressure 2. Weight 3. OBE 4. SRV	RHS return sweepolet (Loop B)
Service Level C (emergency) condition: Eq. 9 < 2.25 S_m & 1.8 S_y	Primary	18,540 psi	35,266 psi	0.53	1. Pressure 2. Weight 3. Chugging 4. SRV	Hanger lug (Loop B)
Service Level D (faulted) condition: Eq. 9 < 3.0 S_m + 2.0 S_y	Primary	34,058 psi	39,184 psi	0.87	1. Pressure 2. Weight 3. SSE 4. AP	RHS return sweepolet (Loop B)

TABLE 3.9B-2m (Cont'd.)

Component/Load Type	Highest Calculated Load	Allowable Load	Ratio Calculated/ Allowable	Loading	Identification of Equipment with Highest Loads
Snubber Load (lb)					
Service Level B	53,924	100,000	0.54	1. OBE 2. SRV	Snubber SB10
Service Level C	20,774	133,000	0.16	1. Chugging 2. SRV	Snubber SB10
Service Level D	72,271	150,000	0.48	1. SSE 2. AP	Snubber SB10
Flange Moment (in-lb)					
Level B	1,164,538	1,527,140	0.76	1. Weight 2. Thermal 3. OBE 4. SRV	Discharge valve (Loop B)
Level C	471,382	1,527,140	0.31	1. Weight 2. Thermal 3. Chugging 4. SRV	Discharge valve (Loop B)
Level D	1,374,232	1,527,140	0.90	1. Weight 2. Thermal 3. SSE 4. AP	Discharge valve (Loop B)
Acceleration (g)					
Horizontal	1.58	9.0	0.18	1. SSE 2. Chugging 3. SRV	Flow control valve (Loop A)
Vertical	1.40	6.0	0.23	1. SSE 2. AP	Flow control valve (Loop B)

TABLE 3.9B-2p

RECIRCULATION PUMP

Summary of Load Classification
High Stress Locations and Limit Criteria Pump Case

Loading Condition ASME Section III	Load Combination: Pressure (psig)	Load Combination: Mechanical Loads	Criteria (ASME Section III NB-3220)	Location	Highest Calc. Stress (psi)/ Usage Factor	Allowable Value	Ratio Act./ All.
Design (NB-3112)	Design pressure = 1,650	1. OBE 2. Pump thrust 3. Deadweight 4. Nozzle loads 5. Gasket seating	Figure NB-3221-1 $P_m \le 1.0\ S_m$ $P_L + P_b \le 1.5\ S_m$	Pump case	28,449 psi	28,838 psi	0.99
Normal (NB-3113.1) and upset (NB-3113.2)	Most severe normal/ upset pressure = 1,313	1. Deadweight 2. Nozzle loads 3. Thermal transient 4. OBE 5. Upset	Figure NB-3222-1 $P_L + P_b + P_e + Q \le 3.0\ S_m$ $P_e \le 3.0\ S_m$	Discharge transition Bolts	48,449 psi u=0.29	58,020 psi 1.0	0.84 0.29
Emergency (NB-3113.3)	Most severe emergency pressure = 1,796	1. Deadweight 2. Nozzle loads 3. Pump thrust 4. Gasket seating 5. OBE	Figure NB-3224-1 $P_m \le (1.2\ S_m$ or $S_y)$ $P_L \le (1.8\ S_m$ or $1.5\ S_y)$ $P_L + P_b \le (1.8\ S_m$ or $1.5\ S_y)$	Crotch	32,317 psi	34,812 psi	0.93
Faulted (NB-3113.4)	Most severe faulted pressure = 1,313	1. Deadweight 2. Nozzle loads 3. SSE 4. Pump thrust 5. Gasket seating	Table F-1322.2-1 $P_m \le 2.4\ S_m$ or $0.7\ S_u$ $+ P \le 1.5\ (2.4\ S_m$ or $0.7\ S_u)$ $P_L + P_b \le 1.5\ (2.4\ S_m$ or $0.75\ S_u)$	Discharge transition	52,563 psi	66,845 psi	0.79

TABLE 3.9B-4

GE-SUPPLIED SEISMIC ACTIVE PUMPS AND VALVES

Component	Master Parts List No.	Standards[1]
Main steam isolation valve	B22-F022 B22-F028	IEEE-323-1974 IEEE-344-1975 IEEE-382-1980 NUREG-0588, Cat. 1 ASME Section III, 1977 Edition, S77 Addenda
Main steam SRV	B22-F013	IEEE-323-1974 IEEE-344-1975 IEEE-382-1980 NUREG-0588, Cat. 1 ASME Section III, 1974 Edition, S76 Addenda
Standby liquid control (explosive) valve	C41-F004	IEEE-323-1974 IEEE-344-1975 NUREG-0588, Cat. 1 ASME Section III, 1977 Edition, S77 Addenda
CRD solenoid valve	C12-F009 C12-F110 C12-F160 C12-F162 C12-F163 C12-F182	IEEE-323-1974 IEEE-344-1975 IEEE-382-1980 NUREG-0588, Cat. 1
CRD globe valve	C12-F010 C12-F011	IEEE-344-1975 ASME Section III, 1971 Edition, S73 Addenda
	C12-F180 C12-F181	IEEE-344-1975 IEEE-382-1980 ASME Section III, 1977 Edition, S77 Addenda
HPCS gate valves	E22-F001 E22-F004 E22-F010 E22-F011	IEEE-323-1974 IEEE-344-1975 IEEE-382-1980 NUREG-0588, Cat. 1

TABLE 3.9B-4 (Cont'd.)

Component	Master Parts List No.	Standards[1]
	E22-F012 E22-F015 E22-F023	ASME Section III, 1971 Edition, W73 Addenda
RCIC turbine	E51-C002	a, b
RCIC pump	E51-C001	d,e,f,h,j
SLC pump and motor	C41-C001	Pump: d,e,f,h,i Motor: a,b,c,d,e, f,g,h,i,j
RHR pump and motor	E12-C002	Pump: d,e,f,h,i Motor: a,b,c,d,e, f,g,h,i,j
LPCS pump and motor	E21-C001	Pump: d,e,f,h,i Motor: a,b,c,d,e, f,g,h,i,j
HPCS pump and motor	E22-C001	Pump: d,e,f,h,i Motor: a,b,c,d,e, f,g,h,i,j

[1] a: IEEE-323-74
b: IEEE-344-75
c: IEEE-334-74
d: RG 1.48
e: RG 1.60
f: RG 1.61
g: RG 1.89
h: RG 1.92
i: RG 1.100
j: RG 1.122

TABLE 3.9B-5

DEFORMATION LIMIT
(FOR SAFETY CLASS REACTOR INTERNAL STRUCTURES ONLY)

Either one of (not both):

	General Limit
1. $\left[\dfrac{\text{Permissible deformation, DP}}{\text{Analyzed deformation causing loss of function, DL}}\right]$	$\leq \dfrac{0.9}{SF_{min}}$
2. $\left[\dfrac{\text{Permissible deformation, DP}}{\text{Experiment deformation causing loss of function, DE}}\right]$[1]	$\leq \dfrac{1.0}{SF_{min}}$

Where:

DP = Permissible deformation under stated conditions of Service Levels A, B, C, or D (normal, upset, emergency, or faulted)

DL = Analyzed deformation that could cause a system loss of function[2]

DE = Experimentally determined deformation that could cause a system loss of function

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[1] Equation 2 is not used unless supporting data is provided to the NRC by GE.

[2] "Loss of function" can only be defined quite generally until attention is focused on the component of interest. In cases of interest, where deformation limits can affect the function of equipment and components, they are specifically delineated. From a practical viewpoint, it is convenient to interchange some deformation condition at which function is assured with the loss of function condition if the required safety margins from the functioning conditions can be achieved. Therefore, it is often unnecessary to determine the actual loss of function condition because this interchange procedure produces conservative and safe designs. Examples where deformation limits apply are: CRD alignment and clearances for proper insertion, core support deformation causing fuel disarrangement or excess leakage of any component.

TABLE 3.9B-6

PRIMARY STRESS LIMIT
(FOR SAFETY CLASS REACTOR INTERNAL STRUCTURES ONLY)

Any one of (no more than one required):	General Limit
1. $\left[\frac{\text{Elastic evaluated primary stresses, PE}}{\text{Permissible primary stresses, PN}}\right]$	$\leq \frac{2.25}{SF_{min}}$
2. $\left[\frac{\text{Permissible load, LP}}{\text{Largest lower bound limit load, CL}}\right]$	$\leq \frac{1.5}{SF_{min}}$
3. $\left[\frac{\text{Elastic evaluated primary stress, PE}}{\text{Conventional ultimate strength at temperature, US}}\right]$	$\leq \frac{0.75}{SF_{min}}$
4. $\left[\frac{\text{Elastic-plastic evaluated nominal primary stress, EP}}{\text{Conventional ultimate strength at temperature, US}}\right]$	$\leq \frac{0.9}{SF_{min}}$
5.* $\left[\frac{\text{Permissible load, LP}}{\text{Plastic instability load, PL}}\right]$	$\leq \frac{0.9}{SF_{min}}$
6.* $\left[\frac{\text{Permissible load, LP}}{\text{Ultimate load from fracture analysis, UF}}\right]$	$\leq \frac{0.9}{SF_{min}}$
7.* $\left[\frac{\text{Permissible load, LP}}{\text{Ultimate load or loss of function load from test, LP}}\right]$	$\leq \frac{1.0}{SF_{min}}$

Where:

PE = Primary stresses evaluated on an elastic basis. The effective membrane stresses are to be averaged through the load-carrying section of interest. The simplest average bending, shear, or torsion stress distribution that supports the external loading is added to the membrane stresses at the section of interest.

PN = Permissible primary stress levels under Service Levels A or B (normal or upset) conditions under ASME Section III.*

LP = Permissible load under stated conditions of Service Levels A, B, C, or D (normal, upset, emergency, or faulted).

TABLE 3.9B-6 (Cont'd.)

CL = Lower bound limit load with yield point equal to 1.5 S_m where S_m is the tabulated value of allowable stress at temperature of ASME Section III or its equivalent. The "lower bound limit load" is here defined as that produced from the analysis of an ideally plastic (nonstrain hardening) material where deformations increase with no further increase in applied load. The lower bound load is one in which the material everywhere satisfies equilibrium and nowhere exceeds the defined material yield strength using either a shear theory or a strain energy of distortion theory to relate multiaxial yield to the uniaxial case.

US = Conventional ultimate strength at temperature or loading that would cause a system malfunction, whichever is more limiting.

EP = Elastic-plastic evaluated nominal primary stress. Strain hardening of the material may be used for the actual monotonic stress strain curve at the temperature of loading or any approximation to the actual stress strain curve that everywhere has a lower stress for the same strain as the actual monotonic curve may be used. Either the shear or strain energy of distortion flow rule may be used.

PL = Plastic instability load defined here as the load at which any load-bearing section begins to diminish its cross-sectional area at a faster rate than the strain hardening can accommodate the loss in area. This type analysis requires a true stress-true strain curve or a close approximation based on monotonic loading at the temperature of loading.

UF = Ultimate load from fracture analyses. For components that involve sharp discontinuities (local theoretical stress concentration <3), the use of a fracture mechanics analysis where applicable utilizing measurements of plane strain fracture toughness may be applied to compute fracture loads. Correction for finite plastic zones and thickness effects as well as gross yielding may be necessary. The methods of linear elastic stress analysis may be used in the fracture analysis where its use is clearly conservative or supported by experimental evidence. Examples where

TABLE 3.9B-6 (Cont'd.)

fracture mechanics may be applied are for fillet welds or end-of-fatigue-life crack propagation.

LE = Ultimate load or loss of function load as determined from experiment. In using this method, account will be taken of the dimensional tolerances that may exist between the actual part and the tested part or parts as well as differences that may exist in the ultimate tensile strength of the actual part and the tested parts. The guide to be used in each of these areas is that the experimentally determined load is adjusted to account for material property and dimension variations, each of which has no greater probability than 0.1 of being exceeded in the actual part.

* Not used unless supporting data are provided.

TABLE 3.9B-7

BUCKLING STABILITY LIMIT
(FOR SAFETY CLASS REACTOR INTERNAL STRUCTURES ONLY)

Any one of (no more than one required):	General Limit
1. $\left[\dfrac{\text{Permissible load, LP}}{\text{Service Level A (normal) permissible load, PN}}\right]$	$\dfrac{2.25}{\leq SF_{min}}$
2. $\left[\dfrac{\text{Permissible load, LP}}{\text{Stability analysis load, SL}}\right]$	$\dfrac{0.9}{\leq SF_{min}}$
3. * $\left[\dfrac{\text{Permissible load, LP}}{\text{Ultimate buckling collapse load from test, SE}}\right]$	$\dfrac{*1.0}{\leq SF_{min}}$

Where:

LP = Permissible load under stated conditions of Service Levels A, B, C, or D (normal, upset, emergency, or faulted).

PN = Applicable Service Level A (normal) permissible load.

SL = Stability analysis load. The ideal buckling analysis is often sensitive to otherwise minor deviations from ideal geometry and boundary conditions. These effects will be accounted for in the analysis of the buckling stability loads. Examples of this are ovality in externally pressurized shells or eccentricity on column members.

SE = Ultimate buckling collapse load as determined from experiment. In using this method, account will be taken of the dimensional tolerances that may exist between the actual part and the tested part. The guide to be used in each of these areas is that the experimentally determined load will be adjusted to account for material property and dimension variations, each of which has no greater probability than 0.1 of being exceeded in the actual part.

*Not used unless supporting data are provided.

TABLE 3.9B-8

FATIGUE LIMIT*
(FOR SAFETY CLASS REACTOR INTERNAL STRUCTURES ONLY)

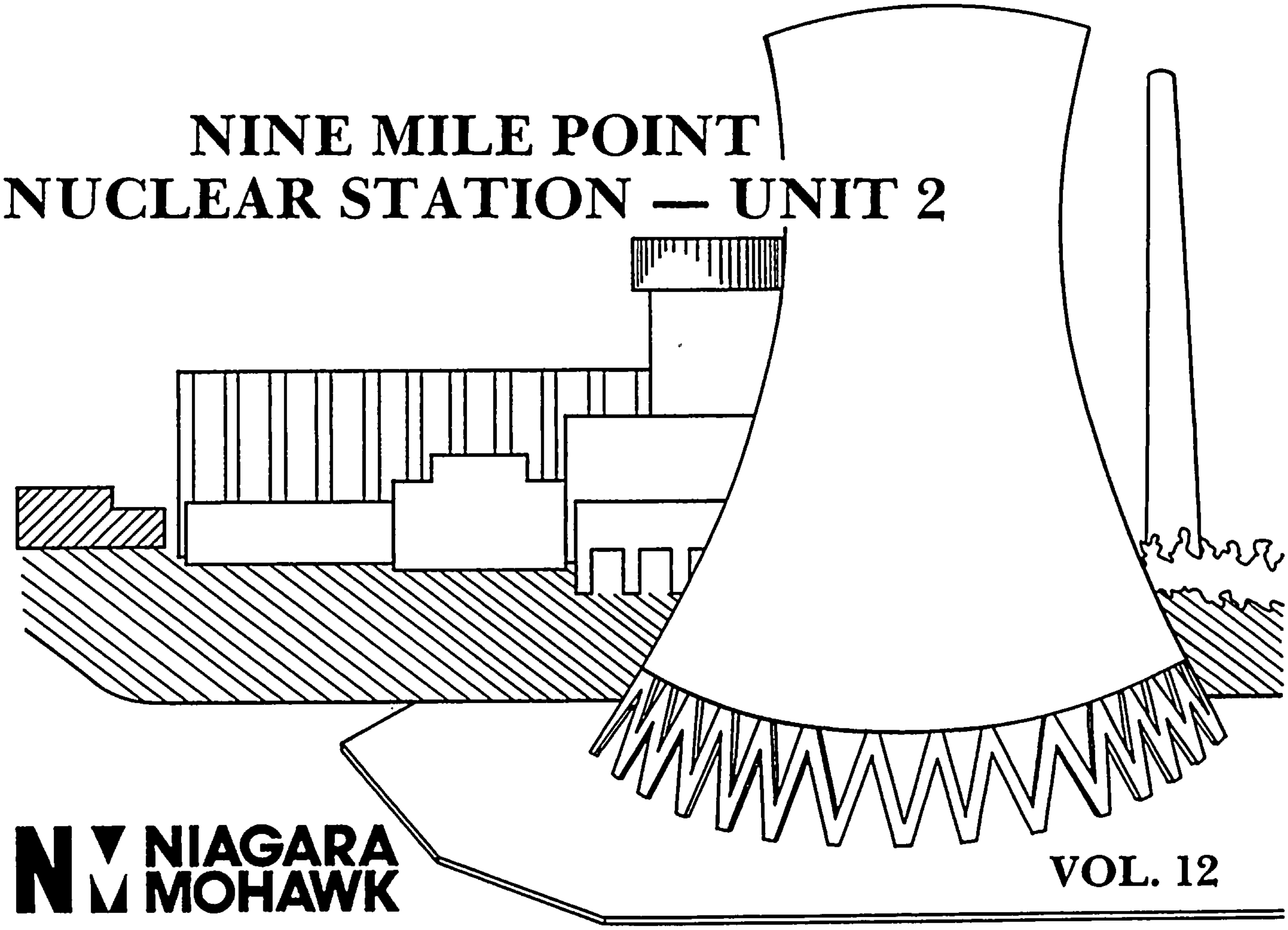
Cumulative Damage in Fatigue	Limit for Service Levels A and B (normal and upset) Design Conditions
Design fatigue cycle usage from analysis using the method of ASME Code	≤ 1.0

* Summation of fatigue damage usage with design and operation loads following Miner hypotheses.

SOURCE: Miner, M.A. Cumulative Damage in Fatigue, Journal of Applied Mechanics, Vol. 12, ASME Vol. 67, pp A159-A164, September 1945.

UPDATED SAFETY ANALYSIS REPORT

NINE MILE POINT NUCLEAR STATION — UNIT 2



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3.10 SEISMIC QUALIFICATION OF CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

Two inputs are provided for Section 3.10: Section 3.10A applies to the SWEC scope of supply, and Section 3.10B applies to the GE scope of supply.

3.10A SEISMIC QUALIFICATION OF CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT (SWEC SCOPE OF SUPPLY)

This section provides the qualification methods for equipment affected by seismic loads. The methods for the qualification of equipment affected by hydrodynamic loads associated with SRV discharge and the postulated LOCA are provided in the DAR, Appendix 6A, Subsection 6A.9.

3.10A.1 Seismic Qualification Criteria

Table 3.10A-1 provides a listing of Category I instrumentation and electrical equipment requiring seismic qualification. Parameters used to develop seismic loadings and criteria for Category I structures, systems, and components are described in Section 3.7A. From the ground input data, a series of response spectrum curves at various building elevations was developed. The magnitude and frequency of the SSE loadings for which each component is qualified vary, depending on their locations within the plant. These seismic data were included in the purchase specifications for Category I equipment and systems. For equipment located at various areas throughout the plant, the purchase specification includes response spectrum curves that envelop the response spectra at all locations where the equipment is used.

For equipment subject to hydrodynamic loads, see the DAR for Hydrodynamic Loads (Appendix 6A) for details.

Seismic qualification and documentation procedures used for Class 1E equipment and/or systems meet the provisions of IEEE-344-1975, as supplemented by RG 1.100.

Category I equipment is divided into two classifications: 1) equipment designed to maintain its functional capability during and after an SSE, and 2) equipment that, although not required to maintain its functional capability, is designed to maintain the pressure boundary integrity of the system of which it is a part, during and after an SSE. The requirements for instrumentation, equipment, and systems required to maintain pressure boundary integrity are in accordance with ASME Section III, 1974 or later, depending on time of purchase of equipment. The performance requirements of Category I electrical and instrumentation items and their respective supports may be structural as well as functional. The structural design is in accordance with applicable codes, as listed in the equipment specification.

It should be noted that certain non-Category I equipment is reviewed for maintenance of structural integrity to ensure that failure of these items or their supports will not jeopardize adjacent Category I equipment.

If no codes are applicable, the stress level for the OBE combined with operating loads is limited to 75 percent of the minimum yield for the material in accordance with the ASTM specification. For the SSE combined with operating loads, the stress level does not exceed the smaller of:

1. 100 percent of the minimum yield strength, or

2. 70 percent of the minimum ultimate tensile strength of the material (at design temperature), in accordance with the ASTM specification.

Seismic analysis, without testing, is performed on equipment whose functional operability is assured by its structural integrity alone. When complete seismic testing is impractical, a combination of tests and analyses is performed. See Table 3.10A-1 for the seismic qualification methods applicable to specific equipment.

3.10A.2 Methods and Procedures for Qualifying Electrical Equipment and Instrumentation

The methods by which the supplier can qualify equipment for compliance with seismic requirements are as follows:

1. Testing.

2. Type-testing (prototype).

3. Analysis.

4. Combination of 1 or 2 and 3.

These methods, including the factors for selection of an analytical or test option, test objectives, and acceptability criteria, are described in Section 3.7A.3.1.1. Qualification and documentation procedures used for Category I equipment and/or systems meet the provisions of IEEE-344-1975, as supplemented by the requirements of RG 1.100.

3.10A.2.1 Testing

Seismic tests are performed by subjecting equipment to vibratory motion that conservatively simulates the seismic loading at the equipment mounting. Such tests are conducted over the range of 1 to 33 Hz. For components susceptible to environmental aging (temperature, humidity, radiation, etc.), seismic testing is performed on environmentally preaged components, following the requirements of IEEE-323-1974.

For equipment subject to hydrodynamic loads, see the DAR for Hydrodynamic Loads (Appendix 6A) for details.

Whenever feasible, seismic qualification tests on equipment are performed while the equipment is subjected to normal operating loads. However, occasionally an operational configuration is difficult to simulate correctly, and where it can be demonstrated that operating loads such as pressure, torque, flow, voltage, current, or temperature do not cause significant stress loads within the equipment, or where such operating loads are not significant to a determination of equipment operability, operation under load is not specified. The equipment is monitored and evaluated during and after the test for malfunction or failure and, upon completion of the test, is tested for proper operation.

In seismic qualification testing, equipment auxiliary components, such as relays, switches, and instruments necessary for proper operation, are mounted similarly to the manner in which they are to be installed, and then tested and qualified along with the equipment. For multicabinet assemblies, the tested prototype unit occasionally consists of a smaller number of frames than the frames in the assembly being provided. In such cases, an evaluation of the responses due to the front-to-back, side-to-side, vertical, and torsional modes of the multicabinet assemblies, with respect to those of the tested unit, are made. This evaluation ensures the adequacy of the qualification of the multicabinet assemblies and of the electrical components located within them.

The input motion is applied to the vertical axis, combined with each one of the principal horizontal axes, unless it can be demonstrated that the equipment response along the vertical direction is not sensitive (coupled) to the vibration motion along the horizontal direction and vice versa. Refer to Section 3.7A.3.1.1 for complete details of testing. The maximum input motion acceleration is equal to, or is in excess of, the maximum seismic acceleration expected at the equipment mounting location. Following the requirements of RG 1.100, it is specified that the test response spectrum closely envelop applicable portions of the required response spectrum in verifying the adequacy of test input motion.

3.10A.2.2 Prototype Testing

In some cases, where groups of equipment have similar characteristics, the test program is based upon testing of a prototype item of equipment. The test reports, furnished by the equipment supplier, are reviewed for assurance that the group of components qualified by the prototype is dynamically similar. If any extrapolation as to dimension or mass is used, the vendor is required to justify similarity of the dynamic characteristics.

3.10A.2.3 Analysis

Analysis without testing is acceptable only if structural integrity alone could assure the design-intended function. Responses are calculated for the three-directional seismic loadings individually and combined by the SRSS method. The seismic response is added to the operating load response on an absolute basis to establish the combined effects, and compared with allowable stress, strain, or deflections, as the basis for acceptable qualification.

3.10A.2.4 Combined Analysis and Testing

When the equipment cannot be practically qualified by analysis or testing alone because of its complexity or size, combined analysis and testing is used. When this procedure is employed, the major component is qualified by analysis, and the motors, operators, and appurtenances necessary for operation are qualified by testing. The auxiliary equipment is tested and qualified to the acceleration level at its mounted location, and its equivalent seismic loading is applied to the major component being analyzed.

3.10A.3 Methods and Procedures of Analysis or Testing of Supports of Electrical Equipment and Instrumentation

A design objective, when feasible, is to provide supports for electrical equipment, instrumentation, and control systems with fundamental natural frequencies above the cutoff frequency of the relevant amplified response spectra curves. This ensures that amplification of floor accelerations through supporting members to mounted equipment is minimized.

The response of racks, panels, cabinets, and consoles is considered in assessing the capability of instrumentation and electrical equipment. Items of electrical equipment and instrumentation are tested, wherever feasible, with their supporting structures in their installed configurations. Intermediate support structures are designed to be rigid to preclude dynamic interaction. When it is impractical to design rigid structures, qualification analysis will include the mass and stiffness characteristics of the support. Mounted components are therefore qualified to acceleration levels consistent with those transmitted by their supporting structures.

Determination of amplification and seismic adequacy of instrumentation and electrical equipment is implemented by the analysis and testing methods outlined in Section 3.7A.3.

The Category I cable tray support systems are analyzed using a modal analysis/response spectra method. Mathematical models include both two- and three-dimensional lumped mass models that are subjected to a support excitation generated by applying the amplified response spectra for that structure for the seismic and/or the hydrodynamic loads events. These conditions were considered in designing the cable tray support system in accordance with the applicable loading combinations described in Section 3.8.4. The boundary conditions used in the analysis assume that the system is fixed (i.e., rigidly attached) or pinned depending on the connection to the main structural steel and concrete members at its support points. The procurement and testing requirements for structural steel tray supports are discussed in Section 3.8.4.6.

3.10A.4 Operating License Review

The results of all seismic tests and analyses performed by outside vendors are reviewed and approved. These results become a permanent onsite record. A summary of seismic test and/or analysis results is given in Table 3.10A-1.

3.10B SEISMIC AND HYDRODYNAMIC QUALIFICATIONS OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT (GE SCOPE OF SUPPLY)

3.10B.1 Dynamic Qualification Criteria

3.10B.1.1 Seismic Category I Equipment Identification

Seismic Category I instrumentation and electrical equipment is listed in Table 3.2-1. "Active" NSSS pumps, motors, valves, and valve-mounted equipment are listed in Table 3.9B-3.

Seismic Category I instrumentation and electrical equipment are designed to withstand the faulted event without functional impairment.

The Class 1E instrumentation electrical equipment and support structures supplied by GE requiring seismic qualification are identified in Table 3.10B-1. The seismic qualification of these instrumentation, equipment, and supports is described in the following subsections.

Section 3.9B.2.2 of this FSAR addresses the dynamic qualification testing and analysis of the Category I mechanical components, equipment, and their supports, including the integral or associated electrical components such as valve-mounted components and pump motors.

3.10B.1.2 Dynamic Design Criteria

3.10B.1.2.1 NSSS Equipment

The seismic criterion used in the design and subsequent qualification of all Class 1E instrumentation and electrical equipment supplied by GE is described in the following paragraph.

The Class 1E equipment is capable of performing its safety-related functions during 1) normal plant operation, 2) anticipated transients, 3) design basis accidents, and 4) postaccident operation while being subjected to, and after the cessation of, the accelerations resulting from the seismic and hydrodynamic loads at the point of attachment of the equipment to the building or supporting structure.

The criteria for each of the devices used in the Class 1E systems depend on the use in a given system, e.g., a relay in one system may have as its safety function to deenergize and open its contacts within a certain time, while in another system it must energize and close its contacts. Since GE supplies many devices for many applications, the approach taken was to test the device in the worst case configuration. In this way, the capability of protective action initiation and the proper operation of fail-safe circuits is ensured.

From the basic input ground motion data, a series of response curves at various building elevations is developed after the building layout is completed. Standard requirement levels that meet or exceed the maximum expected unique plant information are included in the design specifications for Seismic Category I equipment. Equipment is qualified dynamically either by GE or by the supplier; in either case, test data, operating experience, and/or calculations substantiate that the components, systems, etc., do not suffer loss of function during or after exposure to seismic and hydrodynamic loads. The magnitude and frequency of the SSE loadings which each component may experience are determined by its specific location within the plant.

3.10B.2 Methods and Procedures for Qualifying Electrical Equipment and Instrumentation

3.10B.2.1 Methods of Showing NSSS Equipment Compliance with IEEE-344-1975 and Regulatory Guide 1.100

RG 1.100 is not the Unit 2 licensing basis for the GE scope of supply. However, GE dynamically reevaluates the equipment to the requirements of IEEE-344-1975. This is accomplished through the GE Seismic Qualification Review Team (SQRT) program.

Under the GE SQRT program, the qualification of recently qualified equipment or equipment yet to be qualified complies with RG 1.100 and IEEE-344-1975. For equipment originally qualified to IEEE-344-1971, the SQRT methodology is applied to the original test data to demonstrate that requirements of IEEE-344-1975 are satisfied also.

If the SQRT requirements are not satisfied for a specific piece of equipment, the equipment is requalified to IEEE-344-1975 or replaced with a component that is qualified to IEEE-344-1975.

Procedures

GE-supplied Class 1E equipment meets the requirement that the dynamic qualification should demonstrate the capability to perform the required safety function during and after the seismic and hydrodynamic loads. Both analysis and testing were used, but most equipment was tested. Analysis was primarily used to determine the adequacy of mechanical strength, such as mounting bolts and pressure boundaries.

Analysis - GE-supplied Class 1E equipment performing primarily a mechanical safety function (pressure boundary devices, etc.) was analyzed since the passive nature of their critical safety role usually made testing unnecessary. Analytical methods outlined in IEEE-344-1975 were utilized in such cases. (See Table 3.10B-1 for indication of which items were qualified by analysis.)

Testing - GE-supplied Class 1E equipment having an active electrical safety function was tested in compliance with IEEE-344-1975.

Documentation

Available documentation verifies the seismic qualification of GE-supplied Class 1E equipment.

3.10B.2.2 Testing Procedures for Qualifying Electrical Equipment and Instrumentation

The test procedure required that the device be mounted on the table of the vibration machine in a manner similar to its normal, installed configuration. The device was tested in the operating states as if it were performing its Class 1E functions; these states were monitored before, during, and after the test to ensure proper function and absence of spurious function. In the case of the relay example, both energized and deenergized states and normally open and normally closed contact configurations were tested if the relay is used in those configurations in its Class 1E functions.

The dynamic excitation was a random multiple frequency test in which the applied vibration was a sinusoidal table motion at a fixed peak acceleration and a discrete frequency at any given time. The vibratory excitation was applied in two orthogonal axes, horizontal and vertical simultaneously, with the axes chosen as those coincident with the most probable mounting configuration. The device was then rotated 90 deg in the horizontal plane, and the test was repeated. Each device, therefore, has been tested in the three major orthogonal axes.

The first step was usually a search for resonances in each axis since resonances cause amplification of the input vibration and are the most likely cause of malfunction. The resonance search was usually run at low acceleration levels to avoid damaging the test sample if a severe resonance was encountered. The resonance search was performed for the applicable frequency range in accordance with IEEE-344; if the device was large enough, the vibrations were monitored by accelerometers placed at critical locations from which resonances were determined by comparing the acceleration level with that at the table of the vibration machine. Sometimes the devices either were too small for an accelerometer, with their critical parts in an inaccessible location, or had critical parts that will be adversely affected by the mounting of an accelerometer. The vibrations were monitored at the closest location.

Following the frequency scan and resonance determination, the devices were tested to determine their dynamic capability limit. For multifrequency testing, five OBE and one SSE tests were run at the appropriate TRS. In some cases, the TRS was increased gradually until device malfunction occurred or the shake table limit was reached. For single frequency testing, a malfunction limit test was run at each resonant frequency as determined by the frequency scan. In this test, the acceleration level was gradually increased until either the device malfunctioned or the limit of the vibration machine was reached. If no resonances were detected (as was usually the case), the device was considered to be rigid (all parts move in unison) and the malfunction limit was therefore independent of frequency. To achieve maximum acceleration from the vibration machine, rigid devices were malfunction tested at the upper test frequency since that allowed the maximum acceleration to be obtained from deflection-limited machines.

The summary of the tests on the devices used in Class 1E applications given in Table 3.10B-1.

The above procedures were required of purchased devices as well as those made by GE. Vendor test results were reviewed and if unacceptable, the tests were repeated either by GE or the vendor. If the vendor tests were adequate, the device was considered qualified to the limits of the test.

3.10B.2.3 Qualification of Valve Operators

The qualification of valve operators is discussed in Section 3.9B.2.2.

3.10B.2.4 Qualification of NSSS Motors

Seismic qualification of NSSS motors is discussed in Section 3.9B.2.2 in conjunction with the ECCS pump and motor assembly. Seismic qualification of the standby liquid control (SLC) pump motor is discussed in Section 3.9B.2.2.2.10 in conjunction with the SLC pump motor assembly.

3.10B.3 Methods and Procedure of Analysis or Testing of Supports of Electrical Equipment and Instrumentation

3.10B.3.1 Dynamic Analysis and Testing Procedures

3.10B.3.1.1 Panel-Mounted Equipment

The Class 1E equipment supplied by GE is used in many systems on many different plants and is subjected to widely varying dynamic loads. The qualification tests were performed to envelop the applicable frequency range. For supports subjected to seismic loads only, the tested frequencies range from 1 to 33 Hz. Where testing below 5 Hz was limited by capability of the test facility, a combination of test and analysis was used to ensure that there were no untested resonances.

For multicabinet assemblies that are too large for the test table, one or two bays of the assembly are tested, giving representative results in the front-to-back and vertical directions. The side-to-side results are evaluated and generally found to be conservative due to the increased flexibility of the narrower section. If conservatism cannot be established, the panel is modeled accurately and a computer analysis of its structural response is performed.

Some GE-supplied Class 1E devices were qualified by analysis only. Analysis was used for passive mechanical devices and was sometimes used in combination with testing for larger assemblies containing Class 1E devices. For example, a test might have been run to determine natural frequencies in the equipment within the critical frequency range. If the equipment was determined to be free of natural frequencies, it was assumed to be rigid. If it had natural frequencies in the critical frequency range, then calculations of transmissibility and responses to varying input accelerations were determined to see if Class 1E devices mounted in the assembly would operate without malfunctioning. Generally, the testing of Class 1E equipment was accomplished using the following procedure.

Assemblies (i.e., control panels and local racks) containing devices with established seismic and hydrodynamic malfunction limits were mounted on the table of a vibration machine in the manner it was to be mounted when in use. All control panel and local rack tests have been performed according to the requirements of IEEE-344. The initial vibration test in each case was a low-level resonance search. As with the devices, the assemblies were tested in the three major orthogonal axes. The resonance search was run in the same manner as described for devices. If resonances were present, the transmissibility between the input and the location of each Class 1E device was determined by measuring the accelerations at each device location and calculating the magnification between it and the input. Once known, the transmissibilities could be used analytically to determine conservatively the input motion at any Class 1E device location for any given input to the base of the assembly.

The described full acceleration level tests showed that the panel types had more than adequate mechanical strength and that acceptability was just a function of its amplification factor and the malfunction levels of the devices mounted in it. Many devices were mounted in the test panel or rack and qualified as an assembly. Other devices were tested individually as previously described. Sometimes panels were tested at lower acceleration levels and the transmissibilities measured to the various devices. By dividing the device's malfunction levels by the panel transmissibility between the device and the panel input, the panel seismic qualification level could be determined. Several high-level tests have been run on selected generic panel designs to ensure the conservatism in using the transmissibility analysis described.

3.10B.4 Operating License Review

The dynamic test results for safety-related panels and control equipment within the NSSS scope are maintained in a permanent file by GE and can be readily audited in all cases. The equipment used in Class 1E applications at Unit 2 passed the prescribed tests.

A summary of the test results for the devices used in Class 1E applications is given in Table 3.10B-1.

3.11 ENVIRONMENTAL QUALIFICATION OF MECHANICAL AND ELECTRICAL EQUIPMENT

Safety-related equipment and components are qualified to meet performance requirements under normal, abnormal, accident, and post-accident environmental conditions for the length of time they are required to function and to remain in a safe mode after their safety function is performed. The environmental conditions for those portions of the plant containing safety-related equipment are given in the Equipment Qualification Environmental Design Criteria (EQEDC), which is supplied separately from the FSAR.

The methodology used to environmentally qualify equipment located in harsh environments, equipment located in mild environments, and safety-related mechanical equipment is described in the Equipment Qualification Document (EQD). Section 6 of the EQD describes the environmental-qualification-related maintenance/surveillance program.

Equipment qualification documentation described herein, including the EQEDC, EQD, System Component Evaluation Work (SCEW) sheets, and applicable portions of the Unit 2 Master Equipment List, are maintained as part of the Unit 2 Equipment Qualification Program. These documents are maintained separately from the FSAR and are not considered part of the FSAR.

3.11.1 Equipment Identification and Environmental Conditions

Environmentally qualified electrical equipment includes all three categories of 10CFR50.49(b)[1]. Safety-related mechanical equipment includes pumps, MOVs, SRVs, and check valves.

A list of all environmentally qualified electrical and mechanical equipment that is located in a harsh environment area is provided in the Unit 2 MEL.

Environmental conditions for the zones where the equipment is located have been calculated for normal, abnormal, and accident conditions and are reported in the EQEDC. Environmental conditions are listed by zones, each zone defining a specific area in the plant. Environmental parameters include temperature, pressure, relative humidity, beta and gamma radiation dose, dose rate and neutron dose. Where applicable, these parameters are given in terms of a time-based profile. A summary presentation of environmental conditions and qualified conditions for the environmentally qualified equipment located in a harsh environment zone is contained in the SCEW sheets.

3.11.2 Qualification Tests and Analyses

3.11.2.1 Qualification

Environmentally qualified electrical equipment that is located in a harsh environment has been qualified by test or other methods as described in IEEE-323 and permitted by 10CFR50.49(f)[1]. Equipment type test is the preferred method of qualification.

Environmentally qualified mechanical equipment that is located in a harsh environment has been qualified by analysis of materials data which are generally based on tests and operating experience.

The qualification methodology is described in detail in Section 4 of the EQD.

The requirements of GDC 1, 4, 23, and 50 of Appendix A to 10CFR50 and Criterion III of Appendix B to 10CFR50 are met as outlined below:

> GDC 1 of 10CFR50, Appendix A, requirements are achieved by incorporating performance, design, construction, and testing requirements into equipment specifications and by the establishment of a system of reviews to ensure conformance with these specified requirements. Appropriate auditable records are maintained in a permanent file. Refer to Chapter 17 for a further definition of how Criterion III of Appendix B to 10CFR50 is met.
>
> GDC 4 requirements are met for harsh environment equipment by designing and qualifying the equipment for satisfactory operation and proper safety function performance during normal, abnormal, test, and DBA environments.
>
> The protection system meets GDC 23. Failure modes and effects analyses have been performed to prove that no single failure results in a loss of the capability of a system to perform its safety function. Both electrical and mechanical failures have been considered from causes such as loss of power supply, loss of control signal, and failures induced by normal, abnormal, accident, and seismic events. Harsh environment equipment has been environmentally qualified to preclude common mode failures.
>
> GDC 50 requirements are achieved by analysis and testing of pressure boundary components to ensure containment integrity.

A discussion of compliance with Regulatory Guides is provided in Section 1.8.

Design Criteria 1, 4, 23, and 50 of Appendix A, 10CFR50, and Criterion III of Appendix B, 10CFR50.

GDC 1 requirements are achieved by incorporation of performance, design, construction, and testing requirements into equipment specifications and by establishment of a system of reviews to ensure conformance with these specified requirements. Appropriate auditable records are maintained in a permanent file. Refer to Chapter 17 for a further definition of how Criterion III of Appendix B to 10CFR50 is met.

The environmental conditions in areas containing safety-related equipment are given in the EQEDC. GDC 4 requirements are met for Class 1E harsh environment equipment by designing and qualifying the equipment for satisfactory operation and proper safety function performance during normal, abnormal, test, and DBA environments.

The protection system meets GDC 23. Failure modes and effects analyses have been performed to prove that no single failure results in a loss of the capability of a safety-related system to perform its safety function. Both electrical and mechanical failures have been considered from causes such as loss of power supply, loss of control signal, and failures induced by normal, abnormal, accident, and seismic events. Class 1E harsh environment equipment has been environmentally qualified to preclude common mode failures.

The recommendations provided in RGs 1.9, 1.30, 1.40, 1.63, 1.73, 1.89, and 1.131 have been utilized by including these recommendations in appropriate equipment specifications. A discussion of compliance with these regulatory guides is provided in Section 1.8.

3.11.2.2 Method of Qualification of Class 1E Equipment and Components

The date of the construction permit Safety Evaluation Report for Unit 2 is prior to July I, 1974; therefore, Unit 2 is a Category II plant under the guidance of NUREG-0588, July 1981, Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment.

The environmental qualification of Class 1E equipment located in harsh environments meets or exceeds the requirements for Category II qualification in accordance with NUREG-0588, including the guidance provided for incorporation of IEEE-323. The environments for which Class 1E equipment must perform its safety function have been specified and used as the basis for environmental qualification.

Details with respect to the qualification methodology and acceptance criteria are provided in the EQD.

3.11.3 Qualification Test Results

The results of qualification tests for environmentally qualified equipment are maintained in an auditable file. A summary presentation of qualification test results for environmentally-qualified electric equipment that is located in a harsh environment is contained in the SCEW sheets.

3.11.4 Loss of Heating, Ventilating, and Air Conditioning

To ensure that HVAC systems does not adversely affect the operability of safety-related controls and electrical equipment in buildings and areas served by safety-related HVAC systems, the HVAC systems serving these areas meet the single-failure criterion.

The HVAC systems and the respective sections where specific safety evaluation details may be found are as follows:

1. Main control room, relay room, standby switchgear rooms, and electrical tunnels (Section 9.4.1).

2. Reactor building and standby gas treatment filter rooms (Section 9.4.2).

3. Diesel generator building (Section 9.4.6).

4. Service water pump bays (Section 9.4.7.2.2).

3.11.5 Estimated Chemical and Radiation Environment

3.11.5.1 Chemical Environment

The chemical composition and resulting pH to which safety-related equipment is exposed during normal operation and design basis accident conditions is reported in Section 2.3 of the EQD.

Sampling stations are provided for periodic analysis of reactor water, refueling and fuel storage pool water, and suppression pool water to assure compliance with operation limits of the plant technical specifications.

3.11.5.2 Radiation Environment

Safety-related systems and components are designed to perform their safety-related function when exposed to normal operational radiation levels and accident radiation levels. The normal operational exposure is based on the radiation sources provided in Chapters 11 and 12.

Radiation sources associated with the DBA and developed in accordance with NUREG-0588, Revision 1 are provided in Chapter 15.

Integrated doses associated with normal plant operation and the DBA condition for various plant compartments are described in the EQEDC.

Safety-related equipment is qualified in accordance with the methodology described in the EQD for the applicable beta, gamma, and neutron doses.

3.11.6 Submergence

Any safety-related equipment which may be submerged due to a LOCA is identified in Section 2.4 of the EQD. This equipment is qualified for submergence as shown on the EQD SCEW sheets.

3.11.7 References

1. Title 10, Code of Federal Regulations, Paragraph 50.49, Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants. Federal Register Vol. 48, No. 15, January 21, 1983.

2. A. J. Szukiewicz, et al. Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment, NUREG-0588, Revision 1, July 1981.

3. NRC Regulatory Guide 1.89, Environmental Qualification of Electrical Equipment Information to Safety for Nuclear Power Plants. Proposed Revision 1, November 1983.

