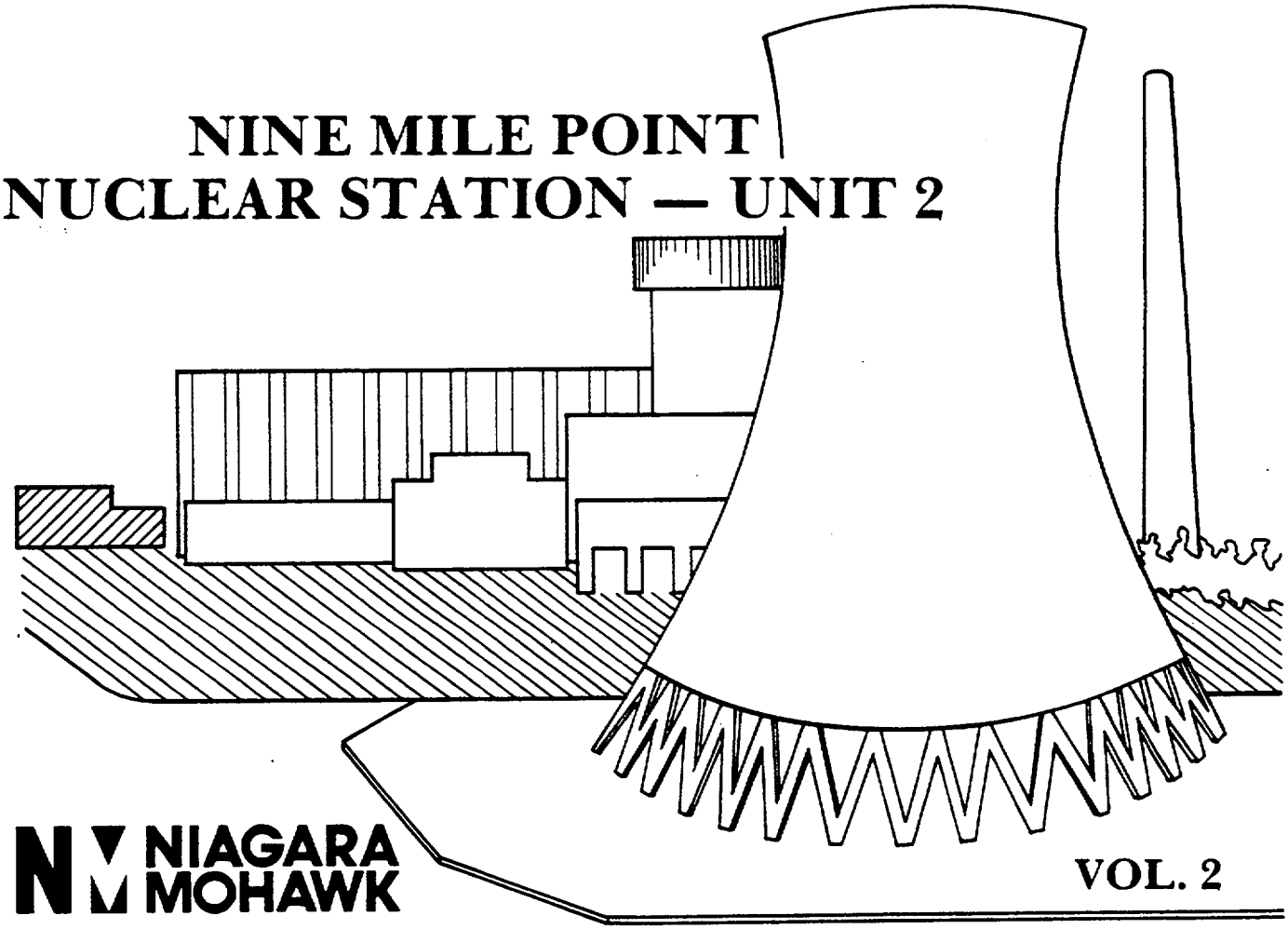
UPDATED SAFETY ANALYSIS REPORT

NINE MILE POINT NUCLEAR STATION — UNIT 2



NIAGARA MOHAWK

VOL. 2

1.8 CONFORMANCE TO NRC REGULATORY GUIDES

Tables 1.8-1 and 1.8-2 indicate the extent of compliance with all applicable NRC Regulatory Guides and the revision number of those guides. A reference to the USAR section(s) in which the applicable design features are described is also provided.

Where the design differs from the regulatory guides, alternative methods of providing an equivalent level of safety have been utilized. These differences are discussed in Tables 1.8-1 and 1.8-2, or reference is made to the appropriate USAR section(s) in which they are discussed.

TABLE 1.8-1 (Cont'd.)

REGULATORY GUIDE 1.10, REVISION 1 (JANUARY 1973)* (cont'd.)

For tests failing the second criterion of two or more splices for any six additional samples, it is considered that the failure rate pertains to the total output of all splicers, and the previous 100 splices are evaluated accordingly.

Conformance to this guide is ensured through a purchase specification.

REGULATORY GUIDE 1.11, REVISION 0 (MARCH 1971) - INSTRUMENT LINES PENETRATING PRIMARY REACTOR CONTAINMENT

FSAR Sections 6.2.4, 7.1.2.3

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Instrument lines penetrating the containment are designed in accordance with this regulatory guide.

REGULATORY GUIDE 1.12, REVISION 1 (April 1974) - INSTRUMENTATION FOR EARTHQUAKES

FSAR Section 3.7A.4.1

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

The technical requirements of the regulatory guide are implemented in the seismic portions of the procurement specification for seismic instrumentation.

REGULATORY GUIDE 1.13, REVISION 1 (DECEMBER 1975) (FOR COMMENT) - SPENT FUEL STORAGE FACILITY DESIGN BASIS

FSAR Sections 9.1.2, 9.1.3, 9.1.4

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

The design of the spent fuel storage facility incorporates the guidance listed in the regulatory position to assure that the fuel storage facility maintains the capability to perform its safety functions. An analysis of tornado protection for spent fuel storage is documented in a GE report entitled, Tornado Protection for Spent Fuel Storage Pool, APED-5696, November 1968.

REGULATORY GUIDE 1.14, REVISION 1 (AUGUST 1975) (FOR COMMENT) - REACTOR COOLANT PUMP FLYWHEEL INTEGRITY

Position This regulatory guide is not applicable to BWRs.

TABLE 1.8-1 (Cont'd.)

REGULATORY GUIDE 1.15, REVISION 1 (DECEMBER 1972)* - TESTING OF REINFORCING BARS FOR CATEGORY I CONCRETE STRUCTURES

FSAR Sections 3.8.3, 3.8.4, 3.8.5

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide. Conformance to this regulatory guide is ensured through the purchase specification for reinforcing bars.

* This regulatory guide was withdrawn on July 8, 1981, by the NRC.

REGULATORY GUIDE 1.16, REVISION 4 (AUGUST 1975) (FOR COMMENT) - REPORTING OF OPERATING INFORMATION - APPENDIX A TECHNICAL SPECIFICATIONS

FSAR Sections Chapter 13, Technical Specifications

Position NMPC complies with Generic Letter 97-02 (May 5, 1997) in meeting the criteria of reporting of operating information.

This is delineated in the Nine Mile Point Nuclear Station Site Administration Procedure that is incorporated in the Technical Specifications for Unit 2.

REGULATORY GUIDE 1.17, REVISION 0 (JUNE 1973) - PROTECTION OF NUCLEAR POWER PLANTS AGAINST INDUSTRIAL SABOTAGE

FSAR Section 13.6

Position The Unit 2 project designs, procures, and installs Unit 2 plant equipment and structures in accordance with this regulatory guide.

The Unit 2 project ensures that compliance is achieved by the following methods:

1. The Unit 2 project Security Design Review Committee reviews the design and arrangement of security-related plant equipment and structures for conformance with the position outlined above.

2. The Unit 2 project controls accessibility to Unit 2 security-related materials.

Tests and operability checks will be performed as required by this regulatory guide.

TABLE 1.8-1 (Cont'd.)

REGULATORY GUIDE 1.58, REVISION 1 (SEPTEMBER 1980) - QUALIFICATION OF NUCLEAR POWER PLANT INSPECTION, EXAMINATION, AND TESTING PERSONNEL

FSAR Section 14.2

Position During the design and construction phase, startup and test personnel involved in testing met the requirements of RG 1.58 and ANSI N45.2.6-1978, with exceptions as discussed in Chapter 14.

Unit 2 plant personnel met the requirements of this regulatory guide as discussed in Chapter 13.

GE startup operations personnel supporting the startup test phase met the requirements of this regulatory guide as discussed in Table 14.2-403.

During the operations phase, the qualification of nuclear power plant inspection, examination, and testing personnel is stated in the NMPC QA Program requirements and is satisfied as specified in the Quality Assurance Program Topical Report (QATR) for Nine Mile Point Nuclear Station Units 1 and 2 - Operations Phase (see Appendix B).

REGULATORY GUIDE 1.59, REVISION 2 (AUGUST 1977) - DESIGN BASIS FLOODS FOR NUCLEAR POWER PLANTS

FSAR Sections 2.4.5, 2.4.3

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide, with the following limitation:

No commitments for compliance are made or implied for the "to be issued" appendices.

The Unit 2 site has hardened protection from flooding by use of a lakefront revetment ditch.

Evaluation of the conditions (Paragraph C.1) resulting in the worst site-related flood probable at the Unit 2 site has been made in conformance with ANSI N170-1976/ANS 2.8. The combined events considered were:

1. Probable maximum surge and seiche with wind wave action and maximum controlled lake level.
2. Probable maximum precipitation and historical maximum lake level.
3. Probable maximum lake level and historical maximum precipitation.

The analysis showed that Unit 2 is designed to withstand these combined events with no safety impact.

REGULATORY GUIDE 1.60, REVISION 1 (DECEMBER 1973) - DESIGN RESPONSE SPECTRA FOR SEISMIC DESIGN OF NUCLEAR POWER PLANTS

FSAR Sections 3.7A, 3.7B

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide as discussed below and in Sections 3.7A and 3.7B.

The design response spectra has been used to generate the seismic data sheets for equipment loadings, for systems and component analysis, and for the structural responses of the various buildings.

TABLE 1.8-1 (Cont'd.)

REGULATORY GUIDE 1.75, REVISION 2 (SEPTEMBER 1978) - PHYSICAL INDEPENDENCE OF ELECTRIC SYSTEMS

FSAR Sections 7.1.2, 7.6.2, 8.3.1

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below and in Section 7.6.2 and 8.3.1.

Regulatory Position C.1 states that "interrupting devices actuated only by fault current are not considered to be isolation devices within the context of this document." In the case of control and instrument circuits, a combination of two interrupting devices actuated by fault current have been used to isolate non-Class 1E circuits from Class 1E circuits. Both of these devices are Class 1E, and both of them are coordinated with the main breaker upstream so that a failure of a non-Class 1E device or circuit will not affect any Class 1E device or system. Any circuit breakers associated with this redundant protection will be tested during each refueling outage.

Regulatory Position C.9 requires that cable splices in raceways be prohibited. Splicing in electrical penetrations for termination is considered to be exempt from this requirement. Also, condulets and junction boxes used as a termination point, at the load, are considered to be exempt from this requirement.

Regulatory Position C.10 requires that the cables be marked at 5-ft intervals. This is a typographical error as confirmed by the former Electrical, Instrument and Control Branch Chief of USNRC, T. A. Ippolito, on October 10, 1975, and the NRC Power Systems Branch Section Leader, R. G. FitzPatrick, on October 30, 1980. The correct distance is 15 ft, which has been followed in Unit 2. Additionally, the cable markings are inspected (100 percent) by Field Quality Control during installation. As of June 1984 more than 50 percent of all cables had been pulled and marked at 15-ft intervals. We believe that mixing the marking of the cables is inappropriate and that marking at 15-ft intervals is sufficient to ensure separation of cables.

The minimum separation distance from 600 V or less nonsafety-related conduit to safety-related open cable trays and cable in free air for any service level is 1 in.

All cables used in Unit 2 are flame retardant. The cable trays are not filled above the side rails. The hazard, in this case, is limited to failure or faults internal to the nonsafety cables in rigid steel conduit. Unit 2 has determined by analysis that 1-in separation between the Class 1E cable tray and non-Class 1E conduit provides adequate protection for the Class 1E cables in the open ladder tray in the event of any failure of the non-Class 1E cables in conduit. This has been established by tests with 600 V levels, as explained later in this section.

Aluminum sheath cables (ALS) used for low-energy 120-V ac systems and 8-hr battery-pack lighting systems, are considered enclosed raceways. These cables have flame-retardant cross-linked polyethylene insulation, chlorosulphonated polyethylene jacket, and polypropylene fillers enclosed in a continuous, impervious aluminum sheath which provides adequate protection. As such, the minimum separation between these cables and Class 1E raceways is 1 in.

The minimum separation between any Class 1E raceway and any lighting cord for drops to the lighting fixtures shall be 1 in. These cords are of size 12 AWG and supply 120/208 V ac low energy in low-density applications. As such, 1-in separation provides adequate protection to the Class 1E circuits in the event of a fault in any lighting cord.

IEEE-384-1974, Section 5.1.1.2, allows lesser separation distances than those specified in Sections 5.1.3 and 5.1.4, if established by analysis. Various tests have indicated that the following minimum separation distances between redundant Class 1E cables and raceways, or between Class 1E and non-Class 1E cables and raceways, 600 V level and below, should be adequate to maintain independence of the redundant systems. NMPC also has verified these minimum separation distances by plant-specific tests (Wyle Test Report No. 47906-02, Electrical Separation Verification Testing).

TABLE 1.8-1 (Cont'd.)

REGULATORY GUIDE 1.78, REVISION 0 (JUNE 1974) - ASSUMPTIONS FOR EVALUATING THE HABITABILITY OF A NUCLEAR POWER PLANT CONTROL ROOM DURING A POSTULATED HAZARDOUS CHEMICAL RELEASE

FSAR Section 6.4.2.3

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

REGULATORY GUIDE 1.79, REVISION 1 (SEPTEMBER 1975) - PREOPERATIONAL TESTING OF EMERGENCY CORE COOLING SYSTEMS FOR PRESSURIZED WATER REACTORS

Position RG 1.79 applies to PWR plants and is not applicable to the Unit 2 project.

REGULATORY GUIDE 1.80 (JUNE 1974) - PREOPERATIONAL TESTING OF INSTRUMENT AIR SYSTEMS

FSAR Section 9.3.1

Position This regulatory guide was withdrawn by the NRC on April 20, 1982. It has been superseded by RG 1.68.3 Revision 0 (April 1982).

REGULATORY GUIDE 1.81, REVISION 1 (JANUARY 1975) - SHARED EMERGENCY AND SHUTDOWN ELECTRIC SYSTEMS FOR MULTI-UNIT NUCLEAR POWER PLANTS

Position RG 1.81 applies to multi-unit nuclear power plants and is not applicable to the Unit 2 project because Unit 2 does not share any emergency or shutdown electric systems.

REGULATORY GUIDE 1.82 (JUNE 1974) - SUMPS FOR EMERGENCY CORE COOLING AND CONTAINMENT SPRAY SYSTEMS

Position RG 1.82 applies to PWR plants and is not applicable to the Unit 2 project.

REGULATORY GUIDE 1.82, REVISION 2 (MAY 1996) - SHARED EMERGENCY AND SHUTDOWN ELECTRIC SYSTEMS FOR MULTI-UNIT NUCLEAR POWER PLANTS

Position RG 1.82 Revision 2 applies to the replacement of the ECCS suction strainers as described in USAR Sections 5.4.7.1.1, 6.1.2.2, 6.2.2.2, 6.2.2.3.2, 6.3.2.2, 6.3.2.2.1 and 6.3.2.2.3.

REGULATORY GUIDE 1.83, REVISION 1 (JULY 1975) - IN-SERVICE INSPECTION OF PWR STEAM GENERATOR TUBES

Position RG 1.83 applies to PWR plants and is not applicable to the Unit 2 project.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.84, Revision 22 (July 1984) - DESIGN AND FABRICATION CODE CASE ACCEPTABILITY - ASME SECTION III DIVISION I

FSAR Section 5.2.1.2

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approaches described below.

RG 1.84 provides a list of ASME Design and Fabrication Code cases that have been generically approved by the regulatory staff. Code cases on this list may, for design purposes, be used until appropriately annulled. Annulled cases are considered "active" for equipment that has been contractually committed to fabrication prior to the annulment.

The various ASME Code cases that were applied to components in the RCPB are listed in Table 5.2-1. All Safety Class 2 and 3 equipment has been designed to ASME Code or ASME-approved Code cases. This provision, together with the quality control programs, provides adequate safety equipment functional assurances.

TABLE 1.8-1 (Cont'd.)

REGULATORY GUIDE 1.130, REVISION 1 (OCTOBER 1978) - DESIGN LIMITS AND LOADING COMBINATIONS FOR CLASS 1 PLATE-AND-SHELL TYPE COMPONENT SUPPORTS

FSAR Section 3.9B.3

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. Paragraph C.2 Ultimate strength temperature correlation of this guide was used in regions adjacent to pipe having high temperatures.

2. Paragraph C.3 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.

REGULATORY GUIDE 1.131, REVISION 0 (AUGUST 1977) (FOR COMMENT) - QUALIFICATION TESTS OF ELECTRICAL CABLES, FIELD SPLICES, AND CONNECTIONS FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

FSAR Section 3.11

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

BOP

Complies with this regulatory guide.

NSSS

This regulatory guide is not applicable for the GE scope of supply because GE-supplied cabling does not experience severe environmental conditions (control room environment) and is qualified as part of the PGCC floor section module.

REGULATORY GUIDE 1.132, REVISION 1 (MARCH 1979)

FSAR Sections 2.5, 3.7A.2

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

This guide was originally issued in September 1977, 6 yr after the field work for the foundation investigation was performed in late 1971. The results were provided in a report* dated May 4, 1972. Although the investigation predates the guide, the work performed was well documented and adequate to support the Construction Permit Application and complies with the intent of the guide.

For any aspect of the Unit 2 site, investigation performed after March 30, 1979, conforms to this regulatory guide.

* Report: Foundation Investigation, Nine Mile Point Nuclear Station, Proposed Unit 2, Scriba, New York, Niagara Mohawk Power Corporation.

TABLE 1.8-1 (Cont'd.)

REGULATORY GUIDE 1.137, REVISION 1 (OCTOBER 1979) - FUEL-OIL SYSTEMS FOR STANDBY DIESEL GENERATORS

FSAR Section 9.5.4

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide except for the alternate approach described in response to Question F430.61, which is incorporated in the Unit 2 Technical Specifications.

REGULATORY GUIDE 1.138, REVISION 0 (APRIL 1978) (FOR COMMENT) - LABORATORY INVESTIGATIONS OF SOILS FOR ENGINEERING ANALYSIS AND DESIGN OF NUCLEAR POWER PLANTS

FSAR Sections 2.5, 3.7A.2

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

The results of laboratory work performed for the foundation investigation were submitted in a report* dated 6 yr prior to the issuance of the guide. The work performed was well documented and adequate to support the Construction Permit Application and complies with the intent of the guide.

Laboratory investigations performed after December 1, 1978, were in conformance with this regulatory guide.

* Report: Foundation Investigation, Nine Mile Point Nuclear Station, Proposed Unit 2, Scriba, New York, Niagara Mohawk Power Corporation.

REGULATORY GUIDE 1.139, REVISION 0 (MAY 1978) (FOR COMMENT) - GUIDANCE FOR RESIDUAL HEAT REMOVAL

FSAR Sections 5.4.7, 6.3

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

Paragraph C.1.b states: "In demonstrating that the system can perform its function assuming a single failure, limited operator action outside the control room would be acceptable if suitably justified." The common RHR shutdown cooling suction line valves are in two divisions (Division I - the outside valve; Division II - the inside valve) to satisfy containment isolation criteria. In the event that the RHR shutdown suction line is not available during shutdown because of a single-valve failure (loss of a division of emergency power), either valve can be opened manually with limited operator action or by establishing an alternate shutdown cooling path.

REGULATORY GUIDE 1.140, REVISION 1 (OCTOBER 1979) - DESIGN, TESTING, AND MAINTENANCE CRITERIA FOR NORMAL VENTILATION EXHAUST SYSTEM AIR FILTRATION AND ADSORPTION UNITS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

FSAR Section 9.4.3

Position Regulatory Guide 1.140 applies only to the radwaste building general area and equipment exhaust systems, each of which is designed to remove only particulate matter. Because charcoal adsorbers are not provided, the sections of the guide relating to adsorbers and iodine adsorption are not addressed. The air filtration units are nonsafety related; however, redundancy is provided for reliability and ease of maintenance.

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approaches described below:

TABLE 1.8-1 (Cont'd.)

REGULATORY GUIDE 1.142, REVISION 1 (OCTOBER 1981) (cont'd.)

The design of the Unit 2 primary containment is in accordance with RG 1.10, 1.15, 1.18, 1.19, 1.55, and 1.103 (as previously described in lieu of RG 1.142).

REGULATORY GUIDE 1.143, REVISION 1 (OCTOBER 1979) - DESIGN GUIDANCE FOR RADIOACTIVE WASTE MANAGEMENT SYSTEMS, STRUCTURES, AND COMPONENTS INSTALLED IN LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

FSAR Sections 15.7.1, 11.4

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

A. Liquid Waste System

The fiberglass tanks purchased for the LWS have been designed in accordance with the National Institute of Standards and Technology (NIST) Product Standard (PS) PS 15-69, Custom Contact-Molded Reinforced-Polyester Chemical-Resistant Process Equipment, as identified in NMP2 PSAR, Table C-10b.

NBS PS 15-69 provides the necessary design and fabrication requirements to ensure the integrity of the tanks without the additional cost of burst testing.

The RWCU phase separator tanks (2LWS-TK6A & 6B), which had been purchased as code-stamped ASME VIII vessels, had their code stamps removed because they were not rehydrotested after a nozzle was added to the top head of each of the two vessels in the field. The vessels still satisfy the intent of the requirements of this regulatory guide in that they are designed and fabricated to the requirements of ASME VIII (including the added nozzles) using materials which meet ASME VIII requirements, and the shop hydrotest established the integrity of the vessels before the nozzles were added. The nozzles were added near the top of the vessels' heads and the vessels see only atmospheric operating conditions (although designed for a nominal 15-psi design pressure). The new nozzles were added to allow improved operation of the vessels' level transmitters and are identical to the original level transmitter nozzles which were blind flanged and abandoned. Therefore, the added nozzles do not affect the proven integrity of the vessels in this application.

The Thermex unit (2LWS-FLT102), which is leased equipment, uses CPVC and PVC components. The Thermex unit is connected to plant components using reinforced, noncollapsible hoses. The equipment vendor has evaluated these components as being suitable for processing low-level radioactive waste. All thermoplastic materials (CPVC and PVC components) comply with ASTM material and dimensional standards. Assembled liquid piping or hoses are hydrostatically tested at 150 percent of the maximum design operating pressure for the limiting assembly. Chemical transfer hoses (nonmetallic) satisfy the requirements of ANSI B31.1-1992 Appendix III and are inspected/tested/replaced on a routine interval to ensure equipment reliability.

B. Offgas System

The charcoal adsorbers of the offgas system are not designed to the seismic requirements of this regulatory guide.

Offsite dose calculations in accordance with Chapter 15.7.1 of the NMP2 FSAR show that release of gaseous activity due to failure of the charcoal adsorbers results in offsite doses less than 0.5 Rem to the whole body. In accordance with RG 1.29, this permits classification as nonseismic. At the time of design and procurement of the offgas system (July 1974), RG 1.29 Revision 1 established the seismic requirements for the radioactive waste processing systems.

TABLE 1.8-1 (Cont'd.)

REGULATORY GUIDE 1.143, REVISION 1 (OCTOBER 1979) (cont'd.)

C. Waste Solidification System

The waste solidification system complies with the requirements of NRC Branch Technical Position ETSB11.1 Revision 1 as outlined in Werner and Pfleiderer Corporation (WPC) Topical Report No. WPC-VRS-001 Revision 1, dated May 1978, with exceptions as discussed in Section 11.4.3. The waste sludge tank is designed, fabricated, examined, and tested (hydrotest at 1.5 times design) in accordance

TABLE 1.8-1 (Cont'd.)

REGULATORY GUIDE 1.147, LATEST APPROVED REVISION - IN-SERVICE INSPECTION CODE CASE ACCEPTABILITY - ASME SECTION XI DIVISION I

FSAR Section 14

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

In accordance with 10CFR50.55a(g)(2), the NMP2 First Ten-Year Interval Inservice Inspection Program was based on the 1983 Edition of ASME Section XI, Summer 1983 Addenda. Code cases included in these programs are identified in the Inservice Inspection Program Plan (First Ten Year). Subsequent Ten-Year Interval ISI, ISPT, and IST programs will be based on the requirements and Code Edition set forth in 10CFR50.55a.

REGULATORY GUIDE 1.148 (MARCH 1981) - FUNCTIONAL SPECIFICATION FOR ACTIVE VALVE ASSEMBLIES IN SYSTEMS IMPORTANT TO SAFETY IN NUCLEAR POWER PLANTS

FSAR Section 5.4

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

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 1.8-2

CONFORMANCE TO DIVISION 8 NRC REGULATORY GUIDE

REGULATORY GUIDE 8.1, REVISION 0 (FEBRUARY 1973) - RADIATION SYMBOL FSAR Section None Position The Unit 2 project complies with Regulatory Position (Paragraph C) of this guide.
REGULATORY GUIDE 8.2, REVISION 0 (FEBRUARY 1973) - GUIDE FOR ADMINISTRATIVE PRACTICES IN RADIATION MONITORING FSAR Section None Position See Section 12.5.3 for an assessment of this Regulatory Guide.
REGULATORY GUIDE 8.3, REVISION 0 (FEBRUARY 1973) - FILM BADGE PERFORMANCE CRITERIA - IF USED FSAR Section None Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.
REGULATORY GUIDE 8.4, REVISION 0 (FEBRUARY 1973) - DIRECT-READING AND INDIRECT-READING POCKET DOSIMETERS FSAR Section None Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.
REGULATORY GUIDE 8.5, REVISION 1 (MARCH 1981) - CRITICALITY AND OTHER INTERIOR EVACUATION SIGNALS FSAR Section None Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.
REGULATORY GUIDE 8.6, REVISION 0 (MAY 1983) - STANDARD TEST PROCEDURES FOR GEIGER-MUELLER COUNTERS FSAR Section None Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.
REGULATORY GUIDE 8.7, REVISION 0 (MAY 1973) - OCCUPATIONAL RADIATION EXPOSURE RECORDS SYSTEMS FSAR Section None Position See Section 12.5.3 and Exhibit 12.1-2 for an assessment of this Regulatory Guide.

TABLE 1.9-1

STANDARD REVIEW PLAN CONFORMANCE TO ACCEPTANCE CRITERIA

SRP Number	Title	Revision	Conformance	Difference
CHAPTER 1: INTRODUCTION AND GENERAL DESCRIPTION OF PLANT				
1.8	Interfaces for Standard Design	1	NA	NA
CHAPTER 2: SITE CHARACTERISTICS				
2.1.1	Site Location and Description	2	X	
2.1.2	Exclusion Area Authority and Control	2	X	
2.1.3	Population Distribution	2	X	
2.2.1-2.2.2	Identification of Potential Hazards in Site Vicinity	2	X	
2.2.3	Evaluation of Potential Accidents	2	X	
2.3.1	Regional Climatology	2	X	
2.3.2	Local Meteorology	2		Note 1
2.3.3	Onsite Meteorological Measurements Programs	2		Note 2
2.3.4	Short-Term Diffusion Estimates for Accidental Atmospheric Releases	1	X	
2.3.5	Long-Term Diffusion Estimates	2	X	
2.4.1	Hydrologic Description	2	X	
	Appendix A	2	X	
2.4.2	Floods	2	X	
2.4.3	Probable Maximum Flood (PMF) on Streams and Rivers	2	NA	NA
2.4.4	Potential Dam Failures	2	NA	NA
2.4.5	Probable Maximum Surge and Seiche Flooding	2	X	
2.4.6	Probable Maximum Tsunami Flooding	2	NA	NA
2.4.7	Ice Effects	2	X	
2.4.8	Cooling Water Canals and Reservoirs	2	NA	NA
2.4.9	Channel Diversions	2	NA	NA
2.4.10	Flood Protection Requirements	2	X	
2.4.11	Cooling Water Supply	2	X	
2.4.12	Groundwater	2		Note 3
	BTP HMB/GSB 1	1	NA	NA
	BTP HGEB 1	2	NA	NA
2.4.13	Accidental Releases of Liquid Effluents in Ground and Surface Waters	2		Note 4
2.4.14	Technical Specifications and Emergency Operation Requirements	2	NA	NA
2.5.1	Basic Geologic and Seismic Information	2	X	
2.5.2	Vibratory Ground Motion	1	X	
2.5.3	Surface Faulting	2	X	
2.5.4	Stability of Subsurface Materials and Foundations	2		Note 5
2.5.5	Stability of Slopes	2		Note 6

TABLE 1.9-1 (Cont'd.)

SRP Number	Title	Revision	Conformance	Difference
CHAPTER 3: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS				
3.2.1	Seismic Classification	1		Note 7
3.2.2	System Quality Group Classification	1		Note 8
3.3.1	Wind Loadings	2		Note 9
3.3.2	Tornado Loadings	2	X	
3.4.1	Flood Protection	2	X	
3.4.2	Analysis Procedures	2	X	
3.5.1.1	Internally-Generated Missiles (Outside Containment)	2	X	
3.5.1.2	Internally-Generated Missiles (Inside Containment)	2	X	
3.5.1.3	Turbine Missiles	2		Note 10
3.5.1.4	Missiles Generated by Natural Phenomena	2	X	
	BTP AAB 3-2	1	NA	NA
	BTP ASB 3-2	2	NA	NA
3.5.1.5	Site Proximity Missiles (Except Aircraft)	1	X	
3.5.1.6	Aircraft Hazards	1	X	
3.5.2	Structures, Systems, and Components to be Protected from Externally-Generated Missiles	2	X	
3.5.3	Barrier Design Procedures	1	X	Note 11
3.6.1	Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	1	X	
	BTP ASB 3-1	1		Note 12
3.6.2A	Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	1		Note 13
	BTP MEB 3-1	1		Note 13
3.6.2B	Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	1		Note 14
	BTP MEB 3-1	1		Note 14
3.7.1A	Seismic Design Parameters	1	X	
3.7.1B	Seismic Design Parameters	1	X	
3.7.2A	Seismic System Analysis	1		Note 15
3.7.2B	Seismic System Analysis	1	X	
3.7.3A	Seismic Subsystem Analysis	1	X	
3.7.3B	Seismic Subsystem Analysis	1		Note 16
3.7.4	Seismic Instrumentation	1		Note 17
3.8.1	Concrete Containment			Note 18
3.8.2	Steel Containment	1	NA	NA
3.8.3	Concrete and Steel Internal Structures of Steel or Concrete Containments	1		Note 19
3.8.4	Other Seismic Category I Structures	1		Note 20
3.8.5	Foundations	1		Note 21
3.9.1A	Special Topics for Mechanical Components	2	X	
3.9.1B	Special Topics for Mechanical Components	2	X	
3.9.2A	Dynamic Testing and Analysis of Systems, Components, and Equipment	2		Note 22
3.9.2B	Dynamic Testing and Analysis of Systems, Components, and Equipment	2		Note 23
3.9.3A	ASME Code Class 1, 2 and 3 Components, Component Supports, and Core Support Structures	1	X	
3.9.3B	ASME Code Class 1, 2 and 3 Components, Component Supports, and Core Support Structures	1		Note 24
3.9.4B	Control Rod Drive Systems	1	X	

TABLE 1.9-1 (Cont'd.)

SRP Number	Title	Revision	Conformance	Difference
3.9.5B	Reactor Pressure Vessel Internals	2		Note 25
3.9.6	In-service Testing of Pumps and Valves	2		Note 26
3.10A	Seismic Qualification of Category I Instrumentation and Electrical Equipment	2		Note 27
3.10B	Seismic Qualification of Category I Instrumentation and Electrical Equipment	2	X	
3.11	Environmental Design of Mechanical and Electrical Equipment	2		Note 28
CHAPTER 4: REACTOR				
4.2	Fuel System Design	2		Note 29
4.3	Nuclear Design	2	X	
	BTP CPB 4.3-1	2	NA	NA
4.4	Thermal and Hydraulic Design	1		Note 30
	Appendix	1	NA	NA
4.5.1	Control Rod Drive Structural Materials	2		Note 31
4.5.2	Reactor Internals and Core Support Materials	2		Note 32
4.6	Functional Design of Control Rod Drive System	1	X	
CHAPTER 5: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS				
5.2.1.1	Compliance with the Codes and Standard Rule 10CFR50.55a	2		Note 33
5.2.1.2	Applicable Codes Cases	2	X	
5.2.2	Overpressurization Protection	1		Note 34
	BTP RSB 5-2	0	NA	NA
5.2.3	Reactor Coolant Pressure Boundary Materials	2		Note 35
	BTP MTEB 5-7	2	NA	NA
5.2.4	Reactor Coolant Pressure Boundary In-service Inspection and Testing	1	X	
5.2.5	Reactor Coolant Pressure Boundary Leakage Detection	1	X	
5.3.1	Reactor Vessel Materials	1		Note 36
5.3.2	Pressure-Temperature Limits	1	X	
	BTP MTEB 5-2	1	X	
5.3.3	Reactor Vessel Integrity	1	X	
5.4	Preface	1	NA	NA
5.4.1.1	Pump Flywheel Integrity (PWR)	1	NA	NA
5.4.2.1	Steam Generator Materials	2	NA	NA
	BTP MTEB 5-3	2	NA	NA
5.4.2.2	Steam Generator Tube In-service Inspection	1	NA	NA
5.4.6	Reactor Core Isolation Cooling System (BWR)	2		Note 37
5.4.7	Residual Heat Removal (RHR) System	2	X	
	BTP RSB 5-1	2	X	
5.4.8	Reactor Water Cleanup System (BWR)	2		Note 38
5.4.11	Pressurizer Relief Tank	2	NA	NA
5.4.12	Reactor Coolant System High-Point Vents	0	X	

TABLE 1.9-1 (Cont'd.)

SRP Number	Title	Revision	Conformance	Difference
CHAPTER 6: ENGINEERED SAFETY FEATURES				
6.1.1	Engineered Safety Features Materials	2	X	
	BTP MTEB 6-1	2	NA	NA
6.1.2	Protective Coating Systems (Paints) - Organic Materials	2		Note 39
6.2.1	Containment Functional Design	2	NA	NA
6.2.1.1A	PWR Dry Containments, Including Subatmospheric Containments	2	NA	NA
6.2.1.1B	Ice Condenser Containments	2	NA	NA
6.2.1.1C	Pressure Suppression Type BWR Containments	4		Note 40
	Appendix I	1		Note 40
6.2.1.2	Subcompartment Analysis	2		Note 41
6.2.1.3	Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents	1		Note 42
6.2.1.4	Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	1	NA	NA
6.2.1.5	Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies	2	NA	NA
6.2.2	Containment Heat Removal Systems	3		Note 43
6.2.3	Secondary Containment Functional Design	2	X	
6.2.4	Containment Isolation System	2	X	
6.2.5	Combustible Gas Control in Containment	2	X	
	Appendix A	2	NA	NA
	BTP CSB 6-2	2	NA	NA
6.2.6	Containment Leakage Testing	2	X	
6.2.7	Fracture Prevention of Containment Pressure Boundary	0		Note 44
6.3	Emergency Core Cooling System			Note 45
	BTP RSB 6-1	1	NA	NA
6.4	Control Room Habitability System	2	X	
	Appendix A	2	X	
6.5.1	Engineered Safety Feature Atmosphere Cleanup System	2		Note 46
6.5.2	Containment Spray as a Fission Product Cleanup System	1	NA	NA
6.5.3	Fission Product Control Systems and Structures	2	X	
6.5.4	Ice Condenser as a Fission Product Cleanup System	2	NA	NA
6.6	In-service Inspection of Class 2 and 3 Components	1	X	
6.7	Main Steam Isolation Valve Leakage Control System (BWR)	2	NA	NA
CHAPTER 7: INSTRUMENTATION AND CONTROLS				
7.1	Instrumentation and Controls - Introduction	2	X	
	Table 7-1 - Acceptance Criteria and Guidelines for Instrumentation and Controls Systems Important to Safety	2	X	
7.2	Reactor Trip System	2	X	
	Appendix A	2	NA	NA
7.3	Engineered Safety Features System	2	X	
	Appendix A	2	NA	NA
7.4	Safe Shutdown Systems	2	X	

TABLE 1.9-1 (Cont'd.)

SRP Number	Title	Revision	Conformance	Difference
CHAPTER 7: INSTRUMENTATION AND CONTROLS (Cont'd.)				
7.5	Information Systems Important to Safety	2	X	
7.6	Interlock Systems Important to Safety	2	X	
7.7	Control Systems	2	X	
Appendix 7-A	Branch Technical Positions (ICSB)	2		
	BTP ICSB 1 (DOR)	2	NA	NA
	BTP ICSB 3	2	X	
	BTP ICSB 4 (PSB)	2	NA	NA
	BTP ICSB 5	2	NA	NA
	BTP ICSB 9	2	NA	NA
	BTP ICSB 12	2	NA	
	BTP ICSB 13	2	NA	NA
	BTP ICSB 14	2	NA	NA
	BTP ICSB 16	2	NA	NA
	BTP ICSB 19	2	NA	NA
	BTP ICSB 20	2	X	
	BTP ICSB 21	2	X	
	BTP ICSB 22	2	X	
	BTP ICSB 25	2	NA	NA
	BTP ICSB 26	2	X	
Appendix 7-B	General Agenda, Station Site Visits	1	NA	NA
CHAPTER 8: ELECTRIC POWER				
8.1	Electric Power - Introduction	2	NA	NA
	Table 8-1 - Acceptance Criteria and Guidelines for Electric Power Systems	2	X	
8.2	Offsite Power System	2	X	
8.3.1	AC Power Systems (Onsite)	2		Note 47
8.3.2	DC Power Systems (Onsite)	2	X	
Appendix 8-A	Branch Technical Positions (PSB)	2		
	BTP ICSB 2 (PSB)	2	NA	NA
	BTP ICSB 4 (PSB)	2	NA	NA
	BTP ICSB 8 (PSB)	2	X	
	BTP ICSB 11 (PSB)	2	X	
	BTP ICSB 15 (PSB)	2	NA	NA
	BTP ICSB 17 (PSB)	2	NA	NA
	BTP ICSB 18 (PSB)	2	X	
	BTP ICSB 21 (PSB)	2	X	
	BTP PSB 1	0	X	
	BTP PSB 2	0	X	
Appendix 8-B	General Agenda, Station Site Visits	0	NA	NA

TABLE 1.9-1 (Cont'd.)

SRP Number	Title	Revision	Conformance	Difference
CHAPTER 9: AUXILIARY SYSTEMS				
9.1.1	New Fuel Storage	2	X	
9.1.2	Spent Fuel Storage	3	X	
9.1.3	Spent Fuel Pool Cooling and Cleanup System	1		Note 48
9.1.4	Light Load Handling System (Related to Refueling)	2	X	
	BTP ASB 9-1	2	NA	NA
9.1.5	Overhead Heavy Load Handling Systems	0		Note 49
9.2.1	Station Service Water System	2	X	
9.2.2	Reactor Auxiliary Cooling Water Systems	1		Note 50
9.2.3	Demineralized Water Makeup Systems	2		Note 51
9.2.4	Potable and Sanitary Water System	2	X	
9.2.5	Ultimate Heat Sink	2	X	
	BTP ASB 9-2	2	X	
9.2.6	Condensate Storage Facilities	2	NA	NA
9.3.1	Compressed Air System	1	NA	NA
9.3.2	Process and Postaccident Sampling System	2		Note 52
9.3.3	Equipment and Floor Drainage System	2	X	
9.3.4	Chemical and Volume Control System (PWR) (Including Boron Recovery System)	2	NA	NA
9.3.5	Standby Liquid Control System (BWR)	2	X	
9.4.1	Control Room Area Ventilation System	2		Note 53
9.4.2	Spent Fuel Pool Area Ventilation System	2		Note 54
9.4.3	Auxiliary and Radwaste Area Ventilation System	2		Note 55
9.4.4	Turbine Area Ventilation System	2	X	
9.4.5	Engineered Safety Feature Ventilation System	2		Note 56
9.5.1	Fire Protection Program	3		Note 57
	BTP CMEB 9.5.1	2		Note 57
9.5.2	Communications System	2	X	
9.5.3	Lighting Systems	2	X	
9.5.4	Emergency Diesel Engine Fuel Oil Storage and Transfer System	2	X	
9.5.5	Emergency Diesel Engine Cooling Water System	2	X	
9.5.6	Emergency Diesel Engine Starting System	2		
9.5.7	Emergency Diesel Engine Lubrication System	2	X	
9.5.8	Emergency Diesel Engine Combustion Air Intake and Exhaust System	2	X	
CHAPTER 10: STEAM AND POWER CONVERSION SYSTEM				
10.2	Turbine Generator	2	X	
10.2.3	Turbine Disk Integrity	1	X	
10.3	Main Steam Supply System	2		Note 58
10.3.6	Steam and Feedwater System Materials	2	X	
10.4.1	Main Condensers	2	X	
10.4.2	Main Condenser Evacuation System	2		Note 59
10.4.3	Turbine Gland Sealing System	2		Note 60
10.4.4	Turbine Bypass System	2	X	

TABLE 1.9-1 (Cont'd.)

SRP Number	Title	Revision	Conformance	Difference
CHAPTER 10: STEAM AND POWER CONVERSION SYSTEM (Cont'd.)				
10.4.5	Circulating Water System	2	X	
10.4.6	Condensate Cleanup System	2	X	
10.4.7	Condensate and Feedwater System	2	X	
	BTP ASB 10-2	2	NA	NA
10.4.8	Steam Generator Blowdown System (PWR)	2	NA	NA
10.4.9	Auxiliary Feedwater System (PWR)	2	NA	NA
	BTP ASB 10-1	2	NA	NA
CHAPTER 11: RADIOACTIVE WASTE MANAGEMENT				
11.1	Source Terms	2	X	
11.2	Liquid Waste Management Systems	2		Note 61
11.3	Gaseous Waste Management Systems	2		Note 62
	BTP ETSB 11-5	0		Note 63
11.4	Solid Waste Management Systems	2	X	
	BTP ETSB 11-3	2	X	
	Appendix 11.4-A	0	NA	NA
11.5	Process and Effluent Radiological Monitoring and Sampling Systems	3	X	Note 79
	Appendix 11.5-A	1	X	
CHAPTER 12: RADIATION PROTECTION				
12.1	Assuring that Occupational Radiation Exposures are as Low as is Reasonably Achievable	2	X	
12.2	Radiation Sources	2		Note 64
12.3-12.4	Radiation Protection Design Features	2		Note 65
12.5	Operational Radiation Protection Program	2		Note 66
CHAPTER 13: CONDUCT OF OPERATIONS				
13.1.1	Management and Technical Support Organization	2	X	
13.1.2-13.1.3	Operating Organization	2	X	
13.2.1	Reactor Operating Training	0	X	Note 78
13.2.2	Training for Non-Licensed Plant Staff	0	X	
13.3	Emergency Planning	2	X	
13.4	Operational Review	2		Note 67
13.5.1	Administration Procedures	0	X	
13.5.2	Operating and Maintenance Procedures	0	X	
13.6	Physical Security	2	X	

TABLE 1.9-1 (Cont'd.)

SRP Number	Title	Revision	Conformance	Difference
CHAPTER 14: INITIAL TEST PROGRAM				
14.1	Initial Plant Test Programs - PSAR	2	NA	NA
14.2	Initial Plant Test Programs - FSAR	2		Note 68
14.3	Standard Plant Designs Initial Test Program Final Design Approval (FDA)	1	NA	NA
CHAPTER 15: ACCIDENT ANALYSIS				
15.0	Introduction	2	NA	NA
15.1	Increase in Heat Removal by the Secondary System			
15.1.1-15.1.4	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	1	X	
15.1.5	Steam System Piping Failures Inside and Outside of Containment (PWR)	2	NA	NA
	Appendix A	2	NA	NA
15.2	Decrease in Heat Removal by the Secondary System			
15.2.1-15.2.5	Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve (BWR), and Steam Pressure Regulator Failure (Closed)	1	X	
15.2.6	Loss of Non-Emergency AC Power to the Station Auxiliaries	1	X	
15.2.7	Loss of Normal Feedwater Flow	1	X	
15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment (PWR)	1	NA	NA
15.3	Decrease in Reactor Coolant System Flow Rate			
15.3.1-15.3.2	Loss of Forced Reactor Coolant Flow Including Trip of Pump and Flow Controller Malfunctions	1	X	
15.3.3-15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	2		Note 69
15.4	Reactivity and Power Distribution Anomalies			
15.4.1	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition	2	X	
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power	2	X	
15.4.3	Control Rod Misoperation (System Malfunction or Operator Error)	2	NA	NA
15.4.4-15.4.5	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	1	X	
15.4.6	Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant (PWR)	1	NA	NA
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	1		Note 70

TABLE 1.9-1 (Cont'd.)

SRP Number	Title	Revision	Conformance	Difference
CHAPTER 15: ACCIDENT ANALYSIS (Cont'd.)				
15.4.8	Spectrum of Rod Ejection Accidents (PWR)	2	NA	NA
	Appendix A	1	NA	NA
15.4.9	Spectrum of Rod Drop Accidents (BWR)	2	X	
	Appendix A	2	X	
15.5	Increase in Reactor Coolant Inventory			
15.5.1-15.5.2	Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	1	X	
15.6	Decrease in Reactor Coolant Inventory			
15.6.1	Inadvertent Opening of a PWR Pressurizer Relief Valve or a BWR Relief Valve	1	X	
15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	2	X	
15.6.3	Radiological Consequences of Steam Generator Tube Failure (PWR)	2	NA	NA
15.6.4	Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)	2		Note 71
15.6.5	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	2		Note 72
15.7	Radioactive Release from a Subsystem or Component			
15.7.1	Waste Gas System Failure	1	NA	NA
15.7.2	Radioactive Liquid Waste System Leak or Failure (Release to Atmosphere)	1	NA	NA
15.7.3	Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	2	X	
15.7.4	Radiological Consequences of Fuel Handling Accidents	1	X	
15.7.5	Spent Fuel Cask Drop Accidents	2	X	
15.8	Anticipated Transients Without Scram			
15.8	Anticipated Transients Without Scram	1		Note 73
CHAPTER 16: TECHNICAL SPECIFICATIONS				
16.0	Technical Specifications	1		Note 74
CHAPTER 17: QUALITY ASSURANCE				
17.1	Quality Assurance During the Design and Construction Phases	2		Note 75
17.2	Quality Assurance During the Operations Phase	2		Note 76
CHAPTER 18: HUMAN FACTORS ENGINEERING				
18.0	Human Factors Engineering/Standard Review Plan Development	0		Note 77

TABLE 1.9-1 (Cont'd.)

KEY: NA = Not Applicable

(1) SRP section has been combined with SRP Section 12.3.
(2) SRP section has been combined with SRP Section 13.1.2.

TABLE 1.9-1 (Cont'd.)

NOTES

1. STANDARD REVIEW PLAN 2.3.2, REVISION 2, JULY 1981 - LOCAL METEOROLOGY

Difference 1 No topographic description is provided in FSAR Section 2.3.2. Effects of terrain modification and plant structures are also not discussed.

Discussion A description of the topography in the site region is provided in Section 2.5.1. The effects of terrain modification and plant structures on local meteorology are not significant and, therefore, are not expected to have any impact on plant operation. They are discussed in Section 2.3.5.

Difference 2 The wind sensors (Bendix Aerovanes) at the 61-m (200-ft), 30-m (100-ft), and 9-m (30-ft) levels of the onsite meteorological tower did not meet the accuracy and starting speed recommended in RG 1.23 for data collected until July 1982.

Discussion The severe weather conditions encountered on the shoreline of Lake Ontario required the choice of very rugged wind speed equipment prior to the tower's installation in late 1973. The Bendix Aerovane was chosen for its proven ability to withstand the climate of the region as opposed to measuring the infrequent calm hours.

The Bendix Aerovane has a starting speed of about 1.2 m/sec (2.6 mph) and continues to operate with speeds of 0.4 to 0.7 m/sec (1 to 1.5 mph). The wind speed accuracy is ±0.4 m/sec (±1.0 mph) above 4.5 m/sec (10 mph) as opposed to the RG 1.23 criterion of ±0.2 m/sec (±0.5 mph) for all wind speeds. More sensitive wind speed sensors available at that time were prone to icing and physical damage from high wind speeds.

Subsequent to July 1982, new instruments were installed to comply with the regulatory guide.

2. STANDARD REVIEW PLAN 2.3.3, REVISION 2, JULY 1981 - ONSITE METEOROLOGICAL MEASUREMENTS PROGRAMS

Difference The wind sensors (Bendix Aerovanes) at the 61-m (200-ft), 30-m (100-ft), and 9-m (30-ft) levels of the onsite meteorological tower did not meet the accuracy and starting speed recommended in RG 1.23 for data collected before July 1982.

TABLE 1.9-1 (Cont'd.)

Discussion See the discussion of Difference 2 for SRP 2.3.2.

3. STANDARD REVIEW PLAN 2.4.12, REVISION 2, JULY 1981 - GROUNDWATER

Difference Section 2.4.12 does not address groundwater.

Discussion The material required by SRP 2.4.12, Groundwater, can be found in FSAR Section 2.4.13. Except for the number change there are no differences noted.

4. STANDARD REVIEW PLAN 2.4.13 - ACCIDENTAL RELEASES OF LIQUID EFFLUENT IN GROUND AND SURFACE WATERS

Difference No chapter/section exists with the required SRP 2.4.13 title.

Discussion Section 2.4.13.3 addresses accidental effects and dilution modeling as required by this SRP acceptance criteria.

5. STANDARD REVIEW PLAN 2.5.4, REVISION 2, JULY 1981 - STABILITY OF SUBSURFACE MATERIALS AND FOUNDATIONS

Difference The acceptance criteria for Section 2.5.4.8, Liquefaction Potential, states that liquefaction potential assessments using both deterministic and probabilistic approaches are desirable. However, only the deterministic method was used.

Discussion All major Category I structures for Unit 2 are founded on sound bedrock.

The deterministic approach discussed in Section 2.5.4.8 for the analysis of liquefaction potential under a few minor structures founded in soil backfill is considered adequate and does not require a supplementary probabilistic analysis.

6. STANDARD REVIEW PLAN 2.5.5, REVISION 2, JULY 1981 - STABILITY OF SLOPES

Difference Only the deterministic method was used in the design and analysis of slopes.

Discussion The revised SRP promotes both deterministic and probabilistic approaches to slope design analysis, indicating that the latter method is desirable rather than

TABLE 1.9-1 (Cont'd.)

mandatory, and that it may be employed by the NRC staff itself. To analyze and design the manmade slopes, which are discussed in FSAR Section 2.5.2.2, only the deterministic approach was utilized. It is considered adequate and does not require a supplementary probabilistic analysis.

7. STANDARD REVIEW PLAN 3.2.1, REVISION 1 - SEISMIC CLASSIFICATION

All differences between seismic classifications in RG 1.29, Revision 3, and the Unit 2 design are indicated in FSAR Table 3.2-1. A discussion of the differences is also included in the table.

8. STANDARD REVIEW PLAN 3.2.2, REVISION 1 - SYSTEM QUALITY GROUP CLASSIFICATION

All differences between System Quality Group Classification in RG 1.26, Revision 3, and the Unit 2 design are indicated in FSAR Table 3.2-1. A discussion of the differences is also included in the table.

9. STANDARD REVIEW PLAN 3.3.1, REVISION 2 - WIND LOADINGS

Difference The acceptance criteria for Revision 1 of SRP 3.3.1 (NUREG-75/087) are allowed using either ASCE Paper No. 3269 or ANSI A58.1-1972 as the basis for wind design. The acceptance criteria for Revision 2 of SRP 3.3.1 (NUREG-0800) considers ANSI A58.1 as the base document while permitting use of ASCE Paper No. 3269 only for cases which ANSI A58.1 does not cover.

Unit 2 is designed using ASCE Paper No. 3269 consistent with the PSAR commitment and with the state of the art available at the time of plant design.

Discussion A review of ANSI A58.1, for derivation of wind pressure for a typical structure, or parts and portions of a structure, indicates that the values thus derived are essentially identical to those derived using ASCE Paper No. 3269.

10. STANDARD REVIEW PLAN 3.5.1.3, REVISION 2 - TURBINE MISSILES

Difference For Unit 2 the following was used in lieu of paragraph II.1:

TABLE 1.9-1 (Cont'd.)

The protection of essential systems located within the low-trajectory missile strike zone is acceptable if the probability of damage summed over all structural cubicles containing such systems is less than 10^{-7} per annum.

Discussion The purpose for using an annual damage probability of 10^{-7} and an annual turbine failure rate of 10^{-4}, rather than directly using the resulting factor of 10^{-3}, is to permit compliance with Section III.2 of SRP 3.5.1.3, Revision 2. This allows the turbine failure rate of 10^{-4} per annum to be subdivided as follows:

$P = 6 \times 10^{-5}$ per turbine year for design speed failures
$P = 4 \times 10^{-5}$ per turbine year for destructive overspeed failures

The reason for evaluating acceptability by summing probabilities over cubicles containing essential systems, rather than by summing over the essential systems themselves, is to simplify the analysis. It is impractical to evaluate the strike probability for each system, considering the complex routing and the possibility for minor layout changes during plant design. By performing the evaluation on the basis of cubicles containing essential systems, this difficulty is avoided while still ensuring that all essential systems are considered in the evaluation.

11. STANDARD REVIEW PLAN 3.5.3, REVISION 1 - BARRIER DESIGN PROCEDURES

Difference The tornado missile spectrum in Table 2 of this SRP is not used.

Discussion Unit 2 is designed to withstand the tornado-generated missiles of Spectrum A of SRP 3.5.1.4, Revision 2.

See Section 3.5.3 for further discussion.

12. CONFORMANCE TO BTP ASB 3-1, REVISION 1, ATTACHED TO STANDARD REVIEW PLAN 3.6.1 - PROTECTION AGAINST POSTULATED PIPING FAILURES IN FLUID SYSTEMS OUTSIDE CONTAINMENT

Difference 1 Section B.1.a.1 states that "even though portions of the main steam and feedwater lines meet the break exclusion requirements of B.1.6 of BTP MEB 3-1, they

TABLE 1.9-1 (Cont'd.)

should be separated from essential equipment. In order for essential equipment to be properly separated, the essential equipment must be protected from the jet impingement and the environmental effects of an assumed longitudinal break of the main steam line and feedwater lines. Each assumed longitudinal break should have a cross-sectional area of at least one square foot and should be postulated to occur at a location that has the greatest effect on essential equipment."

FSAR Section 3.6A.2.1.5 states that "regardless of the fact that all conditions [for break exclusion piping] have been met, a crack is postulated in the main steam or feedwater piping in the main steam tunnel. The crack in the pipe, equal in area to a single-ended pipe rupture, is considered a singular event. Pipe whip and jet impingement are not considered, and a single active failure is not taken as a concurrent event."

Discussion As a result of the issuance of SRPs 3.6.1 and 3.6.2 in 1975, the Unit 2 pipe rupture criteria were revised in a letter to the NRC on July 31, 1978, (Section 3.6, Reference 1) to show compliance to the latest requirements. Subsequently, it was recognized that there were additional concerns in the main steam tunnel. The Unit 2 plant was modified to incorporate the requirements outlined in a letter to the ASLAB from the NRC, dated October 4, 1978, concerning Carolina Power and Light's Shearon Harris plant. The NRC position was as follows:

STEAM TUNNEL DESIGN FOR NUCLEAR PLANTS

We have revised our requirements for the design of nuclear power plants relating to postulated high energy line breaks outside the containment. Specifically the revision will require that the compartment between the containment and the turbine building, which houses the main steam lines and feedwater lines and the isolation valves for these lines, be designed to consider the pressure and environmental effects from an assumed break, equivalent to the flow area of a single-ended pipe rupture in these lines. This revision will require that if this assumed break could cause the collapse of this compartment, then the collapse should not jeopardize the safe shutdown of the plant. Furthermore, it will require that essential equipment located within the compartment, or adjacent to the compartment, be designed to withstand the environmental effects resulting from the above break.

TABLE 1.9-1 (Cont'd.)

The results of a postulated pipe break in a high energy line (one in which either the pressure exceeds 275 psig or the temperature exceeds 200°F) are pipe whip, jet impingement, and the environmental effects of pressure, temperature, humidity and flooding. With the exception of certain break exclusion regions, pipe breaks are postulated at terminal ends and other points of relatively high stress and fatigue. Within the break exclusion region, which is limited to the containment penetration area, a combination of low stress and fatigue design coupled with augmented in-service inspection is used to assure that no pipe breaks will occur due to the design loads. However, the stress is not normally low enough to reasonably preclude the possibility of a postulated pipe crack in the region. As such, it is prudent to require that the surrounding pipe tunnel be designed to withstand the effects of a postulated pipe crack. These effects are of an extreme environmental nature for equipment in the vicinity and include pressure, temperature, humidity and flooding. Because of the augmented in-service inspection and the low stress and fatigue design, it is reasonable to assume that a postulated pipe crack will be detected and repaired before it becomes through wall or, at the latest, in its initial phase of leakage. The flow area for postulated pipe cracks is conservatively selected as the cross-sectional area of the pipe, however, to ensure that the largest possible crack is enveloped by the design.

Implementation of this position will ensure that essential equipment will not be arbitrarily housed in the main steam tunnel such that safety by separation is maintained.

Difference 2 Section B.1.c states that "a program should be developed to ensure that the system stresses due to long-term changes in the system and its supports and restraints, such as due to pipe relaxation and differential settling, will not be adversely affected by the restraints. Details of the methods used to obtain these assurances should be submitted to the staff for review."

Discussion Clearances at pipe whip restraints were extensively reviewed by recording piping displacements at selected whip restraints during startup testing. A comparison between predicted and measured pipe movements was made. It was concluded that piping systems would not

TABLE 1.9-1 (Cont'd.)

experience any additional stress due to these long-term changes provided that:

1. the existing clearances are maintained, or

2. if the maximum pipe movements as predicted by reanalysis are changed, the new displacements at the restraint location are reviewed on a case-by-case basis to ensure that the intent of BTP MEB 3-1 is met.

Difference 3 Section B.2.d states that "piping classification as required by Regulatory Guide 1.26 should be maintained without change until beyond the outboard restraint. If the restraint is located at the isolation valve, a classification change at the valve interface is acceptable." For Unit 2, the piping classification change is made at the valve (not beyond the outboard restraint) in accordance with RG 1.26.

Discussion Although the classification change is made at the valve, the piping between the valve and the first restraint outside containment is B31.1 (for main steam and feedwater) and ASME Section III, Class 2 and 3 (for RCIC and RWCU, respectively). Additionally, the piping meets the stringent break exclusion requirements in Item B.1.b of BTP MEB 3-1. It is therefore concluded that this will not degrade the safety of the plant.

13. STANDARD REVIEW PLAN 3.6.2A, REVISION 1, JULY 1981, BRANCH TECHNICAL POSITION MEB 3-1 - DETERMINATION OF RUPTURE LOCATIONS AND DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

Difference 1 Section B.1.e states that "with the exceptions of those portions of piping identified in B.1.b, leakage cracks should be postulated in ASME Code Section III, Class 1, piping where the stress range by Eq. (10) of Paragraph NB-3653 exceeds 1.20 and in Class 2 or 3 or nonsafety class piping where the stress by the sum of Eq. (9) and (10) of Paragraph NC/ND 3652 exceeds 0.4. Nonsafety piping which has not been evaluated to obtain similar stress information shall have cracks postulated at locations that result in the most severe environmental consequence." For high-energy piping in areas other than the containment penetration, Unit 2 postulates breaks in accordance with Sections B.1.a and B.1.c.

TABLE 1.9-1 (Cont'd.)

Discussion By evaluating the effects of jet impingement, pipe whip, environment, etc., for high-energy piping systems in accordance with Sections B.1.a and B.1.c, any event that could adversely affect the safety of the plant will be considered. Generally this is due to the following:

1. The criteria in Section B.1.a are invoked whenever possible to separate essential equipment from high-energy systems. In this case, breaks are arbitrarily postulated and a stress criterion is meaningless.

2. When it is not possible to separate high-energy piping from essential equipment, redundancy is provided or an evaluation is performed to ensure that the equipment will remain operable.

3. In areas in which high-energy pipe is routed, a sufficient number of breaks will always be postulated such that the effects of jet impingement, pipe whip, environment, etc., which result will envelop any intermediate or additional cracks.

The following discussion shows that for all areas of the plant, an additional criterion to postulate cracks is only repetitive and will not improve the safety of the plant.

High-energy piping not in the reactor building
High-energy pipe is not routed near systems, components, or structures essential to safe shutdown in areas other than the reactor building. For example, there is a significant amount of high-energy piping located in the turbine building; however, there is no essential equipment located there which could compromise the safety of the plant. This piping actually meets the criteria of Section B.1.a where breaks are arbitrarily postulated to ensure separation of high-energy piping and essential equipment. It is therefore concluded that this will not degrade the safety of the plant.

High-energy piping in the reactor building (excluding primary containment and containment penetration areas)
Excluding the main steam tunnel piping, the only systems which qualify as high-energy piping in secondary containment are the RWCU, SLC, CRD, and RCIC systems. Routing of these systems has been controlled so that they are located in well-defined areas (i.e., RCIC pipe chase and turbine room, RWCU pump room, valve room,

TABLE 1.9-1 (Cont'd.)

demineralizer room, heat exchanger room, and pipe chases). The walls of these compartments have been designed for jet impingement loads using a worst-case condition applied at any location. The compartments have also been evaluated for environmental, flood, pressure, etc., effects using the worst-case condition. Breaks in these areas are often arbitrarily postulated, and imposing additional criteria will not really enhance safety. It is therefore concluded that this will not degrade the safety of the plant.

Furthermore, the environmental effects of cracks in high-energy piping are enveloped by the effects of postulated cracks in moderate-energy systems. As discussed in Appendix 3C, all safety-related equipment in the reactor building is reviewed to ensure either operability or functional redundancy, where required, under environmental effects of spray from a crack in the RHR system piping. The resultant environment is equal to or is more severe than the effects of high-energy system piping cracks for both spray and flooding considerations. Therefore, a separate evaluation of high-energy crack environments is not required.

In the main steam tunnel, the effects of jet impingement will govern all cases assuming a minimum break criterion; therefore, this will not degrade the safety of the plant.

Primary containment If the primary containment piping were designed so that pipe stress results indicated that all high-energy systems required a minimum number of breaks to be postulated, approximately 100 breaks would be considered. In light of the separation between the high-energy systems in primary containment, it is reasonable to assume that these high-energy breaks will always govern. Any equipment, systems, or structures must be designed for the extreme environment in primary containment regardless of its particular location. Electrical equipment is routed in conduit or suitable housing so that it is not exposed to the open environment. The combination of separation and redundancy (the preferred method of protection) is also integral to components and piping routed in the primary containment. This is verified in the jet impingement evaluation where breaks are postulated at various elevations and azimuths. Additional investigation is only repetitive. It is therefore concluded that this will not degrade the safety of the plant.

TABLE 1.9-1 (Cont'd.)

Difference 2 Section B.1.c.1.d states that "if intermediate break locations cannot be determined by (b), (B.1.c.1.b) and (c), (B.1.c.1.c) above, two highest stress locations based on equation (10) should be selected." Unit 2 has eliminated arbitrary intermediate breaks (AIBs) in accordance with the intent of GL 87-11.

Discussion Per review of GL 87-11, dated June 19, 1987, and NUREG-1061 Volumes 3 and 5, it is reasonable to conclude that mechanical pipe rupture protection against AIBs in all systems is no longer required.

Difference 3 Section B.1.c.4 states that "if a structure separates a high energy line from an essential component, the separating structure should be designed to withstand the consequences of the pipe break in the high energy line which produces the greatest effect at the structure irrespective of the fact that the above criteria might not require such a break location to be postulated." Unit 2 design structures withstand the consequence of pipe breaks postulated at locations in accordance with Sections B.1.c.1, B.1.c.2, and B.1.c.3.

Discussion A systematic logical method must be used to evaluate the effects of pipe breaks in order to address a finite number of potential load cases. By assuming breaks at highly stressed locations and by requiring a minimum number of locations to be selected, a reasonable margin of safety will evolve.

Requiring breaks to be postulated based on structural capability is not prudent and does not enhance the safety of the plant. Several points are:

1. Pipe whip loadings are very sensitive to the distance over which unrestrained whip could occur, piping geometry, and break orientation. An infinite number of cases would require consideration particularly if splits are arbitrarily postulated along the length of the pipe. Jet impingement does not have this problem since the load is distributed over a reasonable area. However, pipe whip requires evaluation of local effects, which is much more involved.

2. An excessive number of scab plates would be required on all structures which separate high energy and essential systems, thus causing an

TABLE 1.9-1 (Cont'd.)

unreasonable number of scab plates to be installed.

3. By strengthening the weakest part of a structure, the next weakest part would then be the worst case. This is a perpetual cycle.

4. Additional safety is not really obtained by evaluating the least likely events. Since pipe breaks themselves are extremely unlikely, it is reasonable to postulate them only at the higher stressed locations. Additionally, all walls in the proximity of high energy systems are evaluated for a reasonable number of pipe breaks simply due to the number of breaks which must be postulated using the stress criteria.

14. STANDARD REVIEW PLAN 3.6.2B, REVISION 1, JULY 1981 - DETERMINATION OF RUPTURE LOCATIONS AND DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

Difference 1 The method of determining pipe rupture locations inside containment differs in that Equation (12) or (13) is not considered, if Equation (10) is less than or equal to 3.0 Sm and the cumulative usage factor is less than 0.1.