4. NRC Regulatory Guide 1.7, Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident, Revision 2, November 1978.

5. NRC Regulatory Guide 1.97, Revision 3, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.

6. IEEE-323-724, Qualifying Class I Electric Equipment for Nuclear Power Generating Stations.

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APPENDIX 3A

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TABLE 3A.1-1

EXACT AND COMPUTER STRESSES FOR THIN-WALL CYLINDER
SHELL 1 COMPUTER PROGRAM

Variable	Exact	SHELL 1
δR	3.348×10^{-3} in	3.342×10^{-3} in
$\sigma\theta$	3,750 psi	3,750 psi

TABLE 3A.17-3

COMPARISON OF HTLOAD WITH HAND CALCULATION

	HTLOAD	Hand Calculation
Reynolds number	186,700	186,700
Heat transfer coefficient	946.8 Btu/ $^\circ$F-hr-ft^2	946.8 Btu/ $^\circ$F-hr-ft^2

TABLE 3A.17-4

COMPARISON OF HTLOAD WITH CHARTS
OF BROCK AND MCNEILL

Parameter	Charts	HTLOAD
Maximum ΔT_1 (°F)	43.31	45.14
Maximum ΔT_2 (°F)	8.50	8.36

TABLE 3A.17-5

COMPARISON OF HTLOAD WITH TRHEAT

Parameter	TRHEAT	HTLOAD
Maximum ΔT_1(°F)	44.70	45.14
Maximum ΔT_2(°F)	8.69	8.36
Maximum $T_a - T_b$(°F)	19.03	19.08

TABLE 3A.18-1

COMPARISON OF GHOSH-WILSON RESULTS VERSUS THEORETICAL SOLUTIONS FOR A CYLINDER UNDER STATIC INTERNAL PRESSURE

	Radius (R)-ft	Theoretical Solution	Thin-Shell Theory Shell Element	GHOSH-WILSON Results Rectangular Element	GHOSH-WILSON Results Triangular Element
	R=39.5	19.748	-	19.74	19.755
σ_θ	R=40	20.00	20.27	-	-
Ksf	R=40.5	19.249	-	19.24	19.255
σ_r	R=39.5	-0.7357	-	-0.7365	-0.683
Ksf	R=40.5	-0.236	-	-0.2369	-0.188
	R=39	1.82×10^{-3}	-	1.8197×10^{-3}	1.8192×10^{-3}
ΔR	R=40	1.812×10^{-3}	1.87×10^{-3}	1.8111×10^{-3}	1.8110×10^{-3}
ft	R=41	1.804×10^{-3}	-	1.8039×10^{-3}	1.8039×10^{-3}

σ_θ = hoop stress
σ_r = radial stress
ΔR = displacement in radial direction

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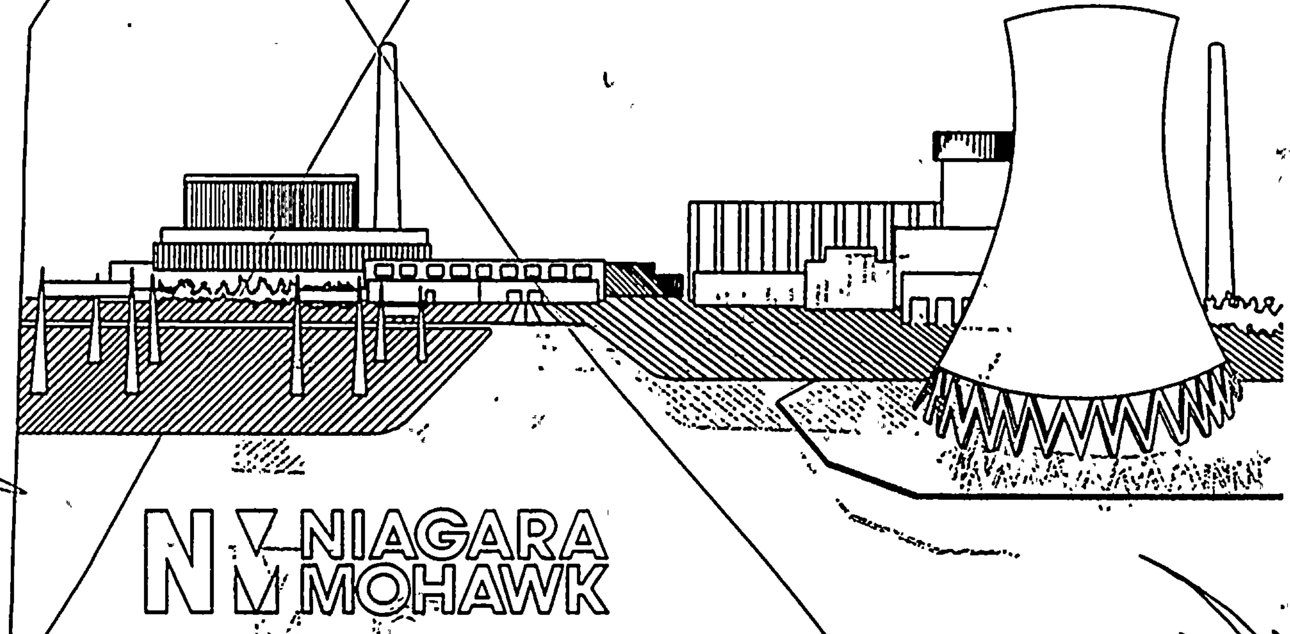

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T 2.5-25	A00	F 2.5-3	A00	F 2.5-48	A00
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T 2B-39 Sh 46	A00	T 2B-39 Sh 92	A00	F 2C-2	A00
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15B.6-2	A26	T 15D-4	R03	15G-iii	R03
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15B.7-1	A26	F 15D-1	R03	15G-2	R03
15B.7-2	R03	F 15D-2	R03	15G-3	R03
15B.8-1	R07	F 15D-3	R03	15G-4	R03
15B.8-2	R07	F 15D-4	R03	15G-5	R03
				15G-6	R03
App 15C	A26	App 15E	A26	15G-7	R03
15C-i	R07	15E-i	R00	15G-8	R03
15C-ii	R07	15E-ii	R00	15G-9	R03
15C-iii	R07	15E-iii	A26	15G-10	R03
15C-1	R07	15E.1-1	A26	15G-11	R03
15C-2	R07	15E.1-2	R00	15G-12	R03
15C-3	R07	15E.2-1	R07	T 15G-1	R03
15C-4	R07	15E.2-2	R07	T 15G-2	R03
15C-5	R07	15E.3-1	R00	T 15G-3	R03
15C-6	R07	15E.3-2	A26	T 15G-4	R03
T 15C-1	R07	15E.3-3	R00	T 15G-5	R03
T 15C-2	R07	15E.3-4	R01	T 15G-6	R03
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F 15C-2	R07	15E.3-6	R01	T 15G-8	R03
F 15C-3	R07	15E.3-7	A26	T 15G-9	R03
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F 15C-6	R07	15E.3-10	R00	T 15G-11 Sh 2	R03
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		17.1-22a	A28	17.1-43	A05
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17.0-3	A15	17.1-23a	A05	17.1-44	A05
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17.1-5b	A05	17.1-31	A09	T 17.1-1 Sh 2	A00
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17.1-7	R00	17.1-32a	A09	T 17.1-1 Sh 4	A00
17.1-8	A28	17.1-32b	A09	T 17.1-1 Sh 5	A00
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18.1-1a	R03	F 18.2-8	R03		
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18.1-2	R03				
18.2-1	R03				
18.2-2	R03				
18.2-3	R03				
18.2-4	R03				
18.2-5	R03				
18.2-6	R03				
18.2-7	R03				
18.2-8	R03				
18.2-9	R03				
18.2-10	R03				
18.2-11	R03				
18.2-12	R03				
18.2-13	R03				
18.2-14	R03				
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6A.1-2a	A09
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6A.1-3	A09
6A.1-4	A00
6A.1-5	A23
T 6A.1-1 Sh 1	R04
T 6A.1-1 Sh 2	R04
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F 6A.1-2	A00
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6A.2-1	A00
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T 6A.2-1a Sh 2	A27
T 6A.2-1b Sh 1	A09
T 6A.2-1b Sh 2	A09
T 6A.2-2 Sh 1	A23
T 6A.2-2 Sh 2	R04
T 6A.2-2 Sh 3	A23
T 6A.2-3 Sh 1	A00
T 6A.2-3 Sh 2	A23
T 6A.2-3 Sh 3	A00
F 6A.2-1	A00
F 6A.2-2	A00
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T 6A.3-3	R05
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T 6A.3-5	R05
T 6A.3-6	R05
F 6A.3-1	R05
F 6A.3-2	R05
F 6A.3-3	R05
F 6A.3-4	R05
F 6A.3-5	R05
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6A.4-2a	R00
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6A.4-3	A00
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6A.4-4a	A21
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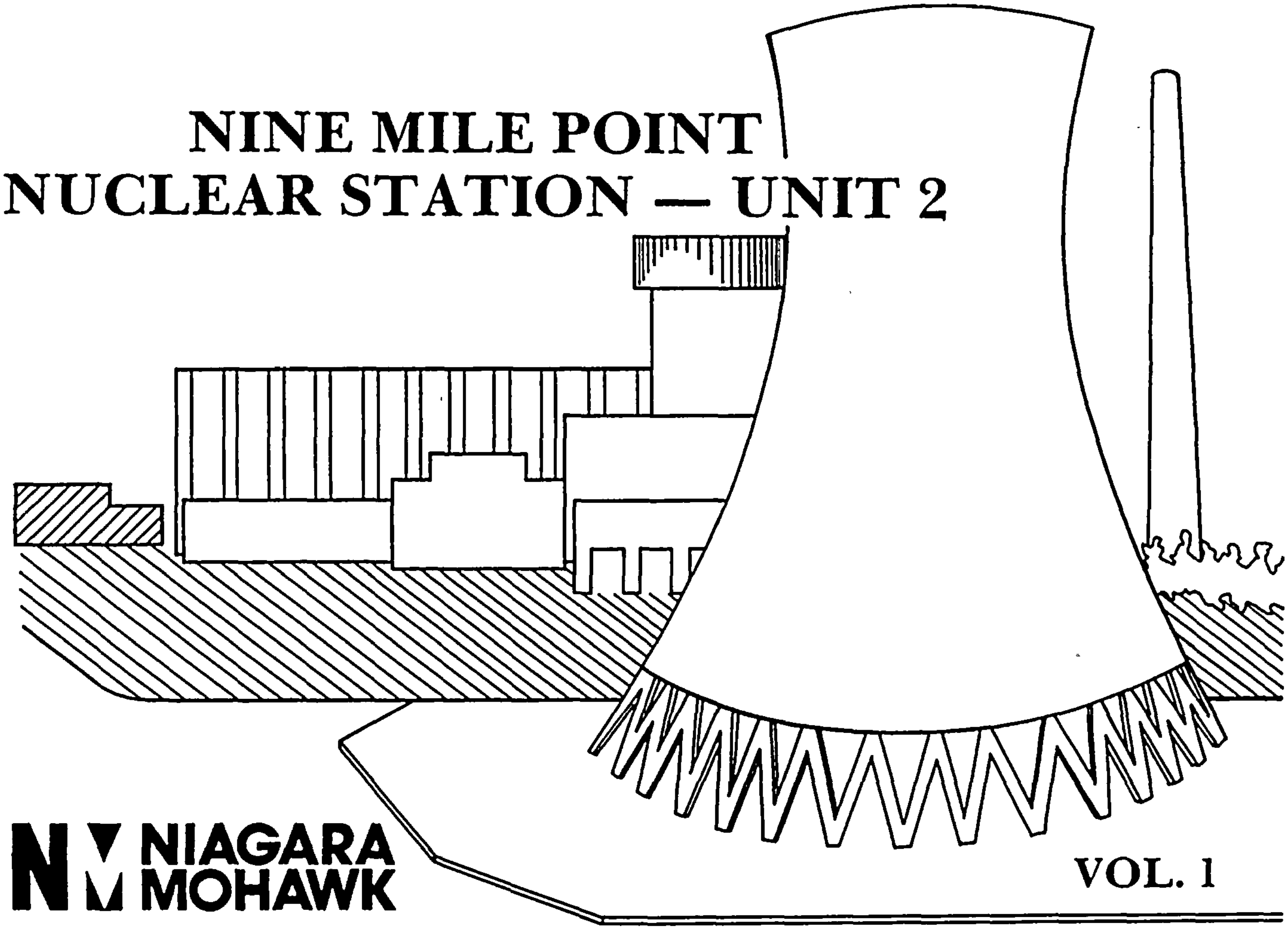

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INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

1.1 INTRODUCTION

This Final Safety Analysis Report (FSAR) is submitted by the Niagara Mohawk Power Corporation (Applicant) and its co-owners (Central Hudson Gas and Electric Corporation, Long Island Lighting Company, New York State Electric and Gas Corporation, and Rochester Gas and Electric Corporation) in support of the application for a Class 103 operating license for the nuclear power station designated Nine Mile Point Nuclear Station - Unit 2 (Unit 2).

Unit 2 is located on a 364-ha (900-acre) site owned by Niagara Mohawk Power Corporation (NMPC), and is situated on the southeast shore of Lake Ontario, Oswego County, NY, approximately 10 km (6.2 mi) northeast of the city of Oswego. Unit 2 and support facilities occupy about 18.2 ha (45 acres), and share the site with the existing Nine Mile Point Nuclear Station - Unit 1 (Unit 1) (Docket No. 50-220) which has been in commercial operation since 1969. The Nine Mile Point site is adjacent to the James A. FitzPatrick Nuclear Power Plant owned by the New York Power Authority (NYPA); Unit 2 is located 274 m (900 ft) east of Unit 1 and about 716 m (2,350 ft) west of the James A. FitzPatrick plant.

Unit 2 employs a nuclear steam supply system (NSSS) consisting of a single-cycle, forced circulating boiling water reactor (BWR). The plant-rated core thermal power level (Figure 1.1-1) is 3,323 MWt corresponding to a net electrical output of 1,080 MWe, and design thermal power of 3,463 MWt corresponding to a gross electrical output of 1,202 MWe. The thermal power used for the plant transient and loss-of-coolant accident (LOCA) analyses is 3,463 MWt. All safety systems have been designed for a thermal power of 3,489 MWt. The NSSS supplier is General Electric Company-Nuclear Energy Operations (GE-NEO). The balance of the plant is designed and constructed by Stone & Webster Engineering Corporation (SWEC). Other plants designed by SWEC that are similar in concept are currently under review by the Nuclear Regulatory Commission (NRC). These are the Shoreham Nuclear Power Station, Brookhaven, Long Island, NY, and the River Bend Station, St. Francisville, LA.

The containment design employs the BWR Mark II concept of over-under pressure suppression with multiple downcomers connecting the reactor drywell to the water-filled pressure suppression chamber. The primary containment is a steel-lined, reinforced concrete enclosure housing the reactor and the suppression pool.

The reactor building completely encloses the primary containment. The structure provides secondary containment when the primary containment is closed and in service, and provides primary containment when the primary containment is open, as during refueling. The reactor building houses the refueling and reactor servicing equipment, new and spent fuel storage facilities, and other reactor auxiliary and service equipment. The primary purpose of the reactor building is to minimize ground-level release of airborne radioactive material.

The outer wall of the reactor building is reinforced concrete up to the crane rail level above the refueling floor. Above the crane rail level, the superstructure is a steel frame using metal wall panels with sealed joints. Access to the building is through airlocks.

The power generation complex includes several contiguous buildings: the reactor building with two auxiliary bays, the control building, the turbine building, and the radwaste building. Other buildings, such as the security facility, are also located in the general plant area. A screenwell for the circulating and service water systems is located approximately 107 m (350 ft) northwest of the centerline of the reactor building.

Condenser cooling for Unit 2 is provided from a counterflow, natural-draft, hyperbolic, concrete cooling tower located approximately 330 m (1,000 ft) south of the centerline of the reactor building. The ultimate heat sink for emergency core cooling is Lake Ontario. Below grade and north of the screenwell building, there are two concrete tunnels that convey the service water intake, service water discharge, and cooling tower blowdown. A safety-related intake pipe is enclosed in each tunnel. The intake pipes extend from the intake shaft approximately 396 m (1,300 ft) northward under Lake Ontario to the submerged intake structures. One tunnel also contains the discharge pipe which extends approximately 550 m (1,800 ft) to the discharge diffuser.

Radionuclides are emitted to the atmosphere from two locations at Unit 2. These are the stack and the combined vent for the radwaste and reactor buildings. Liquid radwaste is stored for decay or concentrated to a solid waste for controlled disposal at regulated storage sites.

The shielding design and plant layout are based on extensive experience of NMPC and SWEC in controlling radiological exposures to as low as reasonably achievable (ALARA) levels. Estimated radiological doses for normal operations and postulated accidents are all fractional parts of the doses listed in federal radiological guidelines for siting and operation of nuclear power plants. Environmental impacts are described in the separate Environmental Report-Operating License Stage (ER-OLS) being submitted for Unit 2.

1.11 ABBREVIATIONS AND ACRONYMS

Table 1.11-1 is a list of abbreviations used in this FSAR.

TABLE 1.11-1

ABBREVIATIONS AND ACRONYMS USED IN FSAR

ADS	Automatic depressurization system
ALARA	As low as reasonably achievable
AOV	Air-operated valve
AP	Annulus pressurization
APRM	Average power range monitor
ARI	Alternate rod insertion
ARMS	Area radiation monitoring system
ATWS	Anticipated transient without scram
BCP	Bottom center pressure
BOC	Beginning of cycle
BSW	Biological shield wall
BTP	Branch technical position
BWR	Boiling water reactor
CAD	Containment atmosphere dilution (device)
CAM	Continuous air monitor
CCW	Closed cooling water
CGCS	Combustible gas control system
CHF	Critical heat flux
CIV	Combined intermediate valve
CIVM	Collision-imported-velocity method
CMFA	Common mode failure analysis
CND	Condensate demineralizer
CO	Condensation oscillation
CPR	Critical power ratio
CRD	Control rod drive
CRDA	Control rod drop accident
CRPI	Control rod position indication
CRVICS	Containment and reactor vessel isolation control system
CUF	Cumulative usage factor
CWS	Circulating water system
DAR	Design Assessment Report for Hydrodynamic Loads
DB	Design basis
DBA	Design basis accident
DBE	Design basis earthquake
DBFL	Design basis flood level
DCDT	Direct current differential transducer
DER	Double-ended rupture
DG	Diesel generator
DRMS	Digital radiation monitoring system
EAB	Exclusion area boundary

TABLE 1.11-1 (Cont)

ECA	Engineering change authorization
ECCS	Emergency core cooling system
ECN	Engineering change notice
EFCV	Excess flow check valve
EHC	Electrohydraulic control
EIC	Energy Information Center
EOC	End of cycle
EOF	Equivalent occurrence factor
EPA	Electric protective assembly
EPZ	Emergency planning zone
EQD	Environmental qualification document
ERF	Emergency response facility
ESF	Engineered safety feature
ETS	Emergency trip system
FA	Full arc (mode of TCV operation)
FAS	Fluid actuator system
FATT	Fracture appearance transition temperature
FCD	Functional control diagram
FDDR	Field deviation disposition request
FLECHT	Full-length emergency cooling heat transfer
FMEA	Failure modes and effects analysis
FMH	Fixture mounting height
FPCC	Fuel pool cooling and cleanup
FPS	Fire protection system
FSAR	Final safety analysis report
GDC	General design criterion
GE	General Electric Company
GETAB	GE thermal analysis basis
HAZ	Heat affected zone
HCU	Hydraulic control unit
HDFM	Heavy density fill material
HELB	High energy line break
HEM	Homogeneous equilibrium model
HEPA	High-efficiency particulate air/absolute (filter)
HEPCO	Hydro-Electric Power Commission of Ontario
HPCI	High pressure coolant injection
HPCS	High pressure core spray
HPU	Hydraulic power unit
HX	Heat exchanger
HVAC	Heating, ventilating, and air conditioning
HVRS	Reactor building ventilation system
IAC	Interim acceptance criteria (NRC)
IAS	Instrument air service

TABLE 1.11-1 (Cont)

IBA	Intermediate break accident
ICC	Inadequate core cooling
IDC	Incident detection circuitry
IDS	Instrument data sheet
IED	Instrument and electrical drawing
IGSCC	Intergranular stress corrosion cracking
ILRT	Integrated leakage rate test
IPCEA	Insulated Power Cables Engineers Association
IRM	Intermediate range monitor
LCO	Limiting condition of operation
LCS	Leakage control system
LDS	Leak-detection system
LFMG	Low frequency motor generator
LHGR	Linear heat generation rate
LOCA	Loss-of-coolant accident
LOFW	Loss of feedwater
LOOP (LOP)	Loss of offsite power
LPAP	Low power alarm point
LPCI	Low pressure coolant injection
LPCS	Low pressure core spray
LPDS	Loose parts detection system
LPRM	Local power range monitor
LPSP	Low power set point
LPZ	Low population zone
LSA	Low specific activity
LSD	Lake survey datum (of 1935)
LSSS	Limiting safety system setting
LTC	Load tap changing (mechanism)
LWS	Liquid radwaste system
MAPLHGR	Maximum average planar linear heat generation rate
MBA	Misplaced bundle accident
MCC	Motor control center
M/CC	Maintenance and calibration communication (system)
MCPR	Minimum critical power ratio
MG	Motor generator set
MLD	Mean low water datum
MLHGR	Maximum linear heat generation rate
MMI	Modified Mercalli intensity
MOI	Method of images
MOV	Motor operated valve
MPC	Maximum permissible concentration
MSIV	Main steam isolation valve
MSIV-LCS	Main steam isolation valve leakage control system

TABLE 1.11-1 (Cont'd)

msl	Mean sea level
MSL	Main steam line
MSLB	Main steam line break
MTV	Mechanical trip valve
NB	Nuclear boiler
NBR	Nuclear boiler rated (power)
NBS	National Bureau of Standards
NDL	Nuclear data link
NDT	Nil ductility transition
NDTT	Nil ductility transition temperature
NED	Nuclear energy division (GE)
NIOSH	National Institute for Occupational Safety and Health
NMS	Neutron monitoring system
NPRDS	Nuclear plant reliability data system
NPSH	Net positive suction head
NRV	Nonreturn valve
NSS	Nonnuclear safety
NSOA	Nuclear safety operational analysis
NSSS	Nuclear steam supply system
NUMAC RWM	Nuclear measurement analysis and control rod worth minimizer
OBE	Operating basis earthquake
OFS	Orificed fuel support
ORE	Occupational radiation exposures
OT	Operational transient
PA	Public address (system)
PAM	Post-accident monitoring
PASNY	Power Authority of the State of New York
PCI	Pellet-cladding interaction
PCIOMR	Preconditioning cladding interim operating management recommendation
PCRVICS	Primary containment and reactor vessel isolation control system
PCS	Process computer system
PCT	Peak cladding temperature
p.f.	Power factor
PGCC	Power generating control center
P&ID	Piping and instrumentation diagram
PLU	Power load unbalance
PMF	Probable maximum flood
PMS	Probable maximum surge
PMWS	Probable maximum windstorm
PP/PA	Page party/public address (system)
PQL	Product quality checklist
PRM	Power range monitor

TABLE 1.11-1 (Cont'd)

PSAR	Preliminary safety analysis report
PSD	Power spectrum density
PTPD	Project test program objectives
PVS	Plant vent stack
PWR	Pressurized water reactor
QA	Quality assurance
QC	Quality control
RAB	Restricted area boundary
RBCLCW	Reactor building closed loop cooling water (system)
RBM	Rod block monitor
RBPC	Reactor building polar crane
RCIC	Reactor core isolation cooling
RCPB	Reactor coolant pressure boundary
RCS	Reactor coolant system
RCSCM	RHR containment spray cooling mode
RDCS	Rod drive control system
RH	Relative humidity
RHR	Residual heat removal
RMCS	Reactor manual control system
RMS	Radiation monitoring system
RMS	Root mean square
RPC	Rod pattern controller
RPIS	Rod position information system
RPS	Reactor protection (trip) system
RPT	Recirculation pump trip
RPV	Reactor pressure vessel
RRCS	Redundant reactivity control system
RSCM	RHR reactor shutdown cooling mode
RSCS	Rod sequence control system
RSO	Reactor system outline
RSPCM	RHR suppression pool cooling mode
RSS	Remote shutdown system
RWCU	Reactor water cleanup
RWP	Radiation work permit
SACF	Single active component failure
SAR	Safety analysis report
SBA	Small break accident
SCA	Single-channel analyzer
SCBA	Self-contained breathing apparatus
SDIV	Scram discharge instrument volume
SDV	Scram discharge volume
SEF	Single equipment failure

TABLE 1.11-1 (Cont'd)

SFC	Spent fuel pool cooling and cleanup system
SGTS	Standby gas treatment system
SLC	Standby liquid control
SMSA	Standard metropolitan statistical area
SOE	Single operator error
SOF	Single operator failure
SORC	Station Operations Review Committee
SPC	Sound-powered communication (system)
SPDS	Safety parameter display system
SPG	Substitute position generator
SRAB	Safety Review and Audit Board
SRDI	Safety-related display instrumentation
SRM	Source range monitor
SRM	Security-related materials
SRP	Standard Review Plan
SRSS	Square root of the sum of the squares
SRV	Safety/relief valve
SRVDL	Safety/relief valve discharge line
SS	Safe shutdown
SSE	Safe shutdown earthquake
SWP	Service water system
TBCLCW	Turbine building closed loop cooling water
TCV	Turbine control valve
TG	Turbine generator
TIP	Traversing incore probe
TLD	Thermoluminescent dosimeter
TSS	Temperature sensor/switch
TSVC	Turbine stop valve closure
UHS	Ultimate heat sink
UPS	Uninterruptible power supply
ZPA	Zero period asymptote

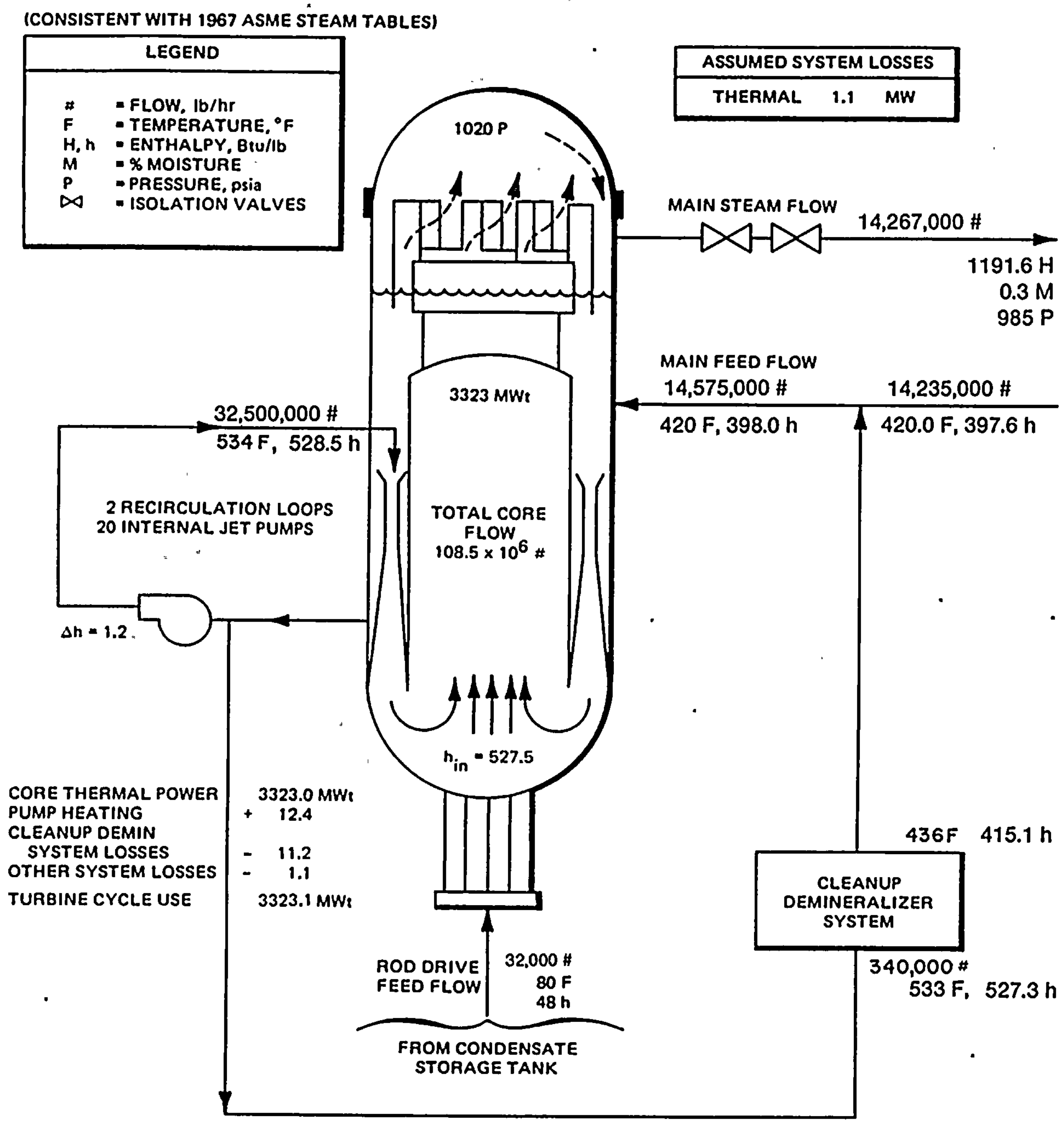

FIGURE 1.1-1

HEAT BALANCE AT RATED POWER

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

accidents that release radioactive material into the primary containment volume.

16. It is possible to test primary containment integrity and leak-tightness at periodic intervals.

17. A reactor building is provided that completely encloses both the primary containment and the fuel storage areas. The secondary containment includes a method for controlling release of radioactive materials from the barrier and includes a capability for filtering radioactive materials within the barrier.

18. The reactor building is designed to act as a radioactive material barrier, if required, when the primary containment is open for expected operational purposes.

19. The primary containment and reactor building, in conjunction with other engineered safeguards, limits radiological effects of accidents resulting in the release of radioactive material to the primary containment volume to significantly less than the requirements of 10CFR100.

20. Provisions are made for removing energy from within the primary containment to maintain the integrity of the primary containment system following accidents that release energy to the primary containment.

21. Piping that penetrates the primary containment structure and serves as a path for the uncontrolled release of radioactive material to the environs is automatically isolated whenever such potential for radioactive material release exists. Such isolation is effected in time to limit radiological effects to significantly less than the requirements of 10CFR100.

22. The emergency core cooling system (ECCS) is provided to limit fuel cladding temperature to 2,200°F as a result of a LOCA.

23. The ECCS provides for continuity of core cooling over the complete range of postulated break sizes in the RCPB.

24. The ECCS is diverse, reliable, and redundant.

25. Operation of the ECCS is initiated automatically when required, regardless of the availability of offsite power.

26. The main control room is shielded against radiation to permit continued occupancy under accident conditions.

27. In the event that the main control room becomes uninhabitable, it is possible to bring the reactor from power range operation to a cold shutdown condition by manipulating local controls and equipment available outside the main control room.

28. Backup reactor shutdown capability is provided independent of normal reactivity control provisions. This backup system shuts down the reactor from any normal operating condition and maintains the shutdown condition.

1.2.1.3 System-by-System Approach

The principal architectural and engineering criteria for design are summarized below on a system-by-system or system group basis. The system-by-system presentation facilitates understanding of the actual design of any one system. Only the most restrictive of any related criteria are stated for a system. Where the most restrictive criterion is classified as a power generation consideration, less restrictive safety criteria may not be stated in the system-by-system presentation. However, the actual design of a system must reflect all criteria that pertain to it.

1.2.1.3.1 Nuclear System Criteria

Principal design criteria for the reactor, ECCS, RCPB, and reactivity control systems are as follows:

1. The nuclear system is designed to support a GE BWR rated at 3,323 MWt.

2. Fuel cladding is designed to retain integrity as a radioactive material barrier throughout the design power range. Fuel cladding is designed to accommodate, without loss of integrity, the pressures generated by the fission gases released from fuel material throughout the design life of the fuel.

3. Fuel cladding, in conjunction with other unit systems, is designed to retain integrity throughout any abnormal operational transient.

4. Those portions of the nuclear system that form part of the RCPB are designed to retain integrity as a radioactive material barrier following abnormal operational transients and accidents.

5. Heat removal systems including the ECCS and makeup water supplies are provided in sufficient capacity, redundancy, and operational adequacy to remove heat generated in the reactor core for the full range of normal operational conditions from unit shutdown to design power and for any abnormal operational transient

The auxiliary boiler building (Figure 1.2-34), located north of the screenwell building, houses the electric boilers and accessories to supply steam to the plant during shutdown.

The standby gas treatment building and railroad access area (Figures 1.2-35 and 1.2-36) house the standby gas treatment filters and associated equipment and allow access for spent fuel shipping.

The condensate storage tank building (Figure 1.2-37) houses the condensate storage tanks and associated equipment.

The natural-draft cooling tower (Figures 1.2-38 and 1.2-39) provides the normal heat sink for heat transferred to the circulating water system from the main condensers.

The auxiliary service building (Figures 1.2-7 and 1.2-8), adjacent to the reactor building, houses the heating, ventilating and air conditioning (HVAC) room and decontamination and shower facilities for personnel.

The decontamination area (Figures 1.2-19 through 1.2-21, 1.2-23, and 1.2-24), immediately south of the radwaste building, provides the facility for decontamination of large tools and equipment, and a sample room. It also houses clean steam reboilers and related equipment.

The hydrogen storage area (for hydrogen cooling of the turbine generator, Figure 1.2-40) is located west of the offgas area. The hydrogen storage bottles are mounted on concrete pads and are in a fenced area.

1.2.4 Nuclear Steam Supply System

The nuclear system includes a direct-cycle, forced circulation, GE BWR that produces steam for direct use in the steam turbine. A heat balance showing the major parameters of the nuclear system for the warranted power condition is shown on Figure 1.1-1.

The NSSS is further discussed in Chapters 4 and 5.

1.2.4.1 Reactor Core and Control Rods

The reactor fuel and core design are described in Section 2 of Reference 5 and Section 1 of Reference 6.

Experience has shown that the control rods are not susceptible to distortion and have an average life expectancy many times the residence time of a fuel loading.

1.2.4.2 Reactor Vessel and Internals

The reactor vessel contains the core and supporting structures; the steam separators and dryers; the jet pumps; the control rod guide tubes; the distribution lines for the feedwater, core sprays, and standby liquid control; the in-core instrumentation; and other components. The main connections to the vessel include the steam lines, coolant recirculation lines, feedwater lines, CRD and in-core nuclear instrument housings, core spray lines, residual heat removal (RHR) lines, standby liquid control line, core differential pressure line, jet pump pressure-sensing lines, and water level instrumentation.

The reactor vessel is designed and fabricated in accordance with applicable codes for a pressure of 1,250 psig. The nominal operating pressure in the steam space above the separators is 1,020 psia. The vessel is fabricated of low-alloy steel and is clad internally with stainless steel (except for the top head nozzles and nozzle weld zones which are unclad).

The reactor core is cooled by demineralized water that enters the lower portion of the core and boils as it flows upward around the fuel rods. The steam leaving the core is dried by steam separators and dryers located in the upper portion of the RPV. The steam is then directed to the turbine through the main steam lines. Each steam line has two isolation valves in series, one on either side of the primary containment barrier.

1.2.4.3 Reactor Recirculation System

The reactor recirculation system consists of two recirculation pump loops external to the RPV. These loops provide the piping path for the driving flow of water to the RPV jet pumps. Each external loop contains one high-capacity motor-driven recirculation pump, two motor-operated maintenance valves, and one hydraulically-operated flow control valve. The variable position hydraulic flow control valve operates in conjunction with a low-frequency motor generator (MG) set to control reactor power level through the effects of coolant flow rate on moderator void content.

The jet pumps are RPV internals. They provide a continuous internal circulation path for the major portion of the core coolant flow. The jet pumps are located in the annular region between the core shroud and the vessel inner wall. Any recirculation line break still allows core flooding to approximately two-thirds of the core height, the level of the inlet of the jet pumps.

1.2.4.4 Residual Heat Removal System

The RHR system is a system of pumps, heat exchangers, and piping that fulfills the following functions:

1. Removes decay and sensible heat during and after plant shutdown.

water temperature, purity, clarity, and level. This process prevents the spent fuel from overheating and the buildup of excessive radioactive materials in the cooling water, thereby minimizing radiation levels.

The system includes two heat exchangers, each of which is capable of removing the full decay heat from a normal refueling offload of spent fuel. A cross-connection to the RHR system provides additional emergency backup cooling and cooling during a full core offload.

Chapter 9 gives further details of the fuel handling and storage system.

1.2.8 Power Conversion System

Chapter 10 provides a detailed discussion of the following equipment systems.

1.2.8.1 Turbine Generator

The turbine is a 1,800-rpm tandem-compound, six-flow, single-stage reheat unit with an electrohydraulic governor control. The turbine generator has an emergency trip system for turbine overspeed. The output of the turbine generator is 1,165,663 kWe at turbine guarantee conditions with 2.0 in Hg abs backpressure and 0 percent makeup.

The generator is a direct-driven, three-phase, 60-Hz, 25,000-V, 1,800-rpm hydrogen inner-cooled, synchronous generator rated at 1,348,400 kVA at 0.90 power factor, 0.58 short-circuit ratio at maximum hydrogen pressure of 75 psig.

1.2.8.2 Main Steam System

The main steam system delivers steam from the nuclear boiler system through four 26-/28-in OD steam lines to the turbine generator, turbine bypass valves, SJAEs, offgas preheaters, steam seal evaporator, and radwaste steam reboiler.

1.2.8.3 Main Condenser

The main condenser maintains 2.0 in Hg abs when operating at turbine guarantee conditions with 67.97°F circulating water inlet temperature. The condenser includes provisions for accepting steam bypassed around the turbine generator. Deaeration of condensate is accomplished in the condenser.

1.2.8.4 Main Condenser Air Removal System

The main condenser air removal system, using air ejectors for normal operation and vacuum hogging pumps for startup, evacuates gases from the main turbine and condenser during plant startup and maintains the condenser essentially free of gases during operation. This system handles all in-leakage of noncondensable gases through the turbine seals, condensate, feedwater, and steam systems, and noncondensables that are generated in the reactor by disassociation of water.

1.2.8.5 Turbine Gland Sealing System

The turbine gland sealing system provides mildly radioactive steam to the seals of the turbine throttle valve stem glands and the turbine shaft glands. The sealing steam is supplied by a clean steam reboiler using condensate. The unit auxiliary boiler provides an auxiliary steam supply for startup and when reactor heating steam is not available. The steam packing exhauster collects and condenses the air and steam mixture and discharges the air and other noncondensables to the plant exhaust duct to the atmosphere, using a motor-driven exhauster.

1.2.8.6 Steam Bypass System and Pressure Control System

A turbine bypass system is provided which passes steam directly to the main condenser under control of the pressure regulator. Steam is bypassed to the condenser whenever the reactor steaming rate exceeds the load passed to the turbine generator. The capacity of the turbine bypass system is 25 percent of the turbine rated steam flow. The pressure regulation system provides main turbine control valve and bypass valve flow demands to maintain a nearly constant reactor pressure during normal plant operation. It also provides demands to the recirculation system to adjust power levels by changing reactor recirculation flow rates.

1.2.8.7 Circulating Water System

The circulating water system (CWS) provides the condenser with a continuous supply of cooling water. The CWS is a pumped closed loop system utilizing an air-cooled natural-draft cooling tower as a heat sink. Six one-sixth capacity circulating water pumps are provided to pump cooling water from the cooling tower basin through the main condenser and back to the top of the cooling tower. Makeup water is provided from Lake Ontario by the service water system.

1.2.8.8 Condensate and Feedwater Systems

The condensate and feedwater systems supply condensate from the condenser hotwell to the RPV. The condensate is pumped by two of the three condensate pumps through the full flow condensate demineralizer system, the intercooler of the air ejectors, and the steam packing exhauster to the condensate booster pumps. The condensate booster pumps pump the flow through three strings consisting of two drain coolers and five stages of low-pressure heaters each. In addition, three heater drain pumps provide approximately one-third of the feedwater flow requirements. The last low-pressure heaters discharge to the suction of three

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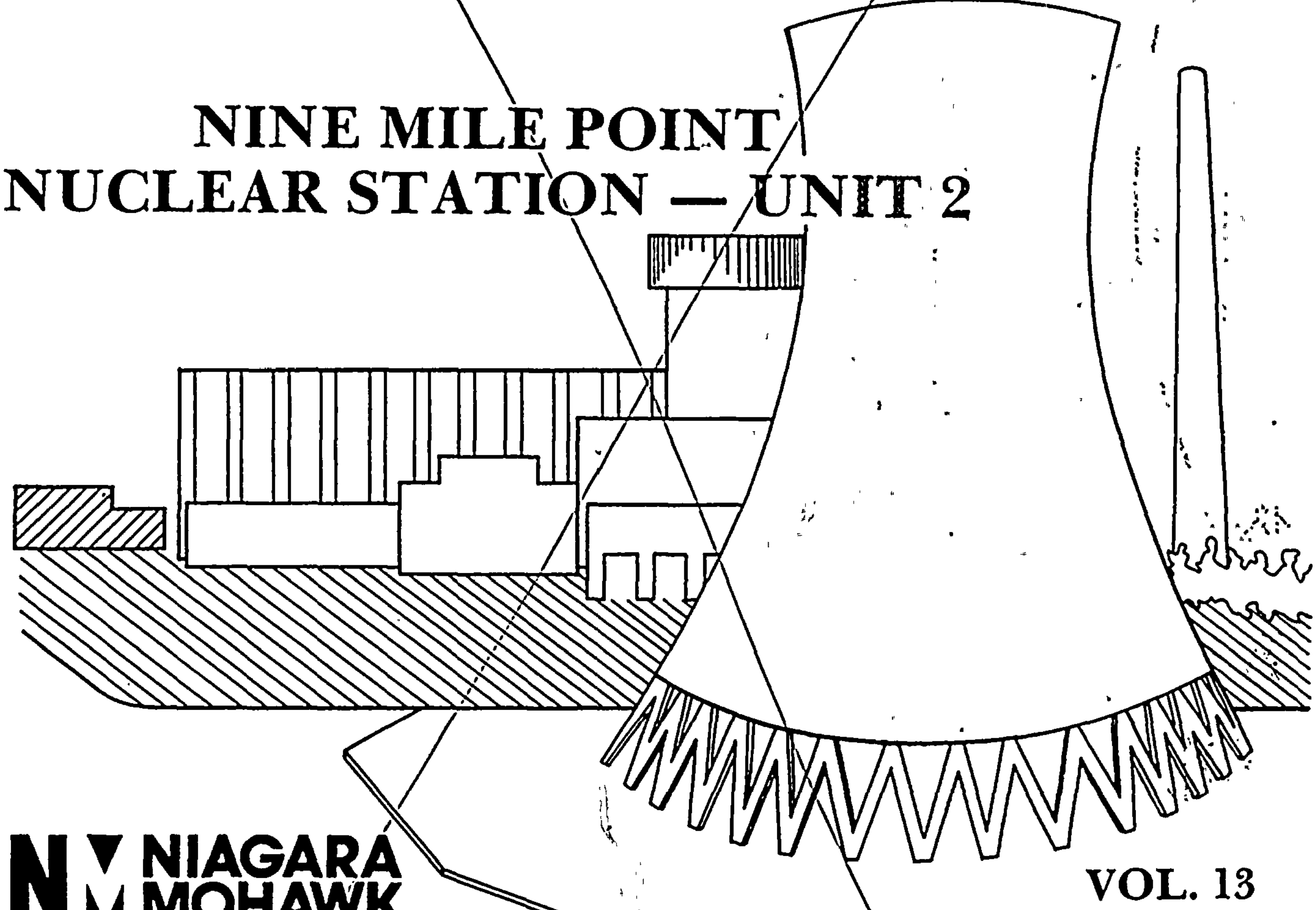
UPDATED SAFETY ANALYSIS REPORT

NINE MILE POINT NUCLEAR STATION — UNIT 2



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CHAPTER 5

REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

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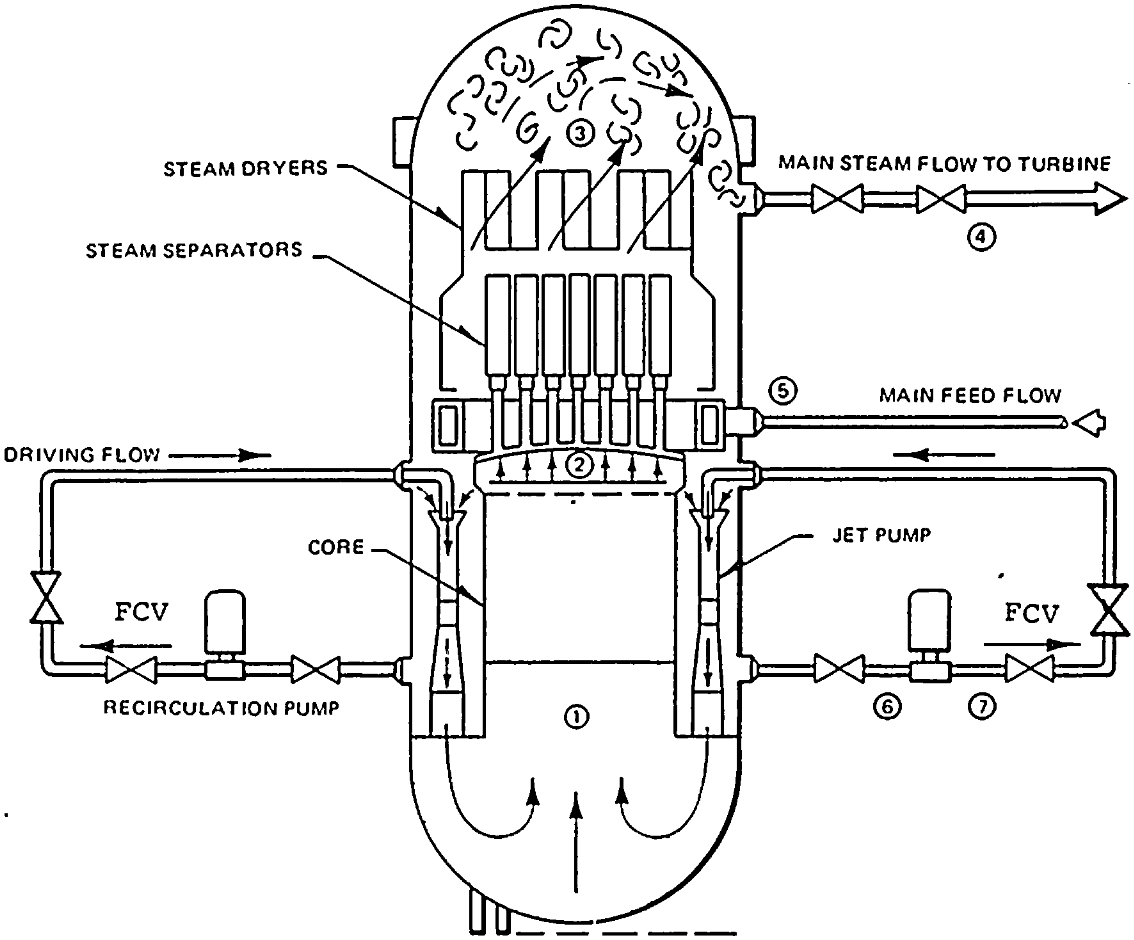


	PRESSURE (psia)	FLOW (lb/hr)	TEMPERATURE (°F)	ENTHALPY (Btu/lb)
1. CORE INLET	1056	108.5×10^6*	533	527.5
2. CORE OUTLET	1031	108.5×10^6	548	632.0
3. SEPARATOR OUTLET (STEAM DOME)	1020	14.3×10^6	547	1191.6
4. STEAM LINE (2ND ISOLATION VALVE)	985	14.3×10^6	543	1191.6
5. FEEDWATER INLET (INCLUDES RWCU RETURN FLOW)	1045	14.6×10^6	420	398.0
6. RECIRCULATING PUMP SUCTION	1025	35.7×10^6	533	527.3
7. RECIRCULATING PUMP DISCHARGE	1303	35.7×10^6	534	528.5

*CHANNEL BYPASS – NOMINALLY 10%

FIGURE 5.1-1a

RATED OPERATING CONDITIONS OF THE BOILING WATER REACTOR

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

Edition up to and including Winter 1972 Addenda. The essential ASME requirements which are all met by this analysis are discussed as follows.