Discussion This method is consistent with the previous revision of the SRP (November 24, 1975).

Difference 2 Criterion B.1.e identified in NUREG-0800 was not used.

Discussion This method was not part of the November 24, 1975, SRP which was the only guidance available during the period this work was being performed. The method used, cracks and breaks, is consistent with the SRP applicable at the time the work was performed.

15. STANDARD REVIEW PLAN 3.7.2A, REVISION 1 - SEISMIC SYSTEM ANALYSIS

Difference Additional seismicity of ±5 percent of the maximum building dimension at the level under consideration was not assumed.

Discussion Since the three-dimensional seismic models used in the dynamic analyses of Category I structures account for the torsional effects, including the effects of eccentricities between the centers of rigidity and the

TABLE 1.9-1 (Cont'd.)

centers of mass of the structural components, the additional eccentricity of ±5 percent of the maximum building dimension is not considered necessary.

16. STANDARD REVIEW PLAN 3.7.3B, REVISION 1 - SEISMIC SUBSYSTEM ANALYSIS

Difference For the determination of the number of earthquake cycles for nuclear steam supply system (NSSS) components and equipment other than piping, 10 equivalent peak operating basis earthquake (OBE) cycles are used, as opposed to the 50 OBE cycles specified in the acceptance criteria.

Discussion Fatigue evaluation due to SSE is not necessary since it is a faulted condition and thus not required by ASME Section III (FSAR Section 3.7B.3.2). The criterion requires that 50 OBE cycles be used, and for NSSS piping, 50 cycles are used. For other NSSS components and equipment, 10 equivalent peak OBE cycles are used (FSAR Section 3.7B.3.2). This 10-cycle approach has been approved by the NRC on the basis of equivalent levels of safety (letter from R. Bosnak [NRC] to R. Artigas [GE], Number of OBE Fatigue Cycles in the BWR NSSS Design, dated February 18, 1982).

17. STANDARD REVIEW PLAN 3.7.4, REVISION 1 - SEISMIC INSTRUMENTATION

Difference Seismic monitoring instrumentation surveillance frequency is not discussed.

Discussion The seismic monitoring instrument surveillance program has been incorporated in the Technical Specifications and Technical Requirements Manual Section 3.3.7.2.

18. STANDARD REVIEW PLAN 3.8.1, REVISION 1 - CONCRETE CONTAINMENT

Difference 1 An analysis was not performed to determine the ultimate capacity of the concrete containment.

Discussion In lieu of performing ultimate capacity analysis of the containment, the structural acceptance test performed prior to the plant operation for 1.15 times the design pressure is considered sufficient assurance for the adequacy of the analysis and design of the concrete containment.

TABLE 1.9-1 (Cont'd.)

Difference 2 A design report was not prepared in accordance with Appendix C to SRP 3.8.4, Revision 1, for concrete containment.

Discussion In lieu of a design report, all the information relevant to the analysis and design of the concrete containment is provided in FSAR Section 3.8.1.

Difference 3 Compliance with RG 1.136 and ASME Section III, Division 2, was not used in designing the concrete containment and containment liner.

Discussion As stated in FSAR Table 1.8-1, RG 1.136, since the containment design precedes the issuance of RG 1.136 and ASME Section III, Division 2, it is not feasible to assure full compliance with these documents. While the loads and loading combinations are in accordance with Table CC-3230-1 of ASME III, Division 2, the acceptance criteria for stresses and strains and the procurement of materials for concrete and steel portions of the containment follows PSAR commitments. Consequently, the design, procurement, and construction of concrete and steel portions of the containment are in accordance with ACI 318, ACI 301, RG 1.94, and ASME Section III, Division 1, respectively. At the Construction Permit Stage this was accepted by the NRC as an adequate basis for the design, procurement, and construction of the containment.

19. STANDARD REVIEW PLAN 3.8.3, REVISION 1 - CONCRETE AND STEEL INTERNAL STRUCTURES OF STEEL OR CONCRETE CONTAINMENTS

Difference 1 Compliance with RG 1.142 and ACI 349-76 was not used in designing concrete internal structures.

Discussion As stated in FSAR Table 1.8-1, RG 1.142, since the design of internal structures precedes the issuance of RG 1.142 and ACI 349-76, it is not feasible to assure full compliance to these documents. Several major provisions of ACI 349-76 are identical to those of ACI 318-71 (and ACI 318-77) which was accepted for Unit 2 by the NRC at the Construction Permit Stage as an adequate basis for the design of Category I concrete structures. ACI 318-71 (and ACI 318-77) is used in designing the concrete internal structures.

Difference 2 A design report as described in Appendix C to Section 3.8.4 for all internal structures was not prepared.

TABLE 1.9-1 (Cont'd.)

Discussion Although the information required by the design report is not provided in the format required by Appendix C of NUREG-0800, SRP 3.8.4, FSAR Section 3.8.3 provides the information necessary for evaluation of internal structures of containment.

20. STANDARD REVIEW PLAN 3.8.4, REVISION 1 - OTHER SEISMIC CATEGORY I STRUCTURES

Difference 1 Compliance with RG 1.142 and ACI 349-76 was not used in designing other Category I structures.

Discussion As stated in FSAR Table 1.8-1, RG 1.142, since the design of Category I structures precedes the issuance of RG 1.142 and ACI 349-76, it is not feasible to assure full compliance to these documents. Several major provisions of ACI 349-76 are identical to those of ACI 318-71 (and ACI 318-77) which was accepted by the NRC for the design of Unit 2 Category I concrete structures at the Construction Permit Stage. ACI 318-71 (and ACI 318-77) is used in designing Category I concrete structures.

Difference 2 A design report, as described in Appendix C, was not prepared for all Category I structures.

Discussion Although the information required by the design report is not provided in the format required by Appendix C, FSAR Section 3.8.4 provides the information necessary for evaluation of Category I structures.

Difference 3 Compliance to safety-related masonry wall criteria, as described in Appendix A to this SRP, was not required.

Discussion As described in FSAR Section 3.8.4.4, Unit 2 does not use masonry walls to support any safety-related structure, system, or component. However, removable, solid concrete blocks contained in position by structural steel supports and adjacent concrete structures are used in Category I structures to provide access for equipment removal and/or installation. Since the supports are designed to withstand all the possible loading combinations and remain in place, the solid concrete blocks cannot endanger adjacent structures, systems, or components. Hence, the criteria in Appendix A to this SRP are not applied to the concrete block designs. The concrete blocks are provided to satisfy the shielding requirements for the area.

TABLE 1.9-1 (Cont'd.)

21. STANDARD REVIEW PLAN 3.8.5, REVISION 1 - FOUNDATIONS

Difference 1 Compliance with RG 1.142 and ACI 349-76 was not used in designing foundations of Category I structures.

Discussion As stated in FSAR Table 1.8-1, since the design of foundations precedes the issuance of RG 1.142 and ACI 349-76, it is not feasible to assure full compliance to these documents. Several major provisions of ACI 349-76 are identical to those of ACI 318-71 (and ACI 318-77) which was accepted by the NRC to provide an adequate basis for the design of Category I concrete structures for Unit 2 at the Construction Permit Stage. ACI 318-71 (and ACI 318-77) is used in designing the foundations of Category I structures.

Difference 2 A design report, as described in Appendix C to Section 3.8.4, was not prepared for all foundations of Category I structures.

Discussion Although the information required by the design report is not provided in the format required by Appendix C of NUREG-0800, SRP 3.8.4, FSAR Section 3.8.5 provides the information necessary for evaluation of foundations of Category I structures.

22. STANDARD REVIEW PLAN 3.9.2A, REVISION 2 - DYNAMIC TESTING AND ANALYSIS OF SYSTEMS, COMPONENTS, AND EQUIPMENT

Difference 1 A list of snubbers on systems which experience sufficient thermal movement to measure snubber travel from the cold to the hot position is not provided.

Discussion This list will be developed prior to the detailed preoperational test program and included in an amendment to the FSAR.

Difference 2 A description of the thermal motion monitoring program, i.e., verification of snubber movement, adequate clearances and gaps, including acceptance criteria and the manner in which motion will be measured, is not provided.

Discussion This will be developed prior to the detailed preoperational test program and included in an amendment to the FSAR.

TABLE 1.9-1 (Cont'd.)

Difference 3 A description of corrective action, to assure that a snubber which did not displace as predicted by analysis is operable, is not provided.

Discussion This will be addressed in the detailed preoperational test program.

Difference 4 The consideration of maximum relative displacements among supports of Category I systems and components is not described in this section.

Discussion Application to piping is discussed in FSAR Section 3.7A.3.8.3.

23. STANDARD REVIEW PLAN 3.9.2B, REVISION 2 - DYNAMIC TESTING AND ANALYSIS OF SYSTEMS, COMPONENTS, AND EQUIPMENT

Difference For the determination of the number of earthquake cycles for NSSS components other than piping, 10 equivalent peak OBE cycles are used. The SRP acceptance criteria require 50 cycles.

Discussion Fatigue evaluation due to SSE is not necessary since it is a faulted condition and thus not required by ASME Section III (FSAR Section 3.7B.3.2). The criterion requires that 50 OBE cycles be used, and for NSSS piping, 50 cycles are used. For other NSSS components and equipment, 10 equivalent peak OBE cycles are used (FSAR Section 3.7B.3.2). The 10-cycle approach has been approved by the NRC on the basis of equivalent levels of safety (letter from R. Bosnak [NRC] to R. Artigas [GE], Number of OBE Fatigue Cycles in the BWR NSSS Design, dated February 18, 1982).

24. STANDARD REVIEW PLAN 3.9.3B, REVISION 1 - ASME CODE CLASS 1, 2, AND 3 COMPONENTS, COMPONENT SUPPORTS, AND CORE SUPPORT STRUCTURES

Difference Combination of loadings does not use the criteria in RG 1.124 and 1.130.

Discussion Load combination and acceptance criteria for Category I component supports are described in FSAR Sections 3.9B.3.1, 3.9B.3.4, and Table 3.9B-2. RG 1.124 and 1.130 apply respectively to Class 1 linear and Class 1 plate and shell component support designs. Their issue dates of January 1978 (RG 1.124, Revision 1) and July 1977 (RG 1.130) are after the Unit 2 Construction Permit docketing date requirement. However, the design utilizes ultimate strength temperature correlations of regulatory

TABLE 1.9-1 (Cont'd.)

position C2 of these guides in regions adjacent to the pipe having high temperatures. Additionally, the critical buckling strength limits of ASME Section III, Appendix XVII, paragraph 2110(b), are observed in RG 1.124. Regulatory position C4 with alternate conservative collapse criteria for plates-shells is being used in lieu of regulatory position C3 in RG 1.130. The remaining design analysis criteria of these regulatory guides are considered to be adequately addressed by conservatisms present in the existing ASME Section III code.

25. STANDARD REVIEW PLAN 3.9.5B, REVISION 2 - REACTOR PRESSURE VESSEL INTERNALS

Difference 1 The design and construction of the core support structures do not conform to the requirements of subsection NG, Core Support Structures, of the ASME Code (Reference 5), and SRP Section 3.9.3.

Discussion Unit 2 core support structures were designed and purchased in 1971 prior to the issue of ASME Section III, subsection NG, in 1974. However, an earlier draft of ASME Section III, subsection NB, was used as a guide in developing the design of these supports. These criteria are presented in Section 3.9B.5.3. Subsequent to the issuance of ASME Section III, subsection NG, comparisons were made to assure that the pre-ASME Section III, subsection NG, design provides the equivalent level of safety as prescribed by ASME Section III, subsection NG, 1974.

Difference 2 The design criteria, loading conditions, and analyses that provide the basis for the design of reactor internals other than the core support structures do not meet the guidelines of NG-3000.

Discussion Unit 2 reactor internals other than core support structures were designed and purchased prior to the initial issuance of ASME Section III, subsection NG. Design guidelines for these components and later safety comparisons against subsection NG criteria were selected as described for core support structures in criteria II.b under Difference 1.

26. STANDARD REVIEW PLAN 3.9.6, REVISION 2 - INSERVICE TESTING OF PUMPS AND VALVES

Difference The acceptance criteria for NUREG-0800 require that pumps and valves not categorized as Code Class 1, 2,

TABLE 1.9-1 (Cont'd.)

or 3 but which are considered to be safety related be added to the inspection program.

Discussion The Unit 2 inservice testing program will conform to these criteria by meeting the relevant requirements set forth in GDC 37, 40, 43, 46, 54, and 10CFR50.55a.

27. STANDARD REVIEW PLAN 3.10A, REVISION 2, JULY 1981 - SEISMIC AND DYNAMIC QUALIFICATION OF MECHANICAL AND ELECTRICAL EQUIPMENT

Difference The position for RG 1.148 is not provided in this section.

Discussion The Unit 2 degree of compliance with RG 1.148 is in FSAR Section 1.8.

28. STANDARD REVIEW PLAN 3.11, REVISION 2 - ENVIRONMENTAL QUALIFICATION OF MECHANICAL AND ELECTRICAL EQUIPMENT

Difference 1 The submittal of the environmental qualification document which demonstrates equipment environmental capability is not included.

Discussion The environmental qualification document is maintained as part of the Nine Mile Point Unit 2 Equipment Qualification Program. This document is maintained separately from the FSAR, and it is not considered a part of the FSAR.

29. STANDARD REVIEW PLAN 4.2, REVISION 2 - FUEL SYSTEM DESIGN

Difference 1 Factors of 2 on stress or 20 on cycles are not used with the current method.

Discussion Design limits for fatigue failure are provided in GESTAR-II, Section 2.5, and have been approved by the NRC.

Difference 2 Allowable fretting wear is not stated in the FSAR.

Discussion See GESTAR-II, Appendix A, Section A.4.2.1.1.3, and Section 2.6.3.

Difference 3 Separate design limits for oxidation, hydriding, and corrosion buildup are not stated in the FSAR.

TABLE 1.9-1 (Cont'd.)

Discussion See GESTAR-II, Appendix A, Section A.4.2.1.1.4.

Difference 4 There is no limit for internal gas pressure stated in the FSAR.

Discussion See GESTAR-II, Appendix A, Section A.4.2.1.1.6. Current methodology has been approved by the NRC.

Difference 5 Allowable fretting wear is not stated in the FSAR.

Discussion See GESTAR-II, Appendix A, Section A.4.2.1.2.3.

Difference 6 There is no centerline melt criterion for abnormal operational events stated in the FSAR.

Discussion See GESTAR-II, Appendix A, Section A.4.2.1.2.5. (Reference subsections 2.4.2.5 and 2.4.1.1, GESTAR-II.)

Difference 7 A description of elastic strain limits is not included in the FSAR. There is no centerline melt criterion for abnormal operational events described in the FSAR.

Discussion The 1 percent plastic strain criterion is applied to all abnormal operating events. No fuel melt criterion is applied. This methodology has been approved by the NRC. See GESTAR-II, Appendix A, Section A.4.2.1.2.7.

Difference 8 The fuel system description does not provide all the information discussed in the acceptance criteria.

Discussion The level of descriptive information and detail in the FSAR is consistent with that previously approved and accepted by the NRC. Quantitative information is provided in GESTAR-II, Chapters 2 and S.2, and is referenced in FSAR Section 4.2.

Difference 9 Surveillance of control rods for boron leaching is not provided in the FSAR.

Discussion Periodic reactivity testing of the control rods (beyond the beginning of cycle shutdown margin demonstration) is performed only if there is reason to

TABLE 1.9-1 (Cont'd.)

suspect control absorber loss or other degradation of the control blades.

30. STANDARD REVIEW PLAN 4.4, REVISION 1 - THERMAL AND HYDRAULIC DESIGN

Difference Compliance with TMI Action Plan requirements (NUREG-0737) is not assessed in this section.

Discussion See FSAR Section 1.10, Tasks II.D.1, II.F.1, and II.F.2.

31. STANDARD REVIEW PLAN 4.5.1, REVISION 2 - CONTROL ROD DRIVE STRUCTURAL MATERIALS

Difference 1 Only those parts of the control rod drive (CRD) forming part of the primary pressure boundary are code materials.

Discussion Jurisdiction of ASME Section III does not extend to the nonpressure parts of the CRD. ASME materials are identified in the materials list of FSAR Section 4.5.1.1.

Difference 2 Some CRD structural materials were not purchased to code requirements, but there is no difference for tempering and aging temperatures.

Discussion Noncode materials are not required to be purchased to code requirements. The materials specified were, however, selected for their compatibility with the reactor coolant. Tempering and aging are done according to standards which are discussed in FSAR Section 4.5.1.

32. STANDARD REVIEW PLAN 4.5.2, REVISION 2 - REACTOR INTERNAL AND CORE SUPPORT MATERIALS

Difference 1 ASME Section III, NG-2000, specifications were not used.

Discussion For core support and reactor internals, the material specifications given in ASME Section III, NG-2000, were not used. Article NG-2000 was not part of Section III at the time these materials were procured for Unit 2. All core support structures were fabricated from ASME- and ASTM-specified materials and designed using ASME Section III as a guide. The other reactor internals are noncoded and are fabricated from ASME or ASTM specification materials. Material requirements in the ASTM specifications are identical to the requirements

TABLE 1.9-1 (Cont'd.)

given in the corresponding ASME specifications. The material specifications for Unit 2 reactor internal and core support materials are given in FSAR Section 4.5.2.1.

Difference 2 ASME Section III, NG-4000 and NG-5000, were not imposed.

Discussion The requirements of Articles NG-4000 and NG-5000 were not part of ASME Section III when fabrication welding was performed for Unit 2. As specified in FSAR Section 4.5.2.2., welding was performed to the requirements of ASME Section IX. Conformance to regulatory guides applicable to welding (i.e., RG 1.31, 1.34, 1.37, 1.44, and 1.71) is presented in FSAR Section 4.5.2.4.

Difference 3 ASME Section III, NG-2500 and NG-5300, were not imposed.

Discussion Articles NG-2500 and NG-5300 were not part of ASME Section III at the time the Unit 2 wrought seamless tubular products and fittings were procured. As contained in FSAR Section 4.5.2.3, wrought seamless tubular products for CRD guide tubes, CRD housings, and peripheral fuel supports were supplied in accordance with applicable ASME material specifications. These specifications require a hydrostatic test on each length of tubing. No other nondestructive testing was specified for the tubes.

33. STANDARD REVIEW PLAN 5.2.1.1, REVISION 2 - COMPLIANCE WITH THE CODES AND STANDARDS RULE, 10CFR50.55a

Difference Differences exist between the Unit 2 design and RG 1.26, Quality Group Classification and Standards.

Discussion Justification for all differences listed in Table 3.2-1 are discussed in the notes to the table.

34. STANDARD REVIEW PLAN 5.2.2, REVISION 2 - OVERPRESSURE PROTECTION

Difference TMI Tasks are not discussed in this section.

Discussion NUREG-0737, Task II.D.3, is discussed in FSAR Sections 1.10 and 5.4.12.

TABLE 1.9-1 (Cont'd.)

35. STANDARD REVIEW PLAN 5.2.3, REVISION 2 - REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

Difference 1 Material Specifications - Part of the RCPB materials comply with ASME Section II, Parts A and C, only.

Not all ASME Code cases used are listed in RG 1.84 and 1.85.

Discussion Section 5.2.3.1 and Table 5.2-5 list components and their material specifications. These specifications comply with ASME Section II, Parts A and C, and are augmented with ASME Section III, Code Cases 1562 and 1572, and Code cases approved by RG 1.84 and 1.85.

Code Cases 1141-1, 1332-6, 1361-2, 1557-2, 1620, and N-1588, which are imposed by RG 1.84 and 1.85, were used as noted in FSAR Table 5.2-1. Other Code cases used but not imposed by these regulatory guides are 1562 and 1572. These Code cases have been annulled and incorporated into ASME Section III.

Difference 2 The FSAR does not address the qualification of welding procedures at the minimum preheat.

Discussion FSAR Section 5.2.3.3.2 indicates that components were either held for an extended time at preheat temperature to assure removal of hydrogen or were maintained at preheat temperature until postweld treatment. Minimum preheat and maximum interpass temperatures were specified and monitored.

Difference 3 Some ferrite tubular products do not meet all requirements of RG 1.66 and ASME Section III, paragraph NB-2550.

Discussion See FSAR Section 5.2.3.3.3. Nondestructive examination of ferrite tubular products met existing ASME Section III and 10CFR50 criteria at order placement, which in some cases predated RG 1.66. CRD housing tubes do meet ASME Section III, paragraph NB-2550.

Difference 4 RG 1.44 was not applied completely by the Unit 2 project design basis for the NSSS.

Discussion For NSSS components, alternate criteria for sensitization controls of stainless steel which satisfy NUREG-0313 are discussed in FSAR Section 5.2.3.4.

TABLE 1.9-1 (Cont'd.)

Difference 5 The NSSS QA program complies with RG 1.37 except for Section 5, paragraph G, which recommends that local rusting on corrosion-resistant alloys be removed by mechanical means.

Discussion GE Topical Report NEDO-11209 (accepted by the NRC) describes the NSSS QA program and does not preclude the use of other than mechanical means for local rust removal.

Difference 6 Some austenitic tubular products were procured prior to the creation of ASME Section III, paragraph NB-2550.

Discussion See FSAR Section 5.2.3.3.3 and the assessment of criterion II.3.c of this SRP for positions on ASME Section III, paragraph NB-2550, requirements.

36. STANDARD REVIEW PLAN 5.3.1, REVISION 1 - REACTOR VESSEL MATERIALS

Difference 1 Special Methods for Nondestructive Examination - Ultrasonic examination methods meet ASME Section XI, 1980 Edition through the Winter 1980 Addenda, Appendix I, rather than ASME Section III requirements.

Discussion FSAR Section 5.3.1.2 describes radiographic examination, which is performed on all pressure-containing welds in accordance with requirements of ASME Section III, subsection NB-5320. FSAR Section 5.3.1.3 indicates that materials and welds on the RPV were examined by methods which meet ASME Section III requirements. Special ultrasonic examination meeting ASME Section XI, 1980 Edition through the Winter 1980 Addenda, Appendix I, requirements using manual techniques were used. Acceptance standards were equal to or greater than those required by ASME Section XI, 1980 Edition through the Winter 1980 Addenda.

Difference 2 The NSSS QA program complies with the referenced regulatory guides except for Section 5, paragraph 6, of RG 1.37, which recommends that local rusting on corrosion-resistant alloys be removed by mechanical means.

Discussion GE Topical Report NEDO-11209 (accepted by the NRC) describes the NSSS QA program and does not preclude the use of other than mechanical means for local rust removal.

TABLE 1.9-1 (Cont'd.)

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, drywell head, biological shield wall, structural steel,

TABLE 1.9-1 (Cont'd.)

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.

Difference 3 A redundant position indicator for each vacuum relief valve and an alarm for vacuum breaker valves are not provided.

TABLE 1.9-1 (Cont'd.)

Discussion Each vacuum breaker flow path has two relief valves mounted horizontally in series to ensure a leak-tight boundary. Three flow paths are required for the vacuum breaker design basis; however, four flow paths (eight valves) are provided. Each vacuum relief valve is provided with three position-sensing devices mounted 120 degrees apart around the circumference of each disc. One of the position-sensing devices is mounted at the bottom of the disc. These devices are designed such that all three positions must be within 0.05 in of the full-closed position before a closed signal can be initiated. Total detectable opening for the vacuum breakers is ≤ 0.044 ft^2 vs. the allowable bypass leakage capacity of 0.05 ft^2, thus providing adequate sensitivity. Although redundant position indication does not exist on each vacuum relief valve, redundancy is achieved due to redundant valves in each flow path. Indication in the control room is achieved by red/green lights.

Difference 4 Visual inspection at each refueling outage for vacuum relief valves and piping is not described.

Discussion 4 This is addressed in the Technical Specification.

Difference 5 Vacuum breaker operability test at monthly intervals is not described.

Discussion 5 This is addressed in the Technical Specification.

41. STANDARD REVIEW PLAN 6.2.1.2, REVISION 2, JULY 1981 - SUBCOMPARTMENT ANALYSIS

Difference The acceptable model for subcompartment initial conditions is to assume air at the maximum allowable temperature, minimum absolute pressure, and zero percent relative humidity.

One of the Unit 2 annulus pressurization analyses assumes 20 percent relative humidity instead of zero percent.

Discussion The governing case for the design of the annulus considers zero percent relative humidity; therefore, Unit 2 meets the intent of the acceptance criteria.

TABLE 1.9-1 (Cont'd.)

42. STANDARD REVIEW PLAN 6.2.1.3, REVISION 1, JULY 1981 - MASS AND ENERGY RELEASE ANALYSIS FOR POSTULATED LOCAs

Difference The ability of the containment and its associated systems, including subcompartments, to withstand calculated pressure and temperature conditions resulting from any LOCA without exceeding design temperature is not discussed in this section.

Discussion See discussion for SRP 6.2.1.1.c (steam bypass temperature of wetwell).

43. STANDARD REVIEW PLAN 6.2.2, REVISION 3, JULY 1981 - CONTAINMENT HEAT REMOVAL SYSTEMS

Difference The spray drop efficiency calculation is not provided.

Discussion An analysis of the spray drop thermal effectiveness was not performed due to the unavailability of drop size test data from the nozzle manufacturer. When the required drop size data become available, the spray thermal effectiveness will be calculated by the method referenced in this SRP.

44. STANDARD REVIEW PLAN 6.2.7, REVISION 0 - FRACTURE PREVENTION OF CONTAINMENT PRESSURE BOUNDARY

Difference Not all of the Class 2 piping and/or components (valves) have had actual impact testing performed.

Ferritic materials of construction for the containment pressure boundary have been toughness tested as follows:

1. All ferritic material of the primary containment liner (e.g., drywell and suppression pool liner plate, equipment and personnel hatches, drywell head, penetration sleeves, etc.) requiring notch toughness have been Charpy impact tested and conform to NE-2300 of ASME Section III. This information may be found in FSAR Section 3.8.1, specifically Item 3.8.1.6.2.

2. Class 1 ferritic process piping has been impact tested and conforms to NB-2300 of ASME Section III.

TABLE 1.9-1 (Cont'd.)

Discussion Class 2 ferritic process piping has not been impact tested, except for that portion included in the penetration assembly which penetrates the containment liner. It has been impact tested and conforms to NB-2300 of ASME Section III.

An initial review indicates that similar construction materials have been used on those items which were not subjected to actual impact testing. This indicates that inherent toughness may be substantiated.

45. STANDARD REVIEW PLAN 6.3, REVISION - EMERGENCY CORE COOLING SYSTEM

Difference The requirements of the following Task Action Plans are not addressed in this section:

1. Task Action Plan II.B.8 of NUREG-0718 (Reference 14).
2. Task Action Plan III.D.1.1 of NUREG-0694 and NUREG-0718.
3. Task Action Plan II.E.2.1 of NUREG-0737.
4. Task Action Plan II.K.3(10) of NUREG-0737 and NUREG-0718.
5. Task Action Plan II.K.3(15) of NUREG-0737 and NUREG-0718.
6. Task Action Plan II.K.3(18) of NUREG-0737 and NUREG-0718.
7. Task Action Plan II.K.3(21) of NUREG-0737 and NUREG-0718.

Discussion Items 1, 2, 5, 6, and 7 are discussed in FSAR Section 1.10. Items 3 and 4 are not applicable to Unit 2.

46. STANDARD REVIEW PLAN 6.5.1, REVISION 2, JULY 1981 - ENGINEERED SAFETY FEATURES ATMOSPHERE CLEANUP SYSTEMS

Difference Exception is taken with compliance to RG 1.52.

Discussion See FSAR Table 1.8-1.

TABLE 1.9-1 (Cont'd.)

47. STANDARD REVIEW PLAN 8.3.1, REVISION 2 - AC POWER SYSTEMS (ONSITE)

Difference The Division III (HPCS) standby diesel generator (GM-EMD) is provided with a standard-duty turbocharger mechanical drive gear assembly.

Discussion The Division III standby diesel generator is retrofitted with a heavy-duty turbocharger drive gear assembly.

48. STANDARD REVIEW PLAN 9.1.3, REVISION 1 - SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

Difference The acceptance criteria of Section II.1.d.(4) require computation of decay heat loads based on one refueling load after 150 hr decay, plus one refueling load after 1 yr decay. FSAR Section 9.1.3.2 describes the conditions for the spent fuel heat load as one refueling load after 288 hr decay plus additional refuelings decayed in multiples of 18 months after reactor shutdown. Also described is the refueling condition with 96 hr decay plus additional refuelings decayed in multiples of 24 months.

Discussion The decay times used to compute the spent fuel heat loads are consistent with expected operating procedures and refueling cycles for Unit 2.

49. STANDARD REVIEW PLAN 9.1.5, REVISION 0 - OVERHEAD HEAVY LOAD HANDLING SYSTEM

There is no FSAR Section 9.1.5. All material relating to this subject is in Section 9.1.4.

50. STANDARD REVIEW PLAN 9.2.2, REVISION 1 - REACTOR AUXILIARY COOLING WATER SYSTEM

Difference 1 Task II.K.2.16 of NUREG-0718 and Task II.K.3.25 of NUREG-0737, as they relate to loss of cooling water to reactor coolant pump seals, are not addressed in this section.

Discussion NUREG-0718 is not applicable to Unit 2. It is applicable to applicants for construction permit or manufacturing license only.

Task II.K.3.25 is addressed in FSAR Section 1.10.

TABLE 1.9-1 (Cont'd.)

Difference 3 The ability of the reactor coolant pumps to withstand a complete loss of cooling water for 20 min is not demonstrated by testing.

Discussion An analysis was used to demonstrate that the cooling water systems have been designed such that cooling water will be provided whenever it is needed.

51. STANDARD REVIEW PLAN 9.2.3, REVISION 2 - DEMINERALIZED WATER MAKEUP SYSTEM

Difference Acceptance Criterion II, 1, is not addressed.

Discussion Unit 2 is in compliance with SRP 9.2.3, Acceptance Criterion II, 1, although it is not addressed. All makeup water system piping in the reactor building is seismically analyzed.

52. STANDARD REVIEW PLAN 9.3.2, REVISION 2 - PROCESS AND POST-ACCIDENT SAMPLING SYSTEMS

Difference 1 The postaccident sampling system is not completely addressed in Section 9.3.2.

Discussion Additional information on the postaccident sampling system is provided in Task II.B.3.

53. STANDARD REVIEW PLAN 9.4.1, REVISION 2, JULY 1981 - CONTROL ROOM AREA VENTILATION SYSTEM

Difference Unit 2 does not meet the guidance of RG 1.52 and 1.140.

Discussion Unit 2 complies with the intent of RG 1.52 and 1.140 (paragraph c of these guides) through the alternate approaches discussed in FSAR Section 1.8.

54. STANDARD REVIEW PLAN 9.4.2, REVISION 2, JULY 1981 - SPENT FUEL POOL AREA VENTILATION SYSTEM

Difference Unit 2 does not meet the guidance of RG 1.52 and 1.140.

Discussion Unit 2 complies with the intent of RG 1.52 and 1.140 (paragraph c of these guides) through the alternate approaches discussed in FSAR Section 1.8.

TABLE 1.9-1 (Cont'd.)

55. STANDARD REVIEW PLAN 9.4.3, REVISION 2, JULY 1981 - AUXILIARY AND RADWASTE AREA VENTILATION SYSTEM

Difference Unit 2 does not meet the guidance of RG 1.140.

Discussion Unit 2 complies with the intent of RG 1.140 (paragraph c) through the alternate approach discussed in FSAR Section 1.8.

56. STANDARD REVIEW PLAN 9.4.5, REVISION 2, JULY 1981 - ENGINEERED SAFETY FEATURE VENTILATION SYSTEM

Difference Unit 2 does not meet the guidance of RG 1.52.

Discussion Unit 2 complies with the intent of RG 1.52 (paragraph c) through the alternate approach discussed in FSAR Section 1.8.

57. STANDARD REVIEW PLAN 9.5.1, REVISION 3, JULY 1981 - FIRE PROTECTION PROGRAM (FIRE PROTECTION SYSTEM)

Deviations to BTP CMEB 9.5-1
Attached to Standard Review Plan 9.5.1
Fire Protection Program
Difference 1 Section C.1.c.(3) states that "the fire suppression system should be capable of delivering water to manual hose stations located within hose reach of areas containing equipment required for safe shutdown following the safe shutdown earthquake (SSE)."

Discussion Unit 2 standpipe and hose connection design is in accordance with Appendix A (dated August 1976) to BTP 9.5-1 (dated May 1, 1976) and Appendix R to 10CFR50, and is not seismically qualified.

The design does not contemplate simultaneous earthquake and fire conditions; therefore, this requirement was not incorporated into the design. Further, justification is that Unit 2 is not in an area of high seismic activity.

Difference 2 Section C.5.a(3)(b) of Unit 2 design incorporates fire boot-type penetration seals (approximately 200 of 11,000 fire-rated seals) for which temperature levels on the unexposed side reached 393°F during the acceptance test.

Discussion Fixed combustibles potentially within close proximity have ignition temperatures of >500°F. Cables are generally installed in raceways (i.e., conduit or cable trays).

TABLE 1.9-1 (Cont'd.)

Difference 3 Section C.5.a(5) - Unit 2 fire doors are administratively supervised to verify that they are in the closed position.

Discussion Fire doors are maintained in the closed position.

The doors are administratively verified to be in the closed position on a daily basis. Additionally, fire doors in areas protected by automatic total-flooding CO_2 systems are provided with heavy-duty door closures. Halon 1301 suppression systems are used in computer rooms and control rooms. Doors to these areas are inherently supervised by the occupants in the area, in addition to the daily inspection, to verify that the doors are in the proper position.

Difference 4 Section C.5.a(14) - Unit 2 floor drains are conservatively sized and are in accordance with the National Plumbing Code. They were not sized based on firefighting water flows.

Discussion Unit 2 fixed water suppression systems incorporate the use of closed-heads and closed-water spray nozzles, which inherently limit the amount of water discharged to the area of involvement during a fire. Refer to Section 9A.3.5.1.12 for the results of an evaluation completed to determine the effects of firefighting water flows on floor drains.

Difference 5 Section C.5.b.(2) - Credit is taken in the Unit 2 reactor building for separation of cables, equipment, and associated circuits of redundant trains of safe shutdown equipment by a horizontal distance of more than 20 ft. Fire detection and automatic suppression systems are provided in the zone. Nonsafe shutdown-related cable trays traverse the 20-ft zone.

Discussion Fire detection, automatic area suppression, and automatic cable tray suppression systems are provided for the cables in this zone in accordance with Section 9A.3.5.5.3. The cables are IEEE-383 qualified.

Difference 6 Section C.5.e.(2) - Unit 2 safety-related cable trays are provided with ionization-type detectors in lieu of line-type and ionization detectors. Unit 2 safety-related cable trays are provided with closed-head preaction sprinkler systems in lieu of open-head deluge or open directional spray nozzle systems.

TABLE 1.9-1 (Cont'd.)

Discussion Safety-related cable trays are provided with ionization-type smoke detectors which provide an earlier warning system than line-type heat detectors. Safety-related cable trays that are not accessible for manual firefighting are protected by zoned automatic closed-head reaction sprinkler systems. Water spray systems that incorporate the use of open directional spray nozzles discharge an excessive amount of water in protected areas, requiring substantially larger drainage and processing capabilities than areas protected by sprinkler systems which minimize the potential for damage to safety-related structures and components.

Difference 7 Section C.5.g(1) - Unit 2 emergency lighting capability is provided by means other than individual 8-hr battery supplies.

Discussion Areas which must be manned during safe shutdown will be supplied with 8-hr battery-packs for access and egress lighting.

Difference 8 Section C.5.g.(3) - The Unit 2 emergency communications system is not independent of the plant communication system.

Discussion Fixed emergency communications systems independent of normal plant communications systems are not necessary because:

1. The systems are connectable to uninterruptible power sources, which provide reliability during emergency conditions.

2. In case of total loss of power to all communication systems, the sound-powered communication (SPC) system can be utilized.

3. The system is set up as described in Section 9.5.2.

4. The system and important components are supervised.

Difference 9 Section C.6.a.(3) - The fire detector spacing criteria for Unit 2 meet the intent of NFPA 72E.

Discussion NFPA 72E recommends one detector per bay for beam depth greater than 8 in and bay width greater than 8 ft. NFPA 72E does not address beam depth greater than 8 in and bay width less than 8 ft. In this situation, the

TABLE 1.9-1 (Cont'd.)

Unit 2 design incorporates one detector for every other bay mounted on the bottom flange of structural steel.

Difference 10 Section C.6.c.(4) - Unit 2 design does not incorporate a cross-connection to the SWP system for firefighting capability post-SSE.

Discussion Standpipes and hose connections for manual firefighting are seismically supported in safety-related areas and in areas containing safety-related equipment. The design bases do not contemplate simultaneous earthquake and fire conditions; therefore, this requirement was not incorporated into the design. Further justification is that Unit 2 is not in an area of high seismic activity.

Difference 11 Section C.7.a.(1), part (c) - During normal operation, the Unit 2 design does not incorporate the use of general area fire detection in the primary containment.

Discussion The Unit 2 containment is inerted during normal operation.

Difference 12 In general, Section C endorses the use of the National Fire Protection Association (NFPA) standards. Unit 2 deviates from a number of these NFPA standards.

Discussion Each Unit 2 deviation to the NFPA standards is described and justified in Table 9.5-3.

Difference 13 Section C.7.b - Unit 2 design incorporates the use of carpet in the control room.

Discussion Carpet exceeds NFPA 101, Class I, interior floor finish requirements and is required to satisfy human factors guidelines.

58. STANDARD REVIEW PLAN 10.3, REVISION 2 - MAIN STEAM SUPPLY SYSTEM

Difference Acceptance Criterion II, 2, is not addressed in Section 10.3 with respect to internally- or externally-generated missiles.

Discussion Unit 2 complies with this criterion as discussed in FSAR Section 3.5.1.

TABLE 1.9-1 (Cont'd.)

59. STANDARD REVIEW PLAN 10.4.2, REVISION 2 - MAIN CONDENSER EVACUATION SYSTEM

Difference RG 1.33 and 1.123 are not addressed in this section.

Discussion RG 1.33 and 1.123 are discussed in Section 1.8.

60. STANDARD REVIEW PLAN 10.4.3, REVISION 2 - TURBINE GLAND SEALING SYSTEM

Difference RG 1.33 and 1.123 are not addressed in this section.

Discussion RG 1.33 and 1.123 are discussed in Section 1.8.

61. STANDARD REVIEW PLAN 11.2, REVISION 2 - LIQUID WASTE MANAGEMENT

Difference The Unit 2 position on RG 1.143 is not addressed in this section.

Discussion The Unit 2 project complies with RG 1.143 through the alternate approach discussed in Section 1.8.

62. STANDARD REVIEW PLAN 11.3, REVISION 2 - GASEOUS WASTE MANAGEMENT SYSTEMS

Difference The Unit 2 position on RG 1.143 is not addressed in this section.

Discussion The Unit 2 project complies with RG 1.143 through the alternate approach discussed in Section 1.8.

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.

TABLE 1.9-1 (Cont'd.)

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 - 249 days Kr - 333 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 .39 Rem); <30 Rem Beta (calculated .31 Rem)

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:

TABLE 1.9-1 (Cont'd.)

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 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.

TABLE 1.9-1 (Cont'd.)

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.

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

TABLE 1.9-1 (Cont'd.)

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.

Discussion The preoperational test descriptions, which will be available for NRC review at least 60 days before the test is to be run, will include plant performance characteristics.

69. STANDARD REVIEW PLANS 15.3.3 AND 15.3.4, REVISION 2 - REACTOR COOLANT PUMP ROTOR SEIZURE AND REACTOR COOLANT PUMP SHAFT BREAK

Difference Accident analysis of these faulted events does not include the assumption of turbine trip and coincident LOOP and coastdown pumps.

Discussion The consequences of this combination would be less severe than the transient analyzed in FSAR Section 15.2.6. The turbine trip, or indirect LOOP, will initiate scram and cause rapid power reduction. The severity of shaft seizure or shaft break, without a LOOP, is evidenced by the fast coastdown of core flow which reduces thermal margin significantly before the L8-initiated scram.

TABLE 1.9-1 (Cont'd.)

70. STANDARD REVIEW PLAN 15.4.7, REVISION 1 - INADVERTENT LOADING AND OPERATION OF A FUEL ASSEMBLY IN AN IMPROPER POSITION

Difference Plant operating procedures will not contain provisions to search for fuel-loading errors with nuclear instrumentation.

Discussion As addressed in FSAR Section 15.4.7.1, the probability of a fuel bundle being misplaced is extremely small.

The Unit 2 approach is to analyze the worst case (misplaced bundle accident) and show compliance with fuel limits. The analysis and results demonstrating compliance with these limits is presented in FSAR Section 15.4.7.3. (See also GESTAR-II, Section S.2.5.4.)

71. STANDARD REVIEW PLAN 15.6.4, REVISION 2 - RADIOLOGICAL CONSEQUENCES OF MAIN STEAM LINE FAILURE OUTSIDE CONTAINMENT (BWR)

Difference The iodine concentration in the primary coolant is stated in NUREG-0800, paragraph III.2.b, to correspond to the following two cases:

1. The concentration is the maximum value permitted and corresponds to the conditions of an assumed preaccident spike (meets the 10CFR100 Dose Guidelines).

2. The concentration is the maximum equilibrium value permitted for continued full-power operation (meets 10 percent of the 10CFR100 Dose Guidelines).

The FSAR presents the results of the main steam line failure analysis performed using only Case 1.

Discussion The main steam line failure analysis performed using the more conservative assumption that the iodine concentration in the primary coolant is the maximum value permitted by the BWR standard technical specifications, results in doses that are less than 10 percent of the limits of 10CFR100.

Therefore, FSAR Section 15.6.4 is considered to meet or exceed the requirements of NUREG-0800, Section 15.6.4, without performing the other analysis.

TABLE 1.9-1 (Cont'd.)

72. STANDARD REVIEW PLAN 15.6.5, REVISION 2 - LOSS-OF-COOLANT ACCIDENTS RESULTING FROM SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

Difference The TMI Task Action Plan requirements for II.E.2.3, II.K.3.25, II.K.3.30, and II.K.3.31 have not been addressed.

Discussion See FSAR Section 1.10 for Tasks II.K.3.25, II.K.3.30, and II.K.3.31. Resolution of Task II.E.2.3 is not addressed in the FSAR but has been generically approved by the NRC.

73. STANDARD REVIEW PLAN 15.8, REVISION 1 - ANTICIPATED TRANSIENTS WITHOUT SCRAM

Difference 1 GDC 10, 15, 26, 27, and 29 are not applied for the ATWS event.

Discussion The postulated ATWS event is so remote that it is outside the range of DBAs to which these GDC apply. The RCPB pressure design has sufficient margin to meet GDC 15.

Difference 2 NUREG-0460, Volume 2, Section IV-4, criterion j, is not applicable to the RPT design.

Discussion The NRC reviewed RPT design features during 1978 and 1979 and, after the publication of Volume 2 of NUREG-0460, determined a set of design criteria to determine RPT acceptability. These criteria are essentially the same as criteria a through i of NUREG-0460, Volume 2, Section IV-4. The NRC has deemed the Monticello and Hatch RPT designs as being acceptable since they meet these criteria as noted in SRP 15.8.

74. SRP DEVIATION WRITEUPS, CHAPTER 16 - TECHNICAL SPECIFICATIONS

The information contained in Chapter 16 was finalized in July 1987 when the full-power license was issued. The results of an analysis to determine conformance to the SRP will be provided in a future update.

TABLE 1.9-1 (Cont'd.)

75. STANDARD REVIEW PLAN 17.1, REVISION 2 - QUALITY ASSURANCE DURING THE DESIGN AND CONSTRUCTION PHASES

This SRP is not applicable to Unit 2. A review of Section 17.1 shows that the program is in conformance to this SRP for the operations phase QA program, as defined in FSAR Appendix B (QA Topical Report).

76. STANDARD REVIEW PLAN 17.2, REVISION 2 - QUALITY ASSURANCE DURING THE OPERATIONS PHASE

QA during the operations phase is discussed in FSAR Appendix B (QA Topical Report). There are no differences noted.

77. STANDARD REVIEW PLAN 18.0, REVISION 0 - HUMAN FACTORS ENGINEERING

SRP acceptance criteria for this section are still being developed. An analysis will be performed when the acceptance criteria are finalized.

78. STANDARD REVIEW PLAN 13.2.1, REVISION 0 - REACTOR OPERATOR TRAINING

Difference The Licensed Operator Candidate and Licensed Operator Requalification Training Programs comply with 10CFR55 requirements.

Discussion The Licensed Operator Candidate and Requalification Training Programs have been certified as using the Systems Approach to Training, which is an acceptable alternative to the line item requirements of this SRP.

79. STANDARD REVIEW PLAN 11.5, REVISION 3 - PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

Difference Process and effluent radiation monitors are not provided for the ADH system.

Discussion A process radiation monitor was not installed on the secondary side of the ADH system because the possibility of the secondary side of the ADH system becoming contaminated has been analyzed and found not to be a credible event.

would not require ECCS actuation, although it may result in leakage beyond Technical Specification limits. On PWRs, the use of new or existing lines whose smallest orifice is larger than the LOCA definition will require a valve-in-series valve that can be closed from the control room to terminate the LOCA that would result if an open vent valve could not be reclosed.

A positive indication of valve position should be provided in the control room.

The reactor coolant vent system shall be operable from the control room.

Since the RCS vent will be part of the RCPB, all requirements for the RCPB must be met, and, in addition, sufficient redundancy should be incorporated into the design to minimize the probability of an inadvertent actuation of the system. Administrative procedures may be a viable option to meet the single-failure criterion. For vents larger than the LOCA definition, an analysis is required to demonstrate compliance with 10CFR50.46.

The probability of a vent path failing to close, once opened, should be minimized; this is a new requirement. Each vent must have its power supplied from an emergency bus. A single failure within the power and control aspects of the reactor coolant vent system should not prevent isolation of the entire vent system when required. On BWRs, block valves are not required in lines with safety valves that are used for venting.

Vent paths from the primary system to within containment should go to those areas that provide good mixing with containment air.

The reactor coolant vent system (i.e., vent valves, block valves, position indication devices, cable terminations, and piping) shall be seismically- and environmentally-qualified in accordance with IEEE-344-1975 as supplemented by RG 1.100 and RG 1.92, and Standard Review Plan (SRP) 3.9.2, 3.9.3, and 3.10. Environmental qualifications are to be in accordance with the May 23, 1980, Commission Order and memorandum (CLI-80-21).

Provisions to test for operability of the reactor coolant vent system should be part of the design. During the First Ten-Year Interval, testing was performed in accordance with Subsection IWV of Section XI, the 1983 Edition with the Summer 1983 Addenda. During the Second Ten-Year Interval and subsequent intervals, inservice testing will be in accordance with 10CFR50.55a and the IST program plan.

It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for Operator error. A human-factor analysis should be performed taking into consideration:

1. The use of this information by an Operator during both normal and abnormal plant conditions.

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2. Integration into emergency procedures.

3. Integration into Operator training.

4. Other alarms during emergency and need for prioritization of alarms.

Specific BWR Design Considerations

Since the BWROG has suggested that the present BWR designs have an inherent capability to vent, a question relating to the capability of existing systems arises. The ability of these systems to vent the RCS of noncondensable gas generated during an accident must be demonstrated. Because of differences among the head vent systems for BWRs, each licensee or applicant should address the specific design features of this plant and compare them with the generic venting capability proposed by the BWROG. In addition, the ability of these systems to meet the same requirements as the PWR vent system must be documented.

In addition to RCS venting, each BWR licensee should address the ability to vent other systems, such as the isolation condenser which may be required to maintain adequate core cooling. If the production of a large amount of noncondensable gas would cause the loss of function of such a system, remote venting of that system is required. The qualifications of such a venting system should be the same as that required for PWR venting systems.

Nine Mile Point Unit 2 Position

The Unit 2 design philosophy is in agreement with the BWROG position on this subject, which is described in detail in NEDO-24782[2].

The Unit 2 design includes 18 main steam SRVs, of which 7 are used for ADS. Redundant divisional power is supplied to the ADS valves. The discharge lines from the ADS valves (as well as the discharge lines from the 11 non-ADS SRVs) run individually to the suppression pool. The ADS valves and discharge lines satisfy the NUREG-0737 requirements for RCS venting.

In addition to the ADS valves, RCS venting can also take place through the RCIC system which directs steam from one of the main steam lines to a turbine-driven pump; the steam then exhausts from the turbine to the suppression pool. The RCIC system can serve as a vent path during hot standby or during reactor isolation.

The reactor vessel top head vent line can also be used to direct steam and noncondensable gases from the reactor upper dome to the suppression pool. Two Class 1E divisionally-powered motor-operated valves (MOVs) are located in series on this line. These MOVs are operated remote manually from the main control room. The reactor vessel top head vent line can be operated over

for concrete, four times the design pressure for steel, and -5 psig for all containments.

Design and qualification criteria are outlined in 10CFR50 Appendix A.

Measurement and indication capability shall extend to 5 psia for subatmospheric containments.

Two or more instruments may be used to meet requirements. However, instruments that need to be switched from one scale to another scale to meet the range requirements are not acceptable.

Continuous display and recording of the containment pressure over the specified range in the control room is required.

The accuracy and response time specifications of the pressure monitor shall be provided and justified to be adequate for their intended function.

II.F.1.5 Containment Water Level Monitor A continuous indication of containment water level shall be provided in the control room for all plants. A narrow-range instrument shall be provided for PWRs and cover the range from the bottom to the top of the containment sump. A wide-range instrument shall also be provided for PWRs and shall cover the range from the bottom of the containment to the elevation equivalent to a 600,000-gal capacity. For BWRs, a wide-range instrument shall be provided and cover the range from the bottom to 5 ft above the normal water level of the suppression pool.

The containment wide-range water level indication channels shall meet the design and qualification criteria as outlined in 10CFR50 Appendix A. The narrow-range channel shall meet the requirements of RG 1.89.

The measurement capability of 600,000 gal is based on recent plant designs. For older plants with smaller water capacities, licensees may propose deviations from this requirement based on the available water supply capability at their plants.

Narrow-range water level monitors are required for all sizes of sumps, but are not required in those plants that do not contain sumps inside the containment.

For BWR pressure-suppression containments, the ECCS suction line inlets may be used as a starting reference point for the narrow-range and wide-range water level monitors, instead of the bottom of the suppression pool.

The accuracy requirements of the water level monitors shall be provided and justified to be adequate for their intended function.

II.F.1.6 Containment Hydrogen Monitor A continuous indication of hydrogen concentration in the containment atmosphere shall be provided in the control room. Measurement capability shall be provided over the range of 0 to 10 percent hydrogen concentration under both positive and negative ambient pressure. Design and qualification criteria are outlined in 10CFR50 Appendix A.

The continuous indication of hydrogen concentration is not required during normal operation. If an indication is not available at all times, continuous indication and recording shall be functioning within 30 min of the initiation of safety injection.

The accuracy and placement of the hydrogen monitors shall be provided and justified to be adequate for their intended function.

Nine Mile Point Unit 2 Position

Human Factor Analysis

For a human factor analysis of the displays and controls of the main control room, refer to Task I.D.1 of this section.

II.F.1.1 Noble Gas Effluent Radiological Monitor Unit 2 has two effluent gaseous release paths: the main stack and the reactor building/radwaste building vent. The main stack receives input from SGT, turbine building ventilation, turbine generator gland seal exhaust, offgas, and mechanical vacuum pump exhaust. The reactor building/radwaste building vent receives input from radwaste building ventilation and normal reactor building ventilation (above and below the refueling floor). Each exhaust path is monitored by an off-line gaseous, particulate, and iodine isotopic radiological monitor, equipped with a high-purity germanium detector. The monitor meets all requirements of NUREG-0737 with a range that meets RG 1.97. Calibration of these monitors is performed using calibration sources which are traceable to the National Institute of Standards and Technology (NIST). In order to determine the particulate and iodine detector efficiencies, sources whose spectrum is distributed over the energy band are used. These sources yield the detector efficiencies directly. In the case of the gas detector, numerous spectra are measured with calibrated point sources at precisely-located positions. Absolute detection efficiencies are obtained at 11 gamma-ray energies for each position. Calculations are then employed to obtain the average efficiencies. These values are subsequently verified using a gas sample of known isotopic content.

The efficiency data determined for the particulate iodine and gas channels are entered manually in the system's computer, which determines detector efficiency curves for each station. Those curves cover the range of energies expected from the emissions of radionuclides collected and analyzed at each channel. Through

the use of a multichannel analyzer in conjunction with each detector, the system is capable of identifying a specific isotope and the amount of that isotope present in a sample containing different radionuclides by collecting data at various discrete energy levels. A detailed description of calibration procedures for these monitors can be found in the Unit 2 operations and maintenance manuals. The monitor is capable of functioning both during and following an accident and provides continuous monitoring of high-level postaccident releases of radioactive noble gases from the plant. See Section 11.5.2.1.1 for further information on these monitors.

II.F.1.2 Sampling and Analysis of Plant Effluents The on-line gaseous, particulate, and iodine isotopic radiological monitor provides continuous sampling of plant gaseous effluent for postaccident releases of radioactive iodines and particulates. The monitor provides the capability for sampling and analysis of effluent without personnel interface. If required for supplemental analysis, or if the isotopic monitor is out of service, the present onsite lab at Unit 1 will have the necessary capability to provide for appropriate facilities.

II.F.1.3 Containment High-Range Radiation Monitor Unit 2 has four high-range in-containment radiation monitors. The range meets the requirements of RG 1.97. The monitors are located 90 deg apart on el 261 ft. These monitors are Category I and are powered via divisional instrument buses. Monitor readouts are displayed continuously and recorded in the main control room.

II.F.1.4 Containment Pressure Monitor Unit 2 instrumentation is provided to monitor containment (drywell) pressure over the range 0-150 psig in accordance with RG 1.97. These instruments are Category I and powered via divisional instrument buses. The pressure is continuously displayed and recorded in the main control room.

II.F.1.5 Containment Water Level Monitor Unit 2 instrumentation is provided to monitor the suppression pool water level from the bottom of the ECCS suction line to 17 ft above normal water level. This range is in accordance with RG 1.97. These instruments are Category 1 and are powered via divisional electrical buses. The containment water level is indicated continuously in the main control room.

II.F.1.6 Containment Hydrogen Concentration Monitor Hydrogen concentration is continuously monitored in the main control room. Sample trees inside the primary containment provide for representative samples. This system consists of two Category I systems. The range of the monitor is 0 to 30 percent in accordance with RG 1.97. These instruments are powered via divisional instrument buses.

II.F.2 INADEQUATE CORE COOLING

FSAR Cross-Reference

Section 4.4

NUREG-0737 Position

Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement existing instrumentation (including primary coolant saturation monitors) in order to provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC). A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

Previous guidance on the design and qualification criteria for upgrading of existing instrumentation was based on RG 1.97, which is still being developed. Detailed design requirements for in-core thermocouples and additional instrumentation were not specified. The pertinent portions of draft RG 1.97 have now been included as Appendix A. Design requirements for in-core thermocouples used in the ICC monitoring system are specified in Attachment 1. The only significant change in design requirements involves a relaxation of qualification requirements for display systems amenable to computer processing. This facilitates procurement of computer systems and makes feasible the use of CRT displays that may be needed for proper interpretation of some reactor-water-level systems under development. This relaxation can be accomplished without compromise of ICC monitoring reliability by requiring 99-percent availability for the display systems, by requiring postaccident maintenance accessibility for nonredundant portions of the system, and by relying on diverse methods of ICC monitoring that include completely qualified display systems.

Design of new instrumentation should provide an unambiguous indication of ICC. This may require new measurements or a synthesis of existing measurements that meet design criteria (Item 7). The evaluation is to include reactor water level indication.

Licensees and applicants are required to provide the necessary design analysis to support the proposed final instrumentation system for ICC and to evaluate the merits of various instruments to monitor water level and to monitor other parameters indicative of core cooling conditions.

The indication of ICC must be unambiguous in that it should have the following properties:

TABLE 1.11-1 (Cont'd.)

DER	Deviation/Event Report
DER	Double-ended rupture
DFFR	Dynamic Forcing Functions Information Report
DFG	Diode function generator
DFW	Floor and equipment drains system
DG	Diesel generator
DLF	Dynamic load factor
DOE	Department of Energy
DOP	Dioctylphthalate
DOT	Department of Transportation
DRMS	Digital radiation monitoring system
DTUC	Digital transponding ultrasonic calibrator
DZO	Depleted zinc oxide
EAB	Exclusion area boundary
ECA	Engineering change authorization
ECCS	Emergency core cooling system
ECN	Engineering change notice
ECP	Electrochemical potential
EDG	Emergency diesel generator
EFCV	Excess flow check valve
EFPY	Effective full-power years
EGA	Air startup standby diesel generator system
EGF	Diesel generator fuel oil system
EHC	Electrohydraulic control
EIC	Energy Information Center
ELLLA	Extended load line limit analysis
ELLLR	Extended load line limit region
EMC	Electromagnetic compatibility
EMS RTU	Emergency management system remote terminal unit
EOC	End of cycle
EOF	Emergency Operations Facility
EOF	Equivalent occurrence factor
EOL	End of life
EOP	Emergency operating procedure
EPA	Electric protective assembly
EPDM	Ethylene-propylene-diene-monomer
EPG	Emergency procedure guideline
EPRI	Electric Power Research Institute
EPZ	Emergency planning zone
EQD	Environmental qualification document
EQEDC	Equipment Qualification Environmental Design Criteria
ER-OLS	Environmental Report-Operating License Stage
ERF	Emergency response facility
ESEERCO	Empire State Electrical and Energy Research Corporation
ESF	Engineered safety feature
EST	Eastern standard time
ESW	Extremely severe weather

TABLE 1.11-1 (Cont'd.)

ETS	Emergency trip system
EWEF	Each way each face
FA	Fire area
FA	Full arc (mode of TCV operation)
FAS	Fluid actuator system
FATT	Fracture appearance transition temperature
FC	Foot-candle
FCB	Flood control berm
FCD	Functional control diagram
FCV	Flow control valve
FDDR	Field deviation disposition request
FHA	Fire Hazards Analysis
FLC	Fuel loading chamber
FLECHT	Full-length emergency cooling heat transfer
FMEA	Failure modes and effects analysis
FMH	Fixture mounting height
FOA	Forced-oil air
FPAPDR	Full power adjusted power density ratio
FPCC	Fuel pool cooling and cleanup
FPQAP	Fire protection quality assurance program
FPS	Fire protection system
FRI	Fuel reliability indicator
FRS	Floor response spectra
FSA	Fire subarea
FSAR	Final Safety Analysis Report
FWCF	Feedwater flow controller failure
FWS	Feedwater system
FZ	Fire zone
GAF	Gain adjustment factor
GDC	General Design Criterion
GE	General Electric Company
GE I&SE	GE Installation & Service Engineering
GE-LSTG	GE-Large Steam Turbine Generator
GE-NEO	GE-Nuclear Energy Operations
GEMS	Gaseous effluent monitoring system
GETAB	GE thermal analysis basis
GGNS	Grand Gulf Nuclear Station
GL	Generic Letter
GRBA	Growth rate based algorithm
GSN	Nitrogen system
GTAW	Gas tungsten arc weld
GTS	Standby gas treatment system
HAZ	Heat-affected zone
HCS	Hydrogen control system
HCU	Hydraulic control unit
HDFM	Heavy density fill material
HELB	High-energy line break

TABLE 1.11-1 (Cont'd.)

HEM	Homogeneous equilibrium model
HEO	Human engineering observation
HEPA	High-efficiency particulate air/absolute (filter)
HEPCO	Hydro-Electric Power Commission of Ontario
HHL	High-high limit
HL	High limit
HPCI	High-pressure coolant injection
HPCS	High-pressure core spray
HPU	Hydraulic power unit
HT	Tritiated gas
HTO	Tritiated oxide
HVAC	Heating, ventilating, and air conditioning
HVC	Control building ventilation system
HVK	Control building chilled water system
HVP	Diesel generator building ventilation system
HVR	Reactor building ventilation system
HVW	Radwaste building ventilation system
HVY	Yard structures ventilation system
HWC	Hydrogen water chemistry system
HX	Heat exchanger
I&C	Instrumentation & control
IAC	Interim acceptance criteria (NRC)
IAS	Instrument air service
IBA	Intermediate break accident
ICC	Inadequate core cooling
ICF	Increased core flow
ICS	Reactor core isolation cooling system
ID	Inner diameter
IDC	Incident detection circuitry
IDS	Instrument data sheet
IE	Office of Inspection and Enforcement (NRC)
IED	Instrument and electrical drawing
IGSCC	Intergranular stress corrosion cracking
IJC	International Joint Commission
ILRT	Integrated leakage rate test
INPO	Institute of Nuclear Power Operations
IOP	Interim operating procedure
IPCEA	Insulated Power Cables Engineers Association
IPE	Individual plant examination
IRM	Intermediate range monitor
ISC	Nuclear boiler instrumentation system
ISEG	Independent Safety Engineering Group
ISI	Inservice inspection
ISPT	Inservice pressure test
ISRM	International Society for Rock Mechanics
IST	Inservice testing
JAERI	Japanese Atomic Energy Research Institute

TABLE 1.11-1 (Cont'd.)

KTG	Karlstein
KWU	Kraftwerk Union

TABLE 1.11-1 (Cont'd.)

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TABLE 1.11-1 (Cont'd.)

LCO	Limiting condition of operation
LCR	Logarithm of the count rate
LCS	Leakage control system
LDS	Leak detection system
LEFM	Leading edge flow meter
LER	Licensee Event Report
LFC	Loaded fuel cell
LFMG	Low-frequency motor generator
LGS	Limerick Generating Station
LHGR	Linear heat generation rate
LL	Low limit
LLL	Low-low limit
LLS	Low-low set
LMS	Leakage monitoring system
LOCA	Loss-of-coolant accident
LOFW	Loss of feedwater
LOOP	Loss of offsite power
LPAP	Low power alarm point
LPCI	Low-pressure coolant injection
LPCS	Low-pressure core spray
LPDS	Loose parts detection system
LPEAC	Loose part event analysis computer
LPMS	Loose parts monitoring system
LPRM	Local power range monitor
LPSP	Low power setpoint
LPZ	Low population zone
LRBPF	Load rejection with bypass failure
LSA	Low specific activity
LSCS	LaSalle County Station
LSD	Lake survey datum (of 1935)
LSMT	Lowest service metal temperature
LSSS	Limiting safety system setting
LTC	Load tap changing (mechanism)
LTM	Low-trajectory missile
LUFC	Loaded uncontrolled fuel cell
LWR	Light-water reactor
LWS	Liquid radwaste system
M&TE	Measuring and testing equipment
MAPLHGR	Maximum average planar linear heat generation rate
MBA	Misplaced bundle accident
M/CC	Maintenance and calibration communication (system)
MCC	Motor control center
MCPR	Minimum critical power ratio
MDAS	Meteorological data acquisition system
MDR	Maximum decay ratio
MG	Motor generator
MLD	Mean low water datum

TABLE 1.11-1 (Cont'd.)

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	Main steam line
msl	Mean sea level
MSLB	Main steam line break
MSR	Moisture-separator reheater
MSS	Main steam system
MTBE	Mean time between events
MTV	Mechanical trip valve
MVD	Multi-vendor data
MWL	Maximum working load
MWR	Maintenance Work Request
MWS	Makeup water system
NB	Nuclear boiler
NBR	Nuclear boiler rated (power)
NDE	Nondestructive examination
NDL	Nuclear data link
NDT	Nil ductility transition
NDT	Nondestructive testing
NDTT	Nil ductility transition temperature
NED	Nuclear energy division (GE)
NEG	GE Nuclear Energy Group
NEMA	National Electrical Manufacturers Association
NFP	NUTMEG fuel preserve
NFPA	National Fire Protection Association
NIOSH	National Institute for Occupational Safety and Health
NIST	National Institute of Standards and Technology
NMPC	Niagara Mohawk Power Corporation
NMS	Neutron monitoring system
NOAA	National Oceanic and Atmospheric Administration
NPCC	Northeast Power Pool Coordination Council
NPRDS	Nuclear plant reliability data system
NPSH	Net positive suction head
NRC	Nuclear Regulatory Commission
NRV	Nonreturn valve
NSOA	Nuclear safety operational analysis
NSS	Nonnuclear safety
NSSS	Nuclear steam supply system
NS^4	Nuclear steam supply shutoff system
NUMAC	Nuclear measurement analysis and control
NUMAC RWM	Nuclear measurement analysis and control rod worth minimizer

TABLE 1.11-1 (Cont'd.)

NVLAP	National Voluntary Laboratory Accreditation Program
NWS	National Weather Service
NYPA	New York Power Authority
NYPP	New York Power Pool
NYSEG	New York State Electric & Gas
NYSERDA	New York State Energy and Resource Development Agency
OBE	Operating basis earthquake
OEA	Operating experience assessment
OFI	Oxygen feedwater injection
OFS	Orificed fuel support
OJT	On-the-job training
OL	Operating license
OLMCPR	Operating limit minimum critical power ratio
OOS	Out of service
OPRM	Oscillation power range monitor
ORE	Occupational radiation exposures
OSC	Operational Support Center
OSHA	Occupational Safety and Health Administration
OT	Operational transient
PA	Public address (system)
PAM	Postaccident monitoring
PASS	Postaccident sampling system
PBA	Period based algorithm
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
PIU	Process interface unit
PLSMT	Permissible lowest service metal temperature
PLU	Power load unbalance
PMF	Probable maximum flood
PMP	Probable maximum precipitation
PMS	Performance monitoring system
PMS	Probable maximum surge
PMW	Probable maximum wind
PMWS	Probable maximum windstorm
PORC	Plant Operations Review Committee
PORV	Power-operated relief valve
PP/PA	Page party/public address (system)

TABLE 1.11-1 (Cont'd.)