It is recognized that the protection of vessels in a nuclear power plant is dependent upon many protective systems to relieve or terminate pressure transients. Installation of pressure-relieving devices may not independently provide complete protection. The safety valve sizing evaluation assumes credit for operation of the scram protective system which may be tripped by either one of two sources, i.e., a direct or flux trip signal. The direct scram trip signal is derived from position switches mounted on the MSIVs, or the turbine stop valves, or from pressure switches mounted on the dump valve of the turbine control valve (TCV) hydraulic actuation system. The position switch settings are less than or equal to 85 percent fully open for MSIVs and less than or equal to 90 percent fully open for the turbine stop valves. The pressure switches are actuated when a fast closure of the TCVs is initiated. Further, no credit is taken for power operation of the SRVs in the relief mode. Credit is taken for the dual-purpose SRVs in the safety mode.

The rated capacity of the SRVs is sufficient to prevent a rise in pressure within the protected vessel of more than 110 percent of the design pressure (1.10 x 1,250 psig = 1,375 psig) for events defined in Chapter 15 and Appendix A.

Full account is taken of the pressure drop on both the inlet and discharge sides of the valves. All SRVs discharge into the suppression pool through a discharge pipe from each valve which is designed to achieve sonic flow conditions through the valve, thus providing flow independence to discharge piping losses. Additional measures to counteract the effects of backpressure in the SRV discharge lines are discussed in Sections 5.2.2.2.3 and 5.2.2.4.1.

The method described in Reference 5 shows that sonic flow is achieved through a SRV with the following dimensions:

Nozzle Bore	4.84 in (d_o)
Valve Discharge Diameter (at outlet flange)	~10 in (d_1)
Valve Inlet Diameter	~8 in (based on sweepolet)

The SRV steam flows from the steam line, a large reservoir, through the sweepolet (same diameter as inlet flange), into the valve nozzle, and out through the outlet flange. The nozzle has a short flow length, and it acts as a standard nozzle or venturi tube. Values of critical pressure ratio, r_c, are found as a function of d_o/d_1 and k on page A-21 of the reference. The value of r_c, the ratio of discharge backpressure, P_2, to inlet pressure, P_1, decreases as d_o/d_1 decreases.

The value of d_o/d_1 is minimized when the outlet valve flange diameter is used as the value of d_1. In this case, $d_o/d_1 = 4.84/10 = 0.484$.

The critical pressure for sonic flow occurs where $r_c = 0.553$ (using k = 1.3 for steam) and, therefore, sonic flow occurs when $P_2 \leq (0.553)\ P_1$ (P_1, P_2, in psia).

SRV discharge lines are required to be designed and configured so that the discharge backpressure at the valve outlet is not greater than 40 percent of the inlet pressure using pressures measured in psig. For absolute pressure, the corresponding limit is less than 41 percent of inlet across the range of operating conditions.

This limit and discharge line design ensures that r_c will not be exceeded. Therefore, sonic flow is ensured.

Table 5.2-3 lists the systems that could initiate during the design basis overpressure event.

5.2.2.2 Design Evaluation

5.2.2.2.1 Method of Analysis

The model used to analyze overpressurization is provided in Section S.2.3 of GESTAR II[1].

5.2.2.2.2 System Design

A parametric study was conducted to determine the required steam flow capacity of the SRVs based on the following assumptions. Cycle-specific information is covered in Appendix A, Section A.5.2.2.2.2. Operation with a single recirculation system loop in operation, or with one MSIV out of service, has also been evaluated. See Appendices 15B and 15D, respectively.

Operating Conditions

The operating conditions are:

1. Operating power = 3,466 MWt (104.3 percent of nuclear boiler rated power).

2. Vessel dome pressure = 1,020 psig.

3. Steam flow = 15.013 x 10^6 lb/hr (105 percent of nuclear boiler rated steam flow).

These conditions are the most severe because maximum stored energy exists at these conditions. At lower power conditions, the transients would be less severe.

A transient analysis study has been performed for a typical BWR to investigate the effects of increasing the initial reactor pressure on the peak transient vessel pressure. Two models, one from the REDY and one from the ODYN codes, were used in the study. The model in the REDY code is more conservative than that in the ODYN code. The conclusion, even for the more conservative model, was that increasing the initial operating pressure up to the high-pressure scram setpoint (analytical upper limit of 1,071 psig) results in an increase of the peak system pressure of less than half the initial pressure increase for the overpressure design transient (i.e., all MSIV closure with indirect high neutron flux scram). The same general trend is expected to exist for Nine Mile Point Nuclear Station - Unit 2 (Unit 2). Since there is a significant margin (107 psi by comparing the peak vessel pressure of 1,268 psig with the ASME Code limit of 1,375 psig) for Unit 2, no safety concern would result from the above-assumed initial dome pressure.

Transients

The overpressure protection system must accommodate the most severe pressurization event described in Section S.3 of GESTAR II(1). Table 5.2-4 lists the sequence of events for this worst-case transient, the MSIV closure with flux scram, based on the installed SRV capacity.

Safety/Relief Valve Transient Analysis Specification

1. Valve groups: Spring-action safety mode - 5 groups

2. Spring pressure setpoint (maximum safety limit) and number of valves per group:

Group 1:	1,177 psig	2 SRVs
Group 2:	1,187 psig	4 SRVs
Group 3:	1,197 psig	4 SRVs
Group 4:	1,207 psig	4 SRVs
Group 5:	1,217 psig	4 SRVs

The setpoints are assumed at a conservatively-high level above the nominal setpoints. This is to account for initial setpoint errors and any instrument setpoint drift that might occur during operation. Typically, the assumed setpoints in the analysis are 1 to 2 percent above the actual nominal setpoints. Conservative SRV response characteristics are also assumed.

Safety/Relief Valve Capacity

Sizing of the SRV capacity is based on establishing an adequate margin from the peak vessel pressure to the vessel code limit (1,375 psig) in response to the reference transients.

Reference 7 provides sufficient information and documentation to show compliance with all requirements of Article NB-7000 of the

ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Division 1, 1971 Edition, with Addenda to and including Winter 1972, in the area of overpressure protection design of the Unit 2 nuclear pressure vessel and other RCPB components. The effects of valve capacity on the pressure transients are shown also.

The overpressure protection analysis also includes the simulation of anticipated transient without scram (ATWS) recirculation pump trip (RPT) on high reactor pressure.

5.2.2.2.3 Evaluation of Results

Safety/Relief Valve Capacity

The required SRV capacity is determined by analyzing the pressure rise from a MSIV closure with flux scram transient as documented in Section S.3 of GESTAR II[1]. Results of this analysis are shown on Figure 5.2-1.

Pressure Drop in Inlet and Discharge

Pressure drop on the piping from the reactor vessel to the valves is taken into account in calculating the maximum vessel pressures. Pressure drop in the discharge piping to the suppression pool is limited by proper discharge line sizing to prevent backpressure on each SRV from exceeding 40 percent of the valve inlet pressure, thus assuring choked flow in the valve orifice and no reduction of valve capacity due to the discharge piping. Each SRV has its own separate discharge line.

Cycle-specific evaluation is covered in Appendix A, Section A.5.2.2.2.3.

5.2.2.3 Piping and Instrument Diagrams

Figures 5.2-2 and 10.1-3 show the schematic location of pressure-relieving devices for:

1. Reactor coolant system.

2. Primary side of the auxiliary or emergency systems interconnected with the primary system.

3. Any blowdown or heat dissipation system connected to the discharge side of the pressure-relieving devices.

The schematic arrangements of the SRVs are shown on Figures 5.2-2 and 5.2-3.

3. Included with the SRV is an instruction manual provided to the customer. Within this manual will be recommended periodic maintenance programs as recommended by the manufacturer based upon his experience.

All SRVs will be subject to the following tests and inspections in accordance with an approved program:

During every refueling outage, at least 50 percent of the installed valves will be tested for verification of set-pressures opening and closing using the pneumatic power actuator, testing of all bolted closures, and testing of pneumatic actuator leakage.

After the preceding testing, the valves will undergo preventative maintenance in accordance with an approved procedure.

All disassembled valves will be inspected for wear, damage, and erosion. All gaskets, seals, and parts will be replaced as needed in accordance with inspection results. Valves will be relapped, as required, and lubricated. All disassembled valves will be retested, and appropriate adjustments will be made prior to use.

It is not feasible to test the SRV setpoints while the valves are in place. The valves are mounted on 1,500-lb primary service rating flanges. They can be removed for maintenance or bench checks and reinstalled during normal plant shutdowns. The valves will be tested to check set-pressure in accordance with the requirements of the plant Technical Specifications. All valves tested will be reset to ±1 percent of the safety set-pressures listed in the Technical Specifications. A value of ±3 percent of the safety set-pressure will be used to determine an increase in the test sample size as specified in the code adopted by the in-service inspection/in-service testing (ISI/IST) program. The external surface and seating of all SRVs are 100-percent visually inspected when the valves are removed for maintenance or bench checks. Valve operability is verified during the preoperational test program as discussed in Chapter 14.

A discussion of SRV operability testing for two-phase flow, in accordance with NUREG-0737, is provided in Section 1.10, Task II.D.1.

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 Material Specifications

Table 5.2-5 lists the principal pressure-retaining components and materials and the appropriate material specifications for the RCPB components.

5.2.3.2 Compatibility with Reactor Coolant

5.2.3.2.1 PWR Chemistry of Reactor Coolant

Not applicable to boiling water reactors (BWRs).

5.2.3.2.2 BWR Chemistry of Reactor Coolant

Materials in the RCS are primarily austenitic stainless steel, carbon steel, and Zircaloy cladding. The reactor water chemistry limits are established to provide an environment favorable to these materials. Limits are placed on conductivity and chloride concentrations. Conductivity is limited because it can be continuously and reliably measured and gives an indication of abnormal conditions and the presence of unusual materials in the coolant. Chloride limits are specified to prevent stress corrosion cracking of stainless steel[2].

The water quality requirements are supported by General Electric Company (GE) stress corrosion test data summarized as follows:

1. Type 304 stainless steel specimens were exposed in a flowing loop operating at 537°F. The water contained 1.5 ppm chloride and 1.2 ppm oxygen at a pH of 7. Test specimens were bent beam strips stressed over their yield strength. After 2,100-hr exposure, no cracking or failures occurred.

2. Welded Type 304 stainless steel specimens were exposed in a refreshed autoclave operating at 550°F. The water contained 0.5 ppm chloride and 1.5 ppm oxygen at a pH of 7. Uniaxial tensile test specimens were stressed at 125 percent of their 550°F yield strength. No cracking or failures occurred at 15,000-hr exposure.

When conductivity is in its normal range, pH, chloride, and other impurities affecting conductivity are also within their normal range[2]. When conductivity becomes abnormal, chloride measurements are made to determine whether they are also out of their normal operating values. Conductivity may be high due to the presence of a neutral salt which does not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are high because of the purposeful use of additives. In BWRs, however, where no additives are used and where near-neutral pH is maintained, conductivity provides a good and prompt measure of the quality of the reactor water. Significant changes in conductivity provide the Operator with a warning mechanism so he can investigate and remedy the condition before reactor water limits are reached. Methods available to the Operator for correcting the out-of-specification condition include operation of the RWCU system or placing the reactor in the cold shutdown condition. The major benefit of cold shutdown is to reduce the temperature-dependent corrosion rates and

Conservative corrosion allowances are provided for all exposed surfaces of carbon and low-alloy steels.

Contaminants in the reactor coolant are controlled to very low limits by the reactor water quality specifications. No detrimental effects will occur to any of the materials from allowable contaminant levels in the high-purity reactor coolant. Expected radiolytic products in the BWR coolant have no adverse effects on the construction materials.

5.2.3.2.4 Compatibility of Construction Materials with External Insulation and Reactor Coolant

Construction materials exposed to external insulation are:

1. Solution-annealed austenitic stainless steels. Types 304, 304L with 0.035-percent maximum of carbon content, 316 and 316K with 0.02-percent weight maximum of carbon content.

2. Carbon and low-alloy steel.

Two types of external insulation are employed on BWRs. Stainless steel reflective metal insulation used does not contribute to any surface contamination and has no effect on construction materials. Similarly, the fibrous (nonmetallic) insulation is encapsulated in metal sheeting which prevents direct contact with the RCS materials. In addition, the fibrous insulation used is assessed to meet the requirements of RG 1.36, and has the proper ratios of leachable sodium and silicate ions to chloride and fluoride ions.

Since there are no additives in the BWR coolant, leakage would expose materials to high-purity, demineralized water. Exposure to demineralized water would cause no detrimental effects.

5.2.3.3 Fabrication and Processing of Ferritic Materials

5.2.3.3.1 Fracture Toughness

Materials in the RCPB, other than the RPV, are required by 10CFR50.55a and Appendix G to meet the fracture toughness requirements of ASME Section III, NB-2300. These fracture toughness requirements for ferritic piping, valve, bolting, and pump materials are met as follows:

1. Piping and weld filler materials are in accordance with ASME Section III, NB-2300, 1974 Edition; field weld filler materials are to 1974 Edition.

2. Valves are in accordance with ASME III, NB-2300, as follows:

a. Motor-operated valves (MOVs), Winter 1975, except as noted in Appendix 5A, 1971 Edition and Winter 1973 Addenda.

b. MSIVs, Summer 1977.

c. Carbon Steel Manual Valves, Winter 1973.

3. Materials for bolts with diameters exceeding 1 in meet the 25-mil lateral expansion requirement of ASME Section III, NB-2300, of the same code date as the associated equipment. In addition, bolting greater than 1 in is required to meet a minimum of 45 ft-lb absorbed energy.

4. There are no ferritic pumps in the RCPB.

The fracture toughness properties of the RPV are discussed in Section 5.3.1 and Appendix 5A.

5.2.3.3.2 Control of Welding

Control of Preheat Temperature Employed for Welding of Low-Alloy Steel (Regulatory Guide 1.50)

RG 1.50 delineates preheat temperature control requirements and welding procedure qualifications supplementing those in ASME Sections III and IX.

The use of low-alloy steel is restricted to the RPV. Other ferritic components in the RCPB are fabricated from carbon steel materials. Preheat temperatures employed for welding of low-alloy steel meet or exceed the recommendations of ASME Section III, Appendix D. Components were either held for an extended time at preheat temperature to assure removal of hydrogen, or preheat was maintained until postweld heat treatment. The minimum preheat and maximum interpass temperatures were specified and monitored.

Control of Electroslag Weld Properties (Regulatory Guide 1.34)

No electroslag welding was performed on RCPB components.

Welder Qualification for Areas of Limited Accessibility (Regulatory Guide 1.71)

Qualification for areas of limited accessibility is discussed in Section 5.2.3.4.2.

These satisfy position c.8.

Plant Technical Specifications comply with position c.9 by specifying limiting conditions for identified and unidentified leakage and by addressing the availability of various types of instruments to assure adequate coverage.

Regulatory Guide 1.22 Assessment

The proper operation of the LDS sensors and logic is verified during the preoperational tests and during plant operation. Each temperature switch (both ambient and differential types) that provides isolation signals is connected to one element of a dual thermocouple. A light illuminates when the temperature exceeds the setpoint. Verification of the thermocouple input is accomplished by comparing the reading from the trip channel with the recorder channel which is connected to the other element of the dual thermocouple. The trip logics are tested by applying a simulated trip signal from an external source to the LDS channel. Keylock test switches are used to prevent the isolation signal from performing its isolating function.

5.2.6 References

1. General Electric Standard Application for Reactor Fuel - United States Supplement, NEDE-24011-P-A-US, (latest approved revision).

2. Skarpelos, J. M. and Bagg, J. W. Chloride Control in BWR Coolants, NEDO-10899, June 1973.

3. GEAP-5620, Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws, by M. B. Reynolds, April 1968.

4. Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants, NUREG-76/067, NRC/PCSG, October 1975.

5. Crane Technical Paper No. 410, 1974 Edition.

6. Pressure Relieving Device Certifications, National Board of Boiler and Pressure Vessel Inspectors, 1979 Edition.

7. General Electric Design Report 22A7122, Overpressure Protection Report, Revision 2.

TABLE 5.2-2

NUCLEAR SYSTEM SAFETY/RELIEF SETPOINTS*

No. of Valves	Spring Set Pressure (psig)	ASME Rated Capacity @ 103% Spring Set Pressure (lb/hr each)	Pressure Setpoint for Power-Actuated Mode (psig)
2	1,148	882,000	1,076
4	1,175	902,000	1,086
4	1,185	910,000	1,096
4	1,195	917,000	1,106
4	1,205	925,000	1,116

NOTE: Seven of the SRVs are used for the automatic depressurization function.

* Cycle-specific values are covered in Appendix A, Table A.5.2-1.

TABLE 5.2-3

SYSTEMS THAT MAY INITIATE DURING OVERPRESSURE EVENT

Cycle-specific information is covered in Appendix A Table A.5.2-2

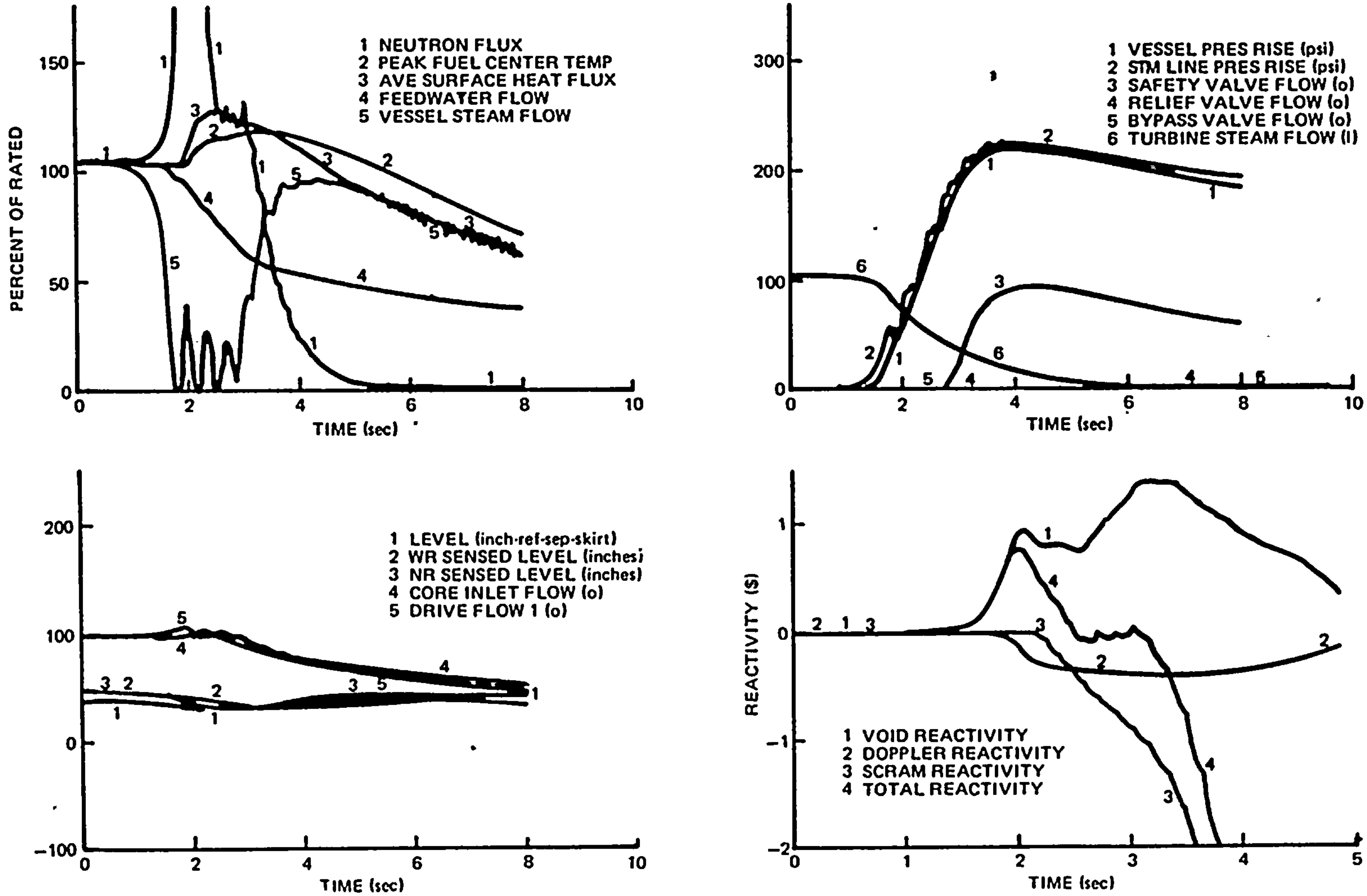
System	Initiating/Trip Signal(s)	Setpoint
RPS	Reactor trips with high flux	
RCIC	ON with reactor low water level	≥L2
	OFF with reactor high water level	≤L8
HPCS	ON with reactor low water level	≥L2
	ON with high drywell pressure	≤2 psig
	OFF with reactor high water level	≤L8
Recirculation	OFF with reactor low water level	<L2
	OFF with reactor high pressure	>1050 psig
RWCU	OFF with reactor low water level	<L2

TABLE 5.2-4

SEQUENCE OF EVENTS FOR FIGURE 5.2-1

Cycle-specific values are covered in Appendix A, Table A.5.2-3

Time (sec)	Event
0	Closure of all MSIVs was initiated.
0.3	MSIVs reached 85% open. Failure of direct position scram was assumed.
1.7	Neutron flux reached APRM flux setpoint and initiated reactor scram.
2.29	Sensed reactor dome pressure reached setpoint of recirculation pump trip.
2.35	Recirculation pump/motor initiated to coast down.
2.7	Steam line pressure reached Group 1 SRVs pressure setpoint (spring-action safety mode), while power-actuated relief mode was ignored. (See Section 5.2.2.2.2.)
3.3	SRVs all opened due to high pressure.
3.68	Vessel bottom pressure reached its peak value.



NOTE: For cycle-specific results see Appendix A, Figure A.5.2-1.

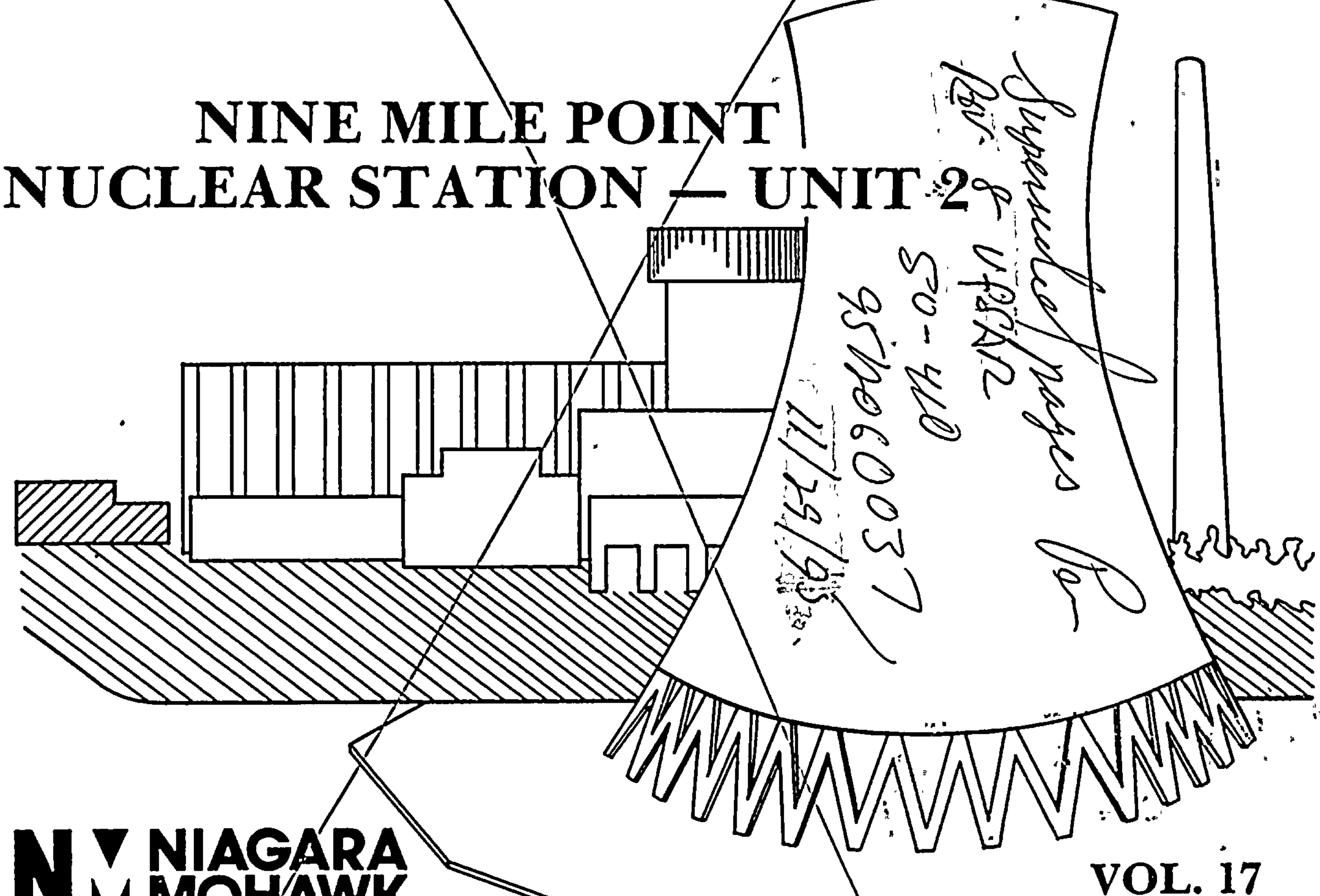
FIGURE 5.2-1

SAFETY RELIEF VALVE CAPACITY SIZING TRANSIENT "MSIV CLOSURE WITH HIGH FLUX TRIP"

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

UPDATED SAFETY ANALYSIS REPORT

NINE MILE POINT NUCLEAR STATION — UNIT 2



NIAGARA MOHAWK

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TABLE 8.3-1 (Cont'd.)

Power Source: 2EJS*US1

					Design Basis Accident Diesel Generator Loading Possibilities									Nonaccident Loading		
					Simultaneous LOOP & LOCA			LOOP With Delayed LOCA			LOCA With Delayed LOOP			LOOP With Unit Trip		
Equip ID	Desc.	Rating Units	Volts Phase	ST PF RUNPF	STKW1 STKVR1	STTIME1	RUNKW1 RUNKVR1	STKW2 STKVR2	STTIME2	RUNKW2 RUNKVR2	STKW3 STKVR3	STTIME3	RUNKW3 RUNKVR3	STKW4 STKVR4	STTIME4	RUNKW4 RUNKVR4
2EJS* X1A	Load Center Xfrmr	1500. KVA	-0- 3	0.2 0.85	3600. 18000.	T = 0	19. 0.	3600. 18000.	T = 0	19. 0.	3600. 18000.	T = 0	19. 0.	3600. 18000.	T = 0	19. 0.
2EJS* X1B	Load Center Xfrmr	1500. KVA	-0- 3	0.2 0.85	3600. 18000.	T = 0	19. 0.	3600. 18000.	T = 0	19. 0.	3600. 18000.	T = 0	19. 0.	3600. 18000.	T = 0	19. 0.
2HCS* PNL22A	Hydrogen Recomb	120. KW	575. 3	1. 1.	120. 0.	T>=2.5 Hr Man	120. 0.	120. 0.	T>=2.5 Hr Man	120. 0.	120. 0.	T>=2.5 Hr Man	120. 0.	0. 0.	Not Req'd	0. 0.
2HVK* CHL1A	Control Bldg. Chiller 1A	180. KW	575. 3	0.35 0.85	378. 1012.	T = 30	153. 95.4	378. 1012.	T = 30	153. 95.4	378. 1012.	T = 30	153. 95.4	378. 1012.	T = 30	153. 95.4
2HVR* UC413A	Reactor Bldg. Unit Cooler A	150. HP	575. 3	0.35 0.85	315. 843.	T = 25	127.5 79.5	315. 843.	T = 25	127.5 79.5	315. 843.	T = 25	127.5 79.5	315. 843.	T = 25	127.5 79.5
2LAR-PNL200	Lighting Panel	200. KVA	600. 3	0.2 0.9	0. 0.	Note 1	0. 0.	-0- -0-	Note 1	-0- -0-	-0- -0-	Note 1	-0- -0-	480. 2400.	T = 0	180. 87.1

TABLE 8.3-1 (Cont'd.)

Total Load On: 2EJS*US1

Design Basis Accident Diesel Generator Loading Possibilities									Nonaccident Loading		
Simultaneous LOOP & LOCA			LOOP With Delayed LOCA			LOCA With Delayed LOOP			LOOP With Unit Trip		
Starting KW KVAR	Running KW KVAR	Time Span	Starting KW KVAR	Running KW KVAR	Time Span	Starting KW KVAR	Running KW KVAR	Time Span	Starting KW KVAR	Running KW KVAR	Time Span
7200 36000	0 0	0<=T<0.1 sec	7200 36000	0 0	0<=T<0.1 sec	7200 36000	0 0	0<=T<0.1 sec	7680 38400	0 0	0<=T<0.1 sec
0 0	38 0	0.1<=T<25 sec	0 0	38 0	0.1<=T<25 sec	0 0	38 0	0.1<=T<25 sec	0 0	218 87	0.1<=T<25 sec
315 843	38 0	25<=T<30 sec	315 843	38 0	25<=T<30 sec	315 843	38 0	25<=T<30 sec	315 843	218 87	25<=T<30 sec
693 1855	38 0	30<=T<31 sec	693 1855	38 0	30<=T<31 sec	693 1855	38 0	30<=T<31 sec	693 1855	218 87	30<=T<31 sec
378 1012	165 79	31<=T<36 sec	378 1012	165 79	31<=T<36 sec	378 1012	165 79	31<=T<36 sec	378 1012	346 167	31<=T<36 sec
0 0	318 175	36<=T<9000 sec	0 0	318 175	36<=T<9000 sec	0 0	318 175	36<=T<9000 sec	0 0	499 262	36 sec<=T
120 0	318 175	9000<=T<9006 sec	120 0	318 175	9000<=T<9006 sec	120 0	318 175	9000<=T<9006 sec	0 0	0 0	
0 0	438 175	9006 sec<=T	0 0	438 175	9006 sec<=T	0 0	438 175	9006 sec<=T	0 0	0 0	

TABLE 8.3-1 (Cont'd.)

Power Source: 2EHS*MCC103

					Design Basis Accident Diesel Generator Loading Possibilities									Nonaccident Loading		
					Simultaneous LOOP & LOCA			LOOP With Delayed LOCA			LOCA With Delayed LOOP			LOOP With Unit Trip		
Equip ID	Desc.	Rating Units	Volts Phase	ST PF RUNPF	STKW1 STKVR1	STTIME1	RUNKW1 RUNKVR1	STKW2 STKVR2	STTIME2	RUNKW2 RUNKVR2	STKW3 STKVR3	STTIME3	RUNKW3 RUNKVR3	STKW4 STKVR4	STTIME4	RUNKW4 RUNKVR4
2RHS* MOV113	Shutdown Cooling Supply Outboard Isolat.	19.2 HP	575. 3	0.6 0.85	115.2 153.6	T = 0	32.64 20.35	115.2 153.6	T = 0	32.64 20.35	115.2 153.6	T = 0	32.64 20.35	115.2 153.6	T=>10 min (MAN)	32.64 20.35
2RHS* MOV12A	Heat Exchgr A to Reactor	0.7 HP	575. 3	0.6 0.85	4.2 5.6	T=>10 min (MAN)	1.2 0.74	4.2 5.6	T=>10 min (MAN)	1.2 0.74	4.2 5.6	T=>10 min (MAN)	1.2 0.74	4.2 5.6	T=>10 min (MAN)	1.2 0.74
2RHS* MOV142	Heat Exchgr B to LWS	0.33 HP	575. 3	0.6 0.85	2. 2.6	T = 0	0.56 0.35	2. 2.6	T = 0	0.56 0.35	2. 2.6	T = 0	0.56 0.35	2. 2.6	T=>10 min (MAN)	0.56 0.35
2RHS* MOV15A	Cont. Spray A	2.6 HP	575. 3	0.6 0.85	15.6 20.8	T=>10 min (MAN)	4.4 2.76	15.6 20.8	T=>10 min (MAN)	4.4 2.76	15.6 20.8	T=>10 min (MAN)	4.4 2.76	15.6 20.8	T=>10 min (MAN)	4.4 2.76
2RHS* MOV1A	Supp. Pool to RHR Pump A	1.6 HP	575. 3	0.6 0.85	9.6 12.8	T=>10 min (MAN)	2.7 1.7	9.6 12.8	T=>10 min (MAN)	2.7 1.7	9.6 12.8	T=>10 min (MAN)	2.7 1.7	9.6 12.8	T=>10 min (MAN)	2.7 1.7
2RHS* MOV22A	Steam Cond. to Exchgr A	1.6 HP	575. 3	0.6 0.85	9.6 12.8	T = 0	2.7 1.7	9.6 12.8	T = 0	2.7 1.7	9.6 12.8	T = 0	2.7 1.7	9.6 12.8	T=>10 min (MAN)	2.7 1.7
2RHS* MOV23A	Steam Cond. to Exchgr A	1.6 HP	575. 3	0.6 0.85	9.6 12.8	T = 0	2.7 1.7	9.6 12.8	T = 0	2.7 1.7	9.6 12.8	T = 0	2.7 1.7	9.6 12.8	T=>10 min (MAN)	2.7 1.7
2RHS* MOV24A	LPCI Inlet A	6.6 HP	575. 3	0.6 0.85	39.6 52.8	T = 0	11.22 7.	39.6 52.8	T = 0	11.22 7.	39.6 52.8	T = 0	11.22 7.	0. 0.	Not Req'd	-0- -0-
2RHS* MOV25A	Cont. Spray A	2.6 HP	575. 3	0.6 0.85	15.6 20.8	T = 0	4.4 2.76	15.6 20.8	T = 0	4.4 2.76	15.6 20.8	T = 0	4.4 2.76	0. 0.	Not Req'd	-0- -0-
2RHS* MOV26A	Heat Exchgr A Vent To Supp Pool	0.13 HP	575. 3	0.6 0.85	0.78 1.	T=>10 min (MAN)	0.22 0.14	0.78 1.	T=>10 min (MAN)	0.22 0.14	0.78 1.	T=>10 min (MAN)	0.22 0.14	0.78 1.	T=>10 min (MAN)	0.22 0.14

TABLE 8.3-1 (Cont'd.)

Total Load On: 2EHS*MCC103

Design Basis Accident Diesel Generator Loading Possibilities									Nonaccident Loading		
Simultaneous LOOP & LOCA			LOOP With Delayed LOCA			LOCA With Delayed LOOP			LOOP With Unit Trip		
Starting KW KVAR	Running KW KVAR	Time Span	Starting KW KVAR	Running KW KVAR	Time Span	Starting KW KVAR	Running KW KVAR	Time Span	Starting KW KVAR	Running KW KVAR	Time Span
792 1681	0 0	0<=T<6 sec	792 1681	0 0	0<=T<6 sec	792 1681	0 0	0<=T<6 sec	515 1311	0 0	0<=T<6 sec
0 0	281 175	6<=T<9 sec	0 0	281 175	6<=T<9 sec	0 0	281 175	6<=T<9 sec	0 0	202 125	6<=T<120 sec
11 15	280 174	9<=T<15 sec	11 15	280 174	9<=T<15 sec	11 15	280 174	9<=T<15 sec	0 0	191 118	120<=T<600 sec
0 0	283 176	15<=T<120 sec	0 0	283 176	15<=T<120 sec	0 0	283 176	15<=T<120 sec	304 408	191 118	600<=T<606 sec
0 0	199 123	120<=T<600 sec	0 0	199 123	120<=T<600 sec	0 0	199 199	120<=T<600 sec	0 0	277 173	606<=T<720 sec
76 105	199 123	600<=T<606 sec	76 105	199 123	600<=T<606 sec	76 105	199 199	600<=T<606 sec	0 0	200 124	720<=T<1800 sec
0 0	221 137	606<=T<720 sec	0 0	221 137	606<=T<720 sec	0 0	221 137	606<=T<720 sec	4 6	200 124	1800<=T<1806 sec
0 0	202 125	720 sec<=T	0 0	202 125	720 sec<=T	0 0	202 125	720 sec<=T	0 0	201 125	1806<=T<1920 sec
0 0	0 0		0 0	0 0		0 0	0 0		0 0	200 124	1920 sec<=T

TABLE 8.3-2 (Cont'd.)

Power Source: 2EHS*MCC303

					Design Basis Accident Generator Loading Possibilities									Nonaccident Loading		
					Simultaneous LOOP & LOCA			LOOP With Delayed LOCA			LOCA With Delayed LOOP			LOOP With Unit Trip		
Equip ID	Desc.	Rating Units	Volts Phase	ST PF RUNPF	STKW1 STKVR1	STTIME1	RUNKW1 RUNKVR1	STKW2 STKVR2	STTIME2	RUNKW2 RUNKVR2	STKW3 STKVR3	STTIME3	RUNKW3 RUNKVR3	STKW4 STKVR4	STTIME4	RUNKW4 RUNKVR4
2RHS* MOV115	SW Inject. to Reactor	1.6 HP	575. 3	0.6 0.85	0. 0.	Not Req'd	0. 0.	0. 0.	Not Req'd	0. 0.	0. 0.	Not Req'd	0. 0.	0. 0.	Not Req'd	0. 0.
2RHS* MOV116	SW Inject. to Reactor	1.6 HP	575. 3	0.6 0.85	0. 0.	Not Req'd	0. 0.	0. 0.	Not Req'd	0. 0.	0. 0.	Not Req'd	0. 0.	0. 0.	Not Req'd	0. 0.
2RHS* MOV12B	Heat Exchgr B to Reactor	0.7 HP	575. 3	0.6 0.85	4.2 5.6	T=>10 min (MAN)	1.2 0.74	4.2 5.6	T=>10 min (MAN)	1.2 0.74	4.2 5.6	T=>10 min (MAN)	1.2 0.74	4.2 5.6	T=>10 min (MAN)	1.2 0.74
2RHS* MOV149	Heat Exchgr B to LWS	0.33 HP	575. 3	0.6 0.85	2. 2.7	T = 0	0.56 0.35	2. 2.7	T = 0	0.56 0.35	2. 2.7	T = 0	0.56 0.35	2. 2.7	T=>10 min (MAN)	0.56 0.35
2RHS* MOV15B	Cont. Spray B	2.6 HP	575. 3	0.6 0.85	15.6 20.8	T=>10 min (MAN)	4.42 2.74	15.6 20.8	T=>10 min (MAN)	4.42 2.74	15.6 20.8	T=>10 min (MAN)	4.42 2.74	-0- -0-	Not Req'd	-0- -0-
2RHS* MOV1B	Supp. Pool to RHR Pump B	4. HP	575. 3	0.6 0.85	24. 32.	T=>10 min (MAN)	6.8 4.2	24. 32.	T=>10 min (MAN)	6.8 4.2	24. 32.	T=>10 min (MAN)	6.8 4.2	24. 32.	T=>10 min (MAN)	6.8 4.2
2RHS* MOV1C	Supp. Pool to RHR Pump C	1.6 HP	575. 3	0.6 0.85	0. 0.	Not Req'd	0. 0.	0. 0.	Not Req'd	0. 0.	0. 0.	Not Req'd	0. 0.	0. 0.	Not Req'd	0. 0.
2RHS* MOV22B	Steam Cond. to Exchgr B	1.6 HP	575. 3	0.6 0.85	9.6 12.8	T = 0	2.72 1.69	9.6 12.8	T = 0	2.72 1.69	9.6 12.8	T = 0	2.72 1.69	9.6 12.8	T=>10 min (MAN)	2.72 1.69
2RHS* MOV23B	Steam Cond. to Exchgr B	1.6 HP	575. 3	0.6 0.85	9.6 12.8	T = 0	2.72 1.69	9.6 12.8	T = 0	2.72 1.69	9.6 12.8	T = 0	2.72 1.69	9.6 12.8	T=>10 min (MAN)	2.72 1.69
2RHS* MOV24B	LPCI Inlet B	6.6 HP	575. 3	0.6 0.85	39.6 52.8	T = 0	11.22 6.95	39.6 52.8	T = 0	11.22 6.95	39.6 52.8	T = 0	11.22 6.95	-0- -0-	Not Req'd	-0- -0-
2RHS* MOV24C	LPCI Inlet C	6.6 HP	575. 3	0.6 0.85	39.6 52.8	T = 0	11.22 6.95	39.6 52.8	T = 0	11.22 6.95	39.6 52.8	T = 0	11.22 6.95	-0- -0-	Not Req'd	-0- -0-

TABLE 8.3-2 (Cont'd.)

Power Source: 2EHS*MCC303

					Design Basis Accident Diesel Generator Loading Possibilities									Nonaccident Loading		
					Simultaneous LOOP & LOCA			LOOP With Delayed LOCA			LOCA With Delayed LOOP			LOOP With Unit Trip		
Equip ID	Desc.	Rating Units	Volts Phase	ST PF RUNPF	STKW1 STKVR1	STTIME1	RUNKW1 RUNKVR1	STKW2 STKVR2	STTIME2	RUNKW2 RUNKVR2	STKW3 STKVR3	STTIME3	RUNKW3 RUNKVR3	STKW4 STKVR4	STTIME4	RUNKW4 RUNKVR4
2RHS* MOV25B	Cont. Spray B	2.6 HP	575. 3	0.6 0.85	15.6 20.8	T->10 min (MAN)	4.4 2.7	15.6 20.8	T->10 min (MAN)	4.4 2.7	15.6 20.8	T->10 min (MAN)	4.4 2.7	-0- -0-	Not Req'd	-0- -0-
2RHS* MOV26B	Heat Exchngr B Vent to Supp. Pool	0.13 HP	575. 3	0.6 0.85	0.78 1.	T->10 min (MAN)	0.22 0.14	0.78 1.	T->10 min (MAN)	0.22 0.14	0.78 1.	T->10 min (MAN)	0.22 0.14	0.78 1.	T = 0	0.22 0.14
2RHS* MOV27B	Heat Exchngr B Vent to Supp. Pool	0.13 HP	575. 3	0.6 0.85	0.78 1.	T->10 min (MAN)	0.22 0.14	0.78 1.	T->10 min (MAN)	0.22 0.14	0.78 1.	T->10 min (MAN)	0.22 0.14	0.78 1.	T = 0	0.22 0.14
2RHS* MOV2B	Reactor to RHR Pump B	0.83 HP	575. 3	0.6 0.85	0. 0.	Not Req'd	0. 0.	0. 0.	Not Req'd	0. 0.	0. 0.	Not Req'd	0. 0.	5. 6.64	T->10 min (MAN)	1.41 0.87
2RHS* MOV30B	RHRB Rtn to Supp. Pool	1.6 HP	575. 3	0.6 0.85	9.6 12.8	T->10 min (MAN)	2.7 1.7	9.6 12.8	T->10 min (MAN)	2.7 1.7	9.6 12.8	T->10 min (MAN)	2.7 1.7	-0- -0-	Not Req'd	-0- -0-
2RHS* MOV32B	Heat Exchngr B to RCIC	0.7 HP	575. 3	0.6 0.85	4.2 5.6	T = 0	1.2 0.74	4.2 5.6	T = 0	1.2 0.74	4.2 5.6	T = 0	1.2 0.74	4.2 5.6	T->10 min (MAN)	1.2 0.74
2RHS* MOV33B	Supp. Pool Spray Header B	0.33 HP	575. 3	0.6 0.85	2. 2.7	T = 0	0.56 0.35	2. 2.7	T = 0	0.56 0.35	2. 2.7	T = 0	0.56 0.35	-0- -0-	Not Req'd	-0- -0-
2RHS* MOV37B	RHR Line B to Supp. Pool	0.33 HP	575. 3	0.6 0.85	2. 2.7	T = 0	0.56 0.35	2. 2.7	T = 0	0.56 0.35	2. 2.7	T = 0	0.56 0.35	2. 2.7	T->10 min (MAN)	0.56 0.35
2RHS* MOV40B	Shutdown Cooling Rtn B	10.3 HP	575. 3	0.6 0.85	61.8 82.4	T = 0	17.51 10.85	61.8 82.4	T = 0	17.51 10.85	61.8 82.4	T = 0	17.51 10.85	61.8 82.4	T->10 min (MAN)	17.51 10.85
2RHS* MOV4B	RHR Min. Flow to Supp. Pool	1.9 HP	575. 3	0.6 0.85	11.4 15.2	T = 9 sec	3.23 2.	11.4 15.2	T = 9 sec	3.23 2.	11.4 15.2	T = 9 sec	3.23 2.	11.4 15.2	T->10 min (MAN)	3.23 2.

TABLE 8.3-2 (Cont'd.)