PQL	Product quality checklist
PRA	Probabilistic Risk Assessment
PRM	Power range monitor
PRMS	Process radiation monitoring system
PRNM	Power range neutron monitor
PSAM	Pool swell analytical model
PSAR	Preliminary Safety Analysis Report
PSD	Power spectrum density
PSTF	GE Pressure Suppression Test Facility
PSTG	Plant-specific technical guideline
P-T	Pressure-temperature
PT	Inservice pressure test
PTPO	Project test program objectives
PVC	Polyvinylchloride
PVS	Plant vent stack
PWR	Pressurized water reactor
QA	Quality assurance
QATR	Quality Assurance Topical Report
QC	Quality control
RAB	Restricted area boundary
RBCLCW	Reactor building closed loop cooling water (system)
RBM	Rod block monitor
RBPC	Reactor building polar crane
RCA	Radiologically-controlled area
RCIC	Reactor core isolation cooling
RCPB	Reactor coolant pressure boundary
RCS	Reactor coolant system
RCSCM	RHR containment spray cooling mode
RDAS	Remote data acquisition system
RDCS	Rod drive control system
RDS	Control rod drive system
RFM	Radwaste fault movement
RFP	Reactor feed pump
RG	Regulatory Guide
RH	Relative humidity
RHR	Residual heat removal
RHS	Residual heat removal system
RMCS	Reactor manual control system
RMS	Radiation monitoring system
RMS	Root mean square
RO	Reactor Operator
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

TABLE 1.11-1 (Cont'd.)

RRS	Required response spectrum
RSCM	RHR reactor shutdown cooling mode
RSCS	Rod sequence control system
RSO	Reactor system outline
RSP	Remote shutdown panel
RSPCM	RHR suppression pool cooling mode
RSS	Remote shutdown system
RTD	Resistance temperature detector
RT_{NDT}	Reference temperature nil ductility transition
RTT	Response time testing
RWCU	Reactor water cleanup
RWE	Rod withdrawal error
RWM	Rod worth minimizer
RWP	Radiation work permit
SACF	Single active component failure
SAP	Site administrative procedure
SAR	Safety analysis report
SAS	Service air system
SBA	Small break accident
SBO	Station blackout
SCA	Single-channel analyzer
SCBA	Self-contained breathing apparatus
SCEW	System component evaluation work
SDIV	Scram discharge instrument volume
SDM	Shutdown margin
SDV	Scram discharge volume
SEF	Single equipment failure
SER	Safety Evaluation Report
SFC	Spent fuel pool cooling and cleanup system
SGTS	Standby gas treatment system
SIL	Service Information Letter
SIM	Safety isolation module
SJAE	Steam jet air ejector
SLC	Standby liquid control
SLO	Single-loop operation
SLS	Standby liquid control system
SMAW	Shielded metal arc weld
SMSA	Standard metropolitan statistical area
SOE	Sequence of events
SOE	Single operator error
SOF	Single operator failure
SOP	Special operating procedure
SORC	Station Operations Review Committee
SORV	Stuck-open relief valve
SOV	Solenoid-operated valve
SPC	Sound-powered communication (system)
SPCM	Suppression pool cooling mode
SPDS	Safety parameter display system

TABLE 1.11-1 (Cont'd.)

SPG	Substitute position generator
SQRT	GE Seismic Qualification Review Team
SQUG	Seismic Qualification Utility Group
SRAB	Safety Review and Audit Board
SRCAS	Safety-related control air systems
SRDI	Safety-related display instrumentation
SRLR	Supplemental Reload Licensing Report
SRM	Security-related materials
SRM	Source range monitor
SRO	Senior Reactor Operator
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
SSA	Safe Shutdown Analysis
SSDS	Safe shutdown system
SSE	Safe shutdown earthquake
SSES	Susquehanna Steam Electric Station
SSS	Station Shift Supervisor
SSW	Radwaste sampling system
STA	Shift Technical Advisor
STRIDE	Standard Reactor Island Design
SVV	Main steam safety relief system
SW	Severe weather
SWEC	Stone & Webster Engineering Corporation
SWP	Service water system
SWR	Radwaste seal water system
TAF	Top of active fuel
TBCLCW	Turbine building closed loop cooling water
TCV	Turbine control valve
TDH	Total developed head
TG	Turbine generator
TIP	Traversing in-core probe
TLD	Thermoluminescent dosimeter
TLTA	Two-loop test apparatus
TME	Gland seal and exhaust steam system
TMI	Three Mile Island
TNDT	Nil ductility transition temperature
TRM	Technical Requirements Manual
TRS	Test response spectrum
TSC	Technical Support Center
TSD	Training System Development
TSS	Temperature sensor/switch
TSVC	Turbine stop valve closure
TVD	Test, vent and drain
UBC	Uniform Building Code
UFC	Uncontrolled fuel cell

TABLE 1.11-1 (Cont'd.)

UHS	Ultimate heat sink
UL	Underwriters' Laboratories Inc.
Unit 1	Nine Mile Point Nuclear Station - Unit 1
Unit 2	Nine Mile Point Nuclear Station - Unit 2
UPS	Uninterruptible power supply
URC	Ultrasonic resin cleaning
U.S.	United States
USAR	Updated Safety Analysis Report
USBM	U.S. Bureau of Mines
USGS	U.S. Geological Survey
USI	Unresolved safety issue
USPHS	U.S. Public Health Service
USLS	U.S. Land Survey
UT	Ultrasonic testing
UTM	Universal Transverse Mercator
V&V	Verification and validation
VWO	Valve wide open
WCS	Reactor water cleanup system
WPPSS	Washington Public Power Supply System
WSS	Water solidification system
WTS	Water treatment system
ZIP	Zinc injection passivation
ZPA	Zero period asymptote
4TCO	Temporary tall test tank-condensation oscillation

Position

The absolute sum method of combining dynamic loads is used for the design of structures. The details of load combinations used in designing the structures are covered in FSAR Section 3.8.

The Unit 2 primary containment liner is evaluated for fatigue to the requirements of ASME Boiler and Pressure Vessel Code Section III, Division 1, Subsection NE.

LICENSING ISSUE: 43 - FLUID/STRUCTURE INTERACTION

Issue

The dynamic forcing functions for various loads have been established through testing on models that are generally more stiff than the actual structures to which the loads will be applied. By directly applying such forcing functions to actual structures in the analysis, the interactive effect between the fluid mass and the structure is neglected. Under certain conditions this effect may be significant. It is proposed that a generic approach to study such effects should be established.

Position

This issue is not directly applicable to the Unit 2 Mark II containment. Since the Unit 2 containment is stiff in the suppression pool region and the dynamic forcing functions are conservatively defined, any interactive effect between the fluid mass and the structure is inherently included.

LICENSING ISSUE: 44 - LONG-TERM POST-LOCA OPERABILITY OF DEEP-DRAFT ECCS PUMPS

Issue

IE Bulletin 79-15, dated July 1979, identified problems with deep-draft ECCS pumps that could threaten their long-term post-LOCA operability. Structure flexibility; shaft/column misalignment; vibrational frequencies near rotation speeds; inlet flow-induced vortices; and dimensional deficiencies such as those discovered with certain LaSalle ECCS pumps, could cause excessive vibration and bearing wear. The NRC staff has asked applicants to define programs and provide data that compare the expected service life with the accumulated operating time and confirm the long-term operability.

Position

There are five safety-related deep-draft pumps utilized in the Unit 2 design. These are three RHR pumps, one LPCS pump, and one HPCS pump.

The inherent design features of the Byron Jackson ECCS pumps in Unit 2 preclude excessive vibration and bearing wear. Each pump is supplied with a casing or suction barrel and is not installed in a wet sump. They do not have long, limber columns; the longest pump is only 24 ft, compared to the 30- to 160-ft pumps described in IE Bulletin 79-15. Also the pump assembly rigidity is enhanced by seismic rings between the assembly and the barrel. The pumps use a double-suction first stage to provide stability over a wide range of flows. Column frequencies are well removed from pump speed. Larger-diameter barrels provide low-flow velocities around pump inlets, and ring seismic restraints act as flow straighteners to suppress vortex formation. The pumps have high-precision, keyed, sleeve-type couplings.

A NRC letter dated December 6, 1983, requested additional information on methods used to qualify long-term operability of deep-draft pumps. The following is a description of the method used.

Long-term operability has been considered in the ECCS pump design. The ECCS pumps' effectiveness is evaluated by acceptance, qualification, and in-plant testing. Long-term operability is assured by preventive maintenance, inservice testing (IST) and surveillance, and vibration monitoring. Scheduled preventive maintenance consists of resistance readings of motor windings; lubrication of critical rotating components; general cleaning and inspection of rotating electrical equipment; and inspection, overhaul, alignment, and adjustment of impeller lift. IST measurements of each pump's differential pressure, flow rate, and vibration, as prescribed by the Code of record for the IST program for pumps and valves, as required by 10CFR50.55a, provide data for engineering analysis to identify performance changes or trends. In addition, vibration data bases, established during the preoperational/startup testing, are compared with functional-testing vibration data to monitor journal bearing wear and shaft whip.

IST and surveillance requirements are specified in Unit 2 Technical Specifications, surveillance procedures, and IST programs. Preventive maintenance and surveillance testing are scheduled at periodic intervals as the IST program test results indicate.

As part of the Unit 2 plant IST program, vibration measurements will be taken in accordance with the Code of record for the IST program for pumps and valves, as required by 10CFR50.55a. The data will be evaluated on a scheduled basis to predict potential bearing and journal failures and establish replacement schedules. Data will be available onsite for inspection.

Vibration limits shall be in accordance with the Code of record for the IST program for pumps and valves, as required by 10CFR50.55a.

Measurements will be performed using an IRD 360 or equivalent equipment, when the motor and pump are operated as a unit over the normal design range of pressure and flow. This limit is based on normal operation. Higher momentary increases may be acceptable during starting or at shutoff. This limit is not based on what the equipment can withstand. The equipment damage threshold is higher: close to 0.020 in, momentary and not sustained.

These deep-draft pumps, due to their relative shortness, demonstrate fewer of the problems associated with longer pumps. The hydraulic design has been developed over the last 40 yr of experience in many applications.

The ECCS pumps contain design features to preclude failure of the impellers, impeller staking, shafts, bearings, wear rings, couplings, and stuffing boxes. The design includes safety factors (loading criteria) based on the expected pressures, temperatures, and loadings defined in the design specification. Lateral restraints are included in the pump to control deflections. Tolerances assuring alignment of the shaft and pumping elements are verified by design calculations. Motor shaft deflections within tolerance are predicted in a static seismic analysis and are verified by a qualification test of a similar motor. A dynamic analysis of the pump and motor is performed to determine resonances and predict loadings throughout the pump and motor.

Tests are performed on each pump delivered. The tests include head versus flow, NPSH, and vibration monitoring. The assembled pumps are checked for proper assembly and low friction by hand turning (rotating) the shaft. Each pump is run for a total of 100 hr during testing. A qualification test of a similar pump motor was performed. This data provides qualification of the Unit 2 pumps motors by a similarity analysis. During the First Ten-Year Interval, pump testing was performed in accordance with Section XI, 1983 Edition through the Summer 1983 Addenda; ASME OM-1987 through the OMa-1988 Addenda, Part 6; and the IST program plan. During the Second Ten-Year Interval and subsequent ten-year intervals, pump testing will be conducted in accordance with the requirements of 10CFR50.55a and the IST program plan.

LICENSING ISSUE: 45 - REPLACE HIGH DRYWELL PRESSURE INTERLOCK ON HPCS TRIP CIRCUITRY WITH LEVEL-8 TRIP TO PREVENT MAIN STEAM LINE FLOODING

Issue

Some designs included an interlock that prevented shutoff of the flow of the HPCS at high water level (8) in the reactor vessel when a high drywell pressure signal was present. Applicants were requested to remove this interlock.

Position

No such system interlock exists in Unit 2.

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LICENSING ISSUE: 46 - ADDITIONAL LOCA BREAK SPECTRUM

Issue

The NRC staff has requested applicants to provide the following additional LOCA analyses to complete the break spectrum:

1. An additional recirculation line break with a discharge coefficient 0.6 times the design bases accident, using the large-break model analysis.

2. An additional recirculation line break with a 0.02-sq ft area, using the small-break model analysis.

Position

The adequacy of the LOCA break spectrum is addressed in Section 6.3.3. Representative analyses done for the LaSalle plant supported by confirmatory plant-unique Appendix K calculations have been found acceptable to the NRC staff without further commitment.

LICENSING ISSUE: 47 - LOCA ANALYSES WITH CLOSURE OF THE RECIRCULATION FLOW CONTROL VALVE

Issue

The ECCS analyses described in Section 6.3 assume the nonsafety-grade, recirculation flow control valve (FCV) locks at its existing position during the LOCA. The NRC staff has requested applicants to provide a discussion of the effects on the analyses if it is assumed the FCV closes at a realistic rate, and of the probability the FCV will fail in this manner.

Position

Using the standard, approved licensing models and an assumed FCV closure rate of 11 percent per second, generic BWR/5 analyses showed an increase in the peak cladding temperature (PCT) of 45°F. The generic ECCS calculations applicable to Unit 2 yield a PCT that can accommodate this increase without violating the 2200°F limit of 10CFR50.46. It is expected that when the Unit 2-unique calculations are completed, the calculated PCT will also be able to accommodate the 45°F increase of FCV closure.

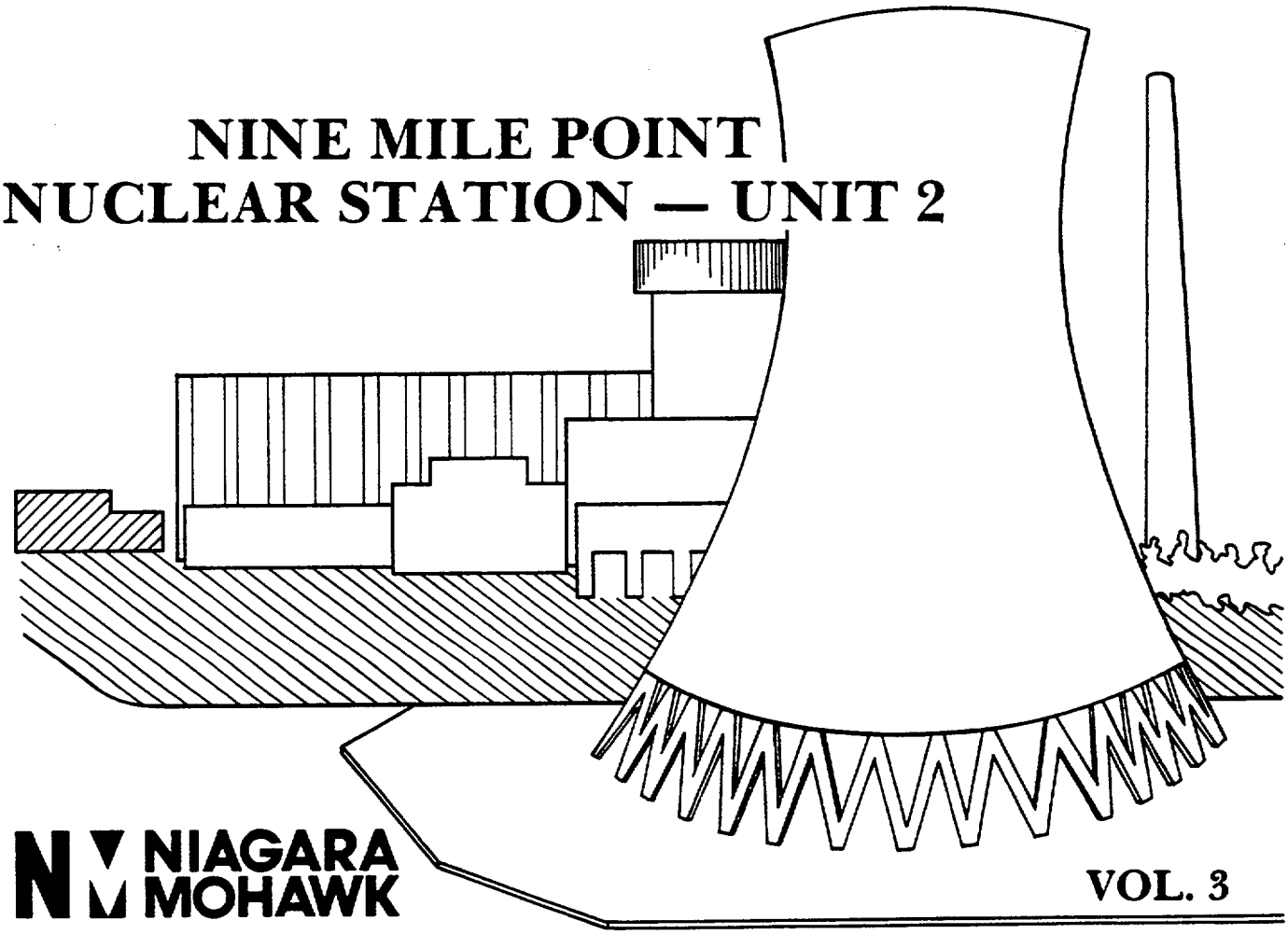
LICENSING ISSUE: 48 - ADEQUATE CORE COOLING MAINTAINED WITH LPCI DIVERSION

Issue

The NRC staff has asked applicants for a demonstration that adequate core cooling would be maintained if the flow of the LPCI

UPDATED SAFETY ANALYSIS REPORT

NINE MILE POINT NUCLEAR STATION — UNIT 2



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relatively high velocity of the air flowing through the tower. The entrained water droplets, called cooling tower drift, are carried from the tower and, subsequently, fall to the ground downwind from the tower.

In order to determine the environmental effects of the cooling tower drift, a mathematical model was developed to determine the downwind distribution of salt and water deposition and the airborne salt concentration resulting from cooling tower operation. A detailed description of the model and results is contained in Appendix 2D. The model takes the following into account:

1. Configuration and performance of the towers.
2. Drift rate.
3. Exit velocity.
4. Total dissolved solids (TDS) level.
5. Droplet size distribution.
6. Evaporation rate.
7. Plume buoyancy.
8. Wind speed.
9. Wind direction.
10. Wet-bulb temperature.
11. Relative humidity.

The maximum amount of drift leaving the cooling tower is assumed to be 0.002 percent of the circulating water flow through the tower. Monthly average TDS concentrations in the blowdown and 3 yr of onsite, hourly average meteorological data (January 1, 1974, through December 31, 1976) were used as input to the salt drift model. Since actual TDS levels in the cooling tower are approximately 2 times higher than those assumed in the drift model, salt deposition and airborne salt concentration results presented in this section and on Figures 2.3-26 and 2.3-28 through 2.3-39 are correspondingly low.

The meteorological input data used in the model consisted of wind speed, wind direction, dry-bulb temperature, wet-bulb temperature, and relative humidity at the 61-m (200-ft) level. The difference between the dry-bulb temperature at 61 m (200 ft) and at 8 m (27 ft) (ΔT) was also used. Normally, the low-level relative humidity would be used to determine tower performance, but due to the large amount of missing data for this parameter, the upper level relative humidity was chosen. A comparison of

the relative humidities at these two levels showed an average difference of only 4.6 percent, which has little effect on the salt drift model results. The results of a sensitivity test of the drift model to relative humidity, using 1 month (December

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1974) of meteorological data, showed an 11-percent decrease in the maximum salt deposition rate and an 8.7-percent decrease in the maximum water deposition rate by using the 61-m (200-ft) relative humidity in place of the 9-m (30-ft) relative humidity.

There was also a substitution of the 30-m (100-ft) wind direction when the 61-m (200-ft) wind direction was missing to ensure that a high percentage of data was used. This practice does not significantly affect the salt drift results because of the very small changes in wind direction with height between these levels.

Average annual salt deposition rates in lb/acre/yr are shown on Figure 2.3-26. The maximum salt deposition rate was predicted to be 0.03 g/sq m/yr (0.27 lb/acre/yr), occurring approximately 2,000 m (6,562 ft) northwest of the tower. This location is over water. The maximum salt deposition rate predicted to occur over land is 0.011 g/sq m/yr (0.099 lb/acre/yr) at a distance approximately 990 m (3,248 ft) west-southwest of the tower.

Figure 2.3-27 presents annual water deposition rates in lb/acre/yr with a maximum value of 77.4 g/sq m/yr (690.6 lb/acre/yr) occurring 2,000 m (6,562 ft) northwest of the tower. This amount corresponds to 0.08 mm (0.003 in) of water per year.

Average monthly salt deposition rates in lb/acre/month are shown on Figures 2.3-28 through 2.3-39. Monthly and seasonal water deposition rates are not shown because the maximum annual amount of 0.08 mm is insignificant compared to annual precipitation at the site.

In addition to the drift deposition rates, airborne salt concentrations at ground level were also calculated. The maximum annual average airborne salt concentration was predicted to be 0.83×10^{-6} mg/cu m (5.18×10^{-14} lb/cu ft) at a distance of 2,400 m (7,874 ft) northwest of the tower, and the highest value over land is predicted to be 0.56×10^{-6} mg/cu m (350×10^{-14} lb/cu ft) at 1,067 m (3,500 ft) south of the tower. A value of 1.22×10^{-3} mg/cu m (7.62×10^{-11} lb/cu ft) was predicted for the maximum hourly airborne salt concentration which occurs over the lake at a distance of 500 m (1,640 ft) west-northwest from the tower. The maximum hourly airborne salt concentration over land is predicted to be 1.19×10^{-3} mg/cu m (7.43×10^{-11} lb/cu ft) at a distance of 1,067 m (3,500 ft) west-southwest of the tower.

2.3.2.3.4 Cloud Development and Cloud Shadowing

The extent to which natural-draft cooling tower plumes contribute to cloud formation can be qualitatively assessed based on observational studies conducted at three operating natural-draft cooling tower sites[37]. At each of these sites, cooling tower plumes were observed to occasionally cause broken cloud decks to become overcast and to make thin clouds thicker. Separate cloud formations were sometimes observed to result from visible plume formation from the cooling towers but usually at altitudes of

primary meteorological tower and the backup tower is shown on Figure 2.3-40. Figure 2.3-40A shows the location of the inland tower, the primary tower, and the backup tower with respect to Unit 2 on a larger-scale map.

The primary tower is instrumented with wind direction and speed sensors at three levels: 9 m (30 ft), 30 m (100 ft), and 61 m (200 ft). Sigma theta is derived for each of the three wind levels. In addition, ambient temperature is measured at the 9 m (30 ft) level and temperature differences are determined between the 61-m (200-ft) and 9-m (30-ft) levels. Actual instrument elevations for these levels can be found in plant design documents[74].

Dew point temperature is obtained at the 9-m (30-ft) level. Near the base of the tower, precipitation and barometric pressure are also measured.

The inland 9-m (30-ft) meteorological tower is located with good exposure in all directions and is situated away from all runways and buildings at the Oswego County Airport. The instrumentation provides wind speed and direction from which sigma theta values are calculated.

The backup wind direction and speed instrumentation is located east of the J. A. FitzPatrick plant on a 27-m (90-ft) utility pole. Data collected coincidentally over a three-year period from the main tower and backup tower have been analyzed[72]. Based upon this analysis and with an earlier study[71,73] by Meteorological Environmental Services, Inc., the backup tower measurements are in general agreement with the main tower and are adequate for use during emergency situations.

2.3.3.2.2 Meteorological Instrumentation

The operability of the meteorological monitoring instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public.

The operational meteorological measurements program is designed to meet the NRC recommendations at the time of installation in midsummer 1982 and is in accordance with RG 1.23, February 1972. Manufacturers' model numbers and specifications for the sensors are shown in Table 2.3-4A. Component errors, as well as sensor and system accuracies, are listed in Table 2.3-5A. Accuracy requirements are in accordance with Section C.4 of proposed Revision 1 to RG 1.23 (September 1980).

<u>Wind Instruments</u> All monitoring locations employ the Teledyne Geotech or Met One Instruments three-cup anemometer and vane.

Temperature The ambient and temperature difference systems consist of Teledyne Geotech or Met One Instruments platinum resistance temperature devices in aspirated housings.

Dew Point The dew point temperature is measured by General Eastern chilled mirror system.

Precipitation Solid and liquid forms of precipitation are measured by a Belfort Instrument Company tipping bucket rain gauge with a heater for subfreezing operations.

Pressure A Yellow Spring Instrument Company aneroid barometer measures Station atmospheric pressure.

2.3.3.2.3 Processing, Storage, Display, and Recording of Meteorological Data

Data Processing and Storage

Digital data processing at each meteorological tower is accomplished by a remote data acquisition system (RDAS) computer. These RDAS computers sample each sensor's analog processor at a rate of once per second and process the data into 1-, 15-, and 60-min averages. Averaged data are transmitted via modem to a central processing system (CPS) computer for access and storage. Each RDAS computer is housed in an environmentally-controlled instrument cabinet at the meteorological towers. The CPS computer is housed in an environmentally-controlled meteorological computer building located at the north end of the Unit 1 parking lot.

Display and Recording

Computer terminals are the interface of the meteorological data acquisition system (MDAS) for the display of digital data. Strip chart recorders display and record analog data. One set of strip chart recorders, which display the parameters from all meteorological towers, is located at the meteorological computer building. The control room at Unit 1, Unit 2, and the J. A. FitzPatrick plant have strip chart recorders for the key parameters: the 61-m (200-ft), 27-m (90-ft) backup, and either the 9-m (30-ft) or 30-m (100-ft) wind direction and speed, as well as sigma theta, temperature, and both temperature differences. In addition, the Technical Support Center has strip chart recorders for the 61-m (200-ft) and either the 9-m (30-ft) or 30-m (100-ft) wind direction and speed.

Wind direction, speed, and sigma theta data from the backup J. A. FitzPatrick tower are displayed in both digital and analog form in the Unit 2 control room. These data will be used as backup to the primary 200-ft tower in the unlikely event that data are unavailable from this system.

2.3.3.2.4 Instrumentation Operability/Surveillance Requirements

The meteorological monitoring instrumentation channels shown in Table 2.3-8 and in Technical Requirements Manual (TRM) Section 3.3.10 shall be operable at all times.

Action

Surveillance and calibration schedules are specified to comply with RG 1.23 recommendations. Equipment checks are performed at least weekly. Charts are changed as required. Each of the required meteorological monitoring instrumentation channels shown in Table 2.3-8 shall be demonstrated operable by the performance of a daily channel check (at least once per 24 hr with a maximum allowable extension not to exceed 25 percent of the surveillance interval), and a semiannual channel calibration (at least once per 184 days with a maximum allowable extension not to exceed 25 percent of the surveillance interval). Component checks and adjustments are performed when required. All meters and other equipment used in calibrations are, in turn, calibrated at scheduled intervals.

Inspection and maintenance of all equipment is accomplished in accordance with procedures in the instrument manufacturer's manuals. Inspection is implemented by qualified technicians that are capable of performing the maintenance, if required. The results of the inspections and maintenance performed are recorded in a log book.

2.3.3.2.5 Data Analysis

All data are subject to quality control checks by a meteorologist prior to tabulation of routine summaries of wind direction, speed, and stability. Other analyses are performed as warranted for special projects, in addition to the routine submittal of data for scheduled reports.

2.3.4 Short-Term (Accident) Diffusion Estimates

2.3.4.1 Objective

The objective is to provide conservative and realistic short-term estimates of relative concentration, X/Q, at specific locations such as the EAB and the outer boundary of the LPZ following a hypothetical release of radioactivity from Unit 2. The

assessment is based on the results of atmospheric diffusion modeling and onsite meteorological data.

2.3.4.2 X/Q Estimates

Three possible locations where accidental radionuclide releases can occur from Unit 2 are the tall main plant stack, the combined radwaste/reactor building vent, and the main steam tunnel blowout panels. Accident X/Q values are assessed at the distances and locations shown in Appendix 2F, Table 2F-1. Both conservative and realistic accident results are reported for the EAB and the LPZ. In addition, accident X/Q values are reported for the Visitor Information Center fence as well as the control room air intakes. Conservative X/Q estimates are made for emergency planning due to a hypothetical release of radioactivity from the main stack. Furthermore, realistic estimates of X/Q values due to a hypothetical release from the main stack and the combined radwaste/reactor building vent are given at the population distances. The accident X/Q values are summarized in the following tables:

1. Table 2F-2 Conservative Short-Term Diffusion Estimates at the EAB, LPZ, Visitor Information Center Fence, and the Control Room Air Intakes for Releases from the Main Stack, the Main Steam Tunnel Blowout Panels, and the Combined Radwaste/Reactor Building Vent.

2. Table 2F-2a Conservative Short-Term Diffusion Estimates at the EAB for Releases from the Main Stack - Coastline and Land Sectors, Using 45-deg Sector Width Distances.

3. Table 2F-2b Conservative Short-Term Diffusion Estimates at the EAB for Releases from the Main Stack - Land Sectors, Using 22 1/2-deg Sector Width Distances.

4. Table 2F-2c Conservative Short-Term Diffusion Estimates at the EAB for Releases from the Combined Radwaste/Reactor Building Vent - Coastline Sectors, Using 22 1/2-deg Sector Width Distance; and Land Sectors, using 45-deg Sector Width Distances.

5. Table 2F-3 Realistic Short-Term Diffusion Estimates at the EAB by Sector for Releases from the Main Stack.

6. Table 2F-4 Realistic Short-Term Diffusion Estimates at the LPZ by Sector for Releases from the Main Stack.

7. Table 2F-5 Realistic Short-Term Diffusion Estimates at the Visitor Information Center Fence for Releases from the Main Stack.

8. Table 2F-6 Emergency Planning Short-Term Diffusion Estimates for Releases from the Main Stack.

2.3.5.3.2 Source Configuration

Radionuclides are routinely released in gaseous effluents from two sources: the main stack and the combined radwaste and reactor building vent. Their source characteristics are given in Table 2G-9.

2.3.5.3.2.1 Main Stack Release

The main stack is more than twice as high as all adjacent structures so that no building downwash is included. However, a correction for the stack tip downwash is made in accordance with RG 1.111 when the vertical exit velocity is less than 1.5 times the horizontal wind speed. This correction for stack tip downwash is given by:

$$C = 3D\ (1.5 - W_o/\bar{u}) \qquad (2.3\text{-}11)$$

Where:

C = Downwash correction factor, m

D = Diameter of the release, m

W_o = Vertical exit velocity, m/sec

$\bar{u}$ = Mean wind speed at release height, m/sec

2.3.5.3.2.2 Vent Release

The combined radwaste and reactor building vent pointing upward is in a rectangular structure between the reactor and turbine buildings and is 4 m (13.2 ft) higher than the top of the reactor building. Therefore, the vent is affected by the nearby building aerodynamics with moderate to strong winds and is treated differently than the main stack.

The entrainment coefficients of RG 1.111 are used to determine the portion of the vent's effluent entrained into the turbine-reactor building wake. The entrainment coefficients are given by the following equations:

$$E_T = 2.58 - 1.58(W_o/\bar{u}) \text{ for } 1 < W_o/\bar{u} \leq 1.5 \qquad (2.3\text{-}12)$$

$$E_T = 0.30 - 0.06(W_o/\bar{u}) \text{ for } 1.5 < W_o/\bar{u} \leq 5.0 \qquad (2.3\text{-}13)$$

Where:

E_T = Entrainment coefficient, dimensionless

When entrainment occurs, the entrained position of the release is assumed to be at ground level and a building wake correction factor (reactor building height squared) of 2,685 sq m (28,900 sq

ft) is used in accordance with the methodology of RG 1.111. The building wake correction factor takes into consideration the initial mixing of the plume within the building cavity.

2.3.5.3.2.3 Site Impacts on the Main Stack and Vent Releases

The final consideration of the source configuration is to determine the effects, if any, of the natural-draft cooling tower on the effluent released from the main stack and combined radwaste and reactor building vent. The natural-draft cooling tower is located 454 m (1,490 ft) south of the main stack and 304 m (996 ft) south of the combined radwaste and reactor building vent. The physical dimensions of the natural-draft cooling tower are:

1. Height above grade = 539.36 ft
2. Base diameter = 415 ft 8 in
3. Throat diameter = 259 ft 10 in
4. Exit diameter = 272 ft 11 in

Field data obtained at Rancho Seco, especially during stable conditions, were used to determine the flow perturbations generated by natural-draft cooling towers. The report states, "The overall interpretation of ground-level concentrations (i.e., crosswind integrated concentrations and sigma-y values) are probably not severely distorted even when the observations are influenced by the cooling tower wakes."[44]

Thus, the effects of the natural-draft cooling tower for both the main stack and the combined radwaste and reactor building vent releases during stable conditions are neglected.

The effect of the cooling tower on the main stack or the combined radwaste and reactor building vent releases during neutral and unstable atmospheric conditions would be to enhance the vertical diffusion through increased mechanical turbulence and thus reduce ground-level concentrations. Therefore, to be conservative in the estimation of ground-level concentrations for neutral and unstable conditions, the wake effect of the cooling tower has been neglected.