<u>Total Load On: 2EHS*MCC303</u>

Design Basis Accident Diesel Generator Loading Possibilities									Nonaccident Loading		
Simultaneous LOOP & LOCA			LOOP With Delayed LOCA			LOCA With Delayed LOOP			LOOP With Unit Trip		
Starting KW KVAR	Running KW KVAR	Time Span	Starting KW KVAR	Running KW KVAR	Time Span	Starting KW KVAR	Running KW KVAR	Time Span	Starting KW KVAR	Running KW KVAR	Time Span
812 1708	0 0	0<-T<6 sec	812 1708	0 0	0<-T<6 sec	812 1708	0 0	0<-T<6 sec	512 1307	0 0	0<-T<6 sec
0 0	287 178	6<-T<9 sec	0 0	287 178	6<-T<9 sec	0 0	287 178	6<-T<9 sec	0 0	201 124	6<-T<120 sec
11 15	287 178	9<-T<14 sec	11 15	287 178	9<-T<14 sec	11 15	287 178	9<-T<14 sec	0 0	189 117	120<-T<600 sec
23 30	287 178	14<-T<15 sec	23 30	287 178	14<-T<15 sec	23 30	287 178	14<-T<15 sec	320 455	189 117	600<-T<606 sec
11 15	290 180	15<-T<20 sec	11 15	290 180	15<-T<20 sec	11 15	290 180	15<-T<20 sec	0 0	282 175	606<-T<720 sec
0 0	293 182	20<-T<120 sec	0 0	293 182	20<-T<120 sec	0 0	293 182	20<-T<120 sec	0 0	206 128	720 sec<-T
0 0	199 123	120<-T<600 sec	0 0	199 123	120<-T<600 sec	0 0	199 123	120<-T<600 sec	0 0	-0- -0-	
129 200	199 123	600<-T<606 sec	129 200	199 123	600<-T<606 sec	129 200	199 123	600<-T<606 sec	0 0	-0- -0-	
0 0	238 147	606<-T<720 sec	0 0	238 147	606<-T<720 sec	0 0	238 147	606<-T<720 sec	0 0	-0- -0-	
0 0	211 130	720 sec<-T	0 0	211 130	720 sec<-T	0 0	211 130	720 sec<-T	0 0	-0- -0-	

TABLE 8.3-3A

HPCS DIESEL GENERATOR CONDITIONS AND CORRESPONDING MAIN CONTROL ROOM ANNUNCIATIONS

HPCS Diesel Generator Condition	Main Control Room Remote Annunciation
Low fuel oil level in day tank	Engine trouble
Low starting air pressure	Engine trouble
Control power failure	DG inoperable 27
Engine in maintenance position	Diesel engine in maintenance position
Diesel engine trip/lockout not reset	Diesel engine trip/trouble
Generator trip/lockout not reset a. Loss of excitation[1] b. Generator reverse power[1]	Generator trip/lockout
Engine overspeed	Engine overspeed
Generator differential	Generator trip/lockout
Manual out-of-service (main breakers or diesel generator inoperable) (control room)	Engine bypassed/inoperable
Low lubrication oil pressure [1]	Engine trouble
High jacket water outlet temperature[1]	Engine trouble
Engine overcrank[1]	Engine trouble

[1]During emergency (LOCA) operation, these conditions are bypassed and do not shutdown the HPCS DG.

TABLE 8.3-4

LIST OF CLASS 1E SAFETY-RELATED LOADS BY POWER SOURCE

Equipment Identity No.	Description[1]	Division[2]	Rating	Volts	Phase	Amps Full Load	Amps Locked Rotor	Power Source Identity No.
2HVC*UC108A	Unit cooler CB standby switchgear room A	G	10 hp	575	3	10.80	64.3	2EJS*PNL102A
2SWP*FV54A	Motor-operated pressure valve	G	0.7 hp	575	3	1.1	29.20	2EHS*MCC103
2SWP*FV54B	Motor-operated pressure valve	Y	0.7 hp	575	3	1.1	29.20	2EHS*MCC303
2VBS*PNLA103	Control room RPS and NS4 distribution panel	G	200 A	120	1	100.0	-	2VBS*PNLA100
2VBS*PNLA104	RPS and N54 distribution panel	O	200 A	120	1	100.0	-	2VBS*PNLA100
2VBS*PNLA105	MSIV distribution panel	G	200 A	120	1	100.0	-	2VBS*PNLA100
2VBS*PNLA106	MSIV distribution panel	Y/W	200 A	120	1	100.00	-	2VBS*PNLA100
2VBS*PNLA110	Control room RPS distribution panel	B/W	200 A	120	1	100.00	-	2VBS*PNLA100
2VBS*PNLB103	Control room RPS distribution panel	Y	200 A	120	1	100.0	-	2VBS*PNLB100
2VBS*PNLB104	RPS and NS4 distribution panel	B	200 A	120	1	100.0	-	2VBS*PNLB100
2VBS*PNLB105	MSIV distribution panel	G/W	200 A	120	1	200.0	-	2VBS*PNLB100
2VBS*PNLB106	MSIV distribution panel	Y	200 A	120	1	200.0	-	2VBS*PNLB100
2VBS*PNLB110	Control room RPS distribution panel	O/W	200 A	120	1	100.0	-	2VBS*PNLB100
2EGS*EG1	DG Div. I	G	5,500 kVA	4,160	3	763.3	-	Diesel generator
2EGS*EG2	DG Div. III	P	3,250 kVA	4,160	3	451	-	Diesel generator
2EGS*EG3	DG Div. II	Y	5,500 kVA	4,160	3	763.3	-	Diesel generator
2BYS*SWG002A	125-V dc switchgear	G	2,000 A	125	0	449	-	2BYS*BAT2A
2BYS*SWG002B	125-V dc switchgear	Y	2,000 A	125	0	283	-	2BYS*BAT2B
2BYS*BAT2A	125-V 1E standby battery Div. I	G	2,500 AH	125	0	330	-	2BYS*CHGR2A1

TABLE 8.3-4 (Cont'd.)

Equipment Identity No.	Description[1]	Division[2]	Rating	Volts	Phase	Amps Full Load	Amps Locked Rotor	Power Source Identity No.
2BYS*SWG002A	125-V dc switchgear	G	2,000 A	125	0	330	-	2BYS*CHGR2A1
2BYS*SWG002A	125-V dc switchgear	G	2,000 A	125	0	330	-	2BYS*CHGR2A2
2BYS*BAT2B	125-V 1E standby battery Div. II	Y	2,500 AH	125	0	330	-	2BYS*CHGR2B1
2BYS*SWG002B	125-V dc switchgear	Y	2,000 A	125	0	330	-	2BYS*CHGR2B1
2BYS*SWG002B	125-V dc switchgear	Y	2,000 A	125	0	330	-	2BYS*CHGR2B2
2BYS*BAT2C	125-V 1E standby battery Div. III	P	60 AH	125	0	55	-	2BYS*CHGR2C1
2BYS*PNL204A	125-V dc distribution panel	G	225 A	125	0	225	-	2BYS*SWG002A
2BYS*PNL204B	125-V dc distribution panel	Y	225 A	125	0	225	-	2BYS*SWG002B
2BYS*PNL201A	125-V dc distribution panel	G	400 A	125	0	200	-	2BYS*SWG002A
2BYS*PNL202A	125-V dc distribution panel	G	225 A	125	0	200	-	2BYS*SWG002A
2DMS*MCCA1	125-V dc MCC reactor building el 240'	G	600 A	125	0	600	-	2BYS*SWG002A
2BYS*PNL201B	125-V dc distribution panel	Y	400 A	125	0	400	-	2BYS*SWG002B
2BYS*PNL202B	125-V dc distribution panel	Y	225 A	125	0	200	-	2BYS*SWG002B
2DMS*MCCB1	125-V dc MCC reactor building el 240'	Y	600 A	125	0	600	-	2BYS*SWG002B
2ICS*FV108	RCIC pump to condensate storage	G	0.36 hp	125	0	4.2	20.8	2DMS*MCCA1
2ICS*MOV116	Lube oil cooling water supply	G	0.36 hp	125	0	4.0	21.00	2DMS*MCCA1
2ICS*MOV120	RCIC steam supply	G	0.72 hp	125	0	8.00	39.00	2DMS*MCCA1
2ICS*MOV122	RCIC turbine exhaust	G	1.8 hp	125	0	14.50	82	2DMS*MCCA1
2ICS*MOV124	RCIC pump to condensate storage	G	0.72 hp	125	0	8.00	39	2DMS*MCCA1
2ICS*MOV126	RCIC pump to reactor	G	4.00 hp	125	0	24.00	104	2DMS*MCCA1
2ICS*MOV129	Condensate storage to RCIC pump	G	0.72 hp	125	0	8.00	39	2DMS*MCCA1
2ICS*MOV136	Suppression pool into RCIC pump	G	0.72 hp	125	0	8.00	39	2DMS*MCCA1

TABLE 8.3-4 (Cont'd.)

Equipment Identity No.	Description[1]	Division[2]	Rating	Volts	Phase	Amps Full Load	Amps Locked Rotor	Power Source Identity No.
2ICS*MOV143	RCIC pump minimum flow to suppression pool	G	0.36 hp	125	0	4.0	21.70	2DMS*MCCA1
2ICS*MOV150	RCIC trip throttle valve	G	0.33 hp	125	0	4.0	21.70	2DMS*MCCA1
2ICS*MOV159	RCIC bypass to steam supply	G	0.13 hp	125	0	2.3	16.6	2DMS*MCCA1
2ICS*MOV164	Vacuum breaker valve outboard	G	0.14 hp	125	0	1.6	15.9	2DMS*MCCA1
2ICS*MOV148	Vacuum breaker valve inboard	Y	0.14 hp	125	0	1.6	15.9	2DMS*MCCB1
2HVY*UC2A	Service water pump pressure indicator transmitter unit cooler	G	40 hp	575	3	40.0	324	2EHS*MCC101
2HVY*UC2C	Service water pump pressure indicator transmitter unit cooler	G	40 hp	575	3	40.0	324	2EHS*MCC101
2SWP*MOV1A	Service water backwash line	G	0.13 hp	575	3	.36	2.5	2EHS*MCC101
2SWP*MOV1C	Service water backwash line	G	0.13 hp	575	3	.36	2.5	2EHS*MCC101
2SWP*MOV1E	Service water strainer backwash	G	0.13 hp	575	3	.36	2.5	2EHS*MCC101
2SWP*MOV3A	Service water to turbine plant	G	4 hp	575	3	5.60	48.0	2EHS*MCC101
2SWP*MOV30A	Motor-operated gate valve	G	1 hp	575	3	2.24	12.8	2EHS*MCC101
2SWP*MOV50A	Service water pump discharge header valve	G	9.90 hp	575	3	15.60	104.00	2EHS*MCC101
2SWP*MOV74A	Service water pump discharge block valve	G	2.6 hp	575	3	4.7	30.0	2EHS*MCC101
2SWP*MOV74C	Service water pump discharge block valve	G	2.6 hp	575	3	4.7	30.0	2EHS*MCC101
2SWP*MOV74E	Service water pump discharge block valve	G	2.6 hp	575	3	4.7	30.0	2EHS*MCC101
2SWP*MOV77A	Motor-operated gate valve	G	0.70 hp	575	3	1.9	10	2EHS*MCC101
2SWP*SSR1A	Bar rack heater	G	3.15 kW	332	1	10.5	-	2EHS*MCC101
2SWP*SSR2A	Bar rack heater	G	3.15 kW	332	1	10.5	-	2EHS*MCC101
2SWP*SSR3A	Bar rack heater	G	3.15 kW	332	1	10.5	-	2EHS*MCC101

TABLE 8.3-4 (Cont'd.)

Equipment Identity No.	Description[1]	Division[2]	Rating	Volts	Phase	Amps Full Load	Amps Locked Rotor	Power Source Identity No.
2SWP*SSR4A	Bar rack heater	G	3.15 kW	332	1	10.5	-	2EHS*MCC101
2SWP*SSR5A	Bar rack heater	G	3.15 kW	332	1	10.5	-	2EHS*MCC101
2SWP*SSR6A	Bar rack heater	G	3.15 kW	332	1	10.5	-	2EHS*MCC101
2SWP*STR4A	Strainer service water	G	3 hp	575	3	3.68	23.0	2EHS*MCC101
2SWP*STR4C	Strainer service water	G	3 hp	575	3	3.68	23.0	2EHS*MCC101
2SWP*STR4E	Strainer service water	G	3 hp	575	3	3.68	24.1	2EHS*MCC101
2EJS*PNL101A	Switchgear room A emergency 600-V panel	G	400 A	600	3	150	-	2EHS*MCC102A
2EJS*PNL103A	AB-N emergency 600-V panel	G	400 A	600	3	150	-	2EHS*MCC102A
2EJS*PNL104A	AB-N emergency 600-V panel	G	400 A	600	3	150	-	2EHS*MCC102A
2FWS*MOV21A	Feedwater to reactor	G	26.4 hp	575	3	29.9	297.0	2EHS*MCC102A
2FWS*MOV21B	Feedwater to reactor	G	26.4 hp	575	3	29.9	297.0	2EHS*MCC102C
2GTS*FN1A	SGTS filter train discharge fan	G	40 hp	575	3	38.5	285.0	2EHS*MCC102A
2GTS*MOV1A	Reactor building ventilation mix plenum to grates	G	0.33 hp	575	3	1.68	5.0	2EHS*MCC102A
2GTS*MOV2A	SGTS filter train A inlet	G	2.0 hp	575	3	2.5	-	2EHS*MCC102A
2GTS*MOV3A	SGTS filter train A discharge	G	2.0 hp	575	3	2.5	-	2EHS*MCC102A
2GTS*MOV4A	Decay heat cool to train A	G	1.0 hp	575	3	2.2	12.5	2EHS*MCC302B
2HCS*MOV1A	Wetwell hydrogen recombiner isolation valve	G	0.7 hp	575	3	1.9	10.0	2EHS*MCC102A
2HCS*MOV2A	Wetwell hydrogen recombiner isolation valve	G	0.33 hp	575	3	0.64	4.64	2EHS*MCC102A
2HCS*MOV3A	Wetwell hydrogen recombiner isolation valve	G	0.7 hp	575	3	1.9	10.0	2EHS*MCC102A
2CMS*P2A	H_2/O_2 analyzer pump	G	1.0 hp	575	3	1.70	-	2EHS*MCC102A

TABLE 8.3-4 (Cont'd.)

Equipment Identity No.	Description[1]	Division[2]	Rating	Volts	Phase	Amps Full Load	Amps Locked Rotor	Power Source Identity No.
2GTS*MOV28A	Cross-bleed LVLV	G	2.0 hp	575	3	2.50	–	2EHS*MCC102A
2HCS*MOV4A	Wetwell hydrogen recombiner isolation valve	G	0.33 hp	575	3	0.64	4.64	2EHS*MCC102A
2HCS*MOV5A	Wetwell hydrogen recombiner isolation valve	G	0.33 hp	575	3	0.64	4.64	2EHS*MCC102A
2HCS*MOV6A	Wetwell hydrogen recombiner isolation valve	G	0.33 hp	575	3	0.64	4.64	2EHS*MCC102A
2MSS*MOV112	Main steam to condenser	G	1.6 hp	575	3	3.2	20.0	2EHS*MCC102A
2MSS*MOV119	Vent valve	G	0.33 hp	575	3	0.64	4.5	2EHS*MCC102A
2MSS*MOV208	Main steam valve	G	0.33 hp	575	3	0.64	4.5	2EHS*MCC102A
2SWP*MOV17A	Service water to RBCLCW	G	1.6 hp	575	3	3.2	20.0	2EHS*MCC102A
2SWP*MOV18A	RBCLCW to service water	G	1.6 hp	575	3	3.2	20.0	2EHS*MCC102A
2SWP*MOV19A	Service water to RBCLCW heat exchanger	G	1.0 hp	575	3	2.2	12.5	2EHS*MCC102A
2SWP*MOV21A	RBCLCW to SFC cooling pool	G	0.33 hp	575	3	0.64	4.5	2EHS*MCC102A
2SWP*MOV33A	RHR heat exchanger A to discharge tunnel	G	0.83 hp	575	3	3.5	10.3	2EHS*MCC102A
2SWP*MOV90A	Service water to RHS heat exchanger	G	0.83 hp	575	3	3.5	10.3	2EHS*MCC102A
2CSL*FV114	LPCS test	G	0.33 hp	575	3	0.66	4.14	2EHS*MCC102C
2CSL*MOV104	LPCS pump to reactor	G	7.80 hp	575	3	9.12	75.00	2EHS*MCC102C
2CSL*MOV107	LPCS minimum flow to RHR	G	1.9 hp	575	3	2.8	21.0	2EHS*MCC102C
2CSL*MOV112	Suppression pool to LPCS pump	G	0.70 hp	575	3	1.8	11.25	2EHS*MCC102C
2CSL*P2	LPCS system pressure pump	G	10 hp	575	3	10.3	56.2	2EHS*MCC102C
2DER*MOV120	Containment isolation valve	G	0.66 hp	575	3	0.86	4.5	2EHS*MCC102C
2DER*MOV131	Reactor building equipment drains TK1 vent	G	0.33 hp	575	3	0.64	4.5	2EHS*MCC102C

TABLE 8.3-4 (Cont'd.)

Equipment Identity No.	Description[1]	Division[2]	Rating	Volts	Phase	Amps Full Load	Amps Locked Rotor	Power Source Identity No.
2DFR*MOV120	Reactor building floor drain discharge isolation valve	G	1.4 hp	575	3	1.75	8.0	2EHS*MCC102C
2DFR*MOV139	Reactor plant floor drain vent isolation valve	G	0.33 hp	575	3	0.64	4.5	2EHS*MCC102C
2ICS*MOV121	Turbine steam supply isolation outboard	G	7.8 hp	575	3	9.12	74.3	2EHS*MCC102C
2ICS*P2	RCIC system pressure pump A	G	10 hp	575	3	10.30	56.2	2EHS*MCC102C
2SLS*MOV1A	Standby liquid control	G	0.33 hp	575	3	0.7	4.8	2EHS*MCC102C
2SLS*MOV5A	SLCS outboard isolation valve	G	0.70 hp	575	3	1.90	10.6	2EHS*MCC102A
2SLS*P1A	Standby liquid pump A	G	40 hp	575	3	38.8	253	2EHS*MCC102C
2WCS*MOV112	RWCU system outboard steam isolation valve	G	5.2 hp	575	3	8.48	48.00	2EHS*MCC102C
2WCS*MOV200	RWCU return isolation valve	G	1.6 hp	575	3	3.20	20.0	2EHS*MCC102A
2CCP*MOV124	Domestic water cooler to RBCLCW outboard I	G	1 hp	575	3	2.2	12.5	2EHS*MCC103A
2CCP*MOV14A	RBCLCW to SFC heat exchanger A	G	1.6 hp	575	3	3.20	20.0	2EHS*MCC103A
2CCP*MOV18A	SFC heat exchanger A to RBCLCW	G	1.6 hp	575	3	3.20	20.00	2EHS*MCC103A
2CCP*MOV265	Isolation valve containment	G	1 hp	575	3	2.2	12.5	2EHS*MCC103A
2CCP*MOV15A	RBCLCW to RCS pump A outboard I	G	0.33 hp	575	3	0.64	4.5	2EHS*MCC103A
2CCP*MOV15B	RBCLCW to RCS pump B outboard I	G	0.33 hp	575	3	0.64	4.5	2EHS*MCC103A
2CCP*MOV17A	To RBCLCW RCS pump A outboard I	G	0.33 hp	575	3	0.64	4.5	2EHS*MCC103A
2CCP*MOV17B	To RBCLCW RCS pump A outboard I	G	0.33 hp	575	3	0.64	4.5	2EHS*MCC103A
2EGA*M1A	DG 1 air comp 1A motor	G	15 hp	575	3	15.80	84.0	2EHS*MCC103A
2EGA*M2A	DG 1 air comp 2A motor	G	15 hp	575	3	15.80	84.0	2EHS*MCC103A
2EGF*P1A	DG 1 fuel oil transfer pump A	G	1.5 hp	575	3	1.82	9.04	2EHS*MCC103A

TABLE 8.3-4 (Cont'd.)

Equipment Identity No.	Description[1]	Division[2]	Rating	Volts	Phase	Amps Full Load	Amps Locked Rotor	Power Source Identity No.
2EGF*P1C	DG 1 fuel oil transfer pump C	G	1.5 hp	575	3	1.76	9.04	2EHS*MCC103A
2EGO*P1A	Lube oil circulation pump	G	15 hp	575	3	14.9	83.5	2EHS*MCC103A
2EGS*P1A	Jacket water circulation pump	G	5 hp	575	3	5.6	34.2	2EHS*MCC103A
2EGT*CH2	Lube oil heater	G	12 kW	575	3	12	–	2EHS*MCC103A
2EGT*CH4	Jacket water heater	G	18 kW	575	3	18	–	2EHS*MCC103A
2HVC*ACU1A	Control room A/C unit 1A	G	40 hp	575	3	39.5	219.0	2EHS*MCC103A
2HVC*ACU2A	Relay room A/C unit 2A	G	40 hp	575	3	39.5	219.0	2EHS*MCC103A
2HVC*ACU3A	Remote shutdown on room A/C unit	G	2 hp	575	3	2.48	16.30	2EHS*MCC103A
2HVC*FN11A	Makeup air switchgear floor	G	7.5 hp	575	3	8.2	48.0	2EHS*MCC103A
2HVC*FN2A	Control room A/C booster fan A	G	10 hp	575	3	10.5	65	2EHS*MCC103A
2HVC*FN4A	Battery room A exchange fan	G	3 hp	575	3	3.7	23.4	2EHS*MCC103A
2HVC*MOV1A	Control room A/C special filter bypass	G	0.25 hp	575	3	0.4	4.5	2EHS*MCC103A
2HVR*CHL1A	Auxiliary oil pump	G	0.75 hp	575	3	0.9	4.83	2EHS*MCC103A
2HVK*P1A	Control building chilled water circulating pump A	G	15 hp	575	3	14.8	94.8	2EHS*MCC103A
2SWP*FV47A	Service water to CWS pumps	G	0.7 hp	575	3	1.10	29.20	2EHS*MCC103A
2SWP*MOV66A	Service water to standby DG coolers E3A	G	1 hp	575	3	2.2	12.8	2EHS*MCC103A
2SWP*MOV67A	Service water to control DG relay room coil	G	0.66 hp	575	3	0.86	4.5	2EHS*MCC103A
2SWP*MOV599	Service water to discharge tunnel isolation	G	1.6 hp	575	3	3.2	20.0	2EHS*MCC103A
2SWP*MOV93A	Service water to discharge tunnel isolation	G	1 hp	575	3	2.20	12.50	2EHS*MCC103A
2SWP*MOV95A	Service water to standby DG coolers	G	1 hp	575	3	2.2	12.5	2EHS*MCC103A

TABLE 8.3-4 (Cont'd.)

Equipment Identity No.	Description[1]	Division[2]	Rating	Volts	Phase	Amps Full Load	Amps Locked Rotor	Power Source Identity No.
2SWP*P2A	Control building chiller service water pump A	G	10 hp	575	3	10.5	64.00	2EHS*MCC103A
2HVP*FN1A	DG 1 exhaust fan 1A	G	30 hp	575	3	32.0	232.0	2EHS*MCC103C
2HVP*FN1C	DG 1 exhaust fan 1C	G	30 hp	575	3	32.0	232.0	2EHS*MCC103C
2RHS*FV38A	Test line A to suppression pool	G	0.33 hp	575	3	0.66	3.8	2EHS*MCC103C
2RHS*MOV80A	Globe valve	G	0.13 hp	575	3	0.4	2.50	2EHS*MCC103C
2RHS*MOV1A	Suppression pool to RHR pump A	G	1.6 hp	575	3	6.0	38.0	2EHS*MCC103C
2RHS*MOV104	Heat spray in outboard isolation	G	0.7 hp	575	3	1.8	9.0	2EHS*MCC103C
2RHS*MOV113	Cooling supply outboard isolation	G	19.2 hp	575	3	20.5	156	2EHS*MCC103C
2RHS*MOV12A	Heat exchanger A to reactor	G	0.7 hp	575	3	1.9	10.0	2EHS*MCC103C
2RHS*MOV142	Heat exchanger B to liquid radwaste system	G	0.33 hp	575	3	0.64	4.5	2EHS*MCC103C
2RHS*MOV15A	Containment spray A	G	2.6 hp	575	3	4.7	30.0	2EHS*MCC103C
2RHS*MOV2A	Reactor to RHR pump A	G	0.83 hp	575	3	3.5	10.3	2EHS*MCC103C
2RHS*MOV22A	Steam condensing to heat exchanger A	G	1.6 hp	575	3	3.2	20.0	2EHS*MCC103C
2RHS*MOV23A	Steam condensing to heat exchanger A	G	1.6 hp	575	3	3.2	20.0	2EHS*MCC103C
2RHS*MOV24A	LPCI inlet A	G	6.6 hp	575	3	8.0	74.2	2EHS*MCC103C
2RHS*MOV25A	Containment spray A	G	2.6 hp	575	3	4.7	30.0	2EHS*MCC103C
2RHS*MOV26A	Heat exchanger A vent to suppression pool	G	0.13 hp	575	3	0.83	2.5	2EHS*MCC103C
2RHS*MOV27A	Heat exchanger A vent to suppression pool	G	0.13 hp	575	3	0.83	2.5	2EHS*MCC103C
2RHS*MOV30A	RHR return to suppression pool	G	1.6 hp	575	3	3.20	20.00	2EHS*MCC103C
2RHS*MOV32A	Heat exchanger A to RCIC	G	0.7 hp	575	3	1.8	9.0	2EHS*MCC103C

TABLE 8.3-4 (Cont'd.)

Equipment Identity No.	Description[1]	Division[2]	Rating	Volts	Phase	Amps Full Load	Amps Locked Rotor	Power Source Identity No.
2RHS*MOV33A	Suppression pool spray header A	G	0.33 hp	575	3	0.64	4.5	2EHS*MCC103C
2RHS*MOV37A	RHR line A to suppression pool	G	0.33 hp	575	3	0.64	4.5	2EHS*MCC103C
2RHS*MOV4A	RHR minimum flow to suppression pool	G	1.9 hp	575	3	2.8	21.00	2EHS*MCC103C
2RHS*MOV40A	Shutdown cooling return A	G	10.3 hp	575	3	12.0	91.4	2EHS*MCC103C
2RHS*MOV67A	RHR shutdown bypass	G	0.33 hp	575	3	0.64	4.50	2EHS*MCC103C
2RHS*MOV8A	Heat exchanger A bypass	G	1.6 hp	575	3	3.2	20.0	2EHS*MCC103C
2RHS*MOV9A	RHR pump A to heat exchanger A	G	0.7 hp	575	3	1.9	10.0	2EHS*MCC103C
2BYS*CHGR2C1	125-V battery charger standby Div. III	P	50 A dc	575	3	12.00	-	2EHS*MCC201
2BYS*CHGR2C2	125-V battery charger	P	50 A dc	575	3	12.00	-	2EHS*MCC201
2CSH*MOV101	Condensate storage to HPCS pump	P	0.7 hp	575	3	1.9	10	2EHS*MCC201
2CSH*MOV105	Flow bypass to suppression pool	P	3.3 hp	575	3	4.2	29.0	2EHS*MCC201
2CSH*MOV107	HPCS pump to reactor	P	19.2 hp	575	3	20.5	156	2EHS*MCC201
2CSH*MOV110	Test bypass to condensate storage	P	13.1 hp	575	3	18.4	131	2EHS*MCC201
2CSH*MOV111	Test bypass to suppression pool	P	9.9 hp	575	3	15.6	104.0	2EHS*MCC201
2CSH*MOV112	Test bypass to condensate storage	P	13.1 hp	575	3	18.4	131	2EHS*MCC201
2CSH*MOV118	Suppression pool to HPCS	P	3.3 hp	575	3	4.2	29.0	2EHS*MCC201
2CSH*P2	Standby water leg pump E22-C003	P	10 hp	575	3	10.3	51	2EHS*MCC201
2EGF*P2A	DG 2 fuel oil transfer pump A	P	1.5 hp	575	3	1.82	9.04	2EHS*MCC201
2EGF*P2B	DG 2 fuel oil transfer pump B	P	1.5 hp	575	3	1.82	9.04	2EHS*MCC201
2EGO*P1	HPCS DG 2 circulating oil pump	P	1 hp	575	3	1.6	-	2EHS*MCC201
2EGT*CH1	HPCS DG 2 immersion heater	P	15 kW	575	3	15.0	-	2EHS*MCC201
2EGT*H1	HPCS DG 2 space heater	P	3.0 kW	575	3	2.15	-	2EHS*MCC201

TABLE 8.3-4 (Cont'd.)

Equipment Identity No.	Description[1]	Division[2]	Rating	Volts	Phase	Amps Full Load	Amps Locked Rotor	Power Source Identity No.
2HVC*UC102	HPCS switchgear room unit cooler	P	5 hp	575	3	5.6	32.9	2EHS*MCC201
2HVP*FN2A	DG 2 exhaust fan 2A	P	30 hp	575	3	32.0	232.0	2EHS*MCC201
2HVP*FN2B	DG 2 exhaust fan 2B	P	30 hp	575	3	32.0	232.0	2EHS*MCC201
2HVP*UC2	DG 2 unit cooler HPCS DG room	P	5 hp	575	3	5.44	–	2EHS*MCC201
2HVR*UC403A	Reactor building HPCS separator cooler el 196′	P	2 motors 15 hp total	575	3	16.00	129.20	2EHS*MCC201
2HVR*UC403B	Reactor building separator cooler el 196′	P	2 motors 15 hp total	575	3	16.00	129.20	2EHS*MCC201
2IAC*XLE03	LTG transformer 600-208Y/120-V	P	30 kVA	600	3	30.0	–	2EHS*MCC201
2SCV*XD200P	Distribution transformer 600-120-V	P	25 kVA	600	1	432.40	–	2EHS*MCC2C1
2SWP*MOV15A	Service water to HPCS unit cooler	P	0.33 hp	575	3	0.64	4.50	2EHS*MCC201
2SWP*MOV15B	Service water to HPCS unit cooler	P	0.33 hp	575	3	0.64	4.50	2EHS*MCC201
2SWP*MOV94A	Service water to standby D/G coolers E3B	P	1 hp	575	3	2.2	12.5	2EHS*MCC201
2SWP*MOV94B	Service water to standby DG coolers E3B	P	1 hp	575	3	2.2	12.5	2EHS*MCC201
2HVY*UC2B	Service water pump PIT unit cooler	Y	40 hp	575	3	40.0	324.00	2EHS*MCC301
2HVY*UC2D	Service water pump PIT unit cooler	Y	40 hp	575	3	40.0	324.00	2EHS*MCC301
2SWP*MOV1B	Service water backwash line	Y	0.13 hp	575	3	0.36	2.5	2EHS*MCC301
2SWP*MOV1D	Service water backwash line	Y	0.13 hp	575	3	0.36	2.5	2EHS*MCC301
2SWP*MOV1F	Service water strainer backwash	Y	0.13 hp	575	3	0.36	2.5	2EHS*MCC301
2SWP*MOV3B	Service water to turbine plant	Y	4 hp	575	3	5.6	48.0	2EHS*MCC301
2SWP*MOV30B	Motor-operated gate valve	Y	1 hp	575	3	2.24	12.8	2EHS*MCC301

TABLE 8.3-4 (Cont'd.)

Equipment Identity No.	Description[1]	Division[2]	Rating	Volts	Phase	Amps Full Load	Amps Locked Rotor	Power Source Identity No.
2SWP*MOV50B	Service water pump discharge header	Y	9.9 hp	575	3	15.60	104.00	2EHS*MCC301
2SWP*MOV74B	Service water discharge block valve	Y	2.6 hp	575	3	4.7	30.0	2EHS*MCC301
2SWP*MOV74D	Service water pump discharge block valve	Y	2.6 hp	575	3	4.7	30.0	2EHS*MCC301
2SWP*MOV74F	Service water pump discharge block valve	Y	2.6 hp	575	3	4.7	30.0	2EHS*MCC301
2SWP*MOV77B	Motor-operated gate valve	Y	0.7 hp	575	3	1.9	10.0	2EHS*MCC301
2SWP*SSR1B	Bar rack heater	Y	3.15 kW	332	1	10.5	-	2EHS*MCC301
2SWP*SSR2B	Bar rack heater	Y	3.15 kW	332	1	10.5	-	2EHS*MCC301
2SWP*SSR3B	Bar rack heater	Y	3.15 kW	332	1	10.5	-	2EHS*MCC301
2SWP*SSR4B	Bar rack heater	Y	3.15 kW	332	1	10.5	-	2EHS*MCC301
2SWP*SSR5B	Bar rack heater	Y	3.15 kW	332	1	10.5	-	2EHS*MCC301
2SWP*SSR6B	Bar rack heater	Y	3.15 kW	332	1	10.5	-	2EHS*MCC301
2SWP*STR4B	Strainer service water	Y	3 hp	575	3	3.68	23.0	2EHS*MCC301
2SWP*STR4D	Strainer service water	Y	3 hp	575	3	3.68	23.0	2EHS*MCC301
2SWP*STR4F	Strainer service water	Y	3 hp	575	3	3.68	23.0	2EHS*MCC301
2EJS*PNL302B	AB-S emergency 600-V panel	Y	400 A	600	3	150	-	2EHS*MCC302B
2EJS*PNL303B	AB-S emergency 600-V panel	Y	400 A	600	3	150	-	2EHS*MCC302B
2EJS*PNL304B	AB-S emergency 600-V panel	Y	400 A	600	3	150	-	2EHS*MCC302B
2GTS*FN1B	GTS filter train discharge fan	Y	40 hp	575	3	39.20	232.00	2EHS*MCC302B
2GTS*MOV1B	HVP mix plenum to GTS	Y	0.33 hp	575	3	1.68	5.0	2EHS*MCC302B
2CMS*P2B	H_2/O_2 analyzer pump	Y	1 hp	575	3	1.70	-	2EHS*MCC302B
2GTS*MOV2B	GTS filter train B inlet	Y	2.0 hp	575	3	2.5	-	2EHS*MCC302B

TABLE 8.3-4 (Cont'd.)

Equipment Identity No.	Description[1]	Division[2]	Rating	Volts	Phase	Amps Full Load	Amps Locked Rotor	Power Source Identity No.
3GTS*MOV3B	GTS filter train B discharge	Y	2.0 hp	575	3	2.5	–	2EHS*MCC302B
2GTS*MOV4B	Decay heat cooler to train B	Y	1.0 hp	575	3	2.2	12.8	2EHS*MCC102A
2HCS*MOV1B	Wetwell hydrogen recombiner isolation valve	Y	0.7 hp	575	3	1.9	10.0	2EHS*MCC302B
2HCS*MOV2B	Wetwell hydrogen recombiner isolation valve	Y	0.33 hp	575	3	0.64	4.64	2EHS*MCC302B
2HCS*MOV3B	Wetwell hydrogen recombiner isolation valve	Y	0.7 hp	575	3	1.9	10.0	2EHS*MCC302B
2HCS*MOV4B	Drywell hydrogen recombiner isolation valve	Y	0.33 hp	575	3	0.64	4.64	2EHS*MCC302B
2HCS*MOV5B	Drywell hydrogen recombiner isolation valve	Y	0.33 hp	575	3	0.64	4.64	2EHS*MCC302B
2HCS*MOV6B	Drywell hydrogen recombiner isolation valve	Y	0.33 hp	575	3	0.64	4.64	2EHS*MCC302B
2SWP*MOV17B	Service water to RBCLCW	Y	1.6 hp	575	3	3.2	20.00	2EHS*MCC302B
2SWP*MOV18B	RBCLCW to service water	Y	1.6 hp	575	3	3.2	20.0	2EHS*MCC302B
2SWP*MOV19B	Service water to RBCLCW heat exchanger	Y	1.0 hp	575	3	2.2	12.5	2EHS*MCC302B
2SLS*MOV5B	SLCS inboard isolation valve	Y	0.70 hp	575	3	1.90	10.00	2EHS*MCC302B
2GTS*MOV28B	Cross-bleed LVLV	Y	2.0 hp	575	3	2.50	–	2EHS*MCC302B
2SWP*MOV21B	RBCLCW to SFC cooling pool	Y	0.33 hp	575	3	0.64	4.5	2EHS*MCC302B
2SWP*MOV33B	RHR heat exchanger B to discharge tunnel	Y	0.83 hp	575	3	3.5	10.3	2EHS*MCC302B
2SWP*MOV90B	RHR heat exchanger inlet	Y	0.83 hp	575	3	3.5	10.3	2EHS*MCC302B
2DER*MOV119	Containment isolation valve	Y	0.66 hp	575	3	0.86	4.5	2EHS*MCC302D
2DER*MOV130	Der TK1 vent	Y	0.33 hp	575	3	0.64	4.5	2EHS*MCC302D

TABLE 8.3-4 (Cont'd.)

Equipment Identity No.	Description[1]	Division[2]	Rating	Volts	Phase	Amps Full Load	Amps Locked Rotor	Power Source Identity No.
2DFR*MOV121	Drywell floor drain discharge isolation valve	Y	1.4 hp	575	3	1.7	8.0	2EHS*MCC302D
2DFR*MOV140	Drywell floor drain discharge isolation valve	Y	0.33 hp	575	3	0.64	4.5	2EHS*MCC302D
2ICS*MOV128	MOV steam supply line	Y	7.80 hp	575	3	9.12	75.4	2EHS*MCC302D
2ICS*MOV170	Bypass of MOV128	Y	0.13 hp	575	3	0.36	2.5	3EHS*MCC302D
2MSS*MOV111	Main steam to condensate inboard isolation	Y	1.6 hp	575	3	3.2	20.0	2EHS*MCC302D
2MSS*MOV118	Vent valve	Y	0.33 hp	575	3	0.64	4.5	2EHS*MCC302D
2SLS*MOV1B	Standby liquid control	Y	0.33 hp	575	3	0.7	4.00	2EHS*MCC302D
2SLS*P1B	Standby liquid pump B	Y	40 hp	575	3	38.8	253.0	2EHS*MCC302D
2WCS*MOV102	RWCU inboard isolation valve	Y	5.2 hp	575	3	8.48	48.00	2EHS*MCC302D
2HVK*CHL1B	Auxiliary oil pump	Y	0.75 hp	575	3	0.9	4.83	2EHS*MCC303B
2CCP*MOV122	Drywell cooler to RBCLCW inboard I	Y	1 hp	575	3	2.2	12.5	2EHS*MCC303B
2CCP*MOV14B	RBCLCW to SFC heat exchanger B	Y	1.6 hp	575	3	3.20	20.00	2EHS*MCC303B
2CCP*MOV18B	SFC heat exchanger to RBCLCW	Y	1.6 hp	575	3	3.20	20.00	2EHS*MCC303B
2CCP*MOV16A	RBCLCW from RCS pump A	Y	1.6 hp	575	3	3.2	20.0	2EHS*MCC303B
2CCP*MOV16B	RBCLCW from RCS pump B	Y	0.33 hp	575	3	0.64	4.5	2EHS*MCC303B
2CCP*MOV273	RBCLCW to drywell cooler inboard isolation	Y	1.0 hp	575	3	2.2	12.5	2EHS*MCC303B
2CCP*MOV94A	Cooling water to P1A	Y	0.33 hp	575	3	0.64	4.50	2EHS*MCC303B
2CCP*MOV94B	Cooling water to P1B	Y	0.33 hp	575	3	0.64	4.50	2EHS*MCC303B
2EGF*P1B	DG 3 fuel oil transformer pump B	Y	1.5 hp	575	3	1.73	9.04	2EHS*MCC303B
2EGF*P1D	DG 3 fuel oil transformer pump D	Y	1.5 hp	575	3	1.73	9.04	2EHS*MCC303B

TABLE 8.3-4 (Cont'd.)

Equipment Identity No.	Description(1)	Division(2)	Rating	Volts	Phase	Amps Full Load	Amps Locked Rotor	Power Source Identity No.
2EGO*P1B	Lube oil circulation pump	Y	15 hp	575	3	14.9	83.5	2EHS*MCC303B
2EGS*P1B	Jacket water circulation pump	Y	5 hp	575	3	5.6	34.2	2EHS*MCC303B
2EGT*CH3	Lube oil heater	Y	12 kW	575	3	12	-	2EHS*MCC303B
2EGT*CH5	Jacket water heater	Y	18 kW	575	3	18	-	2EHS*MCC303B
2HVC*ACU1B	Control room A/C unit 1B	Y	40 hp	575	3	39.5	219.0	2EHS*MCC303B
2HVC*ACU2B	Relay room A/C unit 2B	Y	40 hp	575	3	39.5	219.0	2EHS*MCC303B
2HVC*ACU3B	Remote shutdown on room A/C unit	Y	2 hp	575	3	2.48	16.3	2EHS*MCC303B
2HVC*FN11B	Makeup air switchgear floor	Y	7.5 hp	575	3	8.2	48.0	2EHS*MCC303B
2HVC*FN2B	Control room A/C booster fan B	Y	10 hp	575	3	10.5	65.6	2EHS*MCC303B
2HVC*FN4B	Battery room B exchange fan	Y	3 hp	575	3	3.7	23.4	2EHS*MCC303B
2HVC*MOV1B	Control room A/C special filter bypass	Y	0.25 hp	575	3	4.5	28.1	2EHS*MCC303B
2EGA*M1B	DG 3 air comp 1B motor	Y	15 hp	575	3	15.8	84.00	2EHS*MCC303B
2EGA*M2B	DG 3 air comp 2B motor	Y	15 hp	575	3	15.8	84.00	2EHS*MCC303B
2HVK*P1B	Control building chilled water circulating pump B	Y	15 hp	575	3	14.8	94.8	2EHS*MCC303B
2SWP*FV47B	Service water to CWS pumps	Y	0.70 hp	575	3	1.1	29.2	2EHS*MCC303B
2SWP*MOV66B	Service water to standby DG coolers E3B	Y	1 hp	575	3	2.2	12.5	2EHS*MCC303B
2SWP*MOV67B	Service water to cont ground relay room coil	Y	0.66 hp	575	3	0.86	4.5	2EHS*MCC303B
2SWP*MOV93B	Service water to discharge tunnel isolation	Y	1.0 hp	575	3	2.2	12.5	2EHS*MCC303B
2SWP*MOV95B	Service water to standby DG coolers	Y	1 hp	575	3	2.2	12.5	2EHS*MCC303B
2SWP*P2B	Catch basin chiller service water pump B	Y	10 hp	575	3	10.5	63.2	2EHS*MCC303B

TABLE 8.3-4 (Cont'd.)

Equipment Identity No.	Description(1)	Division(2)	Rating	Volts	Phase	Amps Full Load	Amps Locked Rotor	Power Source Identity No.
2HVP*FN1B	DG 3 exhaust fan 1B	Y	30 hp	575	3	32.0	232.0	2EHS*MCC303D
2HVP*FN1D	DG 3 exhaust fan 1D	Y	30 hp	575	3	32.0	232.0	2EHS*MCC303D
2RHS*FV38B	Test line B to suppression pool	Y	0.33 hp	575	3	0.66	3.8	2EHS*MCC303D
2RHS*FV38C	RHR pump C to suppression pool	Y	0.33 hp	575	3	0.66	3.8	2EHS*MCC303D
2RHS*MOV1B	Suppression pool to RHR pump B	Y	4.0 hp	575	3	6.0	38.0	2EHS*MCC303D
2RHS*MOV1C	Suppression pool to RHR pump C	Y	1.6 hp	575	3	3.2	20.0	2EHS*MCC303D
2RHS*MOV112	Shutdown cooling supply inboard isolation	Y	19.2 hp	575	3	20.5	155	2EHS*MCC303D
2RHS*MOV115	Service water bypass to reactor	Y	1.6 hp	575	3	3.2	20.0	2EHS*MCC303D
2RHS*MOV116	Service water bypass to reactor	Y	1.6 hp	575	3	3.2	20.0	2EHS*MCC303D
2RHS*MOV12B	Heat exchanger B to reactor	Y	0.7 hp	575	3	1.9	10.0	2EHS*MCC303D
2RHS*MOV149	Heat exchanger B to liquid radwaste system	Y	0.33 hp	575	3	0.64	4.5	2EHS*MCC303D
2RHS*MOV15B	Containment spray B	Y	2.6 hp	575	3	4.7	30.00	2EHS*MCC303D
2RHS*MOV2B	Reactor to RHR pump B	Y	0.83 hp	575	3	3.5	10.3	2EHS*MCC303D
2RHS*MOV22B	Steam condensing to heat exchanger B	Y	1.6 hp	575	3	3.2	20.0	2EHS*MCC303D
2RHS*MOV23B	Steam condensing to heat exchanger B	Y	1.6 hp	575	3	3.2	20.0	2EHS*MCC303D
2RHS*MOV24B	LPCI inlet B	Y	6.6 hp	575	3	8.0	74.2	2EHS*MCC303D
2RHS*MOV24C	LPCI inlet C	Y	6.6 hp	575	3	8.0	74.2	EHS*MCC303D
2RHS*MOV25B	Containment spray B	Y	2.6 hp	575	3	4.7	30.0	2EHS*MCC303D
2RHS*MOV80B	Globe valve	Y	0.13 hp	575	3	0.36	2.50	2EHS*MCC303D
2RHS*MOV26B	Heat exchanger B vent to suppression pool	Y	0.13 hp	575	3	0.4	2.5	2EHS*MCC303D

TABLE 8.3-4 (Cont'd.)

Equipment Identity No.	Description[1]	Division[2]	Rating	Volts	Phase	Amps Full Load	Amps Locked Rotor	Power Source Identity No.
2RHS*MOV27B	Heat exchanger B vent to suppression pool	Y	0.13 hp	575	3	0.83	2.5	2EHS*MCC303D
2RHS*MOV30B	RHR return to suppression pool	Y	1.60 hp	575	3	3.20	20.00	2EHS*MCC303D
2RHS*MOV32B	Heat exchanger B to RCIC	Y	0.7 hp	575	3	1.80	9.0	2EHS*MCC303D
2RHS*MOV33B	Suppression pool spray header B	Y	0.33 hp	575	3	0.64	4.5	2EHS*MCC303D
2RHS*MOV37B	RHR line B to suppression pool	Y	0.33 hp	575	3	0.64	4.5	2EHS*MCC303D
2RHS*MOV4B	RHR minimum flow to suppression pool	Y	1.9 hp	575	3	2.8	21.0	2EHS*MCC303D
2RHS*MOV4C	RHR minimum flow to suppression pool	Y	1.9 hp	575	3	2.8	21.0	2EHS*MCC303D
2RHS*MOV40B	Shutdown cooling return B	Y	10.3 hp	575	3	12.0	91.4	2EHS*MCC303D
2RHS*MOV67B	RHR shutdown bypass	Y	0.33 hp	575	3	0.64	4.50	2RHS*MCC303D
2RHS*MOV8B	Heat exchanger B bypass	Y	1.6 hp	575	3	3.2	20.00	2EHS*MCC303D
2RHS*MOV9B	RHR pump B to heat exchanger B	Y	0.7 hp	575	3	1.9	10.0	2EHS*MCC303D
2RHS*P2	RHR system pressure pump	Y	10 hp	575	3	10.3	51	2EHS*MCC303D
2EJA*PNL100A	Reactor building 120-V heater panel	G	150 A	208	3	83.0	-	2EJA*XD100A
2EJA*PNL101A	Control building 120/240-V heater panel	G	150 A	240	1	104.0	-	2EJA*XD101A
2EJA*PNL300B	Reactor building 120-V heater panel	Y	150 A	208	3	83.0	-	2EJA*XD300B
2EJA*PNL301B	Control building 120/240-V heater panel	Y	150 A	240	1	104.0	-	2EJA*XD301B
2BYS*CHGR2A2	125-V battery charger	G	300 A dc	575	3	80	-	2EJS*PNL100A
2EJA*XD100A	Distribution transformer 600V-208Y/120-V	G	30 kVA	600	3	30.0	-	2EJS*PNL100A
2EJA*XD101A	Distribution transformer 600V-120/240-V	G	25 kVA	600	1	43.40	-	2EJS*PNL100A

TABLE 8.3-4 (Cont'd.)

Equipment Identity No.	Description[1]	Division[2]	Rating	Volts	Phase	Amps Full Load	Amps Locked Rotor	Power Source Identity No.
2EJS*PNL102A	Switchgear room A emergency 600-V panel	G	400 A	600	3	100	-	2EJS*PNL100A
2HVC*CH11A	Control building equipment room 306 heater	G	60 kW	575	3	60.30	-	2EJS*PNL100A
2HVC*CH12A	Control building equipment room 288	G	40 kW	575	3	40.20	-	2EJS*PNL100A
2SCV*XD101A	Distribution transformer 600-V-120/240-V	G	25 kVA	600	1	43.40	-	2EJS*PNL100A
2VBA*UPS2A	Div. 1A control UPS	G	25 kVA	575	3	76.00	-	2EJS*PNL100A
2HVR*UC401A	Reactor building space cooler el 175'	G	2 hp	575	3	2.5	16.0	2EJS*PNL101A
2HVR*UC401D	Reactor building space cooler el 175'	G	2 hp	575	3	2.5	16.0	2EJS*PNL101A
2HVR*UC402A	Reactor building space cooler el 175'	G	10 hp	575	3	11.20	78.40	2EJS*PNL101A
2HVR*UC402B	Reactor building space cooler el 175'	G	10 hp	575	3	11.20	78.40	2EJS*PNL101A
2HVR*UC404A	Reactor building space cooler el 196'	G	3 hp	575	3	3.60	36.00	2EJS*PNL101A
2HVR*UC404B	Reactor building space cooler el 196'	G	3 hp	575	3	3.60	25	2EJS*PNL101A
2HVR*UC414A	Reactor building space cooler el 175'	G	3 hp	575	3	3.36	25.00	2EJS*PNL104A
2HVC*XD2A	Spec filter train electric heater	G	15 kVA	575	3	15	-	2EJS*PNL102A
2HVC*UC101A	Standby switchgear room A unit cooler	G	7.5 hp	575	3	8.5	45.7	2EJS*PNL102A
2HVC*UC103A	Chloride room unit cooler	G	1 hp	575	3	1.24	10.5	2EJS*PNL102A
2HVC*UC104	Control building cable tunnel unit cooler	G	15 hp	575	3	15.0	84	2EJS*PNL102A
2HVC*UC106	Cable area base unit cooler	G	15 hp	575	3	15.0	84	2EJS*PNL102A
2HVP*UC1A	DG 1 unit cooler standby DG room	G	5 hp	575	3	5.44	39.20	2EJS*PNL102A
2HVR*UC405	Reactor building space cooler el 198'	G	3 hp	575	3	3.72	23.1	2EJS*PNL103A
2HVR*UC407A	Reactor building space cooler el 215'	G	1.5 hp	575	3	1.85	12.4	2EJS*PNL103A

TABLE 8.3-4 (Cont'd.)

Equipment Identity No.	Description[1]	Division[2]	Rating	Volts	Phase	Amps Full Load	Amps Locked Rotor	Power Source Identity No.
2HVR*UC407B	Reactor building space cooler el 215'	G	1.5 hp	575	3	1.85	12.4	2EJS*PNL103A
2HVR*UC407C	Reactor building space cooler el 215'	G	1.5 hp	575	3	1.85	12.6	2EJS*PNL103A
2HVR*UC415A	SGT space cooler el 261'	G	2 hp	575	3	2.2	16.80	2EJS*PNL103A
2HVC*CAB18A	Cont/relay rooms intake radn	G	1.5 hp	575	3	1.8	10.50	2EJS*PNL102A
2HVC*CAB18C	Cont/relay rooms intake radn	G	1.5 hp	575	3	1.8	10.50	2EJS*PNL102A
2GTS*XD1A	Filter train A heater	G	20 kW	575	3	20.0	-	2EJS*PNL104A
2HVR*UC408A	Reactor building space cooler el 240'	G	5 hp	575	3	5.68	36.70	2EJS*PNL104A
2HVR*UC408B	Reactor building space cooler el 240'	G	5 hp	575	3	5.68	36.70	2EJS*PNL104A
2HVR*UC410A	Reactor building space cooler el 240'	G	1.5 hp	575	3	1.85	12.40	2EJS*PNL104A
2HVR*UC411A	Reactor building space cooler el 261'	G	3 hp	575	3	3.36	25	2EJS*PNL104A
2HVR*UC412A	Reactor building space cooler el 261'	G	3 hp	575	3	3.0	25	2EJS*PNL101A
2BYS*CHGR2B2	125-V battery charger	Y	300 A dc	575	3	80	-	2EJS*PNL300B
2EJA*XD300B	Distribution transformer 600-V-208Y/120-V	Y	30 kVA	600	3	30	-	2EJS*PNL300B
2EJA*XD301B	Distribution transformer 600-V-120/240-V	Y	25 kVA	600	1	43.40	-	2EJS*PNL300B
2EJS*PNL301B	Switchgear room B emergency 600-V panel	Y	400 A	600	3	100	-	2EJS*PNL300B
2HVC*CH11B	Control building equipment room 306 heater	Y	60 kW	575	3	60.30	-	2EJS*PNL300B
2HVR*CAB14A	Reactor building above refuel floor radn	G	1.5 hp	575	3	1.8	10.00	2EJS*PNL104A
2HVR*CAB32A	Reactor building below refuel floor radn	G	1.5 hp	575	3	1.8	10.00	2EJS*PNL104A
2SWP*CAB146B	Service water effluent radiation monitor	Y	1.5 kW	575	3	1.84	-	2EJS*PNL301B

TABLE 8.3-4 (Cont'd.)

Equipment Identity No.	Description[1]	Division[2]	Rating	Volts	Phase	Amps Full Load	Amps Locked Rotor	Power Source Identity No.
2HVC*PNL CH12B	Control building equipment room	Y	40 kW	575	3	40.20	–	2EJS*PNL300B
2SCV*XD301B	Distribution transformer 600-V-120/240-V	Y	25 kVA	600	1	43.40	–	2EJS*PNL300B
2VBA*UPS2B	Div. IIA control UPS	Y	25 kVA	575	3	76.00	–	2EJS*PNL300B
2HVC*XD2B	Transformer 600V-480V	Y	15 kVA	575	3	15	–	2EJS*PNL301B
2HVC*UC101B	Standby switchgear room B unit cooler	Y	7.5 hp	575	3	8.5	45.7	2EJS*PNL301B
2HVC*UC103B	Chloride room unit cooler	Y	1 hp	575	3	1.24	10.5	2EJS*PNL301B
2HVC*UC105	Control building cable tunnel unit cooler	Y	5 hp	575	3	5.4	35	2EJS*PNL301B
2HVC*UC107	Cable area base unit cooler	Y	15 hp	575	3	15.0	84	2EJS*PNL301B
2HVC*UC108B	Control building standby switchgear room B	Y	10 hp	575	3	10.80	64.3	2EJS*PNL301B
2HVP*UC1B	DG 3 unit cooler standby DG room	Y	5 hp	575	3	5.44	39.20	2EJS*PNL301B
2HVR*UC401B	Reactor building space cooler el 175′	Y	2 hp	575	3	2.5	17.10	2EJS*PNL302B
2HVR*UC401C	Reactor building space cooler el 175′	Y	2 hp	575	3	2.5	16	2EJS*PNL302B
2HVR*UC401E	Reactor building space cooler el 175′	Y	2 hp	575	3	2.5	16	2EJS*PNL302B
2HVC*CAB18B	Cont/relay rooms intake radn	Y	1.5 hp	575	3	1.8	10.5	2EJS*PNL301B
2HVC*CAB18D	Cont/relay rooms intake radn	Y	1.5 hp	575	3	1.8	10.5	2EJS*PNL301B
2HVR*UC401F	Reactor building space cooler el 175′	Y	2 hp	575	3	2.5	16	2EJS*PNL302B
2HVR*UC404C	Reactor building space cooler el 196′	Y	3 hp	575	3	3.36	25	2EJS*PNL302B
2HVR*UC404D	Reactor building space cooler el 196′	Y	3 hp	575	3	3.36	25	2EJS*PNL302B
2HVR*UC414B	Reactor building space cooler el 175′	Y	3 hp	575	3	3.36	25.00	2EJS*PNL304B
2HVR*UC406	Reactor building space cooler el 198′	Y	2 hp	575	3	2.4	15.7	2EJS*PNL303B
2HVR*UC407D	Reactor building space cooler el 215′	Y	1.5 hp	575	3	1.85	12.6	2EJS*PNL303B

TABLE 8.3-4 (Cont'd.)

Equipment Identity No.	Description[1]	Division[2]	Rating	Volts	Phase	Amps Full Load	Amps Locked Rotor	Power Source Identity No.
2HVR*UC407E	Reactor building space cooler el 215'	Y	1.5 hp	575	3	1.85	12.6	2EJS*PNL303B
2HVR*UC415B	SGT space cooler el 261'	Y	2 hp	575	3	2.2	16.80	2EJS*PNL303B
2GTS*XD1B	Filter train B heater	Y	20 kW	575	3	20.0	-	2EJS*PNL304B
2HVR*UC409A	Reactor building space cooler el 240'	Y	5 hp	575	3	5.68	34.20	2EJS*PNL304B
2HVR*UC409B	Reactor building space cooler el 240'	Y	5 hp	575	3	5.68	34.2	2EJS*PNL304B
2HTS*XD004	Heat tracing transformer	Y	25 kVA	575	1	43.40	-	2EJS*PNL302B
2HTS*XD003	Heat tracing transformer	Y	25 kVA	575	1	43.40	-	2EJS*PNL302B
2HVR*UC410B	Reactor building space cooler el 240'	Y	1.5 hp	575	3	1.85	12.40	2EJS*PNL304B
2HVR*UC410C	Reactor building space cooler el 240'	Y	1.5 hp	575	3	1.85	12.60	2EJS*PNL304B
2HVR*UC411B	Reactor building space cooler el 261'	Y	3 hp	575	3	3.36	25	2EJS*PNL304B
2HVR*UC411C	Reactor building space cooler el 261'	Y	3 hp	575	3	3.36	25	2EJS*PNL304B
2HVR*UC412B	Reactor building space cooler el 261'	Y	3 hp	575	3	3.36	25	2EJS*PNL302B
2EHS*MCC101	600-V MCC screenwell el 261'	G	600 A	600	3	600	-	2EJS*US1
2EHS*MCC102	600-V MCC reactor building el 240'	G	600 A	600	3	600	-	2EJS*US1
2EHS*MCC103	600-V MCC control building el 240'	G	600 A	600	3	600	-	2EJS*US1
2EJS*PNL100A	Switchgear room A emergency 600-V panel	G	600 A	600	3	600	-	2EJS*US1
2HCS*PNL22A	Hydrogen recombiner power cabinet	G	120 kW	575	3	120	-	2EJS*US1
2HVK*CHL1A	Control building chiller 1A	G	180 kW	575	3	161	725	2EJS*US1
2HVR*UC413A	Reactor building unit cooler A	G	150 hp	575	3	140	782.40	2EJS*US1
2LAC*PNL100A	Control room A emergency lighting panel	G	400 A	600	3	400	-	2EJS*US1
2EHS*MCC301	600-V MCC screenwell el 261'	Y	600 A	600	3	600	-	2EJS*US3

TABLE 8.3-4 (Cont'd.)

Equipment Identity No.	Description[1]	Division[2]	Rating	Volts	Phase	Amps Full Load	Amps Locked Rotor	Power Source Identity No.
2EHS*MCC302	600-V MCC reactor building el 240'	Y	600 A	600	3	600	-	2EJS*US3
2EHS*MCC303	600-V MCC control building el 261'	Y	600 A	600	3	600	-	2EJS*US3
2EJS*PNL300B	Switchgear room B emergency 600-V panel	Y	600 A	600	3	600	-	2EJS*US3
2HCS*PNL22B	Hydrogen recombiner power cabinet	Y	120 kW	575	3	120	-	2EJS*US3
2HVK*CHL1B	Control building chiller 1B	Y	180 kW	575	3	161	725.00	2EJS*US3
2HVK*UC413B	Reactor building unit cooler B	Y	150 hp	575	3	140	782.4	2EJS*US3
2LAC*PNL300B	Control room B emergency lighting panel	Y	400 A	600	3	400	-	2EJS*US3
2EHS*MCC201	600-V MCC HPCS switchgear room	P	600 A	600	3	600	-	2EJS*X2
2CSL*P1	LPCS pump	G	1,500 hp	4,000	3	187.0	1,216	2ENS*SWG101
2EJS*US1	600-V US emergency switchgear room A	G	1,600 A	600	3	962	-	2ENS*SWG101
2RHS*P1A	RHR pump A	G	1,000 hp	4,000	3	126	820	2ENS*SWG101
2SFC*P1A	SFC water circulating pump A	G	450 hp	4,000	3	56	329	2ENS*SWG101
2SWP*P1A	Service water pump A	G	600 hp	4,000	3	77.2	447	2ENS*SWG101
2SWP*P1C	Service water pump C	G	600 hp	4,000	3	76	447	2ENS*SWG101
2SWP*P1E	Service water pump P1E	G	600 hp	4,000	3	76	447	2ENS*SWG101
2CSH*P1	HPCS pump	P	3,050 hp	4,000	3	378	2,457	2ENS*SWG102
2EJS*X2	4160/600-V HPCS transformer	P	225 kVA	4,160	3	31.0	-	2ENS*SWG102
2EJS*US3	600-V emergency switchgear room B	Y	1,600 A	600	3	962	-	2ENS*SWG103
2RHS*P1B	RHR pump B	Y	1,000 hp	4,000	3	126	820	2ENS*SWG103
2RHS*P1C	RHR pump C	Y	1,000 hp	4,000	3	126	820	2ENS*SWG103
2SFC*P1B	SFC water circulating pump B	Y	450 hp	4,000	3	56	329	2ENS*SWG103

TABLE 8.3-4 (Cont'd.)

Equipment Identity No.	Description[1]	Division[2]	Rating	Volts	Phase	Amps Full Load	Amps Locked Rotor	Power Source Identity No.
2SWP*P1B	Service water pump B	Y	600 hp	4,000	3	76	447	2ENS*SWG103
2SWP*P1D	Service water pump D	Y	600 hp	4,000	3	76	447	2ENS*SWG103
2SWP*P1F	Service water pump P1F	Y	600 hp	4,000	3	76	447	2ENS*SWG103
2EPS*SWG002	Emergency switchgear	Y	1,200 A	13.8 kV	3	1200	-	2EPS*SWG001
2EPS*SWG004	Emergency switchgear	Y	1,200 A	13.8 kV	3	1200	-	2EPS*SWG003
2BYS*CHGR2A1	125-V battery charger Div. I	G	300 A dc	575	3	80	-	2LAC*PNL100A
2LAC*XLE01	Lighting transformer 600-208Y/120-V	G	30 kVA	600	3	30.0	-	2LAC*PNL100A
2LAC*XLE04	Lighting transformer 600-208Y/120-V	G	30 kVA	600	3	30.0	-	2LAC*PNL100A
2LAC*XLE06	Lighting transformer 600-208Y/120-V	G	30 kVA	600	3	30	-	2LAC*PNL100A
2SCM*XD101A	Distribution transformer 600-120/240-V	G	25 kVA	600	1	43.40	-	2LAC*PNL100A
2SCM*XD102A	Distribution transformer 600-120/240-V	G	25 kVA	600	1	43.40	-	2LAC*PNL100A
2SCM*XD103A	Distribution transformer 600-120/240-V	G	25 kVA	600	1	43.40	-	2LAC*PNL100A
2VBA*UPS2A	Div. IA control UPS	G	25 kVA	575	1	68.00	-	2LAC*PNL100A
2BYS*CHGR2B1	125-V battery charger standby Div. II	Y	300 A dc	575	3	80	-	2LAC*PNL300B
2LAC*XLE02	Lighting transformer 600-208Y/120-V	Y	30 kVA	600	3	30.0	-	2LAC*PNL300B
2LAC*XLE05	Lighting transformer 600-208Y/120-V	Y	30 kVA	600	3	30.0	-	2LAC*PNL300B
2LAC*XLE07	Lighting transformer 600-208Y/120-V	Y	30 kVA	600	3	30	-	2LAC*PNL300B
2SCM*XD301B	Distribution transformer 600-V-120/240-V	Y	25 kVA	600	1	43.40	-	2LAC*PNL300B
2SCM*XD104A	Distribution transformer	G	25 kVA	600	1	43.40	-	2LAC*PNL100A
2SCM*XD105A	Distribution transformer	G	25 kVA	600	1	43.40	-	2LAC*PNL100A
2SCM*XD302B	120-V distribution panel	Y	25 kVA	600	1	43.40	-	2LAC*PNL300B

TABLE 8.3-4 (Cont'd.)

Equipment Identity No.	Description[1]	Division[2]	Rating	Volts	Phase	Amps Full Load	Amps Locked Rotor	Power Source Identity No.
2SCM*XD303B	120-V distribution panel	Y	25 kVA	600	1	43.40	–	2LAC*PNL300B
2VBA*UPS2B	Div. IIA control UPS	Y	25 kVA	575	1	68.00	–	2LAC*PNL300B
2LAC*PNLE01	Lighting panel	G	100 A	208	3	100 A	–	2LAC*XLE01
2LAC*PNLE02	Lighting panel	Y	100 A	208	3	100 A	–	2LAC*XLE02
2LAC*PNLE04	Lighting panel	G	100 A	208	3	100 A	–	2LAC*XLE04
2LAC*PNLE05	Lighting panel	Y	100 A	208	3	100 A	–	2LAC*XLE05
2LAC*PNLE06	Lighting panel	G	100 A	208	3	100 A	–	2LAC*XLE06
2LAC*PNLE07	Lighting panel	Y	100 A	208	3	100 A	–	2LAC*XLE07
2LAC*PNLE03	Lighting panel	P	100 A	208	3	100 A	–	2LAC*XLE03
2SCM*XD304B	120-V distribution transformer	Y	25 kVA	600	1	43.40	–	2LAC*PNL300B
2SCM*XD305B	120-V distribution transformer	Y	25 kVA	600	1	43.40	–	2LAC*PNL300B
2RCS*MOV10A	Recirculation pump A suction valve	N	4 hp	575	3	5.6	47.80	2NHS-MCC011
2RCS*MOV18A	Recirculation pump A discharge valve	N	4 hp	575	3	5.6	49.7	2NHS-MCC011
2WCS*MOV101	RCS to water cleanup	N	0.70 hp	575	3	1.80	9.00	2NHS-MCC011
2WCS*MOV103	RCS to water cleanup	N	1.6 hp	575	3	3.2	16	2NHS-MCC011
2WCS*MOV104	RCS to water cleanup	N	0.7 hp	575	3	1.9	10.0	2NHS-MCC011
2WCS*MOV105	RCS to water cleanup	N	0.7 hp	575	3	1.8	9.0	2NHS-MCC011
2DER*MOV128	RPV drain isol. valve	N	0.33 hp	575	3	0.64	4.5	2NHS-MCC012
2DER*MOV129	RPV drain isol. valve	N	0.33 hp	575	3	0.64	4.5	2NHS-MCC012
2MSS*MOV108	Vent valve	N	0.33 hp	575	3	0.64	4.5	2NHS-MCC012
2MSS*MOV189	Main steam valve	N	0.33 hp	575	3	0.64	4.5	2NHS-MCC012
2MSS*MOV207	Main steam valve	N	hp	575	3	3.2	20.0	2NHS-MCC012

TABLE 8.3-4 (Cont'd.)

Equipment Identity No.	Description[1]	Division[2]	Rating	Volts	Phase	Amps Full Load	Amps Locked Rotor	Power Source Identity No.
2RCS*MOV10B	Recirculation pump B suction valve	N	4 hp	575	3	5.6	47.8	2NHS-MCC012
2RCS*MOV18B	Recirculation pump B discharge valve	N	4 hp	575	3	6.0	38.0	2NHS-MCC012
2MHR*CRN1	Reactor building polar crane motor el 387'	N	200 hp	575	3	192	1,152	2NJS-US2
2ENS*SWG102	4160-V HPCS switchgear 102	P	1,200 A	4,160	3	1200	-	2NNS-SWG16/17
2ENS*SWG101	4160-V emergency switchgear 101	G	1,200 A	4,160	3	1200	-	2NNS-SWG16/18
2ENS*SWG103	4160-V emergency switchgear 102	Y	1,200 A	4,160	3	1200	-	2NNS-SWG17/18
2EPS*SWG001	Emergency switchgear	G	1,200 A	13.8 kV	3	1200	-	2NPS-SWG001
2EPS*SWG003	Emergency switchgear	G	1,200 A	13.8 kV	3	1200	-	2NPS-SWG003
2SCV*PNL101A	GTS misc. 120/240-V panel	G	150 A	240	1	104	-	2SCV*XD101A
2SCV*PNL200P	HPCS switchgear room 120-V misc. panel	P	225 A	240	1	104	-	2SCV*XD200P
2SCV*PNL301B	GTS misc. 120/240-V panel	Y	150 A	240	1	104	-	2SCV*XD301B
2VBS*PNL101A	120-V UPS distribution panel	G	200 A	120	1	200	-	2VBA*UPS2A
2VBS*PNL102A	120-V UPS distribution panel	G	200 A	120	1	100	-	2VBA*UPS2A
2VBS*PNL301B	120-V UPS distribution panel	Y	200 A	120	1	200	-	2VBA*UPS2B
2VBS*PNL302B	120-V UPS distribution panel	Y	200 A	120	1	100	-	2VBA*UPS2B
2HTS*XD001	Heat tracing transformer	G	25 kVA	575	1	43.40	-	2EJS*PNL103A
2HTS*XD002	Heat tracing transformer	G	25 kVA	575	1	43.40	-	2EJS*PNL103A
2SWP*CAB146A	Service water effluent radiation monitor	G	1.5 hp	575	3	1.84	-	2EJS*PNL102A
2GTS*PNL5A	Reactor building in/out diff press	G	1.5 hp	575	1	3.0	-	2EJS*PNL103A
2CMS*CAB10A	Containment atmosphere leakage radn	G	1.5 hp	575	3	1.8	10.0	2EJS*PNL104A

TABLE 8.3-4 (Cont'd.)

Equipment Identity No.	Description[1]	Division[2]	Rating	Volts	Phase	Amps Full Load	Amps Locked Rotor	Power Source Identity No.
2SWP*CAB23A	RHR service water A radiation monitor	G	1.5 kW	575	3	1.84	10.0	2EJS*PNL104A
2CMS*CAB10B	Containment atmosphere leakage radn	Y	1.5 hp	575	3	1.8	10.0	2EJS*PNL303B
2HVR*CAB14B	Reactor building above refuel floor radn	Y	1.5 hp	575	3	1.8	10.0	2EJS*PNL303B
2HVR*CAB32B	Reactor building below refuel floor radn	Y	1.5 hp	575	3	1.8	10.0	2EJS*PNL303B
2GTS*PNL5B	Reactor building in/out diff press	Y	1.5 hp	575	3	3.0	—	2EJS*PNL303B
2SWP*CAB23B	RHR service water B radiation monitor	Y	1.5 kW	575	3	1.84	10.0	2EJS*PNL303B

[1] KEY TO DESCRIPTION:

A/C - Air conditioning
CWS - Circulating water system
DG - Diesel generator
GTS - Gas treatment system
HPCS - High-pressure core spray
HVR - Reactor building ventilation
LPCS - Low-pressure core spray
MCC - Motor control center
MSIV - Main steam isolation valve
PIT - Pressure indicator transmitter
RBCLCW - Reactor building closed loop cooling water
RCIC - Reactor core isolation cooling
RHR - Residual heat removal
RPS - Reactor protection system
RWCU - Reactor water cleanup
SFC - Spent fuel cooling and cleanup
SFP - Spent fuel pool
SGTS - Standby gas treatment system
SWT - Service water traveling screens, wash and disposal
TBCLCW - Turbine building closed loop cooling water
UPS - Uninterruptible power supply

TABLE 8.3-4 (Cont'd.)

(2) KEY TO DIVISION:

G	=	Green (Division I - ECCS, HVAC, SWP, etc. D1-RPS, NMS, NSSSS Ch-1, D1-MSLIV)
Y	=	Yellow (Division II - ECCS, HVAC, SWP, etc. D2-RPS, NMS, NSSSS Ch-1, D2-MSLIV)
P	=	Purple (Division III - ECCS (HPCS))
O	=	Orange (D3-RPS, NMS, NSSS)
B	=	Blue (D4-RPS, NMS, NSSSS)
G/W	=	Green/White (Ch-2, D1-MSLIV, Ch-A1, D1-RPS trip)
Y/W	=	Yellow/White (Ch-2, D2-MSLIV, Ch-B1, D2-RPS trip)
N	=	Noncolor (Nonsafety systems)
B/W	=	Blue/White (Ch-B2, D4-RPS trip)
O/W	=	Orange/White (Ch-A2, D3-RPS trip)

TABLE 8.3-5

DIVISION I STANDBY DIESEL GENERATOR 2EGS*EG1
LOAD SUMMARY

SIMULTANEOUS LOOP AND LOCA, LOCA WITH DELAYED LOOP

	Start KW	Start KVAR	Run KW	Run KVAR	Total KW	Total KVAR
0 <= T < 0.1 sec	9828	43509	0	0	9828	43509
0.1 <= T < 0.9 sec	1918	3669	163	75	2081	3744
0.9 <= T < 2.9 sec	3036	9201	163	75	3199	9276
2.9 <= T < 5.5 sec	1918	3669	885	383	2803	4052
5.5 <= T < 6 sec	3129	11140	885	383	4014	11523
6 <= T < 8.8 sec	1211	7471	1714	791	2925	8262
8.8 <= T < 9 sec	0	0	2791	1233	2791	1233
9 <= T < 15 sec	11	15	2790	1232	2801	1247
15 <= T < 25 sec	0	0	2793	1234	2793	1234
25 <= T < 30 sec	315	843	2793	1234	3108	2077
30 <= T < 31 sec	693	1855	2793	1234	3486	3089
31 <= T < 32 sec	378	1012	2920	1313	3298	2325
32 <= T < 36 sec	1092	3811	2920	1313	4012	5124
36 <= T < 38 sec	714	2799	3073	1409	3787	4208
38 <= T < 70 sec	0	0	3516	1640	3516	1640
70 <= T < 76 sec	714	2799	3516	1640	4230	4439
76 <= T < 98 sec	0	0	3958	1870	3958	1870
98 <= T < 104 sec	86	228	3958	1870	4044	2098
104 <= T < 2 min	0	0	3993	1891	3993	1891
2 min <= T < 10 min	0	0	3821	1784	3821	1784

TABLE 8.3-5 (Cont'd.)

SIMULTANEOUS LOOP AND LOCA, LOCA WITH DELAYED LOOP (cont'd.)

	Start KW	Start KVAR	Run KW	Run KVAR	Total KW	Total KVAR
10 min <= T < 10.1 min	488	674	3821	1789	4309	2458
10.1 min <= T < 12 min	0	0	3962	1872	3962	1872
12 min <= T < 2 hr	0	0	3842	1796	3842	1796
2 hr <= T < 2 hr 6 sec	590	2018	3842	1796	4432	3814
2 hr 6 sec <= T < 2.5 hr	0	0	4172	1934	4172	1934
T => 2.5 hr	0	0	4292	1934	4292	1934

TABLE 8.3-5 (Cont'd.)

LOOP WITH DELAYED LOCA‡

	Start KW	Start KVAR	Run KW	Run KVAR	Total KW	Total KVAR
0 <= T < 0.1 sec	9828	43509	1165	538	10993	44047
0.1 <= T < 5.5 sec	1918	3669	1328	613	3246	4282
5.5 <= T < 6 sec	3129	11140	1328	613	4457	11753
6 <= T < 8.8 sec	1211	7471	2157	1021	3368	8492
8.8 <= T < 9 sec	0	0	3234	1464	3234	1464
9 <= T < 15 sec	11	15	3234	1464	3244	1478
15 <= T < 25 sec	0	0	3236	1465	3236	1465
25 <= T < 30 sec	315	843	3236	1465	3551	2308
30 <= T < 31 sec	693	1855	3236	14365	3929	3320
31 <= T < 36 sec	378	1012	3363	1544	3741	2556
36 <= T < 70 sec	0	0	3516	1640	3516	1640
70 <= T < 76 sec	714	2799	3516	1640	4230	4439
76 <= T < 98 sec	0	0	3958	1870	3958	1870
98 <= T < 104 sec	86	228	3958	1870	4044	2098
104 <= T < 2 min	0	0	3993	1891	3993	1891
2 min <= T < 10 min	0	0	3821	1784	3821	1784
10 min <= T < 10.1 min	488	674	3821	1784	4309	2458
10.1 min <= T < 12 min	0	0	3962	1872	3962	1872
12 min <= T < 2 hr	0	0	3842	1796	3842	1796
2 hr <= T < 2 hr 6 sec	590	2018	3842	1796	4432	3814

TABLE 8.3-5 (Cont'd.)

LOOP WITH DELAYED LOCA# (Cont'd.)

	Start KW	Start KVAR	Run KW	Run KVAR	Total KW	Total KVAR
2 hr 6 sec <= T < 2.5 hr	0	0	4172	1934	4172	1934
T => 2.5 hr	0	0	4292	1934	4292	1934

TABLE 8.3-5 (Cont'd.)

LOOP WITH UNIT TRIP

	Start KW	Start KVAR	Run KW	Run KVAR	Total KW	Total KVAR
0 <= T < 0.1 sec	10005	45522	0	0	10005	45522
0.1 <= T < 6 sec	1615	3282	343	162	1958	3444
6 <= T < 25 sec	0	0	1089	517	1089	517
25 <= T < 30 sec	315	843	1089	517	1404	1360
30 <= T < 31 sec	693	1855	1089	517	1782	2372
31 <= T < 32 sec	378	1012	1217	597	1595	1609
32 <= T < 36 sec	1092	3811	1217	597	2309	4408
36 <= T < 38 sec	714	2799	1370	692	2084	3491
38 <= T < 70 sec	0	0	1813	922	1813	922
70 <= T < 76 sec	714	2799	1813	922	2527	3721
76 <= T < 2 min	0	0	2255	1153	2255	1153
120 <= T < 10 min	0	0	2169	1099	2169	1099
10 min <= T < 10.1 min	706	964	2169	1099	2875	2063
10.1 min <= T < 12 min	0	0	2371	1226	2371	1226
12 min <= T < 30 min	0	0	2195	1115	2195	1115
30 min <= T < 30.1 min	4	6	2195	1115	2195	1115
30.1 min <= T < 32 min	0	0	2196	1116	2196	1116
32 min <= T < 1 hr	0	0	2195	1115	2195	1115
1 hr <= T < 1 hr 2 sec	1118	5532	2195	1115	3313	6647
1 hr 2 sec <= T < 2 hr	0	0	2917	1422	2917	1422
2 hr <= T < 2 hr 6 sec	590	2018	2917	1422	3507	3440
2 hr 6 sec <= T	0	0	3247	1560	3247	1560

TABLE 8.3-5 (Cont'd.)

NOTES:

Under the scenario of a LOOP with a subsequent LOCA, the Class 1E loads on the standby diesel generators will be administratively controlled by NMP2 Station Blackout operating procedures.

* Time, T, is measured from the instant the diesel generator attains its rated voltage and frequency and is connected to its bus by closing the supply breaker 101-1 (2ENS*SWG101).

TABLE 8.3-6

DIVISION II STANDBY DIESEL GENERATOR 2EGS*EG3
LOAD SUMMARY

SIMULTANEOUS LOOP AND LOCA, LOCA WITH DELAYED LOOP

	Start KW	Start KVAR	Run KW	Run KVAR	Total KW	Total KVAR
0 <= T < 0.1 sec	9749	43058	0	0	9749	43058
0.1 <= T < 0.9 sec	1736	3307	141	50	1877	3357
0.9 <= T < 2.9 sec	2854	8839	141	50	2995	8889
2.9 <= T < 5 sec	1784	3370	863	358	2647	3728
5 <= T < 5.5 sec	2099	4213	863	358	2962	4571
5.5 <= T < 6 sec	3217	9745	863	358	4080	10103
6 <= T < 7.5 sec	1481	6438	1617	746	3098	7184
7.5 <= T < 9 sec	363	906	2339	1054	2702	1960
9 <= T < 11 sec	326	858	2353	1062	2679	1920
11 <= T < 14 sec	11	15	2481	1142	2492	1157
14 <= T < 15 sec	23	30	2481	1142	2504	1172
15 <= T < 20 sec	11	15	2484	1144	2495	1159
20 <= T < 30 sec	0	0	2487	1146	2487	1146
30 <= T < 32 sec	378	1012	2487	1146	2865	2158
32 <= T < 36 sec	1092	3811	2487	1146	3579	4957
36 <= T < 38 sec	714	2799	2641	1241	3354	4040
38 <= T < 70 sec	0	0	3083	1471	3083	1471
70 <= T < 76 sec	714	2799	3083	1471	3797	4270
76 <= T < 98 sec	0	0	3526	1701	3526	1701
98 <= T < 104 sec	86	228	3526	1701	3612	1929

TABLE 8.3-6 (Cont'd.)

SIMULTANEOUS LOOP AND LOCA, LOCA WITH DELAYED LOOP (Cont'd.)

	Start KW	Start KVAR	Run KW	Run KVAR	Total KW	Total KVAR
104 <= T < 120 sec	0	0	3560	1722	3560	1722
120 <= T < 218 sec	0	0	3406	1629	3406	1629
218 <= T < 10 min	0	0	3405	1629	3405	1629
10 min <= T < 10.1 min	209	327	3405	1629	3614	1955
10.1 min <= T < 12 min	0	0	3468	1669	3468	1669
12 min <= T < 1 hr	0	0	3425	1641	3425	1641
1 hr <= T < 1 hr 6 sec	4	6	3425	1641	3429	1647
1 hr 6 sec <= T < 1 hr 2 min	0	0	3426	1642	3426	1642
1 hr 2 min <= T < 2 hr	0	0	3425	1641	3425	1641
2 hr <= T < 2 hr 6 sec	590	2018	3425	1641	4015	3659
2 hr 6 sec <= T < 2.5 hr	0	0	3755	1779	3755	1779
T => 2.5 hr	0	0	3875	1779	3875	1779

TABLE 8.3-6 (Cont'd.)

LOOP WITH DELAYED LOCA#

	Start KW	Start KVAR	Run KW	Run KVAR	Total KW	Total KVAR
0 <= T < 0.1 sec	9749	43058	1165	538	10914	43596
0.1 <= T < 0.9 sec	1736	3307	1306	588	3042	3895
0.9 <= T < 2.9 sec	2854	8839	1306	588	4160	9427
2.9 <= T < 5 sec	1784	3370	2028	896	3812	4266
5 <= T < 6 sec	2099	4213	2028	896	4127	5109
6 <= T < 9 sec	363	906	2782	1284	3145	2190
9 <= T < 11 sec	326	858	2796	1292	3122	2150
11 <= T < 14 sec	11	15	2924	1372	2935	1387
14 <= T < 15 sec	23	30	2924	1372	2946	1402
15 <= T < 20 sec	11	15	2927	1374	2938	1389
20 <= T < 30 sec	0	0	2930	1376	2930	1376
30 <= T < 36 sec	378	1012	2930	1376	3308	2388
36 <= T < 70 sec	0	0	3083	1471	3083	1471
70 <= T < 76 sec	714	2799	3083	1471	3797	4270
76 <= T < 98 sec	0	0	3526	1701	3526	1701
98 <= T < 104 sec	86	228	3526	1701	3612	1929
104 <= T < 120 sec	0	0	3560	1722	3560	1722
120 <= T < 218 sec	0	0	3406	1629	3406	1629
218 <= T < 10 min	0	0	3405	1629	3405	1629
10 min <= T < 10.1 min	209	327	3405	1629	3614	1955

TABLE 8.3-6 (Cont'd.)

LOOP WITH DELAYED LOCA# (Cont'd.)

	Start KW	Start KVAR	Run KW	Run KVAR	Total KW	Total KVAR
10.1 min <= T < 12 min	0	0	3468	1669	3468	1669
12 min <= T < 1 hr	0	0	3425	1641	3425	1641
1 hr <= T < 1 hr 6 sec	4	6	3425	1641	3429	1647
1 hr 6 sec <= T < 1 hr 2 min	0	0	3426	1642	3426	1642
1 hr 2 min <= T < 2 hr	0	0	3425	1641	3425	1641
2 hr <= T < 2 hr 6 sec	590	2018	3425	1641	4015	3659
2 hr 6 sec <= T < 2.5 hr	0	0	3755	1779	3755	1779
T => 2.5 hr	0	0	3875	1779	3875	1779

TABLE 8.3-6 (Cont'd.)

LOOP WITH UNIT TRIP

	Start KW	Start KVAR	Run KW	Run KVAR	Total KW	Total KVAR
0 <= T < 0.1 sec	9449	42657	0	0	9449	42657
0.1 <= T < 5 sec	1436	2906	141	50	1577	2956
5 <= T < 6 sec	1751	3749	141	50	1892	3799
6 <= T < 11 sec	315	843	809	384	1124	1227
11 <= T < 30 sec	0	0	937	424	937	464
30 <= T < 32 sec	378	1012	937	464	1315	1476
32 <= T < 36 sec	1092	3811	937	464	2029	4275
36 <= T < 38 sec	714	2799	1090	559	1804	3358
38 <= T < 70 sec	0	0	1533	789	1533	789
70 <= T < 76 sec	714	2799	1533	789	2247	3588
76 <= T < 2 min	0	0	1976	1019	1976	1019
120 <= T < 10 min	0	0	1904	978	1904	978
10 min <= T < 10.1 min	390	568	1904	978	2294	1546
10.1 min <= T < 12 min	0	0	2019	1051	2019	1051
12 min <= T < 1 hr	0	0	1929	995	1929	995
1 hr <= T < 1 hr 2 sec	1122	5532	1929	995	3051	6527
1 hr 2 sec <= T < 1 hr 6 sec	4	6	2651	1303	2656	1309
1 hr 6 sec <= T < 1 hr 2 min	0	0	2652	1304	2652	1304
1 hr 2 min <= T < 2 hr	0	0	2651	1303	2651	1303
2 hr <= T < 2 hr 6 sec	590	2018	2651	1303	3251	3321

TABLE 8.3-6 (Cont'd.)

<u>LOOP WITH UNIT TRIP</u> (Cont'd.)

	Start KW	Start KVAR	Run KW	Run KVAR	Total KW	Total KVAR
2 hr 6 sec <= T	0	0	2982	1440	2982	1440

NOTES:

- `#` Under the scenario of a LOOP with a subsequent LOCA, the Class 1E loads on the standby diesel generators will be administratively controlled by NMP2 Station Blackout operating procedures.

- * Time, T, is measured from the instant the diesel generator attains its rated voltage and frequency and is connected to its bus by closing the supply breaker 103-14 (2ENS*SWG103).

TABLE 8.3-10

DIVISION III 125-V DC BATTERY 2BYS*BAT2C
LOAD PROFILE

A. Normal Load

Description of Load	0-1 Minute (amp)	1-120 Minutes (amp)
4.16-kV switchgear (one breaker closing)	14.0	-
4.16-kV switchgears control relays/indications	2.1	2.1
Diesel generator field flashing	2	-
Solenoid valves (diesel air start)	2	-
Diesel generator fuel pump	6.3	6.3
Relays and indicator lamps in diesel generator panels	2.5	2.3
Relays and indicator lamps in main control room panels	3.7	3.7
Turbocharger lube oil pump*	7.4	-
Lube oil circulating pump*	6.9	-
Division III transient analysis recorder system local panel (2CES*PNL520)	1	1
Total	47.9	15.4

*These pumps run only during loss of ac power at E22-S002 (HPCS motor control center) when the ac pump motors are not running.

TABLE 8.3-11

NORMAL 125-V DC BATTERY 2BYS-BAT1A
LOAD PROFILE

Description of Load	0-1 Minute (amp)	1-45 Minutes (amp)	45-90 Minutes (amp)	90-120 Minutes (amp)
Circuit breaker trip	274.66	-	-	-
Indicating lights for switchyard and normal 13.8-kV, 4.16-kV, and dc switchgear	13.98	13.98	13.98	13.98
Misc. control and instrumentation loads	67	67	67	67
Essential lighting (2VBB-UPS1C)	545	591	637	-
Normal UPS system: (2VBB-UPS1A)	504	547	590	618
Bearing lube oil pump	684	207.25	-	-
Standby diesel generator fuel pump	30(1)	7.5	7.5	7.5
Random loads (1 min)	-	78	-	-
Total	2118.64(1)	1433.73(2)	1315.48	706.48

(1)30-amp locked rotor used as a random load only.
(2)Does not include the random load of 78 amps which appears for 1 min only at the end of this period.

TABLE 8.3-12

NORMAL 125-V DC BATTERY 2BYS-BAT1B
LOAD PROFILE

Description of Load	0-1 Minute (amp)	1-90 Minutes (amp)	90-120 Minutes (amp)
Circuit breaker trip	274.66	-	-
Indicating lights for switchyard and normal 13.8-kV, 4.16-kV, and dc switchgear	12.93	12.93	12.93
Misc. control and instrumentation loads	60	60	60
Essential lighting (2VBB-UPS1D)	546	638	-
Normal UPS system: (2VBB-UPS3B)	94	110	115
Emergency seal oil pump	370	141	-
Turbine gland seal compressor	145	58	58
Standby diesel generator fuel pump	30(1)	7.5	7.5
Random loads (1 min)	-	71	-
Total	1532.59(1)	1027.43(2)	253.43

(1)30-amp locked rotor used as a random load only.
(2)Does not include the random load of 71 amps which appears for 1 min only at the end of this period.

TABLE 8.3-13

NORMAL 125-V DC BATTERY 2BYS-BAT1C
LOAD PROFILE

Description of Load	0-1 Minute (amps)	1-119 Minutes (amps)	119-120 Minutes (amps)	
Plant computer UPS system (2VBB-UPS1G)	505	562	619	23
Total	505	562	619	23

TABLE 8.3-14

NORMAL 24-V BATTERY 2BWS-3A AND 2BWS-3C
LOAD PROFILE

Description of Load	0-1 Minute (amp)	1-119 Minutes (amp)	119-120 Minutes (amp)
Control	0.3125	0.3125	0.3125
Neutron monitoring	5.00	5.00	5.00
Total	5.3125	5.3125	5.3125

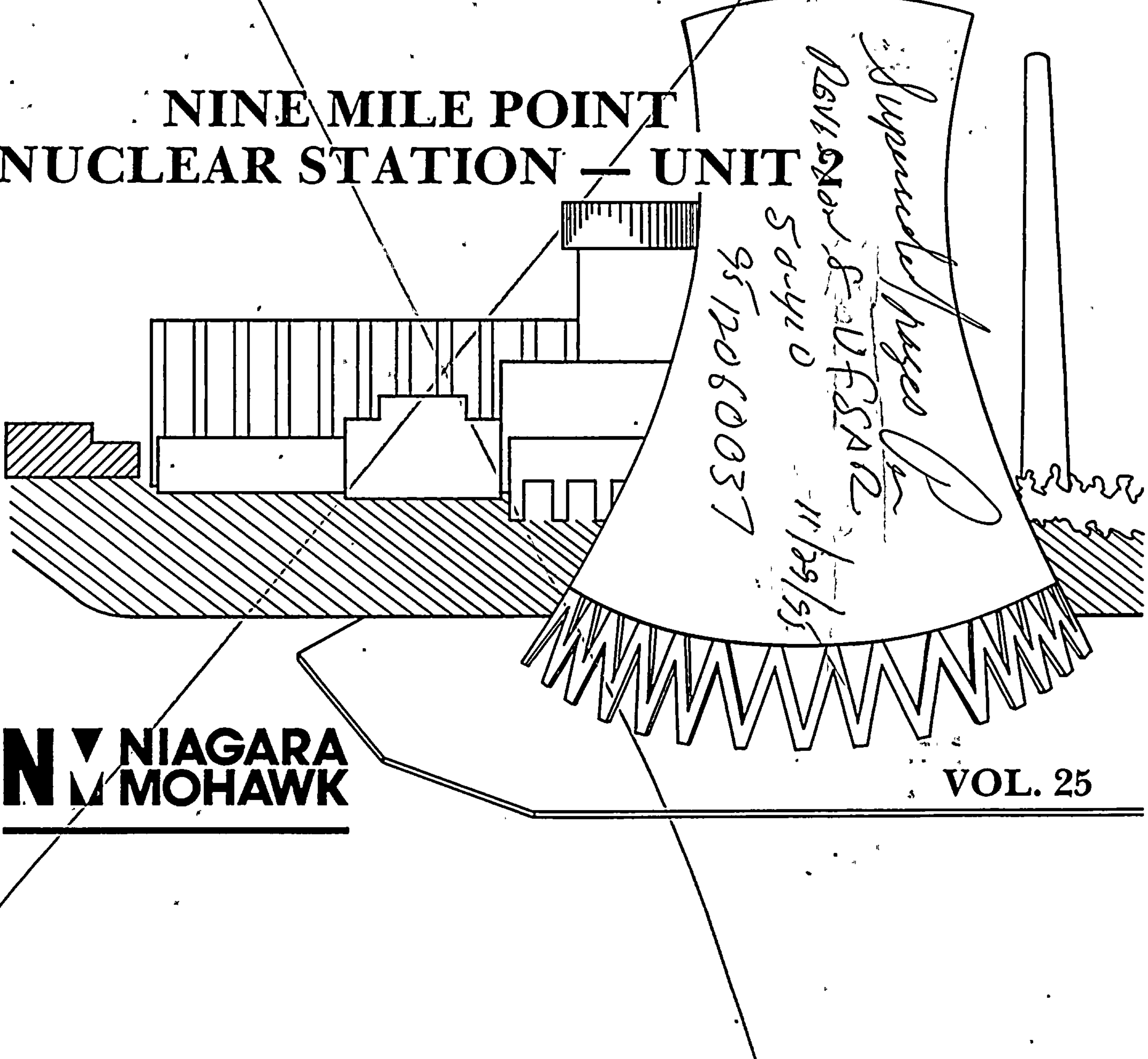
TABLE 8.3-15

NORMAL 24-V BATTERY 2BWS-3B AND 2BWS-3D
LOAD PROFILE

Description of Load	0-1 Minute (amp)	1-119 Minutes (amp)	119-120 Minutes (amp)
Control	0.3125	0.3125	0.3125
Neutron monitoring	5.000	5.000	5.000
Total	5.3125	5.3125	5.3125

UPDATED SAFETY ANALYSIS REPORT

NINE MILE POINT NUCLEAR STATION — UNIT 2



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ventilation and mechanical vacuum pump are based on NUREG-0016 Revision 01. The design annual average release concentrations are shown in Table 11.3-3 as activity in uCi/cc, and as a fraction of maximum permissible concentration.