2.3.5.3.3 Plume Rise

Plume rise is calculated according to the procedures outlined in RG 1.111. For neutral or unstable conditions the following equation is used:

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58. ANSI A58.1, p 100, March 1982.

59. Kramer, M. L. and Halitsky, J. Accident X/Q Values at Category I Fresh Air Intakes and Vital Area Access Zones for Releases from the Main Steam Tunnel, the Combined Radwaste/Reactor Building Vent and the Main Stack at Nine Mile Point Unit 2. Prepared for Niagara Mohawk Power Corporation, January 6, 1984.

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62. DELETED

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68. U.S. Department of Commerce Local Climatological Data - Monthly Summary - Syracuse, New York. NOAA, Environmental Data and Information Service, National Climatic Center, Asheville, NC, 1977 - October 1983.

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TABLE 2.3-7

NORTHEAST STATE SNOWFALL RECORDS ABSTRACTED BY LUDLUM

Location	Storm Duration	Snow Amount (in)
New Haven, CT	24 hr	28.0
Middletown, CT	3 days	50.0
Blue Hill, MA	24 hr	28.2
Peru, MA	4 days	47.0
Randolph, NH	24 hr	56.0
Pinkham Notch, NH	5 days	77.0
Long Branch, NJ	24 hr	29.7
Cape May, NJ	4 days	34.0
Barnes Corners, NY	24 hr	54.0*
Watertown, NY	5 days	69.0
Morgantown, PA	24 hr	38.0
Morgantown, PA	3 days	50.0
St. Johnsburg, VT	24 hr	33.0
Readsboro, VT	5 days	50.0

* Limiting case for deriving the highest 48-hr snowfall.

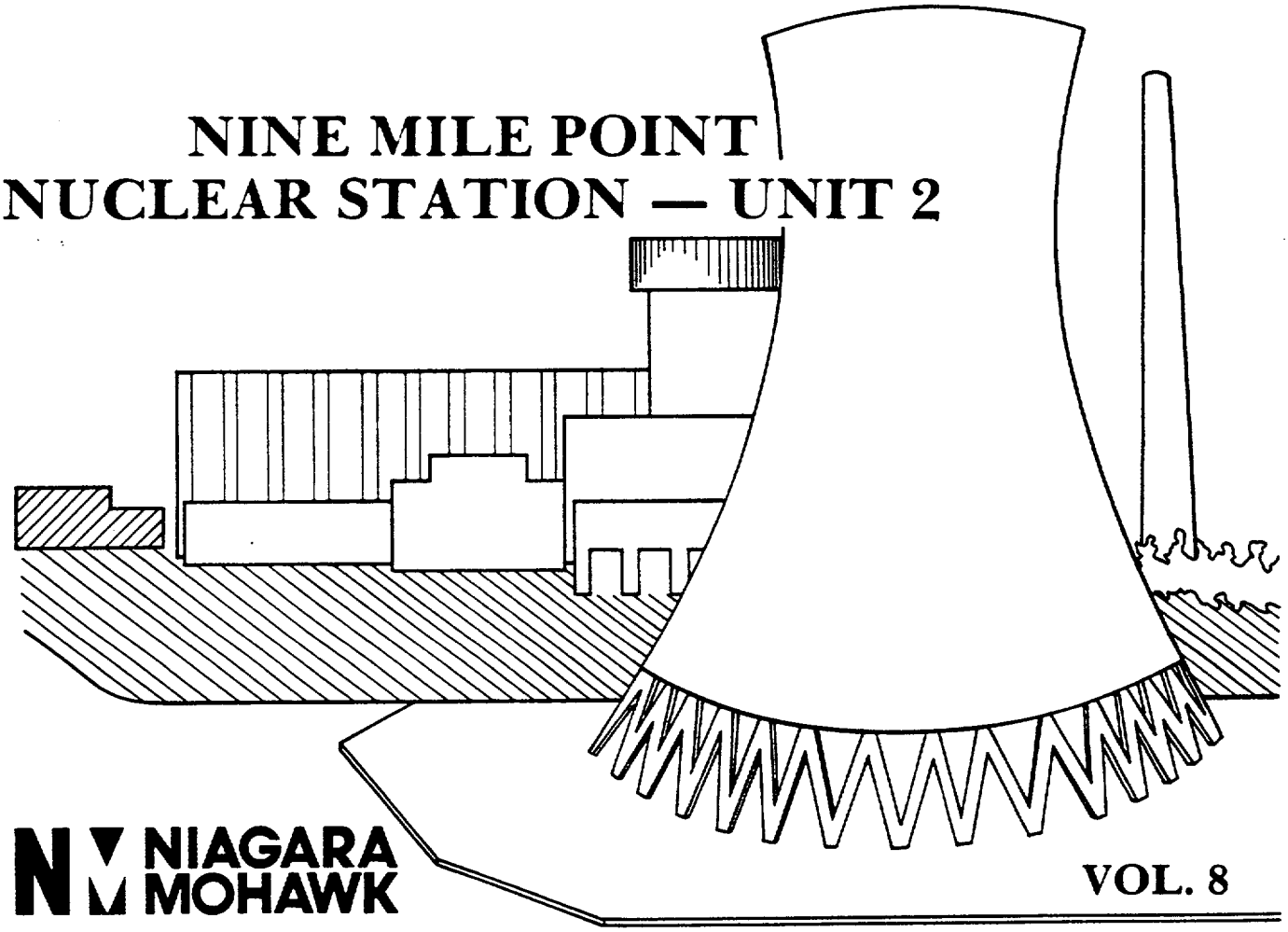
TABLE 2.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

Instrument	Elevation	Minimum Instruments Operable
Wind Speed	30 ft	1
	200 ft	1
Wind Direction	30 ft	1
	200 ft	1
Air Temperature Difference	30/200 ft	1

UPDATED SAFETY ANALYSIS REPORT

NINE MILE POINT NUCLEAR STATION — UNIT 2



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design features built into the CRD system. Response by the RPS is prompt and total scram time is short.

In the unlikely event that more than one control rod fails to insert and the core cannot be maintained in a subcritical condition by the control rods alone as the reactor cools down subsequent to initial shutdown, the SLCS is activated manually to inject soluble boron into the reactor core. The SLCS has sufficient capacity to ensure that the reactor can always be maintained subcritical; hence, only decay heat is generated by the core which can be removed by the RHR system, ensuring that the core is always coolable.

Design of the reactivity control systems assures reliable control of reactivity under postulated accident conditions with appropriate margin for stuck rods. The capability to cool the core is maintained under all postulated accident conditions; thus, Criterion 27 is satisfied.

For further discussion, see the following sections:

Principal Design Criteria	1.2.1
Nuclear Design	4.3
Thermal-Hydraulic Design	4.4
Functional Design of Reactivity Control System	4.6
Reactor Protection (Trip) System	7.2
Control Systems Not Required for Safety	7.7
Accident Analysis	15

3.1.2.28 Reactivity Limits (Criterion 28)

Criterion

"The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the RCPB greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition."

Design Conformance

The control rod system design incorporates appropriate limits on the potential amount and rate of reactivity increase. Control rod withdrawal sequences and patterns are selected to achieve optimum core performance and low individual rod worths. The rod sequence control system (RSCS) prevents withdrawal other than by the preselected rod withdrawal pattern. The RSCS function assists the Operator with an effective backup control rod monitoring routine that enforces adherence to established startup, shutdown, and low-power-level operations control rod procedures.

The control rod mechanical design incorporates a hydraulic velocity limiter in the control rod that prevents rapid rod ejection. This engineered safeguard protects against a high reactivity insertion rate by limiting the control rod ejection velocity to less than 5 fps. Normal rod movement is limited to 6-in increments and the rod withdrawal rate is limited through the hydraulic valve to 3 in/sec.

For Cycle 7, a cycle-specific analysis has been completed for rod withdrawal rates up to 6.0 in per second. For all other cycles, a cycle-generic analysis has been completed for rod withdrawal rates up to 5.0 in per second.

The accident analysis (Chapter 15) evaluates the postulated reactivity accidents, as well as abnormal operational transients. Analyses are included for rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition. The initial conditions, assumptions, calculational models, sequences of events, and anticipated results of each postulated occurrence are covered in detail. Results of these analyses indicate that none of the postulated reactivity transients or accidents result in damage to the RCPB. In addition, the integrity of the core, its support structures, or other reactor pressure vessel internals are maintained so that the capability to cool the core is not impaired for any of the postulated reactivity accidents described in the accident analysis.

The design features of the reactivity control system limit the potential amount and rate of reactivity increase to ensure that Criterion 28 is satisfied for all postulated reactivity accidents.

For further discussion, see the following sections:

Principal Design Criteria	1.2.1
Control Rod Drive Systems	3.9.4.3
Nuclear Design	4.3

Control Rod Drive Housing Supports 4.5.3

Functional Design of Reactivity Control System 4.6

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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
Standby Gas Treatment System								
Filter units, including electrical heating coils	P	M	1E	I	NA	I	P	
Automatic valves	P	M,RB	1E	I	B,C	I	P	(51)
Piping and manual valves, essential	P	M,RB	NA	I	B,C	I	P	(40)(51)
Piping and manual valves, nonessential	P	M,RB	NA	NA	D	NA	P	
All other components, essential	P	M	1E	I	NA	I	P	
All other components, nonessential	P	M,RB	Non-1E	NA	NA	NA	P	
Primary Containment Purge System								
Automatic isolation valves	P	RB	1E	I	B	I	P	
Piping and manual valves, essential	P	RB	NA	I	C	I	P	
Piping and manual valves, nonessential	P	RB	NA	NA	D	NA	P	(50)
All other components, essential	P	RB	Non-1E	I	C	I	P	
All other components, nonessential	P	RB	NA	NA	D	NA	P	
Diesel Generator Systems								
Piping, fuel oil	P	O,S	NA	I	C	I	P	
Valves, fuel oil	P	O,S	1E	I	C	I	P	
Pumps, fuel oil	P	S	NA	I	C	I	P	
Pump motors, fuel oil system	P	S	1E	I	NA	I	P	
Day tanks	P	S	NA	I	C	I	P	
Diesel fuel storage tanks	P	S	NA	I	C	I	P	
Piping, air startup, essential	P	S	NA	I	C	I	P	
Valves, air startup, essential	P	S	1E	I	C	I	P	
Piping, air startup, nonessential	P	S	NA	NA	D	NA	P	
Air dryers	P,GE	S	NA	NA	D	NA	P	
Compressors, air startup	P,GE	S	NA	I	D	NA	P	(47)
Compressor motor	P,GE	S	1E, Non-1E	I(48)	D	NA	P	(47)
Receivers, air startup	P,GE	S	NA	I	C	I	P	(47)
Lube oil cooler	P	S	NA	I	C	I	P	
Piping and valves, cooling water	P	S	NA	I		I	P	(45)
Piping and valves, lube oil	P	S	NA	I	C	I	P	(45)
Pumps, motors	P	S	1E	I	C	I	P	
Standby diesel generators	P	S	1E	I	NA	I	P	
HPCS diesel generator	GE	S	1E	I	NA	I	P	

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
Primary Loop								
Piping	P	RB	NA	NA	D	NA	P	
Valves	P	RB	NA	NA	D	NA	P	
Pumps	P	RB	NA	NA	D	NA	P	
Heat Exchangers	P	RB	NA	NA	D	NA	P	
Pump Motors	P	RB	Non-1E	NA	NA	NA	P	
Secondary Loop								
Secondary Containment Boundary Piping	P	RB	NA	I	C	I	T	
Secondary Containment Boundary Valves	P	RB	NA	I	C	I	T	
Piping (all other)	P	RB,O	NA	NA	D	NA	P,NR	
Valves (all other)	P	RB,O	NA	NA	D	NA	P,NR	
Cooling Towers	P	O	NA	NA	D	NA	NR	
Motor Control Center	P	O	Non-1E	NA	NA	NA	NR	
Pump/Cooling Tower Fan Motors	P	O	Non-1E	NA	NA	NA	NR	
Local Control Panels & Racks								
Cables	P	RB,O	Non-1E	NA	NA	NA	NR	
Controls/Instruments Inside Reactor Building	P	RB	Non-1E	NA	NA	NA	P	
Controls/Instruments in Yard	P	O	Non-1E	NA	NA	NA	NR	
Hydrogen Water Chemistry System								
Piping, Other	GE,P	T	NA	NA	D	NA	NR	
Valves, Other	GE,P	T	NA	NA	D	NA	NR	
Controls/Instruments	GE,P	T	Non-1E	NA	NA	NA	NR	
Electrical Equipment	GE,P	T	Non-1E	NA	NA	NA	NR	

TABLE 3.2-1 (Cont'd.)

Penetration Z-99D

2RCS-001-202-2

Penetration Z-100A

Penetration Z-100B

2RCS-001-203-2

Penetration Z-100C

Penetration Z-100D

2RCS-001-204-2

Penetration Z-319-2

2RCS-750-168-2

Penetration Z-328-3

2RCS-750-151-2

Extension of Primary Containment - Reactor Building Equipment Drain System

(DER) Penetration Z-45

2DER-002-034-2

(47) The equipment and its classification also apply to the air start system associated with the Division III (HPCS) diesel generator.

(48) The GE compressor motor is seismic Category I. The compressor motor starter is seismic Category NA, but it is seismically mounted and evaluated to ensure that if failure occurs it will not cause degradation of safety-related equipment.

(49) The storage pool gate is nonsafety related.

(50) Nonessential portions of these systems within the containment, from the containment penetration up to but not including the safety-related end users, are designed, fabricated and erected to Quality Group C requirements. The safety-related end user components function without or upon loss of nitrogen, or are provided with safety-related

TABLE 3.2-1 (Cont'd.)

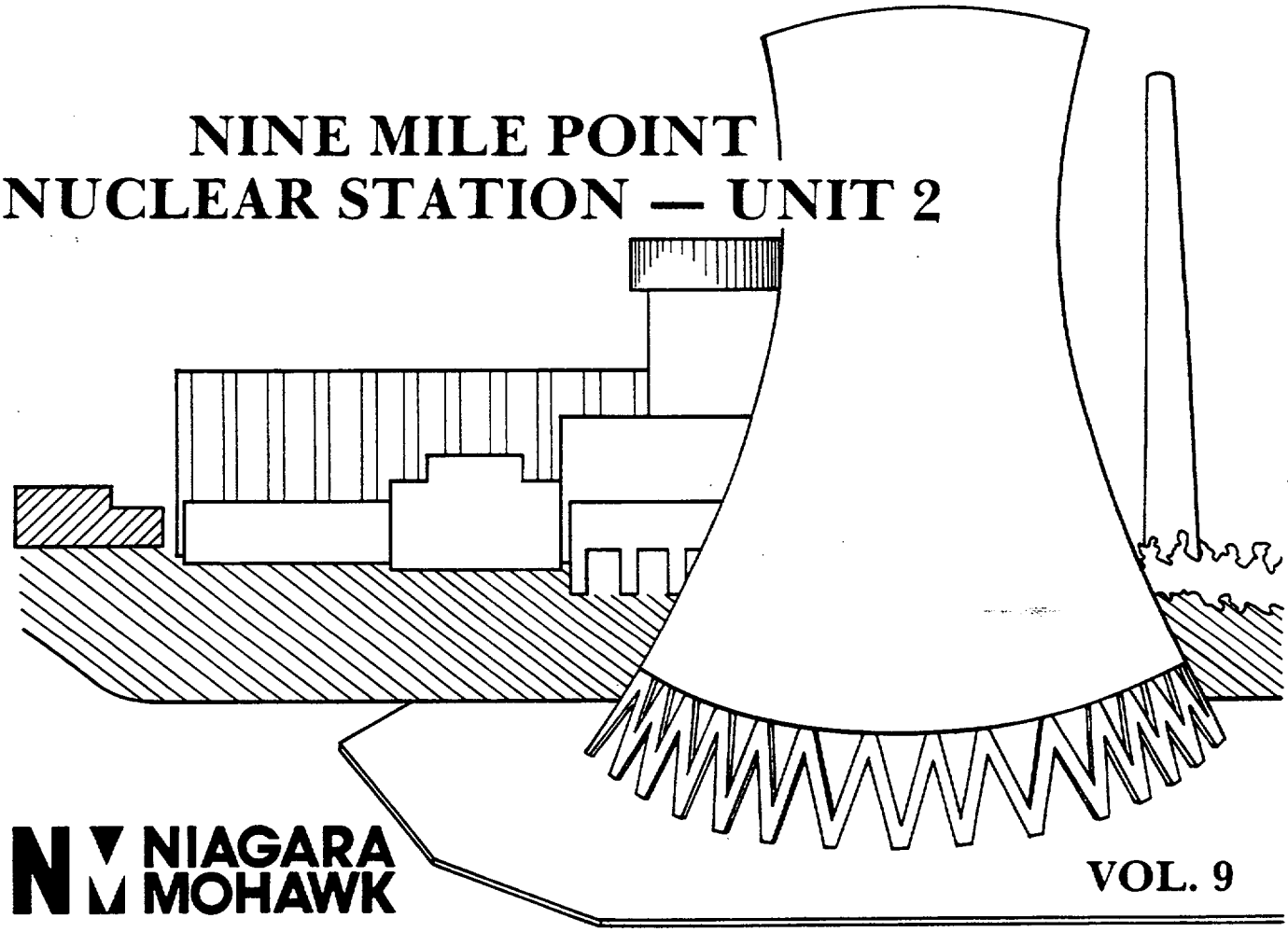
accumulators capable of supplying the required quantities of gas.

(51) The essential components of the GTS air supply system are designed and installed in accordance with the requirements of ASME III, Class 3, except ASME III Code data reports and N-stamping are not required since GTS is not a stamped system (See Note 40).

(1) Lease II User's Manual, "Slope Stability Analysis," by P. J. Trudeau and J. T. Christian, August 1980, Stone & Webster Engineering Corporation.

UPDATED SAFETY ANALYSIS REPORT

NINE MILE POINT NUCLEAR STATION — UNIT 2



NIAGARA MOHAWK

VOL. 9

3.6 PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

Two inputs to Section 3.6 are provided. Section 3.6A is applicable to the Stone & Webster Engineering Corporation (SWEC) scope of supply. Section 3.6B is applicable to the GE scope of supply.

With regard to design for protection against dynamic effects associated with the postulated rupture of piping, the respective GE and SWEC responsibilities are as follows:

1. GE's responsibility includes the reactor recirculation piping only. For the recirculation piping, GE determines the postulated break locations and the blowdown reactions resulting from each postulated break, and provides the restraints to restrict pipe whip in the event that a postulated break occurs.

2. SWEC's responsibility includes the balance of piping inside and outside containment. For all piping, except recirculation piping, SWEC determines the break locations and the resulting blowdown reactions, and provides the required pipe whip restraints. In addition, for all piping <u>including</u> the recirculation piping, SWEC analyzes the jet impingement effects resulting from each postulated break.

This section describes the design for protection against postulated piping failures both inside and outside containment including all high- and moderate-energy piping systems. This section includes or references plant layout drawings and system piping and arrangement drawings, and a description of how the plant structures, systems, and components conform to related design criteria and bases. It also demonstrates the ability to perform a safe shutdown after a postulated piping failure of a high- or moderate-energy system.

This section is consistent with pipe rupture criteria submitted to the NRC by NMPC on July 31, 1978[1]. An excerpt from this letter showing a comparison of the Unit 2 criteria to RG 1.46 and Branch Technical Positions (BTP) MEB 3-1 and APCSB 3-1 is provided in Table 3.6A-1.

3.6A PROTECTION AGAINST EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING (SWEC SCOPE OF SUPPLY)

3.6A.1 Postulated Piping Failures in Fluid Systems Inside and Outside the Containment

3.6A.1.1 Design Bases

Criteria

The pipe failure protection conforms to Appendix A of 10CFR50, GDC 4, Environmental and Missile Design Bases. The overall design for this protection is in compliance with BTP APCSB 3-1 and MEB 3-1. Compliance to RG 1.46 is discussed in Table 1.8-1.

Design Objectives

Protection against pipe failure effects is provided to fulfill the following objectives:

1. To ensure that the reactor can be shut down safely and maintained in a safe shutdown condition.

2. To ensure that radiological doses resulting from a postulated piping failure remain below the limits of 10CFR100.

3. To ensure that containment integrity is maintained.

4. To ensure that a pipe break that is not a loss of reactor coolant does not cause a loss of reactor coolant.

5. To ensure that a postulated piping failure with its direct consequences and a single active component failure do not result in unacceptable consequences except as noted below.

   Where the postulated piping failure is assumed to occur in one of two or more redundant trains of a dual-purpose, moderate-energy, essential system (i.e., one required to operate during normal plant conditions as well as to shut down the reactor and mitigate the consequences of the piping failure), single failures of components in the other train or trains of that system are not assumed since:

   a. The system is designed to Category I standards.

   b. Power is provided from both offsite and onsite sources.

   c. Construction, operation, and inspection are done in accordance with quality assurance, testing, and

in-service requirements appropriate for nuclear safety systems.

Examples of systems that qualify as moderate-energy, dual-purpose, essential systems are the SWP and RHR systems.

6. To ensure that minimum core cooling requirements are maintained for pipe break events, their direct consequences, and any single active failure as specified in Section 3.6A.1.1, the following break areas and break combinations are not exceeded:

   a. For breaks involving recirculation piping, the total effective area of all broken pipes (including the effective area of the recirculation line break) does not exceed the total effective area of the design basis double-ended recirculation line break (DBA).

   b. Breaks involving one recirculation loop do not result in a loss of coolant from the other recirculation loop in excess of that which would result from a break of the attached cleanup connections on the suction side of the loops.

Design Assumptions

The following assumptions are used to determine the protection requirements:

1. Pipe breaks or cracks are postulated to occur during normal plant operation (i.e., reactor startup, operation at power, hot standby, or reactor cooldown to a cold shutdown).

2. Only high-energy piping, as defined in Section 3.6A.2.1.1, is capable of producing breaks. Moderate-energy piping, as defined in Section 3.6A.2.1.2, is capable of producing only cracks.

3. Each longitudinal or circumferential break in high-energy fluid system piping, or leakage crack in moderate-energy fluid system piping, is considered separately as a single postulated initial event occurring during normal plant conditions.

4. Pipe failures (breaks or cracks) inside the containment are not postulated concurrently with pipe failures outside the containment.

5. Offsite power is assumed to be unavailable when a trip of the turbine generator system or RPS is a direct consequence of the postulated piping failure, unless it

is more conservative to assume that offsite power is available (e.g., a feedwater line break with offsite power available leads to a larger inventory of water for flooding considerations).

6. All available systems, including those initiated by Operator actions, are employed to mitigate the consequences of a postulated piping failure.

7. A whipping pipe is not considered capable of rupturing impacted pipes of equal or greater nominal pipe diameter and equal or greater thickness.

8. Pipe whip is assumed to occur in the plane defined by the piping geometry, and to cause movement in the direction of the jet reaction, unless shown to be otherwise by analysis.

9. The fluid internal energy associated with the pipe break reaction takes into account any line restrictions (e.g., flow limiter) between the pressure source and break location and absence of energy reservoirs, as applicable.

10. Damage to the RPV from the surface impact effects of pipe rupture does not occur due to its location relative to piping systems.

11. Initial pipe break events are not assumed to occur in pump and valve bodies because of their greater wall thicknesses.

Approach

Systems, components, and equipment required to safely shut down the plant and mitigate the consequences of postulated piping failures (hereinafter called essential systems, components, and equipment) are reviewed to determine their susceptibility to the pipe failure.

Pipe breaks are evaluated for the effects of pipe whip, jet impingement, flooding, room pressurization, and other environmental effects such as temperature.

Pipe cracks are evaluated for wetting from spray, flooding, and other environmental effects.

Piping system break and crack locations are determined in accordance with Section 3.6A.2. A flow chart of activities, sample model, and typical restraint design are shown on Figures 3.6A-1 through 3.6A-11.

3.6A.1.2 Description of Piping Failures

A list of essential systems, components, and equipment, or portions thereof, is provided in Table 3.6A-72. A list of high-energy lines, as discussed in Section 3.6A.2.1.1, is given in Table 3.6A-73. Moderate-energy piping (Section 3.6A.2.2) is not listed.

Composite drawings (Figures 3.6A-52 through 3.6A-60) show the routing of high-energy piping in relation to compartments. Nearby essential items are discussed in Section 3.6A.2.5.

Pressure response analyses are performed for the subcompartments containing high-energy piping. For a detailed discussion of the line breaks selected, vent paths, room volumes, analytical methods, pressure results, etc., refer to Section 6.2.1.2 for containment subcompartments, and Appendix 3B for subcompartments located outside the containment.

The effects of pipe whip, jet impingement, spraying, and flooding on essential systems, components, and equipment are discussed in Appendix 3C.

There are no high-energy lines located near or in the main control room. Therefore, effects upon the habitability of the main control room from pipe break, including pipe whip, jet impingement, and transport of steam, are not considered. Further discussion of main control room habitability systems is provided in Section 6.4.

There are no high-energy lines located near or in the diesel generator building.

3.6A.1.3 Safety Evaluation

3.6A.1.3.1 Approach

An analysis of pipe failures is performed to identify those safety-related systems, components, and equipment that provide protective actions required to mitigate the consequences of the postulated pipe failure.

Design features such as separation, barriers, and pipe whip restraints are incorporated to ensure that pipe breaks and cracks do not damage essential items to an extent that would impair the integrity or operability of essential systems and components.

Specific design features used for protecting the essential systems, components, and equipment, and the ability of specific safety-related systems to withstand a single active failure concurrent with a postulated event, are discussed in Appendix 3C.

When the pipe layout and plant arrangement drawings show that the effects of postulated breaks/cracks are isolated, physically

remote, or restrained by plant design features from essential systems or components, no further evaluation is performed.

3.6A.1.3.2 Protection Methods

The effects associated with a particular break/crack must be mechanistically consistent with the failure. Thus, actual pipe dimensions, piping layouts, material properties, and equipment arrangements are considered in defining the specific measures for protection against actual pipe movement and other associated consequences of postulated failures. Protection against the dynamic effects of pipe failures is provided in the form of pipe whip restraints, jet impingement shields, barriers, compartments, and physical separation of piping, equipment, and instrumentation. Pipe supports are used as protective measures in isolated cases. The specific method chosen depends on physical limitations such as accessibility, maintenance, and proximity to other essential systems, components, and equipment. Protective measures utilized to meet these requirements consider access requirements for conducting the in-service examinations specified in the ASME Boiler and Pressure Vessel Code, Section XI.

Separation and Enclosure

Separation is achieved to the extent practicable by plant physical layouts that provide sufficient distances so that essential systems and components are separated from other fluid systems. Fluid systems that are not physically separated from essential systems and components are enclosed, when practical, within structures or compartments designed to protect nearby essential systems and components. Alternatively, essential systems and components may be enclosed within structures or compartments designed to withstand the effects of postulated piping failures in nearby fluid systems.

Barriers and Shields

In many cases protection requirements are met by walls, floors, columns, abutments, and foundations. Where adequate protection does not exist due to separation, additional barriers, deflectors, or shields are provided as necessary.

Piping Restraint Protection

Pipe restraints for protection against pipe whip as a result of high-energy pipe breaks are provided except in cases described as follows:

1. The piping is either physically separated (or isolated) from any essential safety-related structure, system, or component required to place the plant in a safe shutdown condition, or the piping is restrained from whipping by plant design features such as concrete

encasement or other structures or compartments designed for jet impingement and pipe whip loads. These loads are applied either at the most critical locations of the protective structures or at locations determined in Appendix 3C.3.1.

2. Following a single break, unrestrained pipe movement of either end of the ruptured pipe could not damage, to an unacceptable level, any structure, system, or component required to place the plant in a safe shutdown condition.

3. The energy associated with the whipping pipe is demonstrated to be insufficient to impair, to an unacceptable level, the safety function of any structure, system, or component required to place the plant in a safe shutdown condition.

The design criteria for restraints are given in Section 3.6A.2.3.2.

3.6A.1.3.3 Specific Protection Measures

Nonessential systems and system components are not required for the safe shutdown of the reactor, nor are they required for the limitation of the offsite release in the event of a pipe rupture. However, while none of this equipment is needed during or following a pipe break event, pipe whip protection is provided in specific cases where the broken pipe in the nonessential system could adversely impact an essential system or component.

The pressure, water level, and flow sensor instrumentation for those essential systems required to function during or after accident conditions are protected from pipe rupture effects.

High-energy fluid system piping restraints and protective measures are designed in such a way that a postulated break in one pipe could not, in turn, lead to rupture of other nearby pipes or components if the secondary rupture could result in consequences that would be considered unacceptable for the initial postulated break.

Pipe rupture restraints are located such that the unrestrained portion will not form a plastic hinge at the restraint. This criterion precludes tip deflection from adversely affecting safety-related equipment.

A postulated rupture in the piping in Unit 1 does not affect the capability for safe shutdown of Unit 2 and vice versa. There are no shared safety systems between Unit 1 and Unit 2.

For any postulated pipe rupture, the structural and leak-tight integrity of the containment structure is maintained.

To maintain the ability to scram the reactor in the event of a pipe rupture, the CRD withdraw lines are protected from pipe break events so that no more than one in any rod array is allowed to be completely crimped (totally blocked). Complete severance of withdraw lines will not affect the scram function. Protection for the CRD insert lines is not required during normal reactor operation since a reactor pressure of 450 psig or higher (CRD insert lines principal backup) could adequately scram the control rods even with a complete loss of insert lines. Routing of high-energy lines in the vicinity of the CRD withdraw lines is strictly controlled by design measures.

The escape of steam, water, combustible, or corrosive fluids, gases, and heat in the event of a pipe rupture does not preclude:

1. Habitability of the main control room.

2. The ability of essential instrumentation, electric power supplies, components, and controls to perform their safety function.

The potential for damage to both independent high-pressure ECCS systems (ADS and HPCS) in the event of a partial break in the pressurized portion of the HPCS has been considered. No portion of the normally-pressurized HPCS system is located within jet impingement distance of any component considered essential to the operation of the ADS, and vice versa.

3.6A.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

3.6A.2.1 Criteria Used to Define Break and Crack Location and Configuration

3.6A.2.1.1 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 temperature exceeds 200°F, or

Maximum pressure exceeds 275 psig.

Normal plant conditions are defined as the plant operating conditions during reactor startup, power plant operation, and reactor cold shutdown, but excluding test modes.

3.6A.2.1.2 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

either in operation or are maintained pressurized under conditions where both of the following are met:

Maximum temperature is 200°F or less, and

Maximum 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 functions, 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 pressure-temperature (P-T) conditions specified for the high-energy fluid system is less than 2 percent of the total operating time for which the system is designed.

3.6A.2.1.3 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 the development of a sudden longitudinal, uncontrolled crack (longitudinal split), and is postulated for the high-energy fluid system only. For moderate-energy fluid systems, pipe breaks are confined to the postulation of controlled cracks in piping and branch runs. These cracks affect the surrounding environmental conditions only, and do not result in whipping of the cracked pipe.

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 includes 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. Internal missiles that might be generated from failures of these components are evaluated as discussed in Section 3.5.1.

A high-energy piping system break is not postulated simultaneously with a moderate energy piping system crack.

Effects of moderate-energy leakage cracks inside the containment are bounded by DBA (large- and small-break LOCA) considerations. Environmental effects, including effects from spraying and flooding as well as from jet impingement and pipe whip, are less severe for moderate-energy pipe cracks than for LOCA events. Therefore, safety-related equipment which is qualified to function post-LOCA is also available to mitigate the consequences of moderate-energy line cracks.

3.6A.2.1.4 Exemptions from Pipe Whip Protection Requirements

Protection from pipe whip need not be provided if any one of the following exists:

1. Piping is classified as moderate-energy piping.

2. Following a single postulated pipe break, the piping for which the unrestrained movement of either end of the ruptured pipe in any feasible direction 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.

3.6A.2.1.5 Postulated Pipe Break Locations

<u>Criteria for Inside the Containment</u>

For ASME Section III, Safety Class 1 piping systems within the containment, design basis piping break locations are selected using the following criteria:

1. At the terminal ends including:

   a. Piping, pressure vessel, or equipment nozzle intersections.

   b. High energy-moderate energy boundary.

   c. A branch connection to a main run unless all the following are met:

      (1) The branch and main runs are of comparable size and fixity (i.e., the nominal size of the branch run is at least one-half that of the main run),

      (2) The intersection is not rigidly constrained by the building structure, and

(3) The branch and main runs are modeled as a common piping system during the pipe stress analysis.

2. At the intermediate locations between the terminal ends selected by either of the following criteria:

   a. At each fitting (e.g., elbow, tee, cross, flange, and nonstandard fitting), welded attachment, and valve, or

   b. At locations where the maximum stress range for the normal and upset plant conditions and for an OBE exceeds 2.4 S_m, calculated by Equation (10) and either Equation (12) or (13) in ASME Section III, Paragraph NB-3653; and at locations where the cumulative usage factor (CUF)*, U, derived from the piping fatigue analysis, under the loadings associated with OBE and operational plant conditions exceeds 0.1. S_m is the allowable stress intensity as specified in ASME Section III, Subparagraph NB-3213.1.