11.3.3.3 Dilution Factors

The atmospheric dilution factor associated with normal plant releases is based on the average annual meteorological conditions applicable to the site as well as the effective release height of the effluent discharge pathway. The site meteorological conditions are given in Section 2.3.

11.3.3.4 Estimated Doses

A summary of the estimated annual radiation doses is presented in Appendix 11A and shows that the estimated annual doses from gaseous effluents are below the dose criteria set forth in 10CFR50 Appendix I and are well below the dose criteria specified in 40CFR190 and 10CFR20.

The maximum hypothetical gamma and beta air doses from noble gas releases occur at the exclusion area boundary (EAB), 1,603 m east of the site. The doses at this location are estimated to be 0.06 mrad/yr gamma and 0.04 mrad/yr beta, as compared with the 10CFR50 Appendix I design objective for gamma and beta air doses of 10.0 mrad/yr and 20.0 mrad/yr, respectively.

TABLE 11.3-1

EXPECTED RADIOACTIVE GASEOUS EFFLUENT FROM ALL SOURCES (CI/YR)

Isotope	Reactor Building	Turbine Building	Radwaste Building	Mechanical Vacuum Pump	Off-Gas System	
I-131	2.3-02	1.1-01	1.8-02	4.3-02	-	
I-132	4.1-01	1.9+00	1.9-01	7.4-01	-	12
I-133	3.0-01	1.6+00	1.5-01	5.9-01	-	
I-134	7.5-01	4.7+00	3.4-01	1.7+00	-	
I-135	3.2-01	1.6+00	1.4-01	6.1-01	-	
Kr-83m	-	-	-	-	7.7-01	
Kr-85m	3.0+00	2.5+01	-	-	7.3+02	
Kr-85	-	-	-	-	2.4+02	
Kr-87	2.1+00	6.1+01	-	-	2.2-02	
Kr-88	3.8+00	9.1+01	-	-	1.7+02	
Kr-89	2.4+00	5.8+02	2.9+01	-	-	
Xe-131m	-	-	-	-	5.3+01	
Xe-133m	-	-	-	-	4.6+00	
Xe-133	1.1+02	1.5+02	2.2+02	1.3+03	5.8+03	
Xe-135m	6.1+01	4.0+02	5.3+02	-	-	
Xe-135	1.3+02	3.3+02	2.8+02	5.0+02	2.2-12	
Xe-137	1.9+02	1.0+03	8.3+01	-	-	
Xe-138	7.3+00	1.0+03	2.0+00	-	-	
Cr-51	9.7-04	9.0-04	7.0-06	-	-	
Mn-54	1.3-03	6.0-04	4.0-05	-	-	
Fe-59	3.5-04	1.0-04	3.0-06	-	-	
Co-58	2.9-04	1.0-03	2.0-06	-	-	
Co-60	5.1-03	1.0-03	7.0-05	-	-	
Zn-65	4.1-03	6.0-03	3.0-06	-	-	
Sr-89	1.7-04	6.0-03	-	-	-	
Sr-90	7.9-06	2.0-05	-	-	-	
Zr-95	6.2-04	4.0-05	8.0-06	-	-	
Nb-95	1.1-02	6.0-06	4.0-08	-	-	
Mo-99	6.3-02	2.0-03	3.0-08	-	-	
Ru-103	5.9-04	5.0-05	1.0-08	-	-	
Ag-110m	2.6-06	-	-	-	-	
Sb-124	2.1-05	1.0-04	7.0-07	-	-	
Cs-134	4.3-03	2.0-04	2.4-05	-	-	
Cs-136	4.9-04	1.0-04	-	-	-	
Cs-137	5.7-03	1.0-03	4.0-05	-	-	
Ba-140	1.4-02	1.0-02	4.0-08	-	-	
Ce-141	7.9-04	1.0-02	7.0-08	-	-	

TABLE 11.3-1 (Cont)

Isotope	Reactor Building	Turbine Building	Radwaste Building	Mechanical Vacuum Pump	Off-Gas System
Ar-41	1.5+01	-	-	-	7.1+01
H-3	2.6+01	2.6+01	-	-	-
C-14	-	-	-	-	9.5+00

NOTE: 3.0+00 = 3.0×10^{0}

TABLE 11.3-2

DATA USED IN CALCULATING ANNUAL RELEASES OF RADIOACTIVE GASEOUS EFFLUENTS

Parameter	Data
Maximum core thermal power	3,489 (MWt)
Total steam flow rate	1.5 x 10^7 lb/hr
Offgas charcoal bed holdup* times per NUREG-0016 (Kr)	29.6 hr
(Xe)	21.75 days
Plant capacity factor	80%
Expected releases source term failed fuel basis	0.05 Ci/sec (after 30 min)
Design releases source term failed fuel basis	0,3489 Ci/sec (after 30 min)
Offgas system charcoal mass/train	48,000 lb
Dynamic adsorption coefficients (Kr)	25 cm^3/gm
(Xe)	440 cm^3/gm
Charcoal delay system normal operating temperature	70°F
Charcoal delay system dew point temperature	-20°F
Ventilation systems	See Section 9.4
Decontamination factors	
Normal operations	
Radwaste building	99% efficient HEPA filter
All other buildings/systems	Unfiltered
Shutdown	
Radwaste building	99% efficient HEPA filter

TABLE 11.3-2 (Cont'd.)

Parameter	Data
Reactor building	10% of effluents - unfiltered 90% of effluents - 90% efficient iodine filter 99% efficient HEPA filter
All other buildings/systems	Unfiltered

* The offgas system holdup times presented are based on NUREG-0016 calculation methods. The holdup time values in this table are used only to calculate annual radioactive gaseous effluent releases from Unit 2. The offgas system holdup times discussed in Section 11.3.2.1 are based on design system parameters.

TABLE 11.3-3

DESIGN ANNUAL AVERAGE GASEOUS RELEASES VS. MPC

Isotope	Main Stack Releases: Continuous Annual Average — Activity at EAB (uCi/cc)	Main Stack Releases: Continuous Annual Average — Fraction of MPC	Main Stack Releases: Intermittent (MVP) — Activity at EAB (uCi/cc)	Main Stack Releases: Intermittent (MVP) — Fraction of MPC	Radwaste/Reactor Building Vent Releases: Annual Average — Activity at EAB (uCi/cc)	Radwaste/Reactor Building Vent Releases: Annual Average — Fraction of MPC	MPC* (uCi/cc)	Total Release (fraction of MPC)	
I-131	3.2-16	3.2-06	7.1-16	7.1-06	1.9-15	1.9-05	1.0-10	3.0-05	
I-132	3.9-15	1.3-06	9.1-15	3.0-06	2.1-14	7.0-06	3.0-09	1.1-05	12
I-133	3.2-15	7.7-06	7.0-15	1.8-05	1.5-14	3.9-05	4.0-10	6.5-05	
I-134	9.8-15	1.6-06	2.0-14	3.4-06	3.8-14	6.3-06	6.0-09	1.1-05	
I-135	3.3-15	3.3-06	7.8-15	7.8-06	1.6-14	1.6-05	1.0-09	2.7-05	12
Kr-83m	1.5-15	5.0-08	-	-	-	-	3.0-08	5.0-08	
Kr-85m	1.5-12	1.5-05	-	-	1.0-13	1.0-06	1.0-07	1.6-05	
Kr-85	4.9-13	1.6-06	-	-	-	-	3.0-07	1.6-06	
Kr-87	1.2-13	6.2-06	-	-	7.6-14	3.8-06	2.0-08	1.0-05	
Kr-88	5.2-13	2.6-05	-	-	1.3-13	6.6-06	2.0-08	3.3-05	
Kr-89	1.2-12	4.0-05	-	-	1.1-12	3.8-05	3.0-08	7.8-05	12
Xe-131m	1.0-13	2.6-07	-	-	-	-	4.0-07	2.6-07	
Xe-133m	9.0-15	3.0-08	-	-	-	-	3.0-07	3.0-08	
Xe-133	1.2-11	4.0-05	1.6-11	5.2-05	1.2-11	4.0-05	3.0-07	1.3-04	
Xe-135m	8.4-13	2.8-05	-	-	2.1-11	7.0-04	3.0-08	7.3-04	
Xe-135	6.7-13	6.7-06	6.0-12	6.0-05	1.5-11	1.5-04	1.0-07	2.2-04	12
Xe-137	2.0-12	6.7-05	-	-	9.7-12	3.2-04	3.0-08	3.9-04	
Xe-138	2.0-12	6.8-05	-	-	3.3-13	1.1-05	3.0-08	7.9-05	
Cr-51	2.6-19	3.3-12	-	-	5.0-18	6.3-11	8.0-08	6.6-11	
Mn-54	2.6-19	2.6-10	-	-	1.0-17	1.0-08	1.0-09	1.0-08	
Fe-59	1.9-19	1.0-10	-	-	1.2-17	6.0-09	2.0-09	6.1-09	
Co-58	1.9-17	9.6-09	-	-	9.6-17	4.8-08	2.0-09	5.8-08	
Co-60	9.0-19	3.0-09	-	-	8.1-17	2.7-07	3.0-10	2.7-07	
Zn-65	1.7-18	8.5-10	-	-	2.1-17	1.1-08	2.0-09	1.2-08	
Sr-89	1.3-16	4.5-07	-	-	6.9-17	2.3-07	3.0-10	6.8-07	
Sr-90	4.9-19	1.6-08	-	-	3.4-18	1.1-07	3.0-11	1.3-07	
Zr-95	1.6-19	1.6-10	-	-	4.2-17	4.2-08	1.0-09	4.2-08	
Nb-95	3.2-20	1.1-11	-	-	7.8-16	2.7-07	3.0-09	2.7-07	
Mo-99	1.6-17	2.3-09	-	-	9.0-15	1.3-06	7.0-09	1.3-06	
Ru-103	1.0-19	3.5-11	-	-	2.1-17	6.9-09	3.0-09	6.9-09	
Ag-110m	2.7-22	9.0-13	-	-	2.0-18	6.5-09	3.0-10	6.5-09	
Sb-124	2.0-19	2.9-10	-	-	7.7-19	1.1-09	7.0-10	1.4-09	
Cs-134	8.9-19	2.3-09	-	-	3.3-16	8.3-07	4.0-10	8.3-07	
Cs-136	4.1-19	6.7-11	-	-	3.5-17	5.9-09	6.0-09	6.0-09	
Cs-137	2.2-18	4.5-09	-	-	2.2-16	4.5-07	5.0-10	4.5-07	
Ba-140	1.6-16	1.6-07	-	-	4.0-15	4.0-06	1.0-09	4.2-06	

TABLE 11.3-3 (Cont)

Isotope	Main Stack Releases: Continuous Annual Average: Activity at EAB (uCi/cc)	Main Stack Releases: Continuous Annual Average: Fraction of MPC	Main Stack Releases: Intermittent (MVP): Activity at EAB (uCi/cc)	Main Stack Releases: Intermittent (MVP): Fraction of MPC	Radwaste/Reactor Building Vent Releases: Annual Average: Activity at EAB (uCi/cc)	Radwaste/Reactor Building Vent Releases: Annual Average: Fraction of MPC	Radwaste/Reactor Building Vent Releases: MPC* (uCi/cc)	Total Release (fraction of MPC)	
Ce-141	2.0-17	4.1-09	-		2.8-17	5.6-09	5.0-09	9.7-09	12
Ar-41	2.5-14	6.3-07	-		-	-	4.0-08	6.3-07	
H-3	7.6-15	3.8-08	-	-	1.3-13	6.5-07	2.0-07	6.9-07	12
C-14	2.8-15	2.8-08	-	-	-	-	1.0-07	2.8-08	
Total		3.2-04		1.5-04		1.3-03		1.8-03	12

*In accordance with 10CFR20, Appendix B

THIS FIGURE HAS BEEN DELETED

FIGURE 11.3-2

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT-UNIT 2

UPDATED SAFETY ANALYSIS REPORT

11.4 SOLID WASTE MANAGEMENT SYSTEM

Power plant operation results in various solid radioactive wastes that require disposal. The radioactive solid waste system is designed to collect, hold, monitor, process, package, and provide temporary storage facilities for radioactive materials prior to shipment offsite and ultimate disposal. The solid waste management system is shown on Figure 11.4-1.

11.4.1 Design Basis

The radioactive solid waste system is designed to the following criteria:

1. The system provides for the solidification and packaging of wet solid wastes with asphalt into shipping containers as a liquid-free homogeneous, immobile mix prior to shipment for offsite disposal.

2. All solid waste containers, shipping casks, and methods of packaging meet applicable state and federal regulations. Wastes will be shipped to a licensed burial site in accordance with applicable NRC and Department of Transportation (DOT) regulations (i.e., 10CFR71 and 49CFR171-178).

3. The filling of containers, solidification, and storage of radioactive solid waste conforms to 10CFR20 and 10CFR50 requirements and RG 8.8 guidelines in terms of ALARA doses to plant personnel and the general public.

4. Remote automatic and/or manual operation is provided by the radwaste solidification system control panel and waste handling control panel.

5. System reliability is emphasized through redundancy in design of primary components, compartmentalization of equipment layout, shielding, containment of possible spills, remote decontamination (if required), accurate process monitoring, and interlocking of process controls.

6. The seismic criteria and analytical procedures for structures housing the solid radwaste system are given in Section 3.2.1 and 3.7. The quality group classification for the system components and piping is given in Table 3.2-1.

11.4.2 System Inputs

Radioactive solid wastes that result from plant operation consist of concentrated liquid wastes from the evaporators in the radioactive LWS, spent resins from all plant demineralizers handling radioactive liquids, filter sludges from LWS filters, phase separators, and miscellaneous solid materials that become contaminated during plant operation and maintenance. Table 11.4-1 is a conservatively high estimate of the expected volumes of both the wet and dry radioactive solid waste generated by the unit.

11.4.2.1 System Inputs Activity

Expected and design wet solid waste activities are given in Table 11.4-2. The prediction of solid radwaste principal nuclide curie inventories was obtained from a mathematical model of expected and design values of reactor coolant and main steam radionuclide concentrations, as discussed in Section 11.1 and found in Table 11.4-3.

Expected and maximum dry solid radwaste inventories and annual curie content for both compactible and noncompactible wastes are provided in Tables 11.4-5 and 11.4-6. The values in these tables are based on data on radionuclides present in dry solid radwaste from operating plants[1], and are calculated for Unit 2 using the parameters in Table 11.4-7.

11.4.3 System Description

The radioactive waste solidification system (WSS) equipment is described in Topical Reports WPC-VRS-001[2] and WPC-VRS-002[3]. It is operated on a batch basis and equipment capabilities are designed to meet design throughput rates. The solidification system consists of a waste fill station, a monitor and capping station, an extruder/evaporator, control console, and piping, metering pumps, and process equipment required for transfer and solidification of wastes. Table 11.4-4 lists the major equipment of the solid waste system.

The solidification equipment requires a minimum of manual action, and in conjunction with the building layout is designed to minimize occupational radiation exposures. The solidified containers are handled by a 30-ton capacity overhead bridge crane with an 8 1/2-ton auxiliary hoist. The crane is remotely operated from the control console by the Operator using closed-circuit TV.

The asphalt handling system consists of a storage tank, redundant recirculation pumps, and redundant recirculation filters. All lines that carry the asphalt are steam heat traced and equipped with relief valves that discharge into the asphalt storage tank. Using a closed loop recirculation path from the storage tank through the pumps and strainers, and back to the tank, this system keeps the molten asphalt in constant recirculation except during periods of system maintenance.

The spent resin and filter sludge handling system consists of a waste sludge tank with agitator, redundant transfer pumps, and a decant pump. The system provides for holdup, recirculation, and sampling of waste, and decanting of excess water prior to transfer to the solidification portion of the system.

The evaporator bottoms handling system includes a concentrated waste transfer pump. In conjunction with the LWS evaporator bottoms tank, this system provides for holdup, recirculation, and sampling of evaporator concentrates prior to transfer to solidification. All lines and components in the waste concentrate system are heat traced (to prevent crystallization of the waste salts in the system) and equipped with relief valves that are discharged to the building floor drains.

The various plant systems that interface with the waste solidification system are shown on Figures 11.4-1a to 11.4-1g and are as follows:

1. LWS provides the waste feed streams.

2. Radwaste auxiliary steam (ASR) provides hot flush water.

3. Steam released from relief valves is discharged to the auxiliary boiler steam (ABM) relief header.

4. Makeup water system (MWS) supplies boiler feedwater to the WSS boiler.

5. Condensate makeup and drawoff water (CNS) is used for equipment flushing.

6. Instrument air system (IAS) supplies air to all instrument and air-operated devices.

7. Service air system (SAS) provides air for clearing lines via hose connections.

8. All equipment is provided with drains to the radwaste building floor and equipment drains system (DFW). All liquid effluent from the WSS is returned to the liquid waste system.

9. Exhaust air from the equipment is vented to the radwaste building ventilation system (HVW).

10. Turbine building closed loop cooling water (CCS) provides component cooling.

11. Radwaste sampling system (SSW) provides process sampling.

12. Radwaste seal water system (SWR) provides seal water for pumps with double mechanical seals. Details are given in Section 11.2.

A compactor is provided to compress dry wastes such as paper, rags, and plastic for packaging in metal boxes. Incompressible solid wastes are packaged in metal boxes, 55-gal drums, or encapsulated in liners ranging in size from 50 to 200 cu ft.

Design of the WSS and equipment is in accordance with Topical Report No. WPC-VRS-001, with the following differences:

1. One waste sludge tank is used to receive both spent resin and filter sludge in lieu of a tank for each feed stream (WPC-VRS-001, Drawing No. SK-VRS-2-R1).

2. The waste sludge tank is designed to ASME Section VIII Division I in compliance with RG 1.143 instead of API 620 (see FSAR Section 1.8).

3. Asphalt storage tank overflow and drain are directed into the tank cubicle instead of outdoors (WPC-VRS-001, Amendment II). The tank cubicle is designed to American Nuclear Insurers (ANI) requirements. It has a 3-hr fire rating and is capable of retaining the tank contents and potential fire suppression water.

4. The ventilation hood at the container fill station is not fitted with HEPA and charcoal filters (WPC-VRS-001, Amendment I, Response 2). Instead, ventilation air from the fill station is processed through the radwaste building HVW system, which is consistent with the gaseous radwaste management design interior discussed in FSAR Section 11.3.

5. Relief steam from the auxiliary steam system will be totally condensed instead of discharged to the atmosphere. Dampers are installed on the asphalt tank and waste sludge tank overflow lines to isolate the tank vapor space from the building atmosphere.

6. Materials for distillate collection tank 2WSS-TK-17 and system piping and valves are in accordance with either the ASME or ASTM specifications.

Items 1, 2, and 6 are changes initiated by the WSS vendor, and all changes are consistent with the requirements of RG 1.143, as discussed in FSAR Section 1.8.

11.4.3.1 Spent Resin/Filter Sludge Packaging

The waste sludge tank receives spent resin/filter sludge from the LWS spent resin tank, floor drain filter, and radwaste filter backwash tank. Excess water may be removed from the sludge tank by the decant pump through retention screens to prevent resin carryover.

When the waste is to be solidified, the agitated waste is recirculated via one of two redundant sludge transfer pumps. Recirculated waste provides suction to one of two redundant sludge metering pumps that feed waste to the evaporator/extruder. At the same time, molten asphalt is metered to the extruder from the asphalt handling system by one of two redundant asphalt metering pumps. In the extruder, water is evaporated and the solid waste is mixed with the asphalt. At the container fill station, the asphalt/waste mixture is discharged from the extruder into approved shipping containers. A ventilation hood directs air from the fill station area to the filtration system of the radwaste building HVW system. The filled containers are moved to the monitoring station where they are capped and radioactively monitored, and monitored for removable contamination by the swipe method before being placed into temporary storage. The evaporated water from the extruder is condensed, cooled, filtered, and then returned to the LWS for further processing.

11.4.3.2 Evaporator Bottoms Packaging

Evaporator bottoms are processed from the LWS evaporator bottoms tank[1]. When the waste is to be solidified, the contents are recirculated by the concentrated waste transfer pump. Recirculated waste provides suction to one of two redundant concentrated waste metering pumps which feeds waste to the extruder/evaporator. At the same time, asphalt is metered to the extruder/evaporator. In the extruder, the water is evaporated, the evaporated water from the extruder is condensed, cooled, filtered, and returned to the LWS, and the crystallized salts are mixed with the asphalt. The asphalt/waste mixture is discharged to shipping containers at the container fill station. The containers are moved to the monitoring station. The containers are capped and radiologically surveyed before being placed into temporary storage.

11.4.3.3 Radwaste Backup System

The radwaste backup system provides an alternate method for the solidification of spent resin/filter sludge and evaporator bottoms by a mobile, skid-mounted solidification system which would be temporarily located in the radwaste building truck bay. In the event that the extruder is out of service, three pipelines will bypass the extruder and go directly to a hose station in the truck bay area. These three pipelines consist of:

1. A 1 1/2-in diameter, electrically-traced line that carries evaporator bottoms waste directly from the evaporator bottoms transfer pump.

2. A 1 1/2-in diameter line that carries spent bead resins and filter sludge directly from the waste sludge tank pumps.

3. A 2-in diameter line that carries spent powder resin from the influent header feed line of the spent resin tank.

Flexible hoses will be used to transfer the spent waste from the hose station to a mobile solidification unit.

The truck bay area is classified as Radiation Zone I/IV, restricted during the radwaste loading operation.

This backup facility consists entirely of transfer piping and valves. It is part of and is designed and constructed to the same criteria as the LWS and WSS systems with which it interfaces. All transfer operations are manually controlled from a control panel located in the truck dock area.

11.4.3.3.1 Radwaste Dewatering System

The radwaste dewatering system provides an alternate method of waste volume reduction by a self-contained, freestanding, portable dewatering system located in the radwaste building truck bay, el 261'. This unit consists of a dewatering skid, a plant connection stand, a control module, a container fillhead, and a waste container and associated interconnecting hoses and cables. Plant services required to properly operate the dewatering system include service air, service water, electrical power, and waste as shown on Figure 11.4-1. The design, operation, and safety evaluation of this dewatering system is described in Chem Nuclear Systems, Inc., Topical Report RDS-25506-01-P/NP[4], which has been reviewed and accepted by Niagara Mohawk Power Corporation (NMPC) and the NRC as applicable.

11.4.3.4 Dry Waste Packaging

Contaminated dry compressible materials are collected at various locations in the plant and transported to the radwaste building for packaging. The dry solid waste, such as paper and rags, is compacted into metal boxes by the waste compactor. The boxes are placed into a temporary storage area prior to shipment to a waste burial site. The boxes are handled by a manually-operated forklift. During the waste compacting operation, the air flow in the vicinity of the compactor is directed by an exhauster through a high-efficiency air filter, and then to the radwaste building HVW system.

11.4.3.5 Incompressible Waste Packaging

Components of low activity, such as contaminated tools, can be packaged in available or specially-designed shipping containers and stored, if necessary, in the radwaste building prior to shipment.

Spent core components with very high activity levels are handled underwater within the reactor refueling cavity and fuel transfer canal, and stored in the fuel pool until adequate packaging is provided for offsite shipment. Refer to Section 9.1 for additional information on spent fuel handling.

11.4.3.6 Waste Packaging Controls

Complete solidification of the processed waste is ensured by preoperational testing and the implementation of a process control program described in WPC-VRS-001[2] and WPC-VRS-002[3]. Waste sludge and evaporator bottoms tanks are provided with means for obtaining representative samples via a shielded sampling station located in the radwaste building. Waste packaging controls and monitoring are provided from the solidification system control panel. The system is designed to prevent external contamination of the containers by instrumentation interlocks that prevent overfilling.

Electrical interlocks are provided with all radioactive waste feed systems to ensure that asphalt is available and flowing to provide the solidification matrix whenever waste is fed to the extruder/evaporator.

11.4.3.7 Waste Handling

Waste handling is provided by a 30-ton, overhead, traveling bridge crane with an 8 1/2-ton auxiliary hoist. Remote handling of processed waste is done by closed-circuit TV monitoring from the waste handling control panel.

11.4.4 Packaging

Filling of containers and storage of radioactive solid wastes conforms with 10CFR20 and 10CFR50 requirements. Packages meet shipping regulations of 49CFR171-178 and 10CFR71, as applicable. The waste packaging procedure is described in Section 11.4.3. Containers used are 50- to 200-cu ft liners, 55-gal drums, and metal boxes with supplementary lead or steel shielding as required for shipment.

11.4.5 Storage Facilities

A limited access storage area is provided for storage of dry, low-level waste packaged by the compactor. The storage capacity of compacted waste in the radwaste building is approximately a 3-month output of packaged trash at expected generation rates. Solidified waste is stored in a shielded area within the radwaste building in liners ranging in size from 50 to 200 cu ft. The storage capacity of processed waste is approximately a 4-month output of packaged waste at expected generation rates.

11.4.6 Shipment

Shipment of radioactive solid wastes conforms with 10CFR50, 10CFR61, 10CFR71, and 49CFR171 through 49CFR178 requirements.

Higher-activity wastes are shipped in shielded casks, as applicable.

Solid waste is transported by a licensed disposal contractor or a common carrier to a licensed burial site.

Tables 11.4-1 and 11.4-2 summarize the annual number of packaged containers, expected number of shipments to be made, and expected and design activities of the wastes.

11.4.7 References

1. Phillips, J., Feizollahi, F., Martineit, R., and Bell, W. Waste Report for Reactor and Fuel-Fabrication Facility Wastes, ONWI-20/NUS-3314, NUS Corporation, March 1979.

2. Radwaste Volume Reduction and Solidification System-Topical Report, Report Number WPC-VRS-001, Revision 1, Werner and Pfleiderer Corporation, May 1978.

3. Topical Report, 10CFR61, Waste Form Conformance Program for Solidified Process Waste Products by a Wastechem Corporation Volume Reduction and Solidification (VRS) System, Report No. VRS-002, Revision 1, August 1987.

4. Topical Report, RDS-1000 Radioactive Waste Dewatering System, Report No. RDS-25506-01-P/NP, Revision 1, Chem Nuclear Systems, Inc.

TABLE 11.4-2

EXPECTED AND DESIGN WET SOLID WASTE ACTIVITIES

Source of Wet Wastes	Expected uCi/cc	Expected Ci/ft³	Total Activity (Ci/yr)	Design uCi/cc	Design Ci/ft³	Total Activity (ci/yr)
Spent resin (radwaste demineralizer, condensate demineralizer)	58.62	1.66	2.01×10^3	497.96	14.10	6.20×10^4
Filter sludges						
1. Radwaste filter	1.58	0.05	43.42	2.60	0.07	133.47
2. Floor drain filter	5.46	0.16	598.95	9.18	0.26	1,884.60
Evaporator bottoms (radwaste and regenerative)	35.50	1.00	8.18×10^3	293.02	8.30	1.22×10^5

TABLE 11.4-4

SOLID WASTE MANAGEMENT SYSTEM MAJOR EQUIPMENT LIST

Component	Parameter
Waste sludge tank 2WSS-TK8	
Number	1
Capacity, gal	1,355
Material of construction	Type 316L stainless steel
Asphalt storage tank 2WSS-TK2	
Number	1
Capacity, gal	10,800
Material of construction	Carbon steel
Extruder/evaporator 2WSS-EV25	
Number	1
Capacity, gpm	Varies ($\underset{\sim}{}$1.0)
Material of construction	Mfg standard
Asphalt metering pump 2WSS-P5A&B	
Number	2
Capacity, gpm	0.3
Material of construction	Cast iron
Asphalt recirc pump 2WSS-P3A&B	
Number	2
Capacity, gpm	20
Material of construction	Cast iron
Waste sludge transfer pump 2WSS-P50A&B	
Number	2
Capacity, gpm	50
Material of construction	High chrome iron
Waste sludge metering pump 2WSS-P12A&B	
Number	2
Capacity, gpm	0.2-0.8
Material of construction	Type 316 stainless steel
Decant pump 2WSS-P10	
Number	1
Capacity, gpm	40
Material of construction	Type 316 stainless steel
Waste concentrate transfer pump 2WSS-P6	
Number	1
Capacity, gpm	50
Material of construction	Alloy 20

TABLE 11.4-4 (Cont'd.)

Component	Parameter
Waste concentrate metering pump 2WSS-P7A&B	
Number	2
Capacity, gpm	0.4-0.8
Material of construction	316TI stainless steel
Steam dome boilout tank 2WSS-TK46	
Number	1
Capacity, gal	3
Material of construction	Type 304 stainless steel
Overhead crane 2MHN-CRN1	
Number	1
Capacity - main hoist	30 tons
auxiliary hoist	8-1/2 tons

TABLE 11.4-5

EXPECTED DRY SOLID RADWASTE ANNUAL NUCLIDE INVENTORIES

Isotope	Compactible (Ci)	Noncompactible (Ci)
Cr-51	5.3-01*	4.0+01
Mn-54	4.3-01	3.3+01
Fe-59	5.1-02	3.9+00
Co-58	1.4-01	1.1+01
Co-60	1.9+00	1.4+02
Zn-65	7.5-01	5.7+01
Zr-95	1.3-02	9.6-01
Nb-95	4.5-02	3.4+00
Cs-134	6.5-01	5.0+01
Cs-137	1.2+00	8.9+01
Total	5.6+00	4.3+02

* 5.3-01 is 5.3×10^{-1}

TABLE 11.4-6

MAXIMUM DRY SOLID RADWASTE ANNUAL NUCLIDE INVENTORIES

Isotope	Compactible (Ci)	Noncompactible (Ci)
Cr-51	8.9-01*	6.7+01
Mn-54	7.2-01	5.5+01
Fe-59	8.6-02	6.6+00
Co-58	2.4-01	1.8+01
Co-60	3.1+00	2.4+02
Zn-65	1.3+00	9.5+01
Zr-95	2.1-02	1.6+00
Nb-95	7.5-02	5.7+00
Cs-134	1.1+00	8.3+01
Cs-137	1.9+00	1.5+02
Total	9.4+00	7.2+02

* 8.9-01 is 8.9×10^{-1}

TABLE 11.4-7

PARAMETERS USED TO CALCULATE DRY SOLID RADWASTE INVENTORIES AND ANNUAL CURIE CONTENT

Average plant waste activity for BWRs (for trash)[1]	Compactible Noncompactible	5.20-03 Ci/MW(e)-yr 3.97-01 Ci/MW(e)-yr
Net (expected) electrical output for Unit 2		1,080 MW(e)
Volume of dry solid radwaste (design) for Unit 2		17,054 ft^3
Volume of dry solid radwaste (expected) for Unit 2		10,234 ft^3

(1) Phillips, J., Feizollahi, F., Martineit, R., and Bell, W., Waste Report for Reactor and Fuel-Fabrication Facility Wastes, ONWI-20/NUS-3314, NUS Corporation, March 1979 (Table 4.2-49 and p. 4-88).

guidelines established in ANSI 13.10. Calibration standards and frequency are described in Section 11.5.2.3.2.

The quality assurance (QA) criteria is consistent with the function of the monitor. Safety-related monitors are procured and designed to 10CFR50 Appendix B criteria. Nonsafety-related monitors are designed and procured under standards that meet the criteria established in RG 1.143.

11.5.2.3.1 Inspection and Tests

Inspection and testing of the monitors specified in Section 11.5.2.3 is described in the Technical Specifications and Station procedures.

11.5.2.3.2 Calibration

The radiation monitor's calibration is traceable to the National Bureau of Standards and is accurate to at least ±15 percent. The source-detector geometry during initial calibration is identical to the sample-detector geometry in actual use. Secondary standards that were counted in a reproducible geometry during the initial calibration are used for calibration after installation. Where applicable, each monitor is calibrated in accordance with the frequencies provided in the plant Technical Specifications during plant operation or during the refueling outage if the detector is not readily accessible.

11.5.2.3.3 Maintenance

The channel detector, electronics, and recorder are serviced and maintained in accordance with manufacturers' recommendations to ensure reliable operations. Such maintenance includes cleaning, lubrication, and assurance of free movement of the recorder in addition to the replacement or adjustment of any components required after performing a test or calibration check. If any work is performed that would affect the calibration, a recalibration is performed at the completion of the work.

11.5.2.4 Sampling

Section 9.3.2 discusses various process and effluent samples periodically taken for chemical and radiochemical analysis.

Liquid process and effluent samples are periodically taken and monitored for radioactivity. Those provisions for sampling not covered in Section 9.3.2 are described in the individual system design sections. Sampling of these fluid systems is via local sampling connections. The Technical Specifications describe various liquid samples and the analysis required, including the sampling frequencies.

Additionally, the process and effluent radiological monitoring system can provide grab samples that are used to locate manually a specific source of high radioactivity when a continuous radiological monitor signals a high radioactivity alarm in the main control room. Tritium in the plant areas is determined on the basis of representative grab samples collected from the effluent points or ventilation exhaust ducts. Grab samples are obtained from locations indicated in Table 11.5-2. Samples are analyzed in the Health Physics laboratory, or by contracted laboratories.

11.5.3 Effluent Monitoring and Sampling

All potentially-radioactive gaseous and liquid effluent discharge paths are either continuously monitored or routinely sampled for radiation level during discharge (Section 11.5.2). Solid waste shipping containers are monitored with gamma-sensitive portable survey instruments. The following gaseous effluent paths are sampled and monitored:

1. Plant main stack exhaust.

2. Combined radwaste/reactor building ventilation exhaust.

The following liquid effluent paths are sampled and monitored:

1. LWS effluent.

2. CWS cooling tower blowdown line.

3. SWP discharge.

All monitor ranges are listed in Table 11.5-1.

An isotopic analysis is performed periodically on samples obtained from each liquid effluent release path to verify the adequacy of effluent processing to meet the discharge limits to unrestricted areas.

This effluent monitoring and sampling program is comprehensive and provides the information for the effluent measuring and reporting programs required by 10CFR50 Section 36a, Appendix A, GDC 64, and Appendix I and RG 1.21 in semiannual reports to the NRC. The frequency of the periodic sampling and analysis described in the Technical Specifications is a minimum and is increased if effluent levels approach Technical Specification limits. Isotopic content of gaseous effluents is continuously monitored by off-line monitors. All potentially-significant radioactive discharge paths are equipped with a control system to isolate the discharge automatically on indication of a high radiation level. These include:

1. Offgas pretreatment.

2. SGTS discharge (isolates containment purge system).

3. Reactor building ventilation exhaust.

4. Liquid radwaste effluent.

The effluent isolation functions for each monitor are given in Table 11.5-1 and in Section 11.5.2.

Radiation levels in radioactive and potentially-radioactive process streams are monitored by the process and effluent monitors given in Table 11.5-1.

Airborne radioactivity in the fuel-handling area and the radwaste building is detected by CAM and area radiation monitors. Airborne radioactivity in the drywell is detected by the drywell atmosphere monitors and the SGTS discharge monitor which isolates the containment purge on high radioactivity. These monitors are also described in Section 12.3.4 since they are used to monitor in-plant airborne radioactivity to protect plant personnel. The area RMS is also described in Section 12.3.4. A system level/qualitative-type failure modes and effects analysis (FMEA) of the MSL radiation monitoring is provided in Appendix 15A. The FMEA for other safety-related radiation monitors is provided in the FMEA Report.

TABLE 11.5-2

GRAB SAMPLES FOR RADIOLOGICAL ANALYSIS

Sample Point Location (No.)	Grab Sample at Sample Station	Local Grab Sample	Grab Sample at Radiation Monitor
Reactor Steam Supply System			
Reactor recirculation system pump discharge	X	-	-
Main steam line	X	-	-
Reactor Water Cleanup System			
Common filter/demineralizer influent	X	-	-
Individual filter/demineralizer effluents (4)	X	-	-
Fuel Pool Cooling and Cleanup System			
Pump discharge	-	-	X
Common filter influent	X	-	-
Individual filter effluents (2)	X	-	-
Individual heat exchanger effluents (2)	X	-	-
Reactor Building Closed Loop Cooling Water			
Cooling water sample (outlet of RWCU and SFC heat exchangers)	-	-	X
Turbine Building Closed Loop Cooling Water			
Cooling water sample (common outlet of radwaste system exchangers)	-	-	X
Residual Heat Removal System			
Individual heat exchanger outlet (service water) (2)	-	-	X
Individual heat exchanger outlet (RHR) (2)	X	-	-
Control Rod Drive System			
Common CRD filter effluent	X	-	-
High-Pressure Core Spray System			
Test return line to condensate storage tank	X	-	-

TABLE 11.5-2 (Cont)

Sample Point Location (No.)	Grab Sample at Sample Station	Local Grab Sample	Grab Sample at Radiation Monitor
Radwaste System			
Individual waste collector tank pump effluents (3)	X	-	-
Individual demineralizer effluents (2)	X	-	-
Filtrate pump effluent	X	-	-
Individual filter effluents (2)	X	-	-
Demineralizer (acid influent)	-	X	-
Demineralizer (caustic influent)	-	X	-
Individual recovery sample pump effluents (2)	X	-	-
Individual floor drain collector pump effluents (2)	X	-	-
Floor drain filter effluent pump discharge	X	-	-
Liquid radwaste final discharge	-	-	X
Regenerant waste pump effluents (2)	X	-	-
Regenerant recirculation pump suction and discharge (2)	X	-	-
Phase separator tank pump discharge	-	X	-
Waste evaporator recirculation pump suction and discharge (2)	X	-	-
Waste evaporator distillate	X	X	-
Individual waste sample pumps, discharge (2)	X	-	-
Regenerant evaporator distillate	X	X	-
Common discharge floor drain collector surge pumps	X	-	-
Common discharge waste collector surge pumps	X	-	-
Individual radwaste auxiliary steam cooler effluents (2)	X	-	-
Water Treating System			
Dilute acid effluent	-	X	-
Dilute caustic effluent	-	X	-
Waste water effluent	-	X	-
Condensate Demineralizer System			
Common demineralizer influent	X	-	-
Common demineralizer effluent	X	-	-
Resin hold tank effluent	X	-	-
Individual demineralizers effluent (9)	X	-	-
Ultrasonic resin cleaner effluent	-	X	-
Resin mix tank effluent	X	-	-
Cation regeneration tank effluent	X	-	-

TABLE 11.5-2 (Cont)

Sample Point Location (No.)	Grab Sample at Sample Station	Local Grab Sample	Grab Sample at Radiation Monitor
Anion regeneration tank effluent	X	-	-
Recovered acid tank effluent	-	X	-
Regeneration system effluent	X	-	-
Ultrasonic resin cleaner resin receiver tank effluent	X	-	-
Ultrasonic resin cleaner resin effluent	X	-	-
Low conductivity waste tank effluent	X	-	-
Demineralizer waste neutralizing tank effluent	X	-	-
Dilute acid effluent	X	X	-
Recovered caustic tank effluent	-	X	-
Dilute caustic effluent	X	X	-
Recovered water sump effluent	-	X	-
Condensate Makeup and Drawoff System			
Condensate transfer line	X	-	-
Makeup Water System			
Demineralizer water transfer line	X	-	-
Condensate System			
Condensate pump discharge	X	-	-
Condenser hotwells (6)	X	-	-
LP heater drains (3)	X	-	-
Common effluent fourth point heaters	X	-	-
LP heater string common effluent	X	-	-
Reactor Feedwater System			
Feedwater (after last heater)	X	-	-
Circulating Water System			
Effluent (blowdown line)	-	X	X
Auxiliary Steam System			
Auxiliary boiler (steam outlet)	X	-	-
Feedwater (pump discharge)	X	-	-
Auxiliary boiler (blowdown)	-	X	-
Auxiliary boiler recirc pump seal heat exchanger outlet and sample cooler discharge (Service Water) (4)	-	X	-

TABLE 11.5-2 (Cont)

Sample Point Location (No.)	Grab Sample at Sample Station	Local Grab Sample	Grab Sample at Radiation Monitor
Sealing Steam System			
Individual clean steam reboiler outlets	X	-	-
Reactor Building Ventilation System			
Reactor/radwaste building ventilation exhaust	-	-	X
Main plant stack exhaust	-	-	X
Containment atmosphere	-	-	X
Containment purge	-	-	X
Standby Gas Treatment System			
SGTS effluent	-	-	X
Control Building Ventilation System			
Main control room intakes	-	X	X
Radwaste Building Ventilation System			
Ventilation exhaust	-	X	X
Radwaste tank vents	-	-	X
Turbine Building Ventilation System			
Ventilation	-	X	-
Mechanical vacuum pump discharge	-	X	-
Off-gas pretreatment (2)	-	-	X
Turbine gland seal discharge	-	X	-
Service Water System			
Final discharge	-	X	X
Storm, Underdrain Water, and Site Sewage Systems			
Final discharge	-	X	-

NOTE: See Section 9.3.2 for details regarding the reactor, turbine, and radwaste sample systems.

APPENDIX 11A

RADIOLOGICAL DOSES

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APPENDIX 11A

RADIOLOGICAL DOSES

11A.1 SUMMARY OF ANNUAL RADIATION DOSES

The calculated annual radiation doses to maximum individuals from normal operation of Unit 2 are presented in Tables 11A.1-1 through 11A.1-16. Table 11A.1-17 demonstrates that the calculated annual radiation doses are below the design objectives of 10CFR50, Appendix I. In providing guidance for the implementation of Appendix I, the NRC has made use of the maximum exposed individual approach. Maximum individuals are characterized as maximum with regard to food consumption, occupancy, and other usage of the region in the vicinity of the plant site and, as such, represent individuals with reasonable deviations from the average individual considered representative of the population in general.

For gaseous radioactive releases the analyzed pathways included: standing on contaminated ground, ingestion of vegetation, and inhalation of and submersion in gaseous effluents. These pathways were considered for all of the resident locations. Resident locations having cows, goats, and meat animals were also analyzed for ingestion of cow milk, goat milk, and beef meat, respectively. Additionally, the doses associated with ingestion of deer were conservatively added to all resident locations analyzed.

The calculated organ dose due to radioiodines and particulates is 1.7 mRem/yr. This represents the dose to the thyroid of an infant living at the residence location 2,350 m east-southeast of the site. The majority of this dose was due to consumption of cow milk. The highest calculated external exposure rates to the whole body and skin from immersion in noble gases at an occupied location were 0.03 and 0.06 mRem/yr, respectively. These occurred at the residence location 1,693 m east of the site.

The highest calculated beta and gamma air doses at an unoccupied location from noble gas releases were 0.04 and 0.06 mrad/yr, respectively. These occurred at the exclusion area boundary (EAB) 1,603 m east of the site.

For liquid releases, the maximum individual consumed fish whose principal habitat was assumed to be the edge of the initial mixing zone. This location was also conservatively used in calculating doses from swimming and boating.

The calculated annual doses to the population residing within an 80-km radius of the site are presented in Table 11A.1-18. Population doses were calculated for a projected population of 1.2 million residing within 80 km of the site in the year 2010.

Gaseous pathways considered in the population dose analysis included: inhalation, exposure to ground deposits, ingestion of vegetation, cow milk, and beef meat, and submersion in noble gases. The calculated annual dose to the population was 0.6 man-Rem/yr whole body and 3.3 man-Rem/yr thyroid.

Liquid pathways considered in the population dose analysis included: ingestion of potable water and fish, shoreline recreation, and swimming and boating. The calculated annual dose to the population was 1.4 man-Rem/yr whole body and 0.06 man-Rem/yr thyroid.

In addition to the 80-km (50-mi) radius population doses, population doses associated with the export of food crops produced within the 80-km (50-mi) region and the atmospheric and hydrospheric transport of the more mobile effluent species, such as noble gases, tritium, and carbon-14, were calculated.

These calculated annual gaseous and liquid doses to the contiguous U.S. population are presented in Table 11A.1-19. For liquid effluents, the calculated doses to the contiguous U.S. population were 1.4 man-Rem/yr whole body and 0.06 man-Rem/yr thryoid. For gaseous effluents, the calculated doses to the contiguous U.S. population were 21.1 man-Rem/yr whole body and 24.9 man-Rem/yr thyroid.