   If, as a result of piping reanalysis of the original design configuration, the highest stress or CUF location shifts for any initially determined arbitrary intermediate break location, the original postulated arbitrary break location need not change unless a redesign of the piping resulting in a change in pipe parameters (diameter, wall thickness, routing) is required, or the dynamic effects from the new (as-built) intermediate break locations are not mitigated by the original pipe whip restraints and jet shields.

For ASME Section III, Safety Class 2 and 3 piping systems, break locations are postulated by the following criteria:

1. At the terminal ends.

2. At the intermediate locations between the terminal ends selected by either of the following criteria:

   a. At each pipe fitting (e.g., elbow, tee, cross, flange, and nonstandard fitting), welded attachment, and valve, or

   b. At each location where the stress associated with normal and upset plant conditions and an OBE event calculated by Equations (9) plus (10) in Paragraph

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* Specified in ASME Section III, Subparagraph NB-3222.4.

NC-3652 of ASME Section III, exceeds 0.8 (1.2 S_h+S_A).

S_h = Allowable stress at the elevated temperature calculated according to ASME Section III, Subarticles NC-3600 and ND-3600 for Safety Class 2 and 3 components, respectively.

S_A = Allowable stress range for the expansion stress calculated according to ASME Section III, Subarticle NC-3600 and ANSI B31.1.

3. If a fatigue analysis is performed at any intermediate locations between the terminal ends where the CUF under the loading associated with OBE and operational plant conditions exceeds 0.1.

Criteria for Outside the Containment

High-Energy Fluid Systems The following criteria are used to define break and crack locations in high-energy fluid systems outside the containment:

1. Fluid Systems Separated from Essential Structures, Systems, and Components Breaks are not postulated in high-energy piping at locations that are isolated or physically remote from essential equipment, structures, and the containment.

2. Fluid System Piping in Containment Penetration Areas Breaks are not postulated in the portions of high-energy piping between the containment isolation valves, outside and inside containment. Breaks are not postulated in the portions of high-energy piping between the isolation valve and the first restraint or groups of restraints designed to protect these portions of piping. Containment isolation valve pipe whip restraints are capable of resisting bending and torsional moments produced by a postulated piping failure outboard of the first restraint or group of restraints beyond the containment isolation valves.

   Restraints are designed to withstand the loadings resulting from a postulated piping failure, so that neither the isolation valve operability nor the leak-tight integrity of the associated containment penetration will be impaired. These portions of piping are designed to meet the requirements of ASME Section III, Subarticle NE-1120, and the following additional design requirements, which are in conformance with Revision 1 (July 1981) of SRP 3.6.2 and BTP MEB 3-1, the documents applicable at the time the analysis was performed:

a. The following design stress and fatigue limits are not exceeded for Safety Class 1 piping:

(1) The maximum stress range between any two load sets (including the zero load set), calculated by Equation (10) in Paragraph NB-3653, ASME Section III, for those loads and conditions thereof for which Level A and Level B stress limits have been specified in the design specification, including an OBE event transient, do not exceed 2.4 S_m. If the calculated maximum stress range of Equation (10) exceeds 2.4 S_m, the stress ranges calculated by both Equations (12) and (13) in Paragraph NB-3653 will meet the limit of 2.4 S_m.

(2) The CUF is less than 0.1.

(3) The maximum stress, as calculated by Equation (9) in Paragraph NB-3652, under the loading resulting from a postulated piping failure beyond these portions of piping does not exceed 2.25 S_m, except that following a failure outside containment, the pipe between the outboard isolation valve and the first restraint may be permitted higher stresses provided a plastic hinge is not formed.

b. The following design stress limits are not exceeded for Safety Class 2 and nonnuclear piping in the break exclusion area:

(1) The maximum stress ranges do not exceed 0.8 (1.2 S_h+S_A), as calculated by Equations (9) and (10) in Paragraph NC-3652, ASME Code Section III, considering normal and upset plant conditions (i.e., sustained loads, occasional loads, and thermal expansion) and an OBE event.

(2) The maximum stresses do not exceed 1.8 S_h, as calculated by Equation (9) in Paragraph NC-3652 under the loadings resulting from a postulated piping failure of fluid system piping beyond these portions of piping.

c. The following design stress limits are not exceeded for Safety Class 3 piping between the outboard isolation valve and the first restraint:

(1) The maximum stress ranges do not exceed 0.8 (1.2 S_h+S_A), as calculated by Equations (9) and (10) in Paragraph NC-3652, ASME Code

Section III, considering normal and upset plant conditions (i.e., sustained loads, occasional loads, and thermal expansion) and an OBE event.

(2) Following a pipe failure outside the containment, the formation of a plastic hinge is not permitted in the piping between the outboard isolation valve and the first pipe whip restraint, or group of restraints, to assure the operability of the isolation valve.

d. Welded attachments for pipe supports or other purposes to these portions of piping are avoided, except where detailed stress analysis demonstrates compliance with the limits discussed in Items 2a and 2b.

e. The number of circumferential and longitudinal piping welds and branch connections is minimized.

f. The length of these portions of piping is reduced to the minimum length practicable.

g. The design of pipe anchors or restraints (e.g., connections to containment penetrations and pipe whip restraints) does not require welding directly to the outer surface of the piping (e.g., flued integrally-forged pipe fittings are used), except where such welds are capable of 100-percent volumetric ISI. This criterion is also applicable to the portion of piping between the containment and the inside containment isolation valves.

h. For these portions of high-energy fluid system piping, preservice and subsequent in-service examinations are performed in accordance with the requirements specified in ASME Section XI. During each inspection interval, as defined in IWA-2400, an ISI is performed on all nonexempt ASME Code Section XI circumferential and longitudinal welds within the break exclusion region for high-energy fluid system piping. These inspections consist of augmented volumetric examinations (nominal pipe size greater than or equal to 4 in) and augmented surface examinations (nominal pipe size less than 4 in) such that 100 percent of the previously defined welds are inspected at each interval. The break exclusion zone consists of those portions of high-energy fluid system piping between the moment limiting restraint(s) outside the outboard containment isolation valve and the moment limiting restraint(s) beyond the inboard

containment isolation valve. The choice of the restraint(s) that define the limits of the break exclusion zone is based upon those restraint(s) which are necessary to ensure the operability of the primary containment isolation valves.

i. Regardless of the fact that all conditions above have been met, a crack is postulated in the main steam or feedwater piping in the main steam tunnel. The crack in the pipe, equal in area to a single-ended pipe rupture, is considered a singular event. Pipe whip and jet impingement are not considered, and a single active failure is not taken as a concurrent event.

3. Balance of Piping Outside the Containment

a. Breaks in ASME Section III, Safety Class 2 and 3 piping and in nonnuclear class piping that is seismically analyzed and supported are postulated at the following locations in each piping and branch run (except those portions of fluid system piping identified in Items 1 and 2):

(1) At terminal ends of the pressurized portions of the runs.

(2) At intermediate locations selected by either of the following criteria:

(a) At each pipe fitting (e.g., elbow, tee, cross, and nonstandard fitting), welded attachment, and valve, or, if the run contains no fittings, at one location at each extreme of the run within the protective structure (a terminal end, if located within a protective structure, may substitute for one intermediate break).

(b) At each location where the stresses associated with normal and upset plant conditions and an OBE event exceed 0.8 ($1.2\ S_h+S_A$), as calculated by Equations (9) and (10), Paragraph NC-3652 of ASME Section III, for Safety Class 2 and 3 piping.

b. Breaks in nonnuclear safety class piping not seismically qualified are postulated at the following locations in each piping or branch run:

(1) At terminal ends of the pressurized portions of the runs.

(2) At each pipe fitting, welded attachment, and valve.

These breaks are sufficient to establish the worst pipe break effects since:

(1) The piping is physically remote from essential equipment and structures, or

(2) There are a large number of pipe breaks postulated on the same line in the same area due to a large number of pipe fittings or attachments on the pipe, or

(3) For nonseismic piping in close proximity to safety-related systems, components, or structures, either the piping is seismically analyzed and supported or other protection is provided.

Moderate-Energy Fluid Systems The following criteria are used to define crack locations in moderate-energy fluid systems outside the containment:

1. For the purpose of satisfying the separation provisions of plant arrangement, a review of the piping layout and plant arrangement drawings is conducted. Safe shutdown systems are isolated or located physically remote from the effects of through-wall leakage cracks, to the extent this is practical.

2. Leakage cracks are not postulated in those portions of piping between the isolation valve and the containment, provided they meet the requirements of ASME Section III, Subsubarticle NE-1120, and are designed so that the maximum stress range associated with normal and upset plant conditions and an OBE event does not exceed 0.4 (1.2 S_h+S_A) (as calculated by Equations (9) and (10), Paragraph NC-3652 of ASME Section III, Safety Class 2 piping).

3. Cracks are not postulated in moderate-energy fluid system piping located in an area in which a break in high-energy piping occurs. Where a postulated leakage crack in the moderate-energy fluid system piping results in more limiting environmental conditions than a break in proximate high-energy fluid system piping, the provisions identified under Item 4 below are applied.

4. Through-wall leakage cracks are postulated in fluid system piping, except where exempted under Items 1, 2, and 3 above, or where the maximum stress range, associated with normal and upset plant conditions and

an OBE event, in these portions of ASME Section III, Safety Class 2 or 3 piping and nonnuclear piping is less than 0.4 (1.2 S_h+S_A) (as calculated by Equations (9) and (10), Paragraph NC-3652 of ASME Section III). The cracks are postulated to occur individually at locations that result in the maximum effects from fluid spray and flooding. Only environmental effects that develop from these cracks are considered.

5. Through-wall leakage cracks, instead of breaks, are postulated in the piping of those fluid systems that qualify as high-energy fluid systems for only short operational periods, but qualify as moderate-energy fluid systems for the major operational period. An operational period is considered short if the fraction of time that the system operates within the P-T conditions specified for high-energy fluid systems is less than 2 percent of the time that the system operates as a moderate-energy fluid system (e.g., systems such as the reactor RHR system qualify as moderate-energy fluid systems).

3.6A.2.1.6 Design Basis Break/Crack Types and Orientation

Circumferential Pipe Breaks

Circumferential breaks are postulated in high-energy fluid system piping at the locations specified in Section 3.6A.2.1.5 except:

1. For nominal pipe size of 1 in or less.

2. Where it is determined by detailed stress analysis that the stress in the circumferential direction is at least 1.5 times that in the axial direction at the location of maximum stress.

Where break locations are selected at pipe fittings without the benefit of stress calculations, breaks are postulated at the piping weld to each fitting, valve, or welded attachment. If detailed stress analyses or tests are performed, the maximum stressed location in the fitting may be selected instead of the pipe-to-fitting weld.

Circumferential breaks are assumed to result in pipe severance and separation amounting to at least one-diameter lateral displacement of the ruptured piping sections unless physically limited by piping restraints, structural members, or piping stiffness as may be demonstrated by inelastic analysis.

The dynamic force of the jet discharge at the break location is based on 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. Limited pipe displacement at the break location, line restrictions, flow

limiters, positive pump-controlled flow, and the absence of energy reservoirs is taken into account, as applicable, in the reduction of jet discharge.

Pipe whipping is assumed to occur in the plane defined by the piping geometry and configuration, and is assumed to cause pipe movement in the direction of the jet reaction, unless shown to be otherwise by analysis.

Longitudinal Pipe Breaks

Longitudinal breaks are postulated in high-energy fluid system piping at the locations specified under circumferential pipe breaks, except:

1. For nominal pipe sizes smaller than 4 in.

2. Where it is determined by detailed stress analysis that the stress in the axial direction is at least 1.5 times that in the circumferential direction at the location of maximum stress.

3. At terminal ends.

4. At intermediate locations where the criterion for a minimum number of break locations must be satisfied, except where it is determined by detailed stress analysis that the stress in the circumferential direction is at least 1.5 times that in the axial direction.

Longitudinal breaks are assumed to result in an axial split without pipe severance. Splits are located (but not concurrently) at two diametrically opposed points on the piping circumference in such a way that a jet reaction causing out-of-plane bending of the piping configuration results. Alternately, a single split may be assumed at the section of highest stress as determined by detailed stress analysis.

The dynamic force of the fluid jet discharge is based on a circular break area equal to the effective cross-sectional flow area of the pipe at the break location, and on a calculated fluid pressure modified by an analytically or experimentally determined thrust coefficient as determined for a circumferential break at the same location. Line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs are taken into account, as applicable, in the reduction of jet discharge.

Pipe movement is assumed to occur in the directions defined by the stiffness of the piping configuration and jet reaction forces, unless limited by structural members or piping restraints.

Through-Wall Leakage Cracks (Outside the Containment Only)

Through-wall leakage cracks are postulated in main steam or feedwater piping systems in containment penetration areas as stated in Section 3.6A.2.1.5 under High-Energy Fluid Systems, Item 2h. The following through-wall leakage cracks are postulated in Moderate-Energy Fluid System piping at the locations specified in Section 3.6A.2.1.5 under Moderate-Energy Fluid Systems:

1. Cracks are postulated in moderate-energy fluid system piping and branch runs exceeding a nominal pipe size of 1 in.

2. Fluid flow from a crack is based on a circular opening of area equal to that of a rectangle one-half the nominal pipe diameter in length and one-half the pipe wall thickness in width.

3. The flow from the crack is assumed to result in an environment that wets all unprotected components within the compartment, with consequent flooding in the compartment and communicating compartments. Flooding effects are determined on the basis of a conservatively estimated time period required to effect corrective actions.

3.6A.2.2 Analytical Methods to Define Forcing Functions and Response Models

3.6A.2.2.1 Introduction

Pipe rupture analyses consist of calculations to determine the fluid forces generated by the blowdown of pressurized lines, complemented by dynamic or energy balance analyses to determine pipe motion and impact effects (Figure 3.6A-1). Restraints for lines 6 in and less in diameter are usually qualified on a generic basis using an energy balance. However, restraints for larger lines are engineered individually for each system, usually using standard design concepts and worst-case dynamic analysis to qualify several similar restraints in different locations. The response of unrestrained lines is analyzed by either inelastic dynamic analysis or energy balance analysis.

Criteria for the response analyses are as follows:

1. An analysis of the pipe run or branch is performed for each postulated longitudinal and circumferential rupture or, alternatively, for a worst case. Worst cases are selected on the basis of gap, fluid force, and piping system stiffness.

2. The loading condition of a pipe run or branch prior to postulated rupture in terms of internal pressure,

temperature, and stress state is that condition associated with reactor operation at 100-percent power.

3. For a circumferential rupture, pipe whip dynamic analyses are only performed for that end (or ends) of the pipe or branch that is (are) connected to a contained fluid energy reservoir having sufficient capacity to develop a jet stream.

4. Dynamic analytical methods, used for calculating the piping or piping/restraint system response to the jet thrust developed after a postulated rupture, adequately account for the effects of the following:

   a. Mass inertia and stiffness properties of the system.

   b. Impact and rebound (if any) as permitted by gaps between piping and restraint.

   c. Elastic and inelastic deformation of piping and/or restraint.

   d. Support boundary conditions.

5. An allowable design strain limit of 0.5 ultimate uniform strain of the restraints is used for tensile energy-absorbing components. For compressive energy-absorbing components, a design limit of 80 percent of energy-absorbing capacity is used.

6. A 10-percent increase of minimum specified yield strength (S_y) may be used to account for strain rate effects in inelastic nonlinear analyses. Alternatively, experimental data may be used to determine the strain rate parameters for use in nonlinear codes that monitor strain rate.

3.6A.2.2.2 Time-Dependent Blowdown Force

The blowdown force calculations are based on the transient pressures, velocities, and other thermodynamic properties of the fluid[2]. To provide the time history of pressure, velocity, etc., the method of characteristics is used to solve the continuity and momentum equations simultaneously. A general description of the method can be found in most gas dynamics textbooks[3-6]. For these one-dimensional fluid mechanics analyses, the pipe is regarded as straight, despite numerous bends. The calculated momentum and pressure forces are applied wherever there is a change in flow direction or cross section of the piping to provide time-dependent loads for pipe dynamic analysis.

The transient forces result from wave propagation and fluid momentum. It is assumed that pipe bends and elbows neither attenuate the traveling pressure waves nor cause reflections. Immediately following the rupture of a pipe, a decompression wave travels from the break at the speed of sound relative to the fluid. The fluids ahead of and behind the wave are at different states. This initial blowdown condition will last until a return signal from a pressure reservoir reaches the break. Repeated wave reflections between the reservoir and break prevail until a steady-state flow condition is established. Boundary conditions that govern the flow at the break end and at the inlet from the vessel to the pipe are applied.

The time histories of transient pressure, mass flow rate, and other thermodynamic properties of the fluid are based on the following equation, which includes static and dynamic effects, to calculate the blowdown force:

$$F = \left[P_e - P_a + \frac{RU_e^2}{144g} \right] A \qquad (3.6A\text{-}1)$$

Where:

F	=	Blowdown force, lb_f
P_e	=	Pressure at exit plane, psia
P_a	=	Ambient pressure, psia
U_e	=	Velocity of fluid at exit plane, fps
R	=	Density of fluid, lb_m/ft^3
A	=	Pipe break area, sq in
g_c	=	Gravitational constant, lb_m-ft/lb_f sec^2

The effects of line friction are included in the evaluation of steady-state blowdown. For the calculation of the transient fluid response, however, friction may or may not be considered.

Subcooled Nonflashing Waterline Blowdown

Transient Flow Immediately following the rupture, a flow disturbance propagates from the break at a speed of sound relative to the fluid, leaving the fluid behind the wave at a thermodynamic state of U_o and $P = P_2$. The governing equation across the wave is:

$$\Delta P = \pm \frac{RC}{g} \Delta U \qquad (3.6A\text{-}2)$$

Where:

ΔP = Differential pressure across wave, psia

ΔU = Differential velocity across wave, fps

C = Speed of sound in fluid, fps

When the disturbance reaches a pressure reservoir, it is reflected and travels toward the break end. The boundary conditions that govern the flow at the break location and at the inlet to the pipe (from the reservoir) are:

$P_e = P_a$

$$P_i = P_o - \frac{RU_i^2}{2g} \qquad (3.6A\text{-}3)$$

Where:

P_i = Pressure at pipe inlet, psia

U_i = Velocity of fluid at pipe inlet, fps

P_e, P_a = Pressure at the break location, psia

P_o = Reservoir pressure, psia

The initial blowdown flow remains constant until the disturbance, which is reflected from the vessel, reaches the break end. Then it is reflected again, and that brings a change of blowdown flow. These repeated wave transmissions and reflections continue until the steady-state flow is established.

Steady-State Flow For steady-state flow, the blowdown forcing function calculations become:

$$F = \left[\frac{2(P_o - P_a)}{P_o} \frac{1}{1 + \frac{fL_e}{D}} \right] P_o A \qquad (3.6A\text{-}4)$$

which is derived by applying Bernoulli's equation across the pipe and by using the expression for the forcing function calculation.

Where:

L_e = Total equivalent length of pipe friction, ft

f = Friction factor (Reynolds number and pipe surface roughness dependent)

D = Pipe inside diameter, ft

Steamline Blowdown Transient Flow Steam is treated as an ideal, single-phase gas with a constant specific heat ratio, k, of 1.3.

Except for the case of steady-state blowdown flow, the flow is assumed to be isentropic with negligible pipe friction. The characteristic method, which is a finite difference approximation using the principle of characteristics, is used as a basis for the numerical solution of the continuity and momentum equations[7,8]. The transient pressure, mass flow rate, and other thermodynamic properties are then used to calculate the transient-state forcing function.

Immediately following the break, a decompression wave travels into the pipe toward the pressure reservoir. The fluid in front of the wave is at a state:

$$U_i = 0$$

$$C_i = C \qquad (3.6A\text{-}5)$$

Where:

U_i = Velocity of fluid in front of the wave, fps

C_o = Speed of sound in fluid with respect to the state of fluid at the reservoir, fps

C_i = Speed of sound in fluid in front of wave

The fluid state at the exit is at the sonic condition, because the initial pressure was sufficiently high[9]:

$$\frac{U_e}{C_o} = \frac{C_e}{C_o} = \frac{2}{k+1} = 0.8695 \; for \; k = 1.3 \qquad (3.6A\text{-}6)$$

The blowdown force can be calculated as:

$$F = \left[\frac{P_e}{P_o} + \frac{R_e C_e^2}{g P_o}\right] P_o A = \left[\frac{P_e}{P_o} + \frac{R_e}{R_o}\left(\frac{C_e}{C_o}\right)^2 \frac{R_o C_o^2}{g P_o}\right] P_o A$$

(3.6A-7)

The pressure ratio across the wave is:

$$\frac{P_e}{P_o} = \left(\frac{T_e}{T_o}\right)^{\frac{k}{k-1}} = \left(\frac{C_e}{C_o}\right)^{\frac{2k}{k-1}} = \left(\frac{2}{k+1}\right)^{\frac{2k}{k-1}} = 0.298 \qquad (3.6A\text{-}8)$$

Where:

T = Temperature, and the density ratio is:

$$\frac{R_e}{R_o} = \left(\frac{P_e}{P_o}\right)^{\frac{1}{k}} = \left(\frac{C_e}{C_o}\right)^{\frac{2k}{k-1}\left(\frac{1}{k}\right)} = \left(\frac{2}{k+1}\right)^{\frac{2}{k-1}}$$

Therefore, the blowdown force can be reformulated as:

$$F = (1+k)\left(\frac{2}{k+1}\right)^{\frac{2k}{k-1}} P_o A = 0.685\, P_o A \qquad (3.6A\text{-}9)$$

The blowdown force is constant until a return signal from the pressure source reaches the break. When the wave reaches the reservoir, it is reflected as a compression wave. The boundary condition at the pressure lies on the steady-state ellipse:

$$\left(\frac{C_i}{C_o}\right)^2 + \frac{k-1}{2}\left(\frac{U_i}{C_o}\right)^2 = 1 \qquad (3.6A\text{-}10)$$

which is the energy equation applying across the vessel-pipe inlet. The boundary condition for this case is:

$$T_o = T_i + \frac{U_i^2}{2C_p} \qquad (3.6A\text{-}11)$$

Where:

C_p = Constant pressure specific heat of a fluid, Btu/lbm °F

i = State at the inlet to the pipe

J = 778 x 32.2 = 25,052 $\frac{Pbm \bullet ft^2}{Btu \bullet Sec^2}$

<u>Steady-State Flow</u> If the steady state is reached, the flow in the pipe is uniform, and if the pressure in the pressure vessel remains high, then the boundary condition at the break always lies on the sonic line, that is:

$$\frac{U^*}{C_o} = \frac{C^*}{C_o} \qquad (3.6A\text{-}12)$$

Then from the critical flow condition:

$$\frac{U^*}{C_o} = \frac{C^*}{C_o} = \sqrt{\frac{2}{k+1}} = 0.9325 \qquad (3.6A\text{-}13)$$

Where:

* = Critical flow condition

Then, the steady-state blowdown force is:

$$F = \left(\frac{P^*}{P_o} + \frac{R^*(U^*)^2}{g}\right) P_o A = (1+k)\left(\frac{2}{k+1}\right)^{\frac{k}{k-1}} P_o A = 1.255\, P_o A$$

(3.6A-14)

For steady-state flow with friction losses, the analysis is based on the theory of compressible flow with friction[9]. The pipe friction is the chief factor bringing about the change of fluid properties in the flow. A curve that describes the variation of steady-state steam blowdown force versus friction parameter fL/D is shown on Figure 3.6A-2.

3.6A.2.2.3 Simplified Blowdown Analysis

A conservative steady-state forcing function may be used for calculations based on the energy balance method. The function has a magnitude of:

$$T = KPA \qquad (3.6A\text{-}15)$$

Where:

P = System pressure prior to pipe break, psia

A = Pipe break area, sq in

K = Thrust coefficient

Theoretical maximum K values are as follows:

1.26 for saturated steam, water, and steam/water mixture

2.00 for nonflashing subcooled water

Where pipe rebound may occur upon impact on the restraint, an amplification factor between 1.0 and 1.1 is applied to the above force. Justification for the use of amplification factors less than 1.1 can be found in Reference 13. In this reference, the results of nonlinear dynamic analysis were compared to the results of the corresponding analyses based on the energy balance

method, which assumes constant blowdown forces and instantaneous development of a dynamic plastic hinge. The pipe will rotate around the hinge as a rigid body. The appropriate correlation factors required to match the energy balance results with the dynamic analyses results were correlated to design parameters such as restraint location from the break, yield strength of energy-absorbing components, and restraint gap. It was shown that the correlation factors were related to a single, nondimensional parameter and could be limited to a value of 1.0 if appropriate design parameters were chosen. Since actual results and designs for Unit 2 were used in the reference analysis, this validates the use of factors less than 1.1. Alternatively, the maximum fluid force during the energy input phase, as determined by the detailed methods of Section 3.6A.2.2.2, may be used. In determining this maximum, a brief initial force of PA may be ignored since the initial pipe velocity is low and the resulting work input is inconsequential. The above amplification factor for rebound is also included.

3.6A.2.2.4 Lumped-Parameter Dynamic Analysis

The piping system is modeled mathematically as a series of beam elements connected at nodes. The geometry of the model matches that of the pipe. The distributed mass of the pipe and contained fluid is modeled as lumped masses located at the nodes. The beam elements have the stiffness properties of the pipe in the elastic range and approximate the plastic behavior after yield.

Before a rupture, the pipe is stressed by internal pressure, but remains in static equilibrium. When initial conditions have a significant effect on the parameters being calculated, such as stresses in break exclusion regions or loads on attached components, this effect is considered.

As a circumferential break propagates before the pipeline breaks up into two sections, the load-carrying metal area at the break of the pipe decreases so that a force unbalance results. The force initially transmitted across the break is assumed to drop linearly to zero in 1 msec. After the break, the forces exerted on the pipe by the fluid are determined by the time-dependent blowdown force derived in Section 3.6A.2.2.2. Similarly, for a longitudinal split, the crack propagation speed limits the rate at which the split opens, so a 1-msec force rise time is assumed. Other break opening times may be used if justified.

Subsequent to a postulated rupture, the inelastic system response is analyzed by the use of an elastic-plastic lumped-mass beam element computer code such as DINASAW or LIMITA (Appendix 3A). The analysis considers the free motion of the pipe through a gap, if one exists, using the appropriate initial conditions and the fluid blowdown forces as calculated in Section 3.6A.2.2.2. The mathematical model includes the restraint or barrier, and sometimes a member simulating the local crush resistance of the pipe. Rebound effects are considered by automatically connecting

and disconnecting that member for impact and rebound, respectively.

Sample Dynamic Analysis

Pipe rupture restraint 2RHS*PRR004 inside the containment limits the motion of the RHR line following a circumferential break at the elbow. The restraint is an omnidirectional-type restraint (Figure 3.6A-10) with a 1.815-in gap between the hot pipe and the restraint.

The analysis of the pipe-restraint interaction used the LIMITA computer codes. The finite-element model is shown on Figure 3.6A-3. The fluid forces depicted on Figure 3.6A-4 were applied to the pipe elbow as shown on Figure 3.6A-3.

The restraint reaction load is shown on Figure 3.6A-5. The maximum restraint load is 388 kips and the maximum deformation of the honeycomb panel is 2.62 in. The corresponding working strain is 0.524 in/in, which is less than the allowable strain of 0.56 in/in.

3.6A.2.2.5 Energy Balance Analysis

The energy balance technique for analyzing pipe impact equates the work done by the escaping fluid to the energy absorbed in deforming the ruptured pipe and the impacted target. A steady-state blowdown force is used for the energy balance analysis. The magnitude of the force is described in Section 3.6A.2.2.3.

The input energy of the system is determined by multiplying the pipe displacement at the break end by the component of the fluid blowdown force in the direction of the displacement.

The input energy is:

$$E = F \times D \qquad (3.6A\text{-}16)$$

Where:

F = Component of blowdown force in direction of pipe displacement, lb

D = Displacement of break end of pipe, in

The strain energy absorbed during pipe whip and impact consists of the energy absorbed by pipe bending, E_{pb}, the energy absorbed by pipe crush during impact, E_{pc}, and the energy absorbed by deformation of the target, E_f.

To determine postimpact target deformation and the peak reaction force, the input energy is equated to the strain energy absorbed by the pipe and target. The energy absorption characteristics of

the pipe crush and target deformation are calculated on the basis of the displacement integral of the appropriate force-deformation curves.

Sample Energy Balance Analysis

The same RHR restraint, 2RHS*PRR004 (Figure 3.6A-6), is analyzed here by the energy balance technique. The energy input from the fluid blowdown force is:

$$E_{input} = F_b\,(g + d)\left(\frac{L_h}{L_h - L}\right) \qquad (3.6A\text{-}17)$$

Where:

F_b = Fluid blowdown force, kips

g = Acceleration gap of restraint, in

d = Restraint deflection, in

L_h = Length from break to plastic hinge, in

L = Length from break to restraint, in

The ratio $L_h/(L_h-L)$ represents the increased pipe displacement at the break, compared to displacement at the restraint, due to the assumed pipe rotation about a plastic hinge.

The fluid force is calculated:

$$F_b = k_r k_f P_o A = 287 \text{ kips} \qquad (3.6A\text{-}18)$$

Where:

k_r = Rebound factor (1.1)

k_f = Thrust coefficient (0.92), reduced from 1.26 after accounting for friction between the source and break

P_o = Initial pressure (1,028 psi)

A = Pipe flow area (252.719 sq in)

This energy may be absorbed in plastic bending of the pipe and in crush of the restraint. The energy absorbed by bending at the plastic hinge is:

$$E_b = M_p = M_p(g+d)/(L_h-L)$$

$$E_b = 932 \text{ in-kips} \qquad (3.6A\text{-}19)$$

Where:

M_b = Plastic moment of the pipe

= σ_y $(D_o^3 - D_i^3)/6$

= 27,510 $(20^3-17.938^3)/6$

= 1.0216×10^4 in-kips

L_h = $3\ M_p/F_b$ = 106.8 in

L = 54 in

g = 1.815 in

d = Allowable crushing of the honeycomb panel

= 0.8 (total crushable depth of honeycomb panel)

= 0.8 x 0.7 x 5.0 = 3.0 in

The energy absorbed in the honeycomb panel is:

$$E_a = F \times d \qquad (3.6A\text{-}19)$$

By equating E_{in} to (E_a+E_b), the restraint reaction force F is found to be in 621 kips, and the honeycomb panel is crushed 3.0 in.

3.6A.2.2.6 Local Pipe Indentation

The local shell indentation stiffness of the pipe is usually considered where other energy-absorbing mechanisms are not available at the point of impact. Examples include impacts into rigid displacement-limiting bumpers, concrete walls, and the omnidirectional restraint weldment (the latter interposes a significant mass between the impacting pipe and the energy absorbers).

Two methods have been used to determine the shell indentation stiffness. The earlier was analytical and tended to overpredict conservatively the indentation stiffness. The other was a series of pseudostatic pipe crush tests covering several crush geometries and a sufficient range of pipe thicknesses and diameters to develop parametric scaling laws[10,11]. This was augmented by analyses to determine the sensitivity to material strength, dynamics, and variations in loading geometry.

3.6A.2.2.7 Concrete Barrier Impact

In a pipe whip impact, the force on the barrier is a complex function of time depending primarily on the sudden deceleration of the pipe wall at the impact point (slug impact), the shell

indentation of the pipe as it locally crushes against the wall, and the force transmitted to the impact point by the more gradual deceleration of the adjacent run of pipe. After impact, the pipe also transmits a more enduring force resulting from the continuing fluid blowdown. The concrete is affected in the same way as in any other missile impact event, the only significant difference being the long-term fluid force. To evaluate this postulated event, the pipe is transformed into an equivalent missile and the concrete is analyzed for scabbing and structural response using the procedure described in Section 3.5.3. The analysis for structural response includes the impulse of the initial impact as well as the subsequent fluid blowdown force and other concurrent loads.

Four basic parameters must be determined to define the equivalent missile: kinetic energy (or impulse), impact velocity, pipe crush stiffness, and bearing area. The kinetic energy and velocity can be found by either of two methods:

1. Simplified Method Use the total input energy (fluid blowdown force times distance of pipe travel) less the energy absorbed in pipe bending prior to impact. Compute the velocity using approximate formulas.

2. Lumped-Parameter Dynamic Analysis (Section 3.6A.2.2.4) This method is especially suited for evaluating the impact of piping systems with complex geometries and can even consider multiple impact points. As an alternative to the kinetic energy, the impact force history (impulse) can be computed.

Regardless of which analysis method is used, the crush resistance of the equivalent missile and the bearing area are derived from the experimental data described in Section 3.6A.2.2.6. These data are modified to account for the effect of dynamics and internal pressure.

3.6A.2.3 Dynamic Analysis Methods to Verify Integrity and Operability

Pipe rupture loads to determine the integrity of mechanical components are determined using the analytical methods described in Section 3.6A.2.2. The applicable load combinations for the components and for break exclusion regions are presented in Sections 3.9A and 3.6A.2.1.5 (Criteria for Outside the Containment, Item 2), respectively. Criteria for rupture restraints are presented in Section 3.6A.2.3.1.