TABLE 11A.1-1

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE ADULT GROUP FROM LIQUID EFFLUENTS

(Annual Dose in mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI Tract
Potable water	3.5E-04	0.0	1.7E-04	4.1E-04	2.9E-04	2.6E-04	2.1E-04	2.0E-04
Fish consumption	5.8E-01	0.0	7.0E-01	8.2E-01	3.6E-03	2.7E-01	8.9E-02	4.3E-02
Shoreline recreation	2.3E-05	2.6E-05	2.3E-05	2.3E-05	2.3E-05	2.3E-05	2.3E-05	2.3E-05
Fresh vegetation	5.9E-05	0.0	4.4E-05	7.5E-05	2.6E-05	3.6E-05	2.2E-05	2.0E-05
Stored vegetation	4.1E-04	0.0	3.0E-04	5.2E-04	1.1E-04	2.5E-04	1.6E-04	1.4E-04
Duck consumption	1.3E-04	0.0	2.1E-03	2.2E-04	1.4E-06	4.7E-05	6.4E-06	2.9E-04
Swimming exposure	3.3E-05	4.2E-05	3.3E-05	3.3E-05	3.3E-05	3.3E-05	3.3E-05	3.3E-05
Boating exposure	3.3E-05	4.2E-05	3.3E-05	3.3E-05	3.3E-05	3.3E-05	3.3E-05	3.3E-05
Total dose	5.8E-01	1.1E-04	7.0E-01	8.2E-01	4.1E-03	2.7E-01	8.9E-02	4.4E-02

NOTE: $3.5E-04 = 3.5 \times 10^{-4}$

TABLE 11A.1-2

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE
TEEN GROUP FROM LIQUID EFFLUENTS

(Annual Dose in mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI Tract
Potable water	2.2E-04	0.0	1.6E-04	3.5E-04	2.2E-04	3.2E-04	1.6E-04	1.4E-04
Fish consumption	3.3E-01	0.0	7.6E-01	8.4E-01	3.3E-03	2.8E-01	1.1E-01	3.3E-02
Shoreline recreation	1.3E-04	1.5E-04	1.3E-04	1.3E-04	1.3E-04	1.3E-04	1.3E-04	1.3E-04
Fresh vegetation	3.2E-05	0.0	3.9E-05	6.3E-05	1.8E-05	5.1E-05	1.7E-05	1.3E-05
Stored vegetation	4.4E-04	0.0	5.3E-04	8.7E-04	1.5E-04	5.6E-04	2.4E-04	1.8E-04
Duck consumption	1.0E-04	0.0	1.8E-03	1.8E-04	9.9E-07	3.0E-04	6.0E-06	1.8E-04
Swimming exposure	3.3E-05	4.2E-05	3.3E-05	3.3E-05	3.3E-05	3.3E-05	3.3E-05	3.3E-05
Boating exposure	3.3E-05	4.2E-05	3.3E-05	3.3E-05	3.3E-05	3.3E-05	3.3E-05	3.3E-05
Total dose	3.3E-01	2.3E-04	7.6E-01	8.4E-01	3.9E-03	2.8E-01	1.1E-01	3.4E-02

NOTE: 2.2-04 = 2.2×10^{-4}

TABLE 11A.1-3

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE CHILD GROUP FROM LIQUID EFFLUENTS

(Annual Dose in mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI Tract
Potable water	3.3E-04	0.0	4.6E-04	6.9E-04	4.7E-04	3.9E-04	3.0E-04	2.5E-04
Fish consumption	1.4E-01	0.0	9.5E-01	7.5E-01	3.3E-03	2.4E-01	8.5E-02	1.4E-02
Shoreline recreation	2.6E-05	3.1E-05	2.6E-05	2.6E-05	2.6E-05	2.6E-05	2.6E-05	2.6E-05
Fresh vegetation	2.5E-05	0.0	6.8E-05	7.9E-05	2.4E-05	3.4E-05	2.0E-05	1.4E-05
Stored vegetation	4.7E-04	0.0	1.3E-03	1.5E-03	2.4E-04	6.3E-04	3.8E-04	2.6E-04
Duck consumption	1.6E-04	0.0	3.4E-03	2.5E-04	1.5E-06	4.2E-05	7.1E-06	1.1E-04
Swimming exposure	1.8E-05	2.4E-05	1.8E-05	1.8E-05	1.8E-05	1.8E-05	1.8E-05	1.8E-05
Boating exposure	1.9E-05	2.4E-05	1.9E-05	1.9E-05	1.9E-05	1.9E-05	1.9E-05	1.9E-05
Total dose	1.4E-01	7.9E-05	9.6E-01	7.5E-01	4.1E-03	2.4E-01	8.6E-02	1.5E-02

NOTE: 3.3E-04 = 3.3×10^{-4}

TABLE 11A.1-4

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE INFANT GROUP FROM LIQUID EFFLUENTS

(Annual Dose in mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Potable water	2.9E-04	0.0	4.7E-04	7.9E-04	6.0E-04	3.9E-04	3.0E-04	2.5E-04
Total dose	2.9E-04	0.0	4.7E-04	7.9E-04	6.0E-04	3.9E-04	3.0E-04	2.5E-04

NOTE: 2.9E-04 = 2.9×10^{-4}

TABLE 11A.1-5

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE ADULT GROUP FROM GASEOUS EFFLUENTS*

At Maximum Residence Location

(Annual Dose in mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground	7.9-03	9.2-03	7.9-03	7.9-03	7.9-03	7.9-03	7.9-03	7.9-03
Inhalation	1.6-04	0.0	1.6-04	2.4-04	1.3-02	3.0-04	2.8-04	2.1-04
Fresh vegetation	7.5-04	0.0	1.7-03	1.1-03	1.3-01	1.2-03	2.1-04	8.0-04
Stored vegetation	2.7-03	0.0	6.3-03	3.2-03	4.5-03	1.8-03	1.1-03	1.8-03
Deer 1,603 m east	1.4-04	0.0	1.7-04	1.9-04	3.2-04	9.2-05	3.1-05	3.3-04
Total dose	1.2-02	9.2-03	1.6-02	1.3-02	1.6-01	1.1-02	9.5-03	1.1-02

*Analysis performed at maximum residence location is 4,106 m (13,471 ft) east.

NOTE: 7.9-03 = 7.9×10^{-3}

TABLE 11A.1-6

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE
TEEN GROUP FROM GASEOUS EFFLUENTS*

At Maximum Residence Location

(Annual Dose in mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground	7.9-03	9.2-03	7.9-03	7.9-03	7.9-03	7.9-03	7.9-03	7.9-03
Inhalation	1.8-04	0.0	2.2-04	2.9-04	1.7-02	3.8-04	3.8-04	2.3-04
Fresh vegetation	5.4-04	0.0	1.6-03	1.0-03	1.1-01	1.1-02	1.9-04	5.8-04
Stored vegetation	3.5-03	0.0	1.1-02	5.7-03	7.3-03	6.0-02	2.0-03	2.7-03
Deer 1,603 m east	7.8-05	0.0	1.4-04	1.5-04	2.4-04	7.6-03	2.7-05	1.9-04
Total dose	1.2-02	9.2-03	2.1-02	1.5-02	1.4-01	8.7-02	1.0-02	1.2-02

*Analysis performed at maximum residence location is 4,106 m (13, 471 ft) east.

NOTE: 7.9-03 = 7.9×10^{-3}

TABLE 11A.1-7

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE
CHILD GROUP FROM GASEOUS EFFLUENTS*

At Maximum Residence Location

(Annual Dose in mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground	7.9-03	9.2-03	7.9-03	7.9-03	7.9-03	7.9-03	7.9-03	7.9-03
Inhalation	1.8-04	0.0	3.0-04	2.8-04	2.1-02	3.6-04	3.3-04	1.9-04
Fresh vegetation	7.1-04	0.0	2.9-03	1.4-03	1.6-01	1.4-03	3.2-04	5.3-04
Stored vegetation	5.7-03	0.0	2.8-02	1.1-02	1.5-02	6.2-03	4.5-03	4.6-03
Deer 1,603 m east	8.6-05	0.0	2.5-04	2.0-04	3.6-04	9.5-05	4.0-05	1.2-04
Total dose	1.5-02	9.2-03	3.9-02	2.1-02	2.0-01	1.6-02	1.3-02	1.3-02

*Analysis performed at maximum residence location is 4,106 m (13,471 ft) east.

NOTE: 7.9-03 = 7.9×10^{-3}

TABLE 11A.1-8

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE
INFANT GROUP FROM GASEOUS EFFLUENTS*

At Maximum Residence Location

(Annual Dose in mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground	7.9-03	9.2-03	7.9-03	7.9-03	7.9-03	7.9-03	7.9-03	7.9-03
Inhalation	1.2-04	0.0	2.2-04	2.2-04	1.9-02	2.3-04	2.4-04	1.1-04
Total dose	8.0-03	9.2-03	8.1-03	8.1-03	2.7-02	8.1-03	8.1-03	8.0-03

*Analysis performed at maximum residence location is 4,106 m (13,471 ft) east.

NOTE: 7.9-03 = 7.9×10^{-3}

TABLE 11A.1-9

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE ADULT GROUP FROM GASEOUS EFFLUENTS*

At Maximum Cow Location

(Annual Dose in mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground	1.3-02	1.6-02	1.3-02	1.3-02	1.3-02	1.3-02	1.3-02	1.3-02
Inhalation	1.4-04	0.0	1.1-04	2.0-04	9.7-03	2.5-04	2.7-04	1.9-04
Fresh vegetation	7.0-04	0.0	1.3-03	1.1-03	1.3-01	1.1-03	1.3-04	7.3-04
Stored vegetation	2.4-03	0.0	4.3-03	3.0-03	4.1-03	1.4-03	6.9-04	1.4-03
Cow milk	2.4-03	0.0	3.0-03	3.7-03	2.2-01	2.7-03	4.5-04	1.1-03
Deer 1,603 m east	1.4-04	0.0	1.7-04	1.9-04	3.2-04	9.2-05	3.1-05	3.3-04
Total dose	1.9-02	1.6-02	2.2-02	2.1-02	3.8-01	1.9-02	1.5-02	1.7-02

*Analysis performed at maximum cow location is 2,350 m (7,710 ft) east-southeast.

NOTE: 1.3-02 = 1.3×10^{-2}

TABLE 11A.1-10

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE
TEEN GROUP FROM GASEOUS EFFLUENTS*

At Maximum Cow Location

(Annual Dose in mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground	1.3-02	1.6-02	1.3-02	1.3-02	1.3-02	1.3-02	1.3-02	1.3-02
Inhalation	1.5-04	0.0	1.5-04	2.4-04	1.3-02	3.1-04	3.6-04	2.1-04
Fresh vegetation	4.8-04	0.0	1.2-03	9.5-04	1.1-01	1.2-02	1.2-04	5.1-04
Stored vegetation	2.8-03	0.0	7.7-03	5.2-03	6.5-03	6.5-02	1.3-03	1.9-03
Cow milk	3.0-03	0.0	5.3-03	6.4-03	3.5-01	4.7-03	8.5-04	1.5-03
Deer 1,603 m east	7.8-05	0.0	1.4-04	1.5-04	2.4-04	7.6-03	2.7-05	1.9-04
Total dose	2.0-02	1.6-02	2.7-02	2.6-02	4.9-02	1.0-01	1.6-02	1.7-02

*Analysis performed at maximum cow location is 2,350 m (7,710 ft) east-southeast.

NCTE: 1.3-02 = 1.3×10^{-2}

TABLE 11A.1-11

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE CHILD GROUP FROM GASEOUS EFFLUENTS*

At Maximum Cow Location

(Annual Dose in mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground	1.3-02	1.6-02	1.3-02	1.3-02	1.3-02	1.3-02	1.3-02	1.3-02
Inhalation	1.5-04	0.0	2.1-04	2.3-04	1.6-02	2.9-04	3.1-04	1.7-04
Fresh vegetation	5.8-04	0.0	2.2-03	1.3-03	1.6-01	1.3-03	1.8-04	3.9-04
Stored vegetation	3.8-03	0.0	1.9-02	9.3-03	1.3-02	4.4-03	2.6-03	2.6-03
Cow milk	4.3-03	0.0	1.3-02	1.1-02	7.1-01	7.9-03	1.6-03	1.8-03
Deer 1,603 m east	8.6-05	0.0	2.5-04	2.0-04	3.6-04	9.5-05	4.0-05	1.2-04
Total dose	2.2-02	1.6-02	4.8-02	3.5-02	9.1-01	2.7-02	1.8-02	1.8-02

*Analysis performed at maximum cow location is 2,350 m (7,710 ft) east-southeast.

NOTE: 1.3-02 = 1.3×10^{-2}

TABLE 11A.1-12

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE INFANT GROUP FROM GASEOUS EFFLUENTS*

At Maximum Cow Location

(Annual Dose in mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground	1.3-02	1.6-02	1.3-02	1.3-02	1.3-02	1.3-02	1.3-02	1.3-02
Inhalation	9.6-05	0.0	1.5-04	1.8-04	1.5-02	1.8-04	2.3-04	9.5-05
Cow milk	6.6-03	0.0	2.3-02	2.2-02	1.7+00	1.3-02	3.2-03	4.7-03
Total dose	2.0-02	1.6-02	3.6-02	3.5-02	1.7+00	2.6-02	1.6-02	1.8-02

*Analysis performed at maximum cow location is 2,350 m (7,710 ft) east-southeast.

NOTE: 1.3-02 = 1.3×10^{-2}

TABLE 11A.1-13

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE
ADULT GROUP FROM GASEOUS EFFLUENTS*

At Maximum Beef Animal Location

(Annual Dose in mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground	2.4-02	2.8-02	2.4-02	2.4-02	2.4-02	2.4-02	2.4-02	2.4-02
Inhalation	2.3-04	0.0	1.3-04	3.1-04	1.2-02	3.6-04	4.9-04	3.2-04
Fresh vegetation	1.6-03	0.0	2.7-03	2.4-03	2.6-01	2.3-03	2.4-04	1.7-03
Stored vegetation	5.7-03	0.0	8.4-03	7.4-03	7.9-03	3.1-03	1.3-03	3.0-03
Beef	9.7-04	0.0	1.7-03	1.4-03	1.5-02	8.1-04	3.2-04	4.4-03
Deer 1,603 m east	1.4-04	0.0	1.7-04	1.9-04	3.2-04	9.2-05	3.1-05	3.3-04 | 25
Total dose	3.2-02	2.8-02	3.7-02	3.6-02	3.2-01	3.1-02	2.6-02	3.3-02

*Analysis performed at maximum beef animal location is 1,693 m (5,555 ft) east.

NOTE: 2.4-02 = 2.4×10^{-2}

TABLE 11A.1-14

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE TEEN GROUP FROM GASEOUS EFFLUENTS*

At Maximum Beef Animal Location

(Annual Dose in mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground	2.4-02	2.8-02	2.4-02	2.4-02	2.4-02	2.4-02	2.4-02	2.4-02
Inhalation	2.4-04	0.0	1.8-04	3.6-04	1.6-02	4.3-04	6.6-04	3.4-04
Fresh vegetation	1.1-03	0.0	2.5-03	2.2-03	2.1-01	3.2-02	2.3-04	1.2-03
Stored vegetation	6.3-03	0.0	1.5-02	1.3-02	1.3-02	1.7-01	2.4-03	4.0-03
Beef	6.1-04	0.0	1.4-03	1.1-03	1.1-02	9.2-02	2.6-04	2.5-03
Deer 1,603 m east	7.8-05	0.0	1.4-04	1.5-04	2.4-04	7.6-03	2.7-05	1.9-04
Total dose	3.2-02	2.8-02	4.3-02	4.1-02	2.7-01	3.2-01	2.8-02	3.2-02

25

*Analysis performed at maximum beef animal location is 1,693 m (5,555 ft) east.

NOTE: 2.4-02 = 2.4×10^{-2}

TABLE 11A.1-15

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE CHILD GROUP FROM GASEOUS EFFLUENTS*

At Maximum Beef Animal Location

(Annual Dose in mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground	2.4-02	2.8-02	2.4-02	2.4-02	2.4-02	2.4-02	2.4-02	2.4-02
Inhalation	2.3-04	0.0	2.4-04	3.4-04	2.0-02	4.0-04	5.6-04	2.6-04
Fresh vegetation	1.2-03	0.0	4.5-03	2.8-03	3.3-01	2.6-03	3.1-04	7.8-04
Stored vegetation	7.6-03	0.0	3.6-02	2.2-02	2.6-02	9.2-03	4.4-03	4.4-03
Beef	7.9-04	0.0	2.5-03	1.4-03	1.7-02	8.8-04	4.4-04	1.6-03
Deer 1,603 m east	8.6-05	0.0	2.5-04	2.0-04	3.6-04	9.5-05	4.0-05	1.2-04
Total dose	3.4-02	2.8-02	6.7-02	5.1-02	4.2-01	3.7-02	3.0-02	3.1-02

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*Analysis performed at maximum beef animal location is 1,693 m (5,555 ft) east.

NOTE: 2.4-02 = 2.4×10^{-2}

TABLE 11A.1-16

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE INFANT GROUP FROM GASEOUS EFFLUENTS*

At Maximum Beef Animal Location

(Annual Dose in mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground	2.4-02	2.8-02	2.4-02	2.4-02	2.4-02	2.4-02	2.4-02	2.4-02
Inhalation	1.4-04	0.0	1.8-04	2.5-04	1.8-02	2.5-04	4.1-04	1.4-04
Total dose	2.4-02	2.8-02	2.4-02	2.4-02	4.2-02	2.4-02	2.4-02	2.4-02

*Analysis performed at maximum beef animal location is 1,693 m (5,555 ft) east.

NOTE: 2.4-02 = 2.4×10^{-2}

TABLE 11A.1-17

COMPARISON OF MAXIMUM CALCULATED DOSES FROM UNIT 2 WITH APPENDIX I DESIGN OBJECTIVES

Criterion	Appendix I Design Objective(1)	Unit 2 Calculated Dose
Gaseous effluents		
Gamma air dose(2), mRad/yr	10	5.8-02
Beta air dose(2), mRad/yr	20	4.2-02
Noble gas - total body(3), mRem/yr	5	2.9-02
Noble gas - skin(3), mRem/yr	15	6.1-02
Iodines and particulates(4)		
Any organ (thyroid), mRem/yr	15	1.7+00
Liquid effluents		
Total body, mRem/yr	3	5.8-01
Any organ(5), mRem/yr	10	9.6-01

NOTE: 5.8-02 = 5.8×10^{-2}

(1) Per reactor.
(2) Calculated at exclusion area boundary 1,603 m (5,259 ft) east.
(3) Calculated at 1,693 m (5,554 ft) east.
(4) Infant thyroid dose from cow milk 2,350 m (7,710 ft) east-southeast.
(5) Child bone dose is calculated to be the highest organ dose.

TABLE 11A.1-18

CALCULATED ANNUAL DOSES FOR
POPULATION WITHIN 80-KM (50-MI) RADIUS

	Whole Body (man-Rem)	Thyroid (man-Rem)
Liquid Effluents		
Ingestion of potable water	3.8-02	3.4-02
Ingestion of fish	1.3+00	4.6-03
Shoreline recreation	2.2-02	2.2-02
Swimming	4.5-05	4.5-05
Boating	2.2-05	2.2-05
Total	1.4+00	6.1-02
Gaseous Effluents		
Submersion	3.3-01	3.3-01
Inhalation	1.4-02	9.9-01
Standing on contaminated ground	7.3-02	7.3-02
Ingestion of fruits, grains, and vegetation	1.7-01	1.2+00
Ingestion of cow milk	4.8-02	6.8-01
Ingestion of meat	3.8-03	7.1-03
Total	6.4-01	3.3+00

NOTES: 1. Based upon a projected 80-km (50-mi) population of 1.2+06 for the year 2010.

2. $3.8\text{-}02 = 3.8 \times 10^{-2}$

TABLE 11A.1-19

CALCULATED POPULATION DOSE COMMITMENT

(Contiguous U.S. Population Dose)

	Annual Dose Per Site	
	Total Body (man-Rem)	Thyroid (man-Rem)
Liquid effluents	1.4+00	6.1-02
Noble gas effluents	1.14+00	1.37+00
Radioiodines and particulates*	1.9+01	2.3+01
Total	2.1+01	2.5+01

NOTE: 1.4+00 = 1.4×10^{0}

*Carbon-14 and tritium have been added to this category.

11A.2 COST-BENEFIT ANALYSIS

This section presents the results of cost-benefit analyses performed in accordance with Section II.D of 10CFR50, Appendix I.

Augments to the liquid and gaseous effluent systems and respective potential reductions to the annual population exposure are taken from Regulatory Guide 1.110. The beneficial savings of each augment were calculated by multiplying the calculated dose reduction by $1,000/man-Rem or $1,000/man-Rem/thyroid.

Augments to the Liquid Effluent Treatment System

Table 11A.2-1 presents the calculated base case annual total body dose (man-Rem) and thyroid dose (man-Rem/thyroid) associated with the operation of the plant liquid radwaste system for the population expected to live within an 80-km radius of the plant for the year 2010. Assuming that each augment is capable of reducing the population doses to zero (an extremely conservative assumption), the maximum benefit to be derived from any augment would be $1,400 for reducing man-Rem exposures to zero and $61 for reducing man-Rem/thyroid exposures to zero.

In an analysis of the annualized procurement, installation, operation, and maintenance costs, the least expensive liquid radwaste augment was found to be $20,000/yr for a plant located in the northeastern United States. Since the benefit from this augment would be less than the corresponding total annualized cost, the cost-benefit ratio is greater than 1. The operation of additional equipment for the purpose of reducing the annual population dose would not be cost effective. Therefore, the most cost-beneficial system has been included in the current plant design.

Augments to the Gaseous Effluent Treatment System

Table 11A.2-2 presents the calculated base case annual total body dose (man-Rem) and thyroid dose man-Rem/thyroid associated with the operation of the gaseous radwaste system for the 80-km radius population.

Assuming that each augment is capable of reducing the population doses to zero, the maximum benefit to be derived from any augment would be $640 for reducing man-Rem exposures to zero and $3,300 for reducing man-Rem/thyroid exposures to zero.

In an analysis of the annualized procurement, installation, operation, and maintenance costs, the least expensive gaseous radwaste augment was found to be $10,580/yr for a plant located in the northeastern United States. Since the benefit from this augment would be less than the corresponding total annualized cost, the cost-benefit ratio is greater than 1. The operation of additional equipment for the purpose of reducing the annual population dose would not be cost effective. Therefore, the most cost beneficial system has been included in the current plant design.

TABLE 11A.2-1

BASE CASE ANNUAL POPULATION DOSES
DUE TO LIQUID EFFLUENTS

Pathway	Total Body Dose (man-Rem)	Thyroid Dose (man-thyroid-Rem)
Ingestion of fish	1.30+00	4.60-03
Ingestion of potable water	3.80-02	3.40-02
Shoreline recreation	2.20-02	2.20-02
Swimming	4.50-05	4.50-05
Boating	2.20-05	2.20-05
Total	1.40+00	6.10-02

NOTE: $1.30+00 = 1.30 \times 10^0$

TABLE 11A.2-2

BASE CASE ANNUAL POPULATION DOSES
DUE TO LIQUID EFFLUENTS

Pathway	Total Body Dose (man-Rem)	Thyroid Dose (man-thyroid-Rem)
Submersion	3.3-01	3.3-01
Inhalation	1.4-02	9.90-01
Standing on contaminated ground	7.30-02	7.30-02
Ingestion of fruits, grains, and vegetation	1.70-01	1.20+00
Ingestion of cow milk	4.8-02	6.80-01
Ingestion of meat	3.8-03	7.10-03
Total	6.40-01	3.30+00

NOTE: 3.3-01 = 3.3×10^{-1}

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RADIATION PROTECTION

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12.2 RADIATION SOURCES

12.2.1 Contained Sources

12.2.1.1 General

Three types of radiation sources that occur in a nuclear plant are discussed in this section: primary radiation from the reactor core, secondary radiation sources resulting from nuclear reactions between the primary radiation and the reactor environment (activation products), and possible release of radioactive material from the reactor core (fission products). During normal operations, secondary sources and released radioactive materials are transported by either the reactor water or the steam to process components in the plant. This section discusses the design sources which are grouped by location and equipment type (e.g., primary containment, core sources). The following sections present the source data for various pieces of equipment throughout the plant. General locations of equipment are shown on the general plant arrangement drawings of Section 1.2.

The biological shield wall and the drywell wall are the principal shields for radiation from the reactor core. The maximum expected neutron flux is a factor in the design of the thickness of these walls. Consideration is also given to gamma rays emitted due to thermalization of neutrons in the shields.

With the exception of the biological shield wall and drywell wall, shielding designs are based on fission product and activation product sources consistent with Section 11.1 and Table 12.2-16. For shielding, it is conservative to design for fission product sources at peak values rather than an annual average, even though experience indicates the annual average is considerably less than the design. Activation products, principally N-16, are the principal source for shielding calculations in most of the primary system. In areas where fission products are significant, conservative allowance is made for transit decay, while at the same time providing for transient increase of the noble gas source, daughter product formation, and energy level of emission. Areas where fission products are significant relative to activation products include: the condenser off-gas system downstream of the steam jet air ejector, liquid and solid radwaste equipment, portions of the reactor water cleanup (RWCU) system, and portions of the feedwater system downstream of the hotwell including condensate treatment equipment. 11

A review of the systems that contain radioactive fluid was
performed to determine whether an increase in source terms
due to crud deposition was required. Based on Unit 2 cal-
culated contact dose rates and measured data[3] (which sum-
11 marizes operating plant data for various systems and
equipment), it was determined that various system components
required an increased source term. Table 12.2-16 provides
the additional contact dose rates which are included in the
shielding design analysis of the corresponding components.

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12.2.1.2 Primary Containment (Drywell)

12.2.1.2.1 Reactor Vessel Sources

This section defines the reactor vessel model for the associated gamma and neutron radiation sources and provides data required for calculations beyond the vessel. The data selected were not chosen for any given computer program, but were chosen to provide information for any of several shield analysis program types.

Physical Data

Table 12.2-1 presents the physical data required to form a reactor vessel model. The data include core material volume fractions, thermal power, average power density, power peaking factors, and noncore average water densities.

Gamma Ray Source Energy Spectra

The energy spectra (presented in this section) include fission gamma rays, fission product gamma rays, and gamma rays resulting from inelastic neutron scattering and thermal neutron capture. The total gamma ray energy release rate in the core is estimated to be accurate to within ±10 percent, while the energy release rate above 6 MeV may be in error by as much as a factor of ±2.

Core Spectrum After Operation Table 12.2-2 gives the gamma ray energy spectrum in MeV/sec-watt in spent fuel at selected time intervals after operation. The data were prepared from tables of fission product decay gamma energies fitted to integral measurements for operation times of 10^8 sec (approximately 3.2 yr). Shutdown sources in the core are found in the same manner as operating sources, by combining the shutdown spectrum with the core thermal power and power distributions. Shutdown sources in a single fuel element can be obtained by using the spectrum and the thermal power that the element contained during operation.

Around the Vessel - Gamma Ray and Fast Neutron Fluxes

Table 12.2-3 presents the calculated gamma ray energy fluxes and fast neutron fluxes with energy greater than 1 MeV at three points outside of the reactor vessel. The calculated data can vary by a factor of ±3. Also, the gamma ray energy fluxes in this section do not include any provisions for scattering from points outside of the vessel nor are there any provisions for gamma ray fluxes from depositions of radioactive isotopes within the vessel.

12.2.1.2.2 Radioactive Sources in the Reactor Water, Steam, and Off Gas

The radioactive sources in the reactor coolant and steam are discussed in Section 11.1, and the off-gas sources are discussed in Section 11.3.3. These sections provide concentrations, during normal operation, of the radioisotopes leaving the reactor vessel and gases leaving the condenser.

12.2.1.3 Reactor Building

12.2.1.3.1 Reactor Water Cleanup (RWCU) System Sources

The radioactive sources in the RWCU system (Section 5.4.8) are the result of the activity in the reactor water in transit through the system or accumulation of radioisotopes removed from the water. Components for this system include regenerative and nonregenerative heat exchangers, pumps, valves, and filter demineralizers. All major components of the cleanup system are located in the reactor building. Activities for the cleanup system are given in Table 12.2-4. The contact dose rates obtained from these activities are 11
augmented by the crud contribution given in Table 12.2-16.

12.2.1.3.2 Main Steam System Sources

All radioactive materials in the main steam system result from radioactive sources carried over from the reactor during plant operation. In most of these components the source is dominated by N-16; where the N-16 has decayed, the other isotopes carried by the steam become significant (Section 12.2.1.4.1).

12.2.1.3.3 Residual Heat Removal (RHR) System Sources

The design gamma source strengths in the RHR system (Section 5.4.7) were evaluated for system operation in the reactor shutdown mode. In this mode, the system recirculates reactor coolant to remove reactor decay heat, operating from approximately 2 hr after shutdown until the end of the shutdown period. When necessary, this system also may supplement the spent fuel pool cooling system capacity. The RHR system is described in Section 5.4.7. The source in the RHR system is the activity in the volume of reactor water contained in the system. The source strengths in the RHR equipment 8 hr after shutdown are shown in Table 12.2-5. The contact dose rates obtained from these 11
activities are augmented by the crud contribution given in Table 12.2-16.

12.2.1.3.4 Reactor Core Isolation Cooling (RCIC) System Sources

The radioactive sources in the RCIC System (Section 5.4.6) were evaluated for the system operating in the test mode. This system may be utilized during reactor shutdown if the main condenser or feedwater systems are unavailable. The system is operated from the time of reactor shutdown for approximately 2 hr until a reactor pressure of 150 psia is achieved. Below 150 psia the RHR system is initiated to achieve cold shutdown.

During routine testing of the system, the source in the equipment is the activity of the steam driving the system turbine. This activity is dominated by N-16. The radiation source data used in the shield design of this system are shown in Table 12.2-6. The contact dose rates obtained from
11 these source data are augmented by the crud contribution given in Table 12.2-16.

12.2.1.3.5 Fuel Pool Cooling and Cleanup (SFC) System Sources

The radiation source for spent fuel is given in Section 12.2.1.2.1, Gamma Ray Source Energy Spectra - Core Spectrum After Operation, and Table 12.2-2. There are two major sources of radioactivity in the spent fuel pool: the mixing of the reactor coolant with the fuel pool water at the start of refueling, and the release of crud from the surface of the spent fuel assemblies during movement.

The activities in the SFC filter/demineralizer are determined from initial reactor coolant activities. First, the corrosion product activities of the reactor coolant are multiplied by 3 to account for an increase in soluble species during the first week following shutdown. Next, the activities are decayed for 24 hr and then diluted by the fuel pool water. This spectrum is fed to the filter/demineralizer (removal efficiency assumed 100 percent) and built up for 7 days. The spectrum obtained here is added to the spectrum generated by the original coolant activities decayed for 7 days and built up in the filter/demineralizer for 23 days. This sum is the activity used for shielding; this two-part method results in a conservative shield design.

The radioactivity in the SFC heat exchanger is arrived at by assuming the SFC filter/demineralizer removal efficiency is 90 percent. The long-lived corrosion product spectrum is determined by the 10 percent passed through. This distribution is normalized to give dose rates consistent with data from other plants due to the plateout and buildup of these corrosion products in the heat exchanger. These and the spent fuel storage pool, surge tank, and other activities shown in Table 12.2-7, are based on liquid activity and crud buildup, and are used for shielding determination.

All SFC components are located within the reactor building. The SFC system is described in Section 9.1.3.

12.2.1.3.6 Liquid Radwaste System Sources (Reactor Building)

Components of the liquid radwaste system (LWS) located in the reactor building include the LWS reactor water cleanup phase separator tanks and pumps (el 289 ft), the LWS spent fuel pool phase separator tank and pump (el 289 ft), and various pipes and valves, including drain discharge piping. All other components are located in the radwaste building (Section 12.2.1.5.1).

12.2.1.3.7 Startup Sources

The four reactor startup sources are tubular source holders each containing a Californium-252 capsule. Californium-252 produces neutrons by spontaneous fission. The Californium-252 (Cf-252) capsules are shipped to the site in a special shielded cask to the source holders. They are then loaded into the reactor while being kept under water. They provide a sufficient source of neutrons in the core to help test the neutron flux monitors (of different ranges) for proper operation and response before and during fuel loading. There are seven alternate locations for various tests. These sources and source holders may be removed from the core at the end of the first fuel cycle.

12.2.1.3.8 Traversing Incore Probe (TIP) System Sources

Five TIPs provide neutron flux readings in the core for calibration of other instruments. The maximum radiation source for this system is given in Table 12.2-8. The radiation source is based upon the probe's location within the core and its residence time. As indicated in the tables, the system is divided into three sections for shielding calculations: the probe's fissionable material, nonfissionable material, and signal cable. Sources are provided for 100-sec irradiation and zero-sec decay.

12.2.1.4 Turbine Building

12.2.1.4.1 Turbine System Sources

Piping and equipment that contain main steam are sources of radiation due to the presence of N-16, the predominant source of activity during operation. Fission product gases (xenon and krypton), water activation products (O-19 and N-17), and the carryover of iodine and other fission products present in steam and condensate are considered additional activity sources. The carryover is conservatively assumed to be 2 percent by weight for halogens and 0.1 percent by weight for other fission products. The N-16 concentrations in the turbine system equipment are listed in Table 12.2-9. Transit decay of the short-lived N-16 has been included in the source terms.

12.2.1.4.2 Condensate and Feedwater System Sources

The sources in the condensate system are based on decayed main steam activities. These systems are described in Section 10.4.

The condensate demineralizer system is designed to maintain high purity of the condensate to assure high quality feedwater to the reactor by removal of various contaminants. Each condensate demineralizer is a tank filled with resins that serves to remove ionic impurities and to filter out suspended solids from the condensate. Resin can be reused if the collected crud is removed and the ion-exchange properties are renewed (regenerated). These are accomplished by ultrasonic resin cleaning and by chemical washing.

Almost all noncondensables entering the condenser are removed by the air ejectors. Because of this and the relatively long holdup time, the N-16 and other gaseous activity is very low in the hotwell and negligible in the remainder of the condensate system. The activities caused by activated corrosion and fission products are shown in Table 12.2-10 for the condensate system.

12.2.1.4.3 Off-Gas System Sources

The gaseous effluent treatment system (Section 11.3.2) is designed to limit offsite doses from routine station releases. The off-gas system contains sources of radiation based on the holdup of noncondensable gases from the main condenser. These gases are removed by the condenser air removal system and are treated through the use of equipment including catalytic recombiners and ambient temperature charcoal adsorption decay beds. The N-16 activity in the air removal system is shown in Table 12.2-9. The N-16 activity in the offgas system (downstream of the holdup pipe) is negligible due to decay. Therefore, the predominant radiation sources are the fission product gases, xenon and krypton, and their daughter products. The source terms used for shielding are given in Table 12.2-11.

12.2.1.5 Radwaste Building

12.2.1.5.1 Liquid Radwaste System Sources (Radwaste Building)

The LWS (Section 11.2.2) is composed of four subsystems designed to collect, treat, and then recycle or discharge different categories of potentially radioactive waste water. The activities from sources in the LWS such as pipes, tanks, filters, demineralizers, and evaporators used in shielding calculations are listed in Table 12.2-12.

12.2.1.5.2 Solid Radwaste System Sources

The solid radwaste system is designed to collect, hold, monitor, process, package, and provide temporary storage facilities for solid radioactive wastes prior to shipment for offsite disposal. This system is described in Section 11.4. Radioisotopic inventories of major components in this system that are used in shielding calculations are listed in Table 12.2-13.

12.2.2 Airborne Radioactive Material Sources

Unit 2 is designed so that the airborne radionuclide concentration in normally occupied areas is well below the limits discussed in 10CFR20. Radiation zoning is discussed in Section 12.3.1.2. Areas that are designated Radiation Zone IV, V, or VI are considered to be normally occupied areas. Table II, column 1 of Appendix B to 10CFR20 provides the criteria used for Zone V and VI areas. Table I, column 1 of Appendix B to 10CFR20 provides the criteria used for a Zone IV radiation area.

Radioactive materials become airborne through evaporation and attachment to suspended water droplets and water vapor. The water vapor comes from leaks in high energy lines (pressurized hot water). Suspended water droplets are created by sprays and splashing. Evaporation occurs wherever there is standing water exposed to air. The level of airborne radioactivity is periodically determined by the radiation protection staff to ensure that radiation exposure is as low as reasonably achievable (ALARA).

9 ### 12.2.2.1 Method for Computing the Airborne Radionuclide Concentration in a Plant Area

The method used for computing the airborne radionuclide concentrations for areas in the reactor, turbine, and radwaste buildings is based on data given in NUREG-0016, Revision 1, and EPRI-495[1,2]. The activity releases (in uCi/yr) from a building are distributed throughout the various building areas specified as follows by evaluating the ventilation system design along with the distributions recommended in NUREG-0016 and EPRI-495. These radionuclide concentrations result in the expected annual releases specified in Table 11.3-1 after credit is taken for filtration of radioiodine and particulates.

For both power operations and shutdown, 100 percent of the reactor building release is assumed to come from the areas containing the RHR, RWCU, and ECCS equipment and components.

For full-power operations, 85 percent of the activity released from the turbine building is from the main condenser area, which is not normally occupied. The
9 remaining 15 percent of the turbine building release is from miscellaneous areas, including the steam jet air ejector area, the turbine operating floor, the feedwater pump area, and the mechanical vacuum pump area. Noble gas concentrations during full-power operations in the turbine operating floor, feedwater pump area, and the mechanical vacuum pump area are expected to be negligible.

The radwaste building is divided into three major areas contributing to the total activity release. These are the waste collection system areas, the floor drain collection system areas (both of which are liquid radwaste system areas), and the solid waste system areas.

The liquid radwaste system consists of four subsystems: waste collection, floor drain, regenerant waste, and phase separator (as discussed in Section 11.2.2). The calculations of airborne radionuclide concentrations are made with the regenerant waste system areas and processing capability considered as part of the waste collection system, and with the phase separator system areas and processing capability considered as part of the solid waste system.

For power operations, 72 percent of the radwaste building releases come from the waste collection system areas; 18 percent, from the floor drain collector system areas; and 10 percent, from the solid waste system areas. The breakdown of releases from the liquid waste system areas is based on the expected processing capability of the subsystems. 9

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12.2.2.2 Production of Airborne Radioactive Material Sources

The primary potential sources of airborne radioactivity during normal operations are:

1. Leakage from process equipment in radioactive systems, such as valves, flanges, and pumps.

2. Evaporation from sumps, drains, tanks, and filter/demineralizer vessels that contain radioactive fluid, except where vapor is hard-piped to an HVAC system.

3. Exhaust from relief valves.

4. Gases released during removal of RPV head and associated internals.

5. Evaporation and gases released from sampling.

6. Airborne radioactivity released from the spent fuel pool water and spent fuel movement.

7. Maintenance activities such as decontamination.

Sections 12.2.2.2.1 to 12.2.2.2.6 discuss each of these sources and their effect on the airborne radionuclide concentrations in normally-occupied areas. Design features that serve to reduce these concentrations are also discussed. Tables 12.2-15a and 12.2-15b present the airborne radioactivity concentrations expected in reactor building, radwaste building, and turbine building areas for both power operations and shutdown.

Abnormal occurrences that can cause airborne radiation include: (1) spills (i.e., overflows and splashing), (2) failure of a ventilation system, (3) cracks in piping, (4) failures of pump and valve seals, and (5) malfunctioning equipment.

Airborne radioactivity is expected in the decontamination area, occasionally in labs, and during refueling on the refueling floor. The airborne radioactivity is caused by leaks, spills, venting, decontamination, etc.; concentrations are calculated for the occurrences that are the most common, leaks and venting.

12.2.2.2.1 Effect of Leakage from Process Equipment in Radioactive Systems

Normally there is no leakage from equipment in radioactive systems. If a leak occurs, its effect is determined by three items.

If a leak occurs from a component located in a high radiation area, it does not contribute to the airborne radionuclide concentration in a normally occupied area because the plant HVAC systems are designed to provide airflow from areas of lesser to progressively greater potential radioactive contamination prior to final exhaust. Separate HVAC systems are provided for each building to aid in the isolation of contamination. Areas where radiation levels vary are regarded as having a high potential for airborne radioactivity. Air contained in these areas is treated as if the highest potential of radioactivity exists, so that it will not affect the airborne radionuclide concentration of a normally occupied area.

The system operating pressure affects the possible leakage rate. A system such as the main steam system, which operates at high pressure, is expected to have a higher leakage rate than a system that operates at atmospheric pressure. However, radioactive systems that operate at high pressure are located in high radiation areas. Thus, these systems do not significantly contribute to the airborne radioactivity level in normally occupied areas due to the HVAC airflow discussed earlier.

A system that can leak highly radioactive fluid such as reactor coolant or main steam is of greater concern, initially, than a system that can leak a less contaminated fluid such as condensate storage tank water. Systems that can leak highly radioactive fluid are located in high radiation areas, and this activity is not transported to lower radiation areas by HVAC airflow, as discussed earlier.

There are many systems containing radioactive fluids that are located in low radiation areas. Each system has been evaluated to determine the effective potential of creating an airborne radiation hazard. Design features have been incorporated to reduce the possibility of hazard. These measures include hard piping all relief valves on the
25 | auxiliary boiler steam system to a contact condenser. Gaseous effluents from this tank are vented to the auxiliary boiler building ventilation system, which is ducted to a monitored release point.

12.2.2.2.2 Effect of Sumps, Drains, and Tanks

The floor and equipment drains in the reactor, radwaste, and turbine buildings are designed to collect and transport various types of waste to the LWS for processing. These drain systems convey waste by gravity to their respective sumps; waste is pumped from the sumps to the radwaste building. Drains and sumps in the systems noted are not significant sources of airborne radionuclides for the following reasons:

1. Each sump is covered with a steel plate. The free volume in the sump is maintained at a negative pressure with respect to the surrounding area by the use of a riser vent, which is connected to the HVAC system in the building of concern. The steel plate covering the sump does not provide an airtight seal. Air is drawn into the sump around the edges of the steel cover and exhausted through the riser into the HVAC system. Any radionuclides that escape into the free volume of the sump are discharged to the HVAC system and do not escape into the area surrounding the sump.

2. To prevent crud buildup, the drains empty by gravity with no water traps or level pipe runs. Air is drawn through the drains to the sump by the same riser vent discussed in the previous paragraph and out to the HVAC system.

Sump 2DFT-SUMP2H receives condensate effluent intermittently from the main steam line drain to the condenser. This sump is not vented to the building ventilation system because there is no flashing concern which could result in airborne radioactivity due to the temperature being less than 212°F. Also, noble gases, which could become airborne even in the absence of flashing, are expected to be negligible.

The holding tanks and filter/demineralizer units that contain significant inventories of radionuclides are hard piped to the HVAC system. These tanks and filter/demineralizer vessels are located in high radiation areas. Even if any airborne radionuclides were released from these components, there would be no effect on normally occupied areas due to the HVAC system design features discussed in Section 12.2.2.3.1.

12.2.2.2.3 Effect of Relief Valve Exhaust

The relief valves found in the various plant systems which can exhaust radioactive fluids are not considered a significant source of airborne radioactivity in normally occupied areas for the following reasons:

1. The exhaust of many relief valves is piped directly to the condenser, with no access to the atmosphere.

2. The exhaust of other relief valves is either piped directly to the suppression pool or to the floor or equipment drains. These drains lead to the LWS (Section 12.2.2.2.2).

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12.2.2.2.4 Effect on Removing RPV Head and Associated Internals

Experience at BWR plants has shown that an inventory of radioactive gases will accumulate inside the reactor vessel head between the time of shutdown and head removal. These gases consist primarily of the longer-lived radiohalogens and noble gases. To prevent these gases from being released to the refueling area, provisions are made for the venting of the gases to the HVAC system prior to RPV head removal. These gases are vented through an 8-in diameter duct connected between the reactor head and reactor head evacuation filter assembly containing particulate and charcoal filter units.

Experience at Dresden and Quad-Cities stations has shown that some airborne radioactivity can result from the following:

1. When the reactor water in the reactor cavity goes above 100°F, noticeable increases in the I-131 airborne activity result, increasing with temperature.

2. When the reactor water level in the vessel is low, previously covered metallic surfaces dry out. If cobalt dioxide (CoO_2) is plated out on these surfaces, air moving across these surfaces can dislodge fine particles of CoO_2. The dryer and separator are also susceptible to this phenomeon.

These two airborne activity problems have been solved by maintaining water temperatures below 100°F and by making provisions for clean water services to the RPV cavity area.

This permits wetting of the RPV cavity and components. By following these procedures, it is anticipated that RPV head and reactor internals removal will have a minimal effect on the airborne radionuclide level in the spent fuel area.

12.2.2.2.5 Effect of Sampling

The possibility of releasing radionuclides that could become airborne during sampling operations is recognized. Design features are incorporated into the sample system to limit the radionuclide release. Radioactive fluids that require frequent grab sampling are piped via sample lines to fume hoods located in sample rooms. Grab sampling will be accomplished in the fume hoods. During sampling, an inflow air velocity of approximately 100 ft/min will be maintained to sweep any airborne radioactive particles to the exhaust duct. Administrative control is used by following procedures when process fluids are sampled. This minimizes the release of radioactive fluids and, hence, exposure to personnel during the sampling process.

12.2.2.2.6 Effect of Spent Fuel Movement

Experience at operating BWR plants has shown that fuel movement normally does not present any unusual radiological problems. The expected level of radioactivity in the spent fuel pool water is listed in Table 12.2-7; this includes activity due to crud buildup. Evaporation of the spent fuel pool water is the major possible contributor to airborne activity, but is not expected to be significant.

12.2.2.2.7 Effect of Solid Radwaste Handling Areas

The solid radwaste handling equipment located in the radwaste building is designed for semi-remote operation. Entry for maintenance activities will normally entail shutdown and flushing of systems and equipment.

The ventilation supply for the radwaste building is filtered outside air. The air flow around all components that are possible sources of airborne radioactive contamination is ducted directly to the HVAC system. Expected airborne
9 radioactivity concentrations in the solid radwaste handling areas are provided in Table 12.2-15.

12.2.2.2.8 Effect of Liquid Radwaste Handling Areas

Low maintenance type equipment is designed for the LWS, located in the radwaste building. All components that contain radioactive materials and are located near normally occupied areas are shielded. The ventilation of the liquid radwaste handling areas is similar to that of the solid radwaste system. The supply air is filtered outside air, and the air flow around components that are possible sources of airborne radioactivity is return ducted either individually or by cubicle directly to the HVAC system. Expected airborne radioactivity concentrations in liquid waste handling areas are provided in Table 12.2-15. 9

12.2.2.3 References

1. Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR-GALE Code), NUREG-0016, Revision 1, January 1979.

2. Sources of Radioiodine at Boiling Water Reactors, EPRI 495, February 1978.

3. Hazzan, M. J.; Stocknoff, M. S.; Barcomb, D.; Irving, T. Radiation Levels Due to Crud Deposition in Boiling Water
11 Reactors. Presented at American Nuclear Society 1983 Winter Meeting, San Francisco, CA.

TABLE 12.2-1

BASIC REACTOR DATA

A. Reactor thermal power* 3,489 MW

B. Average power density 50.7 watts/cm^3

C. Core power peaking factors:

1. At core center:

$$\left.\frac{P_{avg}}{P_{max}}\right|_Z \quad \text{(axial)} \quad 1.5$$

$$\left.\frac{P_{avg}}{P_{max}}\right|_R \quad \text{(radial)} \quad 1.4$$

2. At core boundary:

$$\left.\frac{P_{avg}}{P_{max}}\right|_Z \quad \text{(axial)} \quad 0.5$$

$$\left.\frac{P_{avg}}{P_{max}}\right|_R \quad \text{(radial)} \quad 0.7$$

D. Core volume fractions:

Material	Density (g/cc)	Volume Fraction
UO_2	10.4	0.254
Zr	6.4	0.140
H_2O	1.0 (liquid)	0.274
Void	0	0.332

E. Average water density between core and vessel and below the core 0.74 g/cc

F. Average water-steam density above core:

1. In the plenum region 0.23 g/cc
2. Above the plenum (homogenized) 0.6 g/cc

G. Average steam density 0.036 g/cc

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* A power level of 105 percent of rated power (3,323 MW) is chosen to ensure that the calculated design bases sources bound the expected sources in the reactor at the licensed rated power condition.

TABLE 12.2-2

POST OPERATION GAMMA SOURCES IN CORE*

(MeV/sec-watt)

Energy Bounds (MeV)	Time After Shutdown			
	0 sec	1 day	1 week	1 month
4.0-6.0	8.2+9	<1.0+6	<1.0+6	<1.0+6
3.0-4.0	1.8+10	7.0+6	4.6+6	<1.0+6
2.6-3.0	1.1+10	5.7+6	3.7+6	<1.0+6
2.2-2.6	1.7+10	2.9+8	1.7+8	<2.0+7
1.8-2.2	2.1+10	4.5+8	4.0+7	4.0+7
1.4-1.8	3.3+10	3.1+9	2.1+9	6.4+8
0.9-1.4	3.7+10	2.3+9	1.6+9	1.1+9
0.4-0.9	5.1+10	7.5+9	3.8+9	2.1+9
0.1-0.4	1.2+10	1.8+9	8.7+8	3.6+8

*Operating history of 3.2 yr.

NOTE: 8.2+9 = 8.2×10^9

TABLE 12.2-3

RADIATION LEVELS INSIDE DRYWELL

Location	Gamma Flux (Mev/cm^2-sec)* 1 Mev	1.5 Mev	2.3 Mev	3 Mev	5 Mev	7 Mev	Fast Neutron Flux (>1 Mev) (n/cm^2-sec)
A	9.4+2	6.9+3	9.4+4	9.3+5	1.2+6	4.7+6	4.3-1
B	3.0+7	1.7+8	1.1+9	7.4+9	2.2+9	4.4+9	1.0+7
C	7.4-4	3.2-1	1.2+2	4.5+3	4.0+4	3.1+5	4.7-7

*The flux levels represent direct core fluxes. Contributions caused by scattering from walls and surfaces outside the reactor vessel are not included.

NOTE: 9.4+2 = 9.4×10^2

KEY (All on outer surface):

A = Top of RPV
B = Side of RPV (at highest core axial flux height)
C = Bottom of RPV

TABLE 12.2-4

REACTOR WATER CLEANUP SYSTEM SOURCES

(uCi/cc)

Isotope	RWCU Regenerative Heat Exchanger	RWCU Nonregenerative Heat Exchanger	RWCU Filter/ Demineralizer
Kr-83m	1.23-04	1.50-04	4.20+00
Xe-131m	4.93-09	6.00-09	4.99+01
Xe-133m	8.68-07	1.06-06	6.00+00
Xe-133	1.22-05	1.48-05	2.06+02
Xe-135m	9.92-04	1.20-03	2.61+01
Xe-135	1.56-04	1.90-04	8.70+01
N-17	1.84-06	2.88-07	-
O-19	1.87-01	1.40-01	-
Nb-98	1.48-02	1.48-02	9.64-01
Tc-104	3.19-01	3.17-01	7.14+00
Br-83	2.29-02	2.29-02	4.20+00
Br-84	2.85-02	2.84-02	1.14+00
Br-85	1.38-02	1.32-02	4.39-02
I-129	2.76-18	3.37-18	2.61-13
I-131	1.30-02	1.30-02	1.36+02
I-132	2.19-01	2.19-01	3.16+02
I-133	1.60-01	1.60-01	2.56+02
I-134	3.86-01	3.85-01	2.55+01
I-135	1.70-01	1.70-01	8.70+01
Np-239	2.40-01	2.40-01	1.03+03
Rb-89	2.12-02	2.10-02	3.96-01
Sr-89	3.10-03	3.10-03	5.06+01
Sr-90	2.30-04	2.30-04	4.11+00
Sr-91	6.89-02	6.89-02	5.13+01
Sr-92	1.10-01	1.10-01	2.26+01
Y-90	3.54-08	4.31-08	3.02+00
Y-91m	4.72-04	5.74-04	3.08+01
Y-91	1.10-04	1.10-04	6.17+01
Y-92	1.93-02	1.93-02	2.27+01
Y-93	1.20-02	1.20-02	7.72-02
Y-94	-	-	2.73-08
Zr-97	4.00-05	4.00-05	4.13-02
Nb-95	4.20-05	4.20-05	7.53-01
Nb-97m	1.69-05	1.95-05	3.96-02
Nb-97	9.34-08	1.30-07	4.13-02
Mo-99	2.20-02	2.20-02	1.09+00
Tc-99m	2.80-01	2.80-01	1.30+02
Tc-101	3.65-01	3.61-01	6.35+00
Ru-103	5.40-05	5.40-05	8.59-01

TABLE 12.2-4 (Cont)

Isotope	RWCU Regenerative Heat Exchanger	RWCU Nonregenerative Heat Exchanger	RWCU Filter/ Demineralizer
Ru-105	6.19-03	6.18-03	2.10+00
Ru-106	8.40-06	8.40-06	1.49-01
Rh-103m	5.10-07	6.20-07	8.56-01
Rh-105m	9.45-04	1.07-03	2.12+00
Rh-105	1.39-06	1.72-06	4.74-03
Rh-106	5.83-06	6.42-06	1.49-01
Te-129m	1.10-04	1.10-04	1.72+00
Te-129	6.04-07	7.35-07	1.70+00
Te-131m	2.80-04	2.80-04	6.46-01
Te-131	1.44-06	1.75-06	1.29-01
Te-132	4.90-02	4.90-02	2.78+02
Cs-134	1.60-04	1.60-04	1.61+00
Cs-136	1.10-04	1.10-04	2.76-02
Cs-137	2.40-04	2.40-04	4.28+00
Cs-138	1.87-01	1.86-02	1.84-03
Ba-137m	4.69-05	5.58-05	3.94+00
Ba-139	1.59-01	1.59-01	1.69+01
Ba-140	9.00-03	9.00-03	1.13+02
Ba-141	1.65-01	1.63-01	3.73+00
Ba-142	1.61-01	1.59-01	2.09+00
La-140	2.20-06	2.68-06	9.82+01
La-141	4.19-04	5.08-04	3.99+00
La-142	1.80-02	1.81-02	1.99+00
Ce-141	8.40-05	8.40-05	2.32+00
Ce-143	8.40-05	8.40-05	2.13-01
Ce-144	3.50-05	3.50-05	6.17-01
Pr-143	1.10-04	1.10-04	1.51+00
Pr-144	1.18-06	1.43-06	6.15-01
Nd-147	1.40-05	1.40-05	1.67-01
Pm-147	5.99-12	7.30-12	9.65-04
N-13	4.71-02	4.65-02	-
F-18	3.98-03	3.97-03	7.12+00
Na-24	4.10-03	4.10-03	4.73+00
P-32	7.80-05	7.80-05	1.02+00
Cr-51	2.30-03	2.30-03	3.48+01
Mn-54	4.00-05	4.00-05	7.06-01
Mn-56	4.98-02	4.98-02	9.85+00
Fe-55	3.90-04	3.90-04	6.95+00
Fe-59	8.00-05	8.00-05	1.29+00
Co-58	5.00-03	5.00-03	8.38+01
Co-60	5.00-04	5.00-04	8.93+00
Ni-65	2.99-04	2.99-04	5.85-02
Cu-64	1.20-02	1.20-02	1.19+01

TABLE 12.2-4 (Cont)

Isotope	RWCU Regenerative Heat Exchanger	RWCU Nonregenerative Heat Exchanger	RWCU Filter/ Demineralizer
Zn-65	7.80-05	7.80-05	1.37+00
Zn-69m	8.19-04	8.19-04	8.67-01
Ag-110m	6.00-05	6.00-05	1.05+00
Ag-110	5.96-07	6.45-07	2.10-02
W-187	3.00-03	3.00-03	5.51+00
N-16	4.78-03	1.61-03	-
Zr-95	-	-	7.40-01
Nb-95m	-	-	3.44-03
Zn-69	8.46-06	1.03-05	-
Kr-85m	3.37-05	4.02-05	-
Kr-85	3.86-13	5.65-13	-

NOTE: 1.23-04 = 1.23×10^{-4}

TABLE 12.2-5

RESIDUAL HEAT REMOVAL SYSTEM PUMP AND HEAT EXCHANGER SOURCE TERMS

Isotope	Activity[1] (uCi/cc)	
Br-83	2.1-3[2]	9
I-131	1.2-2	
I-132	5.7-2	
I-133	1.2-1	9
I-134	6.2-4	
I-135	7.0-2	
Nb-95	3.9-5	
Nb-97	2.3-5	
Nb-97m	2.1-5	9
Nb-98	2.1-5	
Ag-110	1.1-6	
Ag-110m	5.6-5	
W-187	2.2-3	
Np-239	2.0-1	
La-140	1.1-3	9
La-141	3.3-3	
La-142	4.4-4	
Ce-141	1.2-4	
Ce-143	6.6-5	
Ce-144	3.2-5	9
Pr-143	1.0-4	
Pr-144	3.2-5	9
Nd-147	1.3-5	
F-18	1.9-4	
Na-24	2.6-3	9
P-32	7.1-5	
Cr-51	2.1-3	
Mn-54	3.7-5	
Mn-56	5.4-3	
Fe-55	3.7-4	9
Fe-59	7.4-5	
Co-58	4.6-3	
Co-60	4.6-4	
Ni-65	3.2-5	9
Cu-64	7.2-3	
Zn-65	7.2-5	
Zn-69m	5.1-4	9
Zr-95	3.7-5	
Zr-97	2.2-5	9
Sr-89	2.9-3	
Sr-90	2.1-4	

TABLE 12.2-5 (Cont)

Isotope	Activity (uCi/cc)	
Sr-91	3.6-2	
Sr-92	1.3-2	
Y-90	1.7-5	
Y-91	2.8-4	9
Y-91m	2.4-2	
Y-92	3.0-2	
Y-93	6.4-3	
Mo-99	1.8-2	
Tc-99m	1.1-1	
Ru-103	5.0-5	
Ru-105	1.6-3	9
Ru-106	7.8-6	
Te-129	9.9-5	
Te-129m	9.9-5	9
Te-131	4.4-5	
Te-131m	2.2-4	
Te-132	4.2-2	
Cs-134	1.5-4	
Cs-136	9.8-5	9
Cs-137	2.2-4	
Cs-138	5.8-6	
Ba-137m	2.0-4	
Ba-139	2.7-3	
Ba-140	8.2-3	
Rh-103m	5.0-5	9
Rh-105m	1.6-3	
Rh-106	7.8-6	

(1) Values less than $1x10^{-6}$ are assumed to be negligible. 9
(2) $2.1-3 = 2.1x10^{-3}$

TABLE 12.2-6

REACTOR CORE ISOLATION COOLING SYSTEM
DESIGN ACTIVITIES

Isotope	Activity (uCi/cc)
Kr-83m	2.95-04
Kr-85m	5.16-04
Kr-85	1.65-06
Kr-87	1.78-03
Kr-88	1.79-03
Kr-89	1.02-02
Kr-90	1.42-02
Xe-131m	1.27-06
Xe-133m	2.44-05
Xe-133	7.05-04
Xe-135m	2.25-03
Xe-135	1.97-03
Xe-137	1.17-02
Xe-138	7.35-03
Br-83	1.60-05
Br-84	2.51-05
Br-85	1.44-05
I-131	1.22-05
I-132	1.50-04
I-133	1.22-04
I-134	3.60-04
I-135	1.27-04
Rb-88	3.17-05
Rb-89	1.01-06
Sr-89	1.46-07
Sr-90	1.08-08
Sr-91	3.24-06
Sr-92	5.16-06
Y-90	8.89-13
Y-91m	8.09-09
Y-91	5.18-09
Y-92	8.99-07
Y-93	5.64-07
Zr-95	1.88-09
Zr-97	1.50-09
Nb-95m	2.27-15
Nb-95	1.97-09
Nb-97m	4.00-10
Nb-97	1.16-12
Mo-99	1.03-06
Tc-99m	1.32-05
Tc-101	1.75-05

TABLE 12.2-6 (Cont)

Isotope	Activity (uCi/cc)
Ru-103	2.54-09
Ru-105	2.91-06
Ru-106	4.00-10
Rh-103m	1.39-11
Rh-105m	1.00-06
Rh-106	1.87-10
Te-129m	5.17-09
Te-129	2.37-11
Te-131m	1.32-08
Te-131	3.34-11
Te-132	2.30-06
Cs-134	7.52-09
Cs-136	5.17-09
Cs-137	1.13-08
Cs-138	1.61-04
Ba-137m	1.21-09
Ba-139	7.49-06
Ba-140	4.23-07
Ba-141	7.85-06
Ba-142	7.76-06
La-140	5.54-11
La-141	1.08-08
La-142	7.96-07
Ce-141	3.95-09
Ce-143	3.95-09
Ce-144	1.65-09
Pr-143	5.17-09
Pr-144	2.99-11
Nd-147	6.60-10
Pm-147	1.51-16
N-13	3.19-04
N-16	3.24-01
N-17	1.00-05
O-19	1.98-02
F-18	1.87-04
Na-24	1.93-07
P-32	3.67-09
Cr-51	1.08-07
Mn-54	1.88-09
Mn-56	2.35-06
Fe-55	1.83-07
Fe-59	3.76-09
Co-58	2.35-07
Co-60	2.35-08

TABLE 12.2-6 (Cont)

Isotope	Activity (uCi/cc)
Ni-65	1.41-08
Cu-64	5.64-07
Zn-65	3.67-09
Zn-69m	3.85-08
Ag-110m	2.82-09
Ag-110	3.03-11
W-187	1.41-07
Nb-98	7.01-07
Tc-104	1.52-05
Np-239	1.13-05

NOTE: 2.95-04 = 2.95×10^{-4}.

TABLE 12.2-8

TRAVERSING INCORE PROBE SYSTEM MATERIALS
AND RADIATION SOURCES

A. Material Composition of TIP System Components as Used in Activation Calculations

Material	Weight (g)
Detector Region	
AISI 304 stainless steel	4.0
Commercially pure titanium	3.0
Fosterite ceramic	0.5
Nichrome	0.02
Uranium-235	0.001

Cable Region	Weight/Length (g/in)
AISI 3041 stainless steel	0.12
AISI C1070 carbon steel	2.10
Magnesium oxide	0.12

B. TIP Detector Decay Gamma Activities in Mev/sec of 0.001 g of U-235

Decay Time = 0 sec
Activation Time = 10^2 sec

Energy-MeV	MeV/Sec
0.1-0.4	3.4+9
0.4-0.9	1.5+10
0.9-1.35	1.2+10
1.35-1.8	1.1+10
1.8-2.2	8.0+9
2.2-2.6	6.4+9
2.6-3.0	5.6+9
3.0-3.5	5.1+9
3.5-4.0	4.3+9
4.0-4.5	2.5+9
4.5-5.0	1.5+9
5.0-5.5	7.9+8

TABLE 12.2-8 (Cont)

C. TIP Detector Decay Gamma Activities of Materials in Detector (Excluding U-235) in uCi in Irradiated Detector

Decay Time = 0 sec
Activation Time = 10^2 sec

Activated Isotopes	uCi
Fe-59	1.1+1
Mn-56	1.7+5
Cr-51	7.0+1
Mn-54	2.1+0
Co-58m	3.5+3
Co-58	2.2-2
Ni-57	1.1-1
Co-57	6.0-7
Ni-65	4.0+2
Co-60m	7.6+3
Co-60	1.8-3
Co-61	9.6+0
Si-31	2.9+1

D. Decay Gamma Activities of Materials in the Cable in uCi/in of Irradiated Cable

Decay Time = 0 sec
Activation Time = 10^2 sec

Activated Isotopes	uCi/in
Fe-59	8.2+0
Mn-56	7.4+4
Cr-51	3.7+0
Mn-54	1.6+0
Co-58m	1.0+2
Co-58	6.5-4
Ni-57	3.3-3
Co-57	1.8-8
Ni-65	1.2+1
Co-60m	2.2+2
Co-60	5.1-5
Co-61	2.8-1
Si-31	8.7-1

NOTE: 3.4+9 = 3.4×10^9

TABLE 12.2-9

AVERAGE N-16 ACTIVITIES IN EQUIPMENT
IN THE TURBINE BUILDING

Component	Activity (uCi/g)
Equalizer tube	8.56+01
High-pressure turbine	7.70+01
Moisture separator/reheater, tube side	5.49+01
Moisture separator/reheater, shell side	6.79+01
Moisture separator drain tank	4.80-02
Reheater drain tank	3.40-02
Low-pressure turbine	7.03+01
Condenser	2.97+01
Air removal piping	2.68+00*
Second point heater drain cooler	2.06-04
Third point heater drain cooler	2.02-03
First point heater, shell side	6.26+01
Second point heater, shell side	5.75+01
Third point heater, shell side	8.46+00
Fourth point heater, shell side	1.17+00
Fifth point heater, shell side	3.83+00
Sixth point heater, shell side	4.07+00
Clean steam reboiler	4.98+01

NOTE: 8.56+01 = 8.56×10^1
* uCi/cc

TABLE 12.2-10

CONDENSATE SYSTEM SOURCE TERMS

Isotope	Hotwell (5-min Delay) and Piping Upstream of Condensate Demineralizer Activity (uCi/cc)	Condensate Demineralizer Activity (uCi/cc)*
Kr-83m	0.0	1.25-01
Kr-85m	0.0	7.69-04
Kr-85	0.0	1.59-06
Xe-131m	0.0	7.83-02
Xe-133m	0.0	2.47-01
Xe-133	0.0	8.49+00
Xe-135m	0.0	4.31-01
Xe-135	0.0	2.80+00
Br-83	3.32-04	1.25-01
Br-84	4.84-04	4.05-02
Br-85	1.02-04	7.69-04
I-129	0.0	1.35-11
I-131	2.60-04	7.86+00
I-132	3.12-03	1.74+00
I-133	2.59-03	8.50+00
I-134	7.21-03	9.96-01
I-135	2.68-03	2.80+00
Rb-89	1.75-05	7.00-04
Sr-89	3.10-06	3.18-01
Sr-90	2.30-07	3.38-02
Sr-91	6.86-05	1.03-01
Sr-92	1.08-04	4.62-02
Y-90	2.07-10	3.14-02
Y-91m	2.67-06	5.99-02
Y-91	1.11-07	6.15-02
Y-92	2.05-05	5.76-02
Y-93	1.19-05	1.89-02
Zr-95	4.00-08	4.41-03
Zr-97	3.19-08	8.50-05
Nb-95m	1.86-13	2.87-05
Nb-95	4.20-08	5.62-03
Nb-97m	2.92-08	8.05-05
Nb-97	1.10-09	8.53-05
Mo-99	2.20-05	2.29-01
Tc-99m	2.78-04	4.65-01
Tc-101	2.98-04	1.11-02
Ru-103	5.40-08	5.05-03
Ru-105	6.12-06	4.29-03
Ru-106	8.40-09	1.17-03

TABLE 12.2-10 (Cont)

Isotope	Hotwell (5-min Delay) and Piping Upstream of Condensate Demineralizer Activity (uCi/cc)	Condensate Demineralizer Activity (uCi/cc)*
Rh-103m	2.90-09	4.55-03
Rh-105m	1.70-06	1.20-03
Rh-105	9.42-09	4.34-03
Rh-106	8.39-09	1.17-03
Te-129m	1.10-07	9.59-03
Te-129	3.46-09	6.23-03
Te-131m	2.79-07	1.32-03
Te-131	7.94-09	2.91-04
Te-132	4.90-05	6.05-01
Cs-134	1.60-07	2.29-02
Cs-136	1.10-07	5.18-03
Cs-137	2.40-07	3.52-02
Cs-138	1.71-04	1.45-02
Ba-137m	1.68-07	3.33-02
Ba-139	1.53-04	3.32-02
Ba-140	9.00-06	4.15-01
Ba-141	1.41-04	6.78-03
Ba-142	1.23-04	3.46-03
La-140	1.29-08	4.12-01
La-141	2.25-06	8.18-03
La-142	2.17-05	8.74-03
Ce-141	8.41-08	1.29-02
Ce-143	8.39-08	4.37-04
Ce-144	3.50-08	4.81-03
Pr-143	1.10-07	5.74-03
Pr-144	6.33-09	4.81-03
Nd-147	1.40-08	5.69-04
Pm-147	3.50-14	1.69-05
Na-24	4.08-06	9.69-03
P-32	7.80-08	3.95-03
Cr-51	2.30-06	1.82-01
Mn-54	4.00-08	5.53-03
Mn-56	4.89-05	1.99-02
Fe-55	3.90-07	5.62-02
Fe-59	8.00-08	7.89-03
Co-58	5.00-06	5.66-01
Co-60	5.00-07	7.28-02
Ni-65	2.93-07	1.17-04
Cu-64	1.20-05	2.40-02

TABLE 12.2-10 (Cont)

Isotope	Hotwell (5-min Delay) and Piping Upstream of Condensate Demineralizer Activity (uCi/cc)	Condensate Demineralizer Activity (uCi/cc)*
Zn-65	7.80-08	1.06-02
Zn-69m	8.17-07	1.77-03
Ag-110m	6.00-08	8.17-03
Ag-110	7.80-10	1.06-04
W-187	2.99-06	1.13-02
Nb-98	1.40-05	1.88-03
Tc-104	2.72-04	1.29-02
Np-239	2.40-04	2.14+00
Zn-69	4.81-08	1.78-03
Pu-239	-	8.88-06
H-3	1.00-02	1.46+03
N-13	4.95-03	1.30-01
F-18	3.88-03	1.15+00

*Per cc of resin volume.

NOTE: 1.25-01 = 1.25×10^{-1}

TABLE 12.2-11

OFF-GAS SYSTEM SOURCE ACTIVITIES

(Ci)

Isotope	Charcoal Adsorber Activities
Xe-131m	6.30+01
Xe-133m	2.78+02
Xe-133	1.87+04
Xe-135m	1.17+02
Xe-135	3.92+03
Xe-137	1.21+02
Xe-138	3.41+02
Cs-138	3.41+02
Kr-83m	1.15+02
Kr-85m	4.74+02
Kr-85	9.97+00
Kr-87	4.65+02
Kr-88	1.04+03
Kr-89	8.37+01
Rb-88	1.04+03

NOTE: 6.30+01 = 6.30×10^1

TABLE 12.2-13

SOLID RADWASTE SYSTEM COMPONENTS
DESIGN ACTIVITIES

(uCi/cc)

Isotope	Waste Sludge Tank	Extruder Evaporator Waste Casks
Br-83	1.56-1	2.33-2
Br-84	4.22-2	6.30-2
Br-85	1.63-3	2.43-3
I-129	1.09-11	1.63-11
I-131	1.94+1	2.90+1
I-132	2.12+1	3.17+1
I-133	1.01+1	1.51+1
I-134	9.43-1	1.41+0
I-135	3.22+0	4.81+0
Rb-89	1.47-2	2.19-2
Sr-89	2.68+1	4.00+1
Sr-90	3.49+0	5.21+0
Sr-91	1.90+0	2.84+0
Sr-92	8.35-1	1.25+0
Y-90	3.43+0	5.12+0
Y-91m	1.15+0	1.72+0
Y-91	3.49+1	5.21+1
Y-92	8.41-1	1.26+0
Y-93	2.87-3	4.28-3
Y-94	1.01-9	1.51-9
Zr-95	-	-
Zr-97	1.58-3	2.36-3
Nb-95m	-	-
Nb-95	2.91-1	4.34-1
Nb-97m	1.51-3	2.25-4
Nb-97	1.58-3	2.36-3
Mo-99	6.93-2	1.03-1
Tc-99m	4.85+0	7.24+0
Tc-101	2.35-1	3.51-1
Ru-103	4.06-1	6.06-1
Ru-105	7.78-2	1.16-1
Ru-106	1.18-1	1.76-1
Rh-103m	4.06-1	6.06-1
Rh-105m	7.84-2	1.17-1
Rh-105	2.26-4	3.37-4
Rh-106	1.18-1	1.76-1
Te-129m	7.44-1	1.11+0

TABLE 12.2-13 (Cont)

Isotope	Waste Sludge Tank	Extruder Evaporator Waste Casks
Te-129	7.44-1	1.11+0
Te-131m	2.80-2	4.18-2
Te-131	5.60-3	8.36-3
Te-132	1.95+1	2.91+1
Cs-134	1.32+0	1.97+0
Cs-136	6.19-3	9.24-3
Cs-137	3.64+0	5.43+0
Cs-138	6.82-5	1.02-4
Ba-137m	3.35+0	5.00+0
Ba-139	6.25-1	9.33-1
Ba-140	2.40+1	3.58+1
Ba-141	1.38-1	2.06-1
Ba-142	7.73-2	1.15-1
La-140	2.60+1	3.88+1
La-141	1.48-1	2.21-1
La-142	7.39-2	1.10-1
Ce-141	9.88-1	1.48+0
Ce-143	9.53-3	1.42-2
Ce-144	4.79-1	7.15-1
Pr-143	3.40-1	5.08-1
Pr-144	4.79-1	7.15-1
Nd-147	3.13-2	4.67-2
Pm-147	2.05-3	3.06-3
F-18	2.64-1	3.94-1
Na-24	1.79-1	2.67-1
P-32	2.37-1	3.54-1
Cr-51	1.34+1	2.00+1
Mn-54	5.53-1	8.26-1
Mn-56	3.65-1	5.45-1
Fe-55	5.76+0	8.60+0
Fe-59	6.46-1	9.64-1
Co-58	5.04+1	7.52+1
Co-60	7.50+0	1.12+1
Ni-65	2.16-3	3.22-3
Cu-64	4.47-1	6.67-1
Zn-65	1.05+0	1.57+0
Zn-69m	3.26-2	4.87-2
Ag-110m	8.10-1	1.21+0
Ag-110	1.62-2	2.42-2
W-187	2.24-1	3.34-1
Nb-98	3.57-2	5.33-2
Tc-104	2.64-1	3.94-1
Np-239	5.92+1	8.84+1

TABLE 12.2-13 (Cont)

Isotope	Waste Sludge Tank	Extruder Evaporator Waste Casks
Nb-98	3.57-2	5.33-2
Tc-104	2.64-1	3.94-1
Np-239	5.92+1	8.84+1

NOTE: 1.56-1 = 1.56×10^{-1}

TABLE 12.2-14

THE INFORMATION ON THIS PAGE HAS BEEN DELETED. 7

TABLE 12.2-16

CRUD RADIATION LEVELS

Component/System	Additional Contact Dose Rate Due to Crud (mrem/hr)
RCIC piping/pump	60
RHR piping/heat exchanger	600
CRD	
discharge header	600
pumps/filter	50
RWCU	
piping/pumps	2,000
heat exchanger	2,500
Reactor building equipment drain*	
tank	6,000
cooler	3,000
pump	2,000
Drywell equipment drain*	
tank	30,000
cooler	15,000
pump	10,000

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* The analyses performed for these systems are based entirely on the crud levels quoted.

6. Assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

Additional criteria are found in Section 11.5.1.2.

The primary design criteria for the nonsafety-related in-plant airborne radioactivity monitoring systems are to:

1. Provide continuous data output in the main control room of radiation levels in selected building exhaust systems.

2. Permit checking the operational availability of each channel during reactor operation with provision for calibration function and instrument checks.

3. Perform their intended functions under normal operating conditions for the design life of the plant.

Additional criteria are found in Section 11.5.1.2.

12.3.4.2.2 Criteria for Airborne Radioactivity Monitor Locations

The following criteria for locating airborne radioactivity monitors are dependent upon the point of leakage, the ability to identify the source of radioactivity so that corrective action may be performed, and the possibility for exposing personnel to airborne radioactivity:

1. Airborne radioactivity monitors sample the drywell atmosphere for reactor pressure boundary leak detection.

2. The outside air intake ducts for the main control room area are monitored to measure the possible introduction of radioactive materials into the main control room to ensure habitability of those areas requiring personnel occupancy for safe shutdown.

3. Exhaust ducts servicing an area containing processes which, in the event of a major leakage, could result in concentrations within the plant approaching the limits established by 10CFR20 for plant workers are monitored.

Monitor sensitivity criteria are noted in Section 12.3.4.2.5.

Airborne process and effluent radiation monitor locations and functions are summarized in Table 12.3-2. ANSI N13.1 was used as a guide in locating monitors and sample points. Monitor locations are shown on the shielding arrangement and facilities drawings, Figures 12.3-1 through 12.3-33.

12.3.4.2.3 System Description (Airborne Radioactivity Monitors)

Monitors Required for Safety

Drywell Atmosphere Monitoring The drywell atmosphere radiation monitors are designed for early RCPB leak detection in accordance with RG 1.45.

Redundant off-line gas and particulate monitors located in the reactor building are dedicated to sampling the drywell atmosphere. Samples are drawn from the various elevations of drywell air by the use of sampling trees, are pumped through the monitoring system, and then are returned to the drywell. Each sample is continuously monitored for particulate and gaseous activities. A complete monitor description is found in Section 11.5.2.1.1. A removable iodine cartridge filter, which may be used for laboratory analysis, is provided between the moving particulate filter and the gas sample chamber. Alarms are provided for alert or high radiation levels for each channel. Alarms are also provided for channel or sampling system component failure. All alarms are annunciated locally at the monitor and in the main control room. Recorders are provided in the main control room to maintain a permanent record of drywell radiation levels.

Reactor Building Ventilation Exhaust One off-line gas and particulate and one off-line gas monitor (Section 11.5.2.1.1) are provided on the reactor building ventilation exhaust air ductwork, both above and below the refueling floor. Their function is to indicate the airborne levels of activity in the reactor building. Sampling is performed by an isokinetic sampling system. On a high radiation alarm signal the reactor building ventilation intake and exhaust air is isolated and the reactor building air is recirculated, with a small fraction being diverted through the SGTS and exhausted after treatment.

Main Control Room Ventilation The main control room ventilation radiation monitors are designed to measure the radiation levels in the junction of the two inlet ducts of the main control room ventilation system, and automatically divert the air through the emergency filter system on detection of high radiation. Dual redundant off-line gas monitors (Section 11.5.2.1.1) are provided in the main control room outside air intake ducts' junction. Sampling is performed by a sampling system with probes and returns located near the intakes. The monitors for this intake are located in the control building.

The main control room radiation monitors provide a single channel for gaseous activity only. Fixed-particulate and iodine filters are located upstream of the gas sample chamber and can be removed for analysis in the health physics laboratory. Alert and high radiation levels, channel failure, and sampling system failure are alarmed locally and in the main control room.

TABLE 12.3-3

PERSONNEL DOSES FOR REQUIRED OCCUPANCY TIMES IN VITAL AREAS

Vital Area	Task Performed	Occupancy Time	Dose (rem)	Notes
Main control room and relay and computer room	Execute the safe shutdown of the plant	Continuous for 30 days	1.73+0	30-day average dose rate = 5.50 mRem/hr
Health physics/ counting room Unit 1	Perform routine health physics functions and analyze radioactive grab samples	8 hr	2.37-1	Dose based upon continuous occupancy for an 8-hr workday at the time of maximum dose rate
Radwaste sample room/ Unit 1 chemistry lab (PASS)(1,2)	a)Obtain and perform general isotopic and Boron analysis of dilute reactor coolant sample(3)	1 hr, 5 min	1.84+0 2.10+0	Whole body Extremity
	b)Obtain and perform isotopic analysis of containment atmosphere sample(3)	52 min	2.07+0 3.73+0	Whole body Extremity
	c)Determine level of dissolved gases (e.g, H_2) in reactor coolant	1 hr, 45 min	6.38-1 6.38-1	Whole body Extremity
	d)Obtain and perform chloride analysis of undiluted reactor coolant sample(3)	1 hr, 18 min	4.41+0 1.60+1	Whole body Extremity
Turbine building offline isotopic monitor	Replace large liquid nitrogen dewar(4)	22 min	3.62-1	Dose includes dose received for one round trip between the OSC and the monitor location
Main stack offline isotopic monitor	a) Replace large liquid nitrogen dewar and refill sample cartridge feed hopper(4)	24 min	2.44+0	Dose includes dose received for one round trip between the OSC and the monitor location.
	b) Manual sampling(5)	53 min	2.73+0 2.79+0	Whole body Extremity
Radwaste control room	a) Turn off reactor building equipment and floor drain pumps	12 min	5.96-1	Dose includes dose received for one round trip between the OSC and the radwaste control room

TABLE 12.3-3 (Cont)

Vital Area	Task Performed	Occupancy Time	Dose (rem)	Notes
	b) Service ERF computer system	22 min	8.05-1	
A one-way trip between the OSC and the control room emergency zone	For information only	6 min	1.92-1	
Technical support center	Per NUREG-0696	Continuous for 30 days in accordance with the Site Emergency Plan procedures	4.30+0	Whole body

(1)Maximum dose rate for each subtask was used to develop the maximum dose for the task.
(2)See Section 1.10, Item II.B.3, for specific information on the post-accident sampling system and Table II.B.3-1 for a breakdown of the tasks and required occupancy times.
(3)Dose includes exposure received for one round trip from the OSC, to the radwaste sample room, to the Unit 1 chem lab, and back to the OSC.
(4)This assumes that the spare dewar is stored at the monitor location.
(5)Dose includes exposure received for one round trip from the OSC, to the main stack, to the Unit 1 counting room and back to the OSC.

TABLE 12.3-4

DOSE RATE (REM/HR) AT LOCATION:*

Time Post-LOCA (Hr)	A	B	C	D	E	F	G	H	I
1	7.31+1	7.33+1	7.40+0	7.42+0	7.42+0	7.46+1	9.23+1	8.97+2	8.98+2
3	2.25+0	2.49+0	3.42+0	3.73+0	3.73+0	4.31+0	4.59+0	3.99+0	4.07+0
6	2.34+0	2.64+0	3.45+0	3.86+0	3.86+0	4.57+0	4.89+0	4.52+0	4.56+0
9	2.44+0	2.74+0	3.63+0	4.05+0	4.05+0	4.83+0	5.17+0	4.81+0	4.85+0
12	1.77+0	2.09+0	2.74+0	3.19+0	3.19+0	4.00+0	4.26+0	4.19+0	4.17+0
18	1.69+0	2.03+0	2.64+0	3.11+0	3.11+0	3.95+0	4.21+0	4.22+0	4.20+0
24	1.67+0	2.01+0	2.57+0	3.06+0	3.06+0	3.91+0	4.17+0	4.27+0	4.22+0
50	1.39+0	1.72+0	2.16+0	2.61+0	2.61+0	3.42+0	3.67+0	3.86+0	3.79+0
100	1.38+0	1.71+0	2.14+0	2.60+0	2.60+0	3.44+0	3.69+0	3.91+0	3.83+0
200	1.34+0	1.67+0	2.07+0	2.53+0	2.53+0	3.35+0	3.60+0	3.83+0	3.76+0
400	8.18-1	1.02+0	1.27+0	1.55+0	1.55+0	2.05+0	2.21+0	2.35+0	2.31+0
550	4.73-1	5.85-1	7.27-1	8.89-1	8.89-1	1.18+0	1.27+0	1.35+0	1.32+0
720	2.19-1	2.73-1	3.38-1	4.13-1	4.13-1	5.49-1	5.92-1	6.31-1	6.18-1

Time Post-LOCA (Hr)	J	K	L	M	N	O	P	Q	R
1	8.98+2	8.97+2	8.97+2	8.96+2	8.96+2	8.97+2	1.06+2	1.81+2	1.80+2
3	3.76+0	3.12+0	2.13+0	2.00+0	2.00+0	3.63+0	4.13+0	3.21+0	4.00+0
6	4.18+0	3.41+0	2.21+0	2.11+0	2.11+0	4.08+0	4.69+0	4.82+0	4.64+0
9	4.44+0	3.60+0	2.30+0	2.20+0	2.20+0	4.34+0	4.97+0	5.10+0	4.92+0
12	3.75+0	2.87+0	1.52+0	1.47+0	1.47+0	3.70+0	4.29+0	3.95+0	3.93+0
18	3.76+0	2.86+0	1.46+0	1.42+0	1.42+0	3.72+0	4.34+0	3.98+0	3.97+0
24	3.78+0	2.86+0	1.43+0	1.40+0	1.40+0	3.75+0	4.37+0	4.00+0	3.99+0
50	3.39+1	2.53+0	1.19+0	1.18+0	1.18+0	3.38+0	3.96+0	3.62+0	3.62+0
100	3.42+0	2.55+0	1.19+0	1.19+0	1.19+0	3.42+0	4.02+0	3.67+0	3.67+0
200	3.26+0	2.48+0	1.15+0	1.15+0	1.15+0	3.26+0	3.94+0	3.59+0	3.59+0
400	2.06+0	1.53+0	7.04-1	7.04-1	7.04-1	2.06+0	2.42+0	2.21+0	2.21+0
550	1.18+0	8.73-1	4.02-1	4.02-1	4.02-1	1.18+0	1.39+0	1.27+0	1.27+0
720	5.49-1	4.10-1	1.88-1	1.88-1	1.88-1	5.49-1	6.47-1	5.92-1	5.92-1

*Refer to Figure 12.3-69.

TABLE 12.3-4 (Cont)

Time Post-LOCA (Hr)	S	T	U	V	W	X	Y	Z
1	1.80+2	1.80+2	1.79+2	1.79+2	1.78+2	1.78+2	9.15+1	9.09+1
3	4.00+0	3.81+0	3.23+0	2.50+0	1.92+0	1.80+0	3.23+0	2.24+0
6	4.64+0	4.42+0	3.71+0	2.88+0	2.17+0	2.02+0	3.50+0	2.30+0
9	4.92+0	4.68+0	3.90+0	2.99+0	2.23+0	2.07+0	3.68+0	2.38+0
12	3.93+0	3.67+0	2.86+0	1.90+0	1.10+0	9.27-1	3.05+0	1.70+0
18	3.97+0	3.71+0	2.87+0	1.89+0	1.07+0	8.91-1	2.99+0	1.59+0
24	3.99+0	3.73+0	2.88+0	1.90+0	1.05+0	8.71-1	2.95+0	1.52+0
50	3.62+0	3.37+0	2.56+0	1.69+0	9.05-1	7.56-1	2.97+0	1.63+0
100	3.67+0	3.41+0	2.57+0	1.69+0	8.91-1	7.19-1	2.56+0	1.20+0
200	3.59+0	3.34+0	2.52+0	1.66+0	8.65-1	6.95-1	2.49+0	1.15+0
400	2.21+0	2.05+0	1.55+0	1.02+0	5.31-1	4.27-1	1.53-0	7.05-1
550	1.27+0	1.18+0	8.87-1	5.85-1	3.04-1	2.44-1	8.75-1	4.04-1
720	5.92-1	5.49-1	4.13-1	2.73-1	1.42-1	1.14-1	4.10-1	1.88-1

Time Post-LOCA (Hr)	AA	AB	AC	AD	AE	AF	AG	AH	AI
1	8.97+2	9.15+1	9.15+1	9.19+1	9.26+1	1.06+2	9.23+1	9.22+1	9.23+1
3	2.13+0	3.23+0	3.23+0	3.87+0	4.72+0	4.90+0	4.77+0	4.59+0	4.77+0
6	2.21+0	3.50+0	3.50+0	4.27+0	5.05+0	5.28+0	5.11+0	4.89+0	5.11+0
9	2.30+0	3.68+0	3.68+0	4.52+0	5.34+0	5.57+0	5.40+0	5.17+0	5.40+0
12	1.52+0	3.05+0	3.05+0	3.93+0	4.43+0	4.60+0	4.50+0	4.26+0	4.50+0
18	1.46+0	2.99+0	2.99+0	3.89+0	4.39+0	4.57+0	4.45+0	4.21+0	4.45+0
24	1.43+0	2.95+0	2.95+0	3.87+0	4.36+0	4.54+0	4.44+0	4.17+0	4.44+0
50	1.19+0	2.97+0	2.97+0	3.83+0	3.83+0	4.01+0	3.91+0	3.67+0	3.91+0
100	1.19+0	2.56+0	2.56+0	3.43+0	3.85+0	4.04+0	3.93+0	3.69+0	3.93+0
200	1.15+0	2.49+0	2.49+0	3.26+0	3.76+0	3.94+0	3.84+0	3.60+0	3.84+0
400	7.04-1	1.53+0	1.53+0	2.06+0	2.31+0	2.42+0	2.35+0	2.21+0	2.35+0
550	4.02-1	8.75-1	8.75-1	1.18+0	1.32+0	1.39+0	1.35+0	1.27+0	1.35+0
720	1.88-1	4.10-1	4.10-1	5.49-1	6.18-1	6.47-1	6.31-1	5.92-1	6.31-1

*Refer to Figure 12.3-69.

TABLE 12.3-4 (Cont)

Time Post-LOCA (Hr)	AJ	AK	AL	AM	AN	AO	AP	AQ
1	9.21+1	9.23+1	9.22+1	2.07+0	1.85+0	1.85+0	7.34+1	7.40+1
3	4.30+0	4.77+0	4.59+0	4.11+0	3.73+0	3.73+0	2.78+0	3.42+0
6	4.80+0	5.11+0	4.89+0	4.31+0	3.86+0	3.86+0	2.96+0	3.45+0
9	5.08+0	5.40+0	5.17+0	4.55+0	4.05+0	4.05+0	3.13+0	3.63+0
12	4.52+0	4.50+0	4.26+0	3.72+0	3.19+0	3.19+0	2.48+0	2.74+0
18	4.51+0	4.45+0	4.21+0	3.65+0	3.11+0	3.11+0	2.45+0	2.64+0
24	4.49+0	4.44+0	4.17+0	3.61+0	3.06+0	3.06+0	2.43+0	2.57+0
50	4.41+0	3.91+0	3.67+0	3.14+0	2.61+0	2.61+0	2.12+0	2.16+0
100	4.03+0	3.93+0	3.69+0	3.15+0	2.60+0	2.60+0	2.12+0	2.14+0
200	3.94+0	3.84+0	3.60+0	3.07+0	2.53+0	2.53+0	2.07+0	2.07+0
400	2.42+0	2.35+0	2.21+0	1.88+0	1.55+0	1.55+0	1.27+0	1.27+0
550	1.39+0	1.35+0	1.27+0	1.08+0	8.89-1	8.89-1	7.25-1	7.27-1
720	6.47-1	6.31-1	5.92-1	5.04-1	4.13-1	4.13-1	3.38-1	3.38-1

Time Post-LOCA (Hr)	AR	AS	Health Physics/ Counting Room Unit 1	Turbine Building Radwaste Sample Room	Turbine Building Offline Isotopic Monitor	Main Stack Offline Isotopic Monitor
1	7.34+1	7.31+1	2.46+0	1.41+1	9.15+1	4.44+0
3	2.78+0	2.25+0	2.96-2	3.50-1	3.23+0	4.71+0
6	2.96+0	2.34+0	2.81-2	3.18-1	3.50+0	6.32+0
9	3.13+0	2.44+0	2.81-2	2.97-1	3.68+0	5.43+0
12	2.48+0	1.77+0	7.37-3	1.43-1	3.05+0	5.53+0
18	2.45+0	1.69+0	5.24-3	1.04-1	2.99+0	5.85+0
24	2.43+0	1.67+0	4.06-3	8.22-2	2.95+0	4.99+0
50	2.12+0	1.39+0	1.02-3	3.27-2	2.97+0	5.54+0
100	2.12+0	1.38+0	6.07-4	2.13-2	2.56+0	5.74+0
200	2.07+0	1.34+0	1.15-4	1.43-2	2.49+0	5.44+0
400	1.27+0	8.17-1	5.23-5	8.90-3	1.53+0	2.90+0
550	7.25-1	4.73-1	2.08-5	5.18-3	8.75-1	1.72+0
720	3.38-1	2.19-1	7.59-6	2.46-3	4.10-1	1.17+0

*Refer to Figure 12.3-69

TABLE 12.3-4 (Cont)

Time Post-LOCA (Hr)	Radwaste Control Room	Unit 1 Chemistry Lab
1	3.51+1	5.12+0
3	1.25+0	6.50-2
6	1.21+0	6.19-2
9	1.19+0	6.17-2
12	8.25-1	1.77-2
18	7.18-1	1.26-2
24	6.14-1	9.71-3
50	4.67-1	2.62-3
100	4.10-1	1.53-3
200	3.79-1	4.35-4
400	2.34-1	2.34-4
550	1.35-1	1.18-4
720	6.35-2	5.18-5

*Refer to Figure 12.3-69