3.6A.2.3.1 Jet Impingement Analysis

Jet impingement loadings are determined as follows:

1. Jet forces are represented by time-dependent forcing functions. The effects of the piping geometry,

capacity of the upstream energy reservoir, source pressure, and fluid enthalpy are considered in these forcing functions.

2. The steady-state jet force has a magnitude of:

$$T_j = K_j PA \qquad (3.6A\text{-}20)$$

Where:

P = System pressure prior to pipe break, psia

A = Pipe break area, sq in

K_j = Jet coefficient

The following K_j values are used whenever the reservoir pressure is constant, pipe friction is negligible, and there are no upstream flow restrictions:

1.26 for saturated steam, saturated water, and steam/water mixture.

2.00 for nonflashing subcooled water

3. In calculating the jet impingement load on a distant object or target, the retarding action of the surrounding air along the jet path is neglected. The jet impingement pressure on an effective target is calculated by taking the jet force as being constant at all effective distances from, and normal to, the break area and by assuming that the jet stream diverges conically at a solid angle of 20 deg. However, for those cases where the 20-deg divergence assumption is shown to be unnecessarily conservative for the blowdown of steam or steam-water mixtures, Moody's asymptotic jet expansion model is adopted[12].

4. The proportion of the total jet force acting on the target is determined from the fraction of the jet intercepted and by the shape factor of the target. For a target with its flat surface area normal to the center of the jet stream, the impingement load is the product of the pressure and the intercepted jet area. For those cases where the target area is such that the intercepted jet stream is deflected rather than totally stopped, a shape factor that is less than unity and is a function of the target geometry is used in calculating the total jet impingement load.

Since the jet impingement force is a dynamically applied load, the target is analyzed either by static methods using an appropriate dynamic load factor, or dynamically using elastic

structural response computer codes such as STARDYNE or inelastic structural response computer codes such as DINASAW or LIMITA (Appendix 3A). The load combinations and design allowables are given in Sections 3.8 and 3.9A.

3.6A.2.3.2 Pipe Rupture Restraints

Two basic restraint types are used, elastic and energy-absorbing. Energy-absorbing restraints are used where the primary objective is to dissipate the energy of a ruptured pipe.

The elastic portions of pipe rupture restraints and intermediate structures (auxiliary steel) are designed in accordance with the loads, loading combinations, and stress limits outlined in Section 3.8.4.3 for steel structures except for the following: for abnormal loading conditions, the allowable stress for combination 7 is 1.6S, and for combination 8 the allowable stress is 1.7S.

Only energy-absorbing components (e.g., honeycomb panels and studs) of pipe rupture restraints are designed for strain beyond the elastic limit.

The design of pipe rupture restraints and supporting structures includes the forces induced by SSE; therefore, failure cannot occur during a seismic event.

The portion of the pipe rupture restraint that contacts and supports the process pipe is designed to meet the pipe support requirements of ASME Section III (Section 3.9).

Elastic Restraints

Since elastic restraints are used to minimize displacements of the broken pipe, they are close-gapped. For some applications, this requires that they contact the pipe during conditions other than a postulated rupture, in which case they are designed as a pipe support. If an elastic restraint only contacts the pipe following a rupture, it is designed according to the criteria for structural steel (Section 3.8.3).

Energy-Absorbing Restraints

Several approaches are used for energy absorption in pipe rupture restraints. In tension, stainless steel studs or straps are used, with a design limit of 50 percent of uniform ultimate strain. In compression, honeycomb panels or pipe are used. Compressive components are designed to 80 percent or less of their energy absorption capacity. Other energy-absorbing devices that may be used are designed to these limits.

One or more of the above energy-absorbing mechanisms are utilized in each of the typical restraints described below. When a single

energy-absorbing mechanism is utilized, the design limits are met for the design range of loading directions.

Elastic components of energy-absorbing restraints are designed to the criteria for structural steel (Section 3.8).

Pipe Crush Bumper The pipe crush bumper absorbs impact energy in a direction toward the supporting structure. The energy absorber is a length of pipe placed normal to the axis of the process pipe. Subsequent to a rupture, the bumper pipe is crushed between its support structure and the moving process pipe. This absorbs energy and forms a retaining recess in the bumper pipe. The bumper pipe is attached to its support by welding and bolting (Figure 3.6A-7).

Laminated Strap Restraint The laminated strap restraint is capable of absorbing impact loads in the outward direction from the supporting structure (Figures 3.6A-8 and 3.6A-9). The energy-absorbing component is a U-shaped strap that consists of multiple strips (depending on energy to be absorbed) of highly ductile material (Type 304 stainless steel).

This laminated design results in great flexibility. If the process pipe contacts the sides of the restraint during an event other than pipe rupture, only negligible loads are transmitted. The design also minimizes bending strains, permitting the strap to act mainly as a membrane during the rupture event.

Omnidirectional Restraint The omnidirectional restraint is capable of absorbing impact loads applied in any direction in the plane of the restraint (Figure 3.6A-10). This restraint consists of a base weldment, an arch, ductile holddown studs on each side of the base weldment, and a honeycomb panel. The primary function of the studs is to absorb energy from impact loads acting outward from the support structure. The honeycomb panel absorbs energy from impact loads acting in an inward direction. Side load impacts are absorbed by the combined action of the studs and honeycomb. A limit stop (Figure 3.6A-11) is a restraint whose design varies slightly with that of an omnidirectional restraint. This restraint is designed primarily to absorb energy from the impact load in the inward impact direction.

Combinations of pipe crush bumpers and laminated straps may also be used to achieve energy absorption over a range of impact directions up to a full 360 deg.

3.6A.2.4 Guard Pipe Assembly Design Criteria

Guard pipes are not used at Unit 2, so this section is not applicable.

3.6A.2.5 Material for the Operating License Review

Pipe break and crack locations are obtained in accordance with the criteria of Section 3.6A.2.1. High-energy piping with break locations identified are provided in isometric drawings (Reference 14). High-energy piping composites (Figures 3.6A-52 through 3.6A-60) have been provided to show graphically the pipes in relation to their branch piping. The stress results that were originally utilized to determine the break types and locations were submitted to the NRC in References 14, 15, and 16. If there are changes in the pipe routing, restraint locations, or stress analysis as a result of plant modifications, the effects on pipe break analyses are evaluated in accordance with established programs and procedures. The locations of cracks are adequately defined in the FMEA (Appendix 3C).

The augmented ISI plan is discussed in Section 6.6. Pipe whip restraints are designed as discussed in Section 3.6A.2.3.1. The restraint locations and orientation and break locations are shown on Figures 3.6A-12 through 3.6A-49. Jet thrust and impingement forces are determined in accordance with Section 3.6A.2.3.

The effects of breaks and cracks are discussed in detail in Appendix 3C. The results of this appendix are based on the protection evaluation criteria of Section 3.6A.1. Any protective measures to assure a safe shutdown (i.e., barriers, separation, and restraints) are also discussed.

3.6A.3 References

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15. Letter from Mr. C. V. Mangan, NMPC, to Ms. Elinor G. Adensam, NRC, dated August 8, 1986 (NMP2L 0809).

16. Letter from Mr. C. V. Mangan, NMPC, to Ms. Elinor G. Adensam, NRC, dated August 21, 1986 (NMP2L 0832).

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:

$$F = \left[(P - P_a) + \frac{\rho v^2}{g_c} \right] A \qquad (3.6B\text{-}1)$$

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:

$$C_T = 3.0 - 0.861h^{*2};\ 0 \le h^* \le 0.75$$

$$C_T = 3.22 - 3.0h^* + 0.97h^{*2};\ 0.75 < h^* \le 1.0$$

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) Design steady-state load 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) Dynamic material mechanical properties 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. Type II Restraint (e.g., clevises, brackets, pins)

a. Materials 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. Inspection 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-10313. 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.

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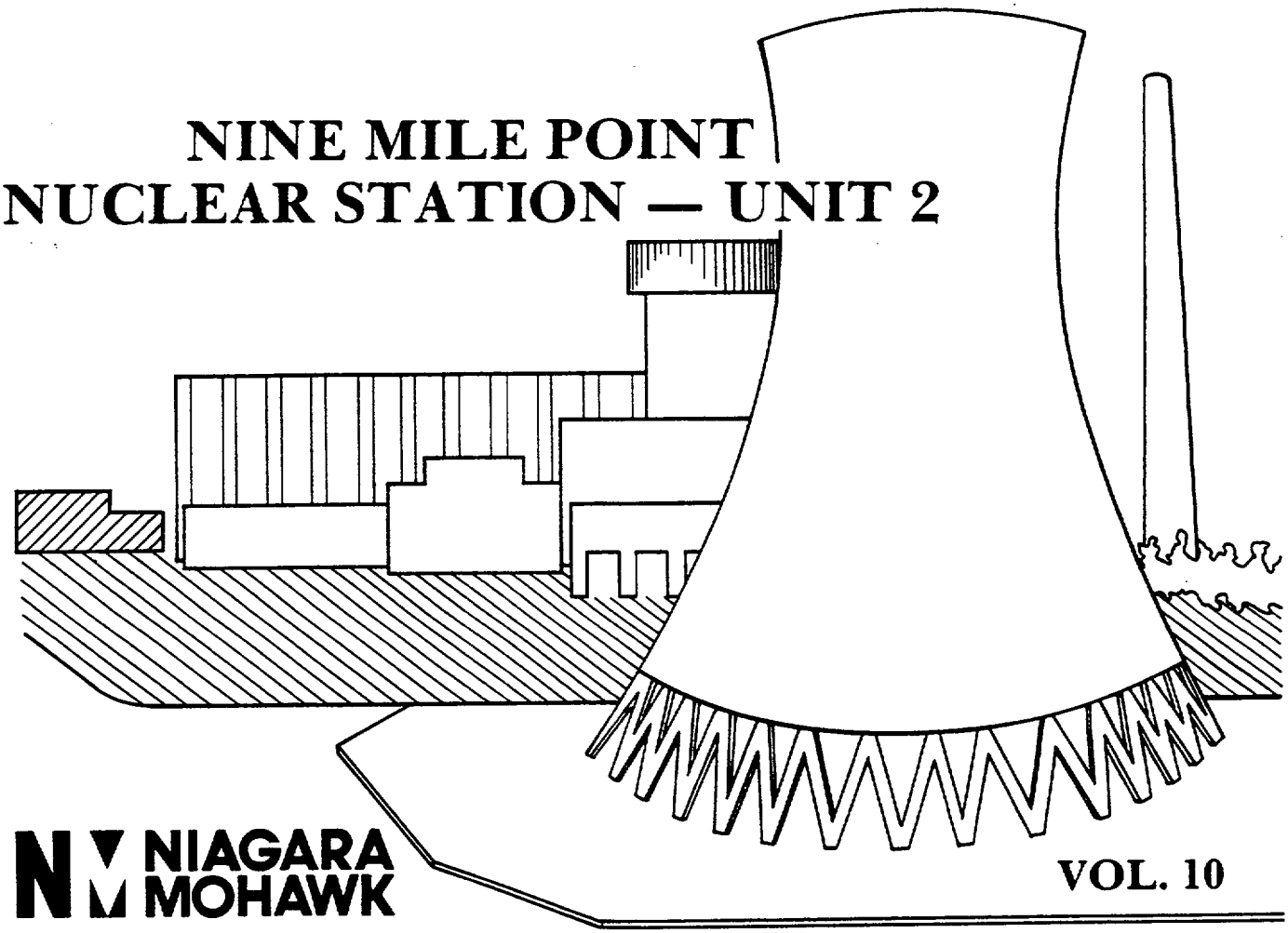

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TABLE 3.7A-10

TABULATION OF STRUCTURAL GAP AND WORST-CONDITION GAP FOR SSE CONDITION

Structural Gap Provided Between Building A and Building B		Structural Gap (in)	Maximum Deflection at Roof Elev. of Shorter Bldg.		Cumulative Deflection (in)	Worst Computed Gap (in)
			Bldg. A (in)	Bldg. B (in)		
Reactor	Auxiliary Service	3	0.285	0.088	0.373	2.627
Reactor	Standby Gas Treatment	3	0.466	0.162	0.628	2.372
Reactor	Radwaste Tunnel	3	0.155	0.037	0.192	2.808
Reactor	Main Steam Tunnel	3	0.466	0.070	0.536	2.464
Control	Normal Switchgear	3	0.031	0.055	0.086	2.914
Control	Turbine	3	0.037	0.081	0.118	2.882
Control	Auxiliary Service	3	0.021	0.075	0.096	2.904
South Auxiliary Bay	Control	3	0.119	0.015	0.134	2.866
South Auxiliary Bay	Auxiliary Service	3	0.119	0.024	0.143	2.857
North Auxiliary Bay	Standby Gas Treatment	3	0.155	0.003	0.158	2.842
North Auxiliary Bay	Radwaste Tunnel	3	0.119	0.024	0.143	2.857
Radwaste	Turbine	3	0.050	0.081	0.131	2.869
Radwaste	Screenwell	12	0.148	0.059	0.207	11.793

TABLE 3.8-3

LOAD COMBINATIONS FOR ACCESS LOCKS AND HATCHES

PART I

Where Structure Is Integral and Continuous

Design Category	Load Combination	Stress Comparisons	ASME Paragraph
Design I	D + PD + OBE + SRV + LOCA	Pm ≤1.0 Sm @ TD P ≤1.5 Sm @ TD Pl + Pb ≤1.5 Sm @ TD	NE-3131 (C)
Design II	D + PD + SSE + SRV + LOCA	Pm ≤ larger of 1.2 Sm or Sy @ TD Pl ≤ larger of 1.8 Sm or 1.5 Sy @ TD Pl + Pb ≤ larger of 1.8 Sm or 1.5 Sy @ TD	NE-3131 (C)
Design III	D + PE + OBE	Pm ≤1.0 Sm @ TE Pl ≤1.5 Sm @ TE Pl + Pb ≤1.5 Sm @ TE ASME III instability criteria	NE-3131 (C) NE-3133, NE-3131 (C)
	D + PE + SSE	Pm ≤ larger of 1.2 Sm or 1.0 Sy @ TE Pl ≤ larger of 1.8 Sm or 1.5 Sy @ TE Pl + Pb ≤ larger of 1.8 Sm or 1.5 Sy @ TE 1.2 (Instability value from NE-3133)	NE-3131 (C) NE-3133, NE-3131 (C)
Design IV	D + PF + OBE	Pm ≤ 1.5 Sm @ TF Pl ≤ larger of 1.8 Sm or 1.5 Sy @ TF Pl + Pb ≤ larger of 1.8 Sm or 1.5 Sy @ TF	
Operating I	D + Poc + To + OBE + SRV + LOCA	Pl + Pb + Q ≤3 Sm Pl + Pb + Q + F (fatigue evaluation)	NE-3222
Operating II	D + Poc + To + SSE + SRV + LOCA	Pl + Pb + Q ≤3 Sm	NE-3222
Test	D + PT	Pm ≤0.85 Sy @ TT Pl + Pb ≤1.25 Sy @ TT	NE-6322

TABLE 3.8-6

LOAD COMBINATIONS AND STRESS LIMITS FOR SAFETY CLASS 2 PIPE PENETRATIONS

Plant Condition	Load Combinations	Stress Limits	ASME Paragraph
Design operating	a. PD + D	$\frac{P_D D_o}{4t} + 0.75I \frac{M_A}{z} \leq 1.0S_h$	NC-3652.1 Eq 8
Normal and upset	b. Pmax + D + (FLT+OBEI+SRV)^{SRSS}	$\frac{P_{max} D_o}{4tn} + 0.75I \frac{M_A + M_B}{z} \leq 1.2S_h$	NC-3652.2 Eq 9
	c. (To+OBEA) + (PD+D)	$\frac{iM_c}{z} + \left(\frac{P_o D_o}{4tn} + 0.75i \frac{M_A}{Z}\right) \leq S_A + S_h$	NC-3652.3 Eq 11
Emergency	d. PE + D + (FLT+OBEI+SRV+LOCA)^{SRSS}	$\frac{P_E D_o}{tn} + 0.75i \frac{M_A + M_B}{Z} \leq 1.8S_h$	NC-3611.3(c)
Faulted	e. PF + D + (FLT+SSEI+SRV+RT+LOCA)^{SRSS}	$\frac{P_F D_o}{4tn} + 0.75i \frac{M_A + M_B}{Z} \leq 2.4S_h$ *(See Note)*	ASME Section III Code Case N-53 (1606-1)
	f. PF + D + (FLT+SSEI+RT+AP)^{SRSS}	$\frac{P_F D_o}{4tn} + 0.75i \frac{M_A + M_B}{Z} \leq 2.4S_h$ *(See Note)*	
Test	g. Pt + D	$\frac{P_t D_o}{4tn} + 0.75i \frac{M_A}{Z} \leq 1.35S_y$	NC-3218.1
Pipe break exclusion			
Normal and upset	P_{max} + D + (FLT+OBEI+SRV)^{SRSS} + (To+OBEA)	$\frac{P_{max} D_o}{4t_n} + 0.75i \frac{M_A + M_B}{Z} + i \frac{M_c}{Z} \leq 0.8\ (1.2S_h + S_A)$	NRC Branch Technical Position MEB 3-1
	PF + D + (FLT+SSEI+SRV+LOCA+RT)^{SRSS}	$\frac{P_f D_o}{4t_n} + 0.75i \frac{M_A + M_B}{Z} \leq 1.8S_h$	
	PF + D +(FLT+SSEI+RT+AP)^{SRSS}		

TABLE 3.8-6 (Cont'd.)

LOAD AND SYMBOL DEFINITIONS:

Mi (i=A,B,C)	=	Resultant moment, i.e., $Mi = (Mix^2+Miy^2+Miz^2)^{1/2}$, for each load combination except Mc which is due to (To+OBEA).
Z	=	Pipe section modulus per NC-3654.4.
i	=	Stress intensification factor, if 0.75 i <1, take 0.75 i = 1.
Do	=	Outside diameter of pipe.
tn	=	Nominal wall thickness of component.
PD, PE, PF, Po	=	Internal pressure for design, emergency, faulted, and normal/upset operating conditions, respectively.
Pmax	=	Maximum of Po.
Pt	=	Test pressure.
D	=	Deadweight and other sustained loads.
FLT	=	Pipe load due to fluid transient, such as thrust from relief and safety valve loads from pressure and flow transients.
RT	=	Pipe rupture loads and local and general effects from jet impingement if applicable.
LOCA	=	Dynamic loads from suppression pool due to the postulated loss-of-coolant accident.
OBEI, OBEA	=	Loads due to OBE from structure or piping inertia and piping anchor movement effects, respectively.
AP	=	Effect of vibration of reactor pressure vessel and biological shield wall due to annulus pressurization.
SRV	=	Dynamic loads from suppression pool due to safety/relief valve blowdown.
SSEI	=	Load due to safe shutdown earthquake (SSE) from structure or piping inertia effect.

Superscripts:

ABS	=	Absolute sum.
SRSS	=	Square root of sum of squares.
Sy	=	Yield strength from Appendix I of ASME Section III.
Sh	=	Allowable stresses from Table I-7.0 of ASME Section III at maximum (hot) temperature.
SA	=	Allowable stress range for expansion stresses (see ASME Section III, NC-3611.2(c)).

NOTE: The faulted stress limits and analysis techniques specified in ASME Section III, Appendix F, can be applied in lieu of Code Case N-53 (1606-1). Inelastic methods can be used as allowed by the Code.

TABLE 3.8-13

MAJOR STRUCTURES FOUNDATION SYSTEMS

Structure	Foundation Method	Approx. Contact Area (ft x ft)	Approx. Founding Elevation (ft, msl)	Depth of Burial of Structure Below Finished Grade (ft)
Reactor building including auxiliary bays	Reinforced concrete mat	182 ft diameter[1]	164	97
Radwaste building	Reinforced concrete mat	150 x 110	232	29
Control building	Reinforced concrete mat	96 x 134	210	51
Diesel generator building	Reinforced concrete mat and spread footings	-	242	19
Turbine building[2]	Reinforced concrete mat and spread footings	-	236 except at C.W. enclosure	25
Screenwell building[2]	Reinforced concrete mat and spread footings	-	222 except at C.W. enclosure	39
Intake structures	Tremie concrete slab with tie-downs in rock	Hexagonal with face-to-face dimension of 33'-6"	222	Structures in lake
Intake tunnels	Reinforced concrete mat	14' x 1,340'±	Varies from 132 to 146	Structures in lake
Main stack	Reinforced concrete mat	60 ft diameter	232'-5"	28.5
Offgas, regeneration, and condensate demineralizer area[2]	Reinforced concrete mat, spread footings, and grade beam arrangement	-	236	25
Standby gas treatment building and railroad access lock[2]	Reinforced concrete mat and spread footings and partially supported on tunnel	-	247 except 233 at top of tunnel	14 or 28
Auxiliary service building[2]	Slab supported on the top of tunnel	-	233 (top of tunnel)	28

(1) Actual contact area at auxiliary bays exceeds 182 ft diameter.
(2) Certain portions of these structures (including foundations) are designated Category I, whereas the remaining portions are designated non-Category I (Section 3.8.4).

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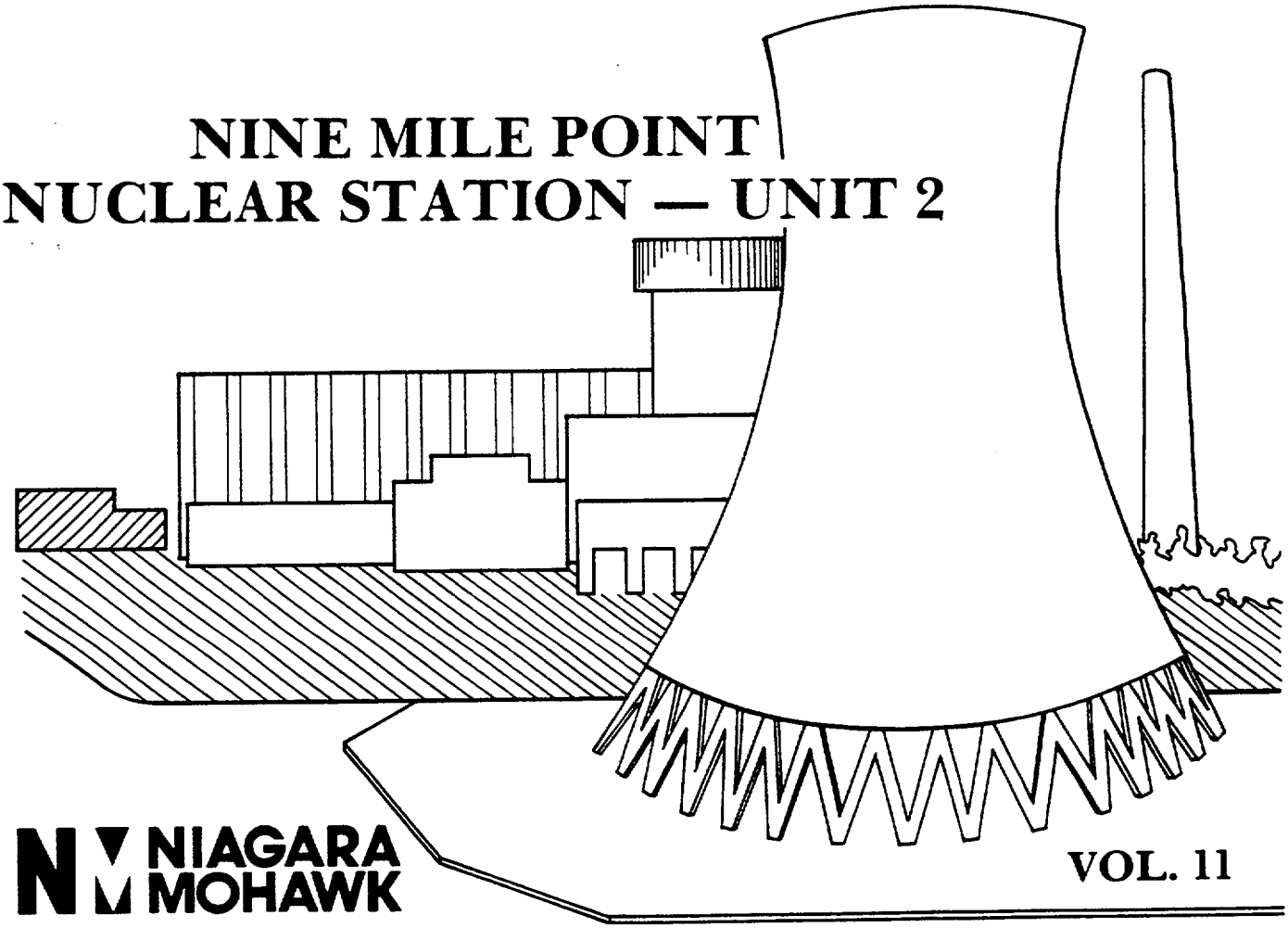

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3.9 MECHANICAL SYSTEMS AND COMPONENTS

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. Stress limits for inelastic system (and/or component) analysis are in accordance with ASME Section III, Appendix F.

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.

The stress limits and techniques specified in ASME Section III, Appendix F, Summer 1983 Addenda, may be used in evaluating faulted loaded conditions for Class 1, 2 and 3 valves.

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:

$$KU = P-Q \qquad (3.9A-1)$$

Where:

K	=	Stiffness matrix for assembled system
U	=	Nodal displacement vector
P	=	External forces, weights, etc.
Q	=	Equivalent thermal forces = $\int_0^L AE \propto \overline{T} d\ell$
A	=	Cross-section area
E	=	Young's Modulus
$\propto$	=	Thermal expansion coefficient
$\overline{T}$	=	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.

Dead Loads (Weight, Pressure) and Live Loads

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:

$$E \propto \Delta T_1, E \propto \Delta T_2, and\ E_{ab}\ (\alpha_a T_a - \alpha_b T_b) \qquad (3.9A\text{-}2)$$

Based on geometry, fluid type, insulation, thermal transients, environmental data:

$$\propto \Delta T_1, \propto \Delta T_2, and\ E_{ab}\ (\alpha_a T_a - \alpha_b T_b) \qquad (3.9A\text{-}3)$$

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:

$$M\ddot{U} + C\ddot{U} + KU = F(t) \qquad (3.9A\text{-}4)$$

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.

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* 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 envelop the applicable frequency range of the RRS with the required 10-percent margin in accordance with IEEE-323-1974. Where the TRS does not envelop 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. Operability requirements are described in Technical Requirements Manual (TRM) Section 3.4.2.

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.

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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. Stress limits for inelastic system (and/or component) analysis are in accordance with ASME Section III, Appendix F.

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.

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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 inservice inspections and inservice 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 post-faulted condition operation is not impaired by the seismic event.

In addition to the post-faulted 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 inservice inspections and inservice tests 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 [SOV]) 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. Safety Relief Valves

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 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, 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 applicable abnormal environmental conditions, 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.

Anchors 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 Inservice 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. Revision 0 of the First Ten-Year Interval IST Program was submitted to the NRC on March 31, 1988. Additional information was provided by letter dated September 30, 1988, and February 8, 1989. The NRC approved the IST program by letter dated October 29, 1990 (TAC No. 63429). The First Ten-Year Interval Program became effective on April 5, 1988, and was based on the ASME Boiler and Pressure Vessel Code, Section XI, 1983 Edition through the Summer 1983 Addenda. The purpose of the IST program is to ensure that certain ASME Class 1, 2, and 3 pumps provided with an emergency power source, and ASME 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 inservice pump and valve test program is based on the Code of record in accordance with 10CFR50.55a. The IST program plan will be periodically updated in accordance with 10CFR50.55a.

3.9A.6.1 Inservice Testing of Pumps

The IST program for certain ASME Class 1, 2, and 3 pumps that have an emergency power source is in accordance with the requirements of 10CFR50.55a and the Code of record for the IST program plan. 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 the IST program plan and its implementing documents.

Pumps are tested periodically during plant operation and during shutdown periods, in accordance with the IST program plan.

3.9A.6.2 Inservice Testing of Valves

The IST program for all ASME Class 1, 2, and 3 valves that are required to perform a safety function will be in accordance with the requirements of 10CFR50.55a. All valves requiring inservice testing will be listed in the valve testing section of the Unit 2 IST program plan. Each valve in the plan is categorized in accordance with the requirements of the approved Code Edition (i.e., the Edition and Addenda that are incorporated by reference in 10CFR50.55a, or that may be specifically approved for use at Unit 2 by the NRC). Test methods for each valve tested under the IST program plan will be described in the valve test procedures.

The valves which separate the RCPB, identified in Table 3.4.6-1 of the Technical Requirements Manual, Section 3.4.6, from interfacing low-pressure systems shall be leak tested in accordance with Technical Specifications.

These pressure isolation valves (PIV) are included in the Unit 2 IST program. The Unit 2 leak testing requirements for these valves are specified in the IST program plan and described by valve testing procedures.

1. 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 to the acceptance criteria for compliance with the Technical Specification requirement.

2. The periodic leak test will be performed during refueling outages.

3. After maintenance which can affect leak-tightness of the valve, leak testing will be performed in accordance with the IST program plan and the Technical Specification, prior to returning the valve to service.

P&IDs were supplied with the FSAR and Preservice and Inservice Inspection Plans which described inservice testing of the RCS pressure isolation valves. Procedures to support Technical Specification testing were generated in accordance with the startup schedule. During the Second Ten-Year Interval and successive intervals, the P&IDs are incorporated by reference into the IST program plan. The pump and valve testing procedures required to implement the IST program plan are part of the overall IST program and, like the P&IDs, they are separately controlled and maintained.

3.9A.6.3 Relief Requests

Proposed alternatives and requests for relief from the requirements of the ASME Code for IST of pumps and valves will be processed and submitted to the NRC, as required by 10CFR50.55a and the Unit 2 Technical Specifications. Implementation of relief requests and NRC-approved alternatives will be in accordance with Technical Specifications and 10CFR50.55a.

3.9A.6.4 Pipe Welds Within Break Exclusion Area

During each inspection interval, as defined in IWA-2400, an ISI is performed on all nonexempt ASME Code Section XI circumferential and longitudinal welds within the break exclusion region for high-energy fluid system piping. These inspections consist of augmented volumetric examinations (nominal pipe size greater than or equal to 4 in) and augmented surface examinations (nominal pipe size less than 4 in) such that 100 percent of the previously defined welds are inspected at each interval. The break exclusion zone consists of those portions of high-energy fluid system piping between the moment limiting restraint(s) outside the outboard containment isolation valve and the moment limiting restraint(s) beyond the inboard containment isolation valve. The choice of the restraint(s) that define the limits of the break exclusion zone is based upon those restraint(s) which

are necessary to ensure the operability of the primary containment isolation valves.

TABLE 3.9A-2 (Cont'd.)

PART II

ASME Code Class 2 and 3 Systems[1,2]

ASME III (1974) NC-3600 ND-3600 Equations (Class 2 & Class 3)	P	DW	OBEI [3]	OBEA [3]	SSEI [3]	Fluid Trans [3]	SRV [3-5]	SBA IBA Max of Chug or CO_a [3,4]	DBA Max of Bubble Form or Chug or CO_b [3,4]	CO_b	DBA AP [3]	Jet Imp [3,6]	Th	Anch Move Non-repeated	RHRB [3,4]	RHR Chug [3,4,14]	RHR CO [3,4,14]	Plant Conditions	ASME III Allow-ables [7,12,15]
Eq. 8 NC-3652.1 ND-3652.1	X[12]	X[12]																Normal	1.0 S_h
Eq. 9 XNC-3652.2 ND-3652.2 [5,9]	X	X	X			X	X											Upset	1.2 S_h
	X	X	X			X	X	X				X						Emergency (SBA, IBA)	1.8 S_h
	X	X	X			X				X		X						Emergency (SBA, IBA)	1.8 S_h
	X	X	X			X									X			Emergency	1.8 S_h
	X	X	X			X												Emergency	1.8 S_h
	X	X	X			X												Emergency	1.8 S_h
	X	X			X	X	X	X				X						Faulted (SBA, IBA)	2.4 S_h
	X	X			X	X				X		X						Faulted (SBA, IBA)	2.4 S_h
	X	X			X	X					X	X						Faulted (AP)	2.4 S_h
	X	X			X	X			X			X						Faulted (DBA)	2.4 S_h
Eq. 10 NC-3652.3 ND-3652.3				X									X					Normal & upset	S_A

TABLE 3.9A-2 (Cont'd.)

PART II (Cont'd.)

ASME III (1974) NC-3600 ND-3600 Equations (Class 2 & Class 3)	P	DW	OBEI (3)	OBEA (3)	SSEI (3)	Fluid Trans (3)	SRV (3-5)	SBA IBA Max of Chug or CO_a (3,4)	DBA Max of Bubble Form or Chug or CO_b (3,4)	CO_b	DBA AP (3)	Jet Imp (3,6)	Th	Anch Move Non-repeated	RHRB (3,4)	RHR Chug (3,4, 14)	RHR CO (3,4, 14)	Plant Conditions	ASME III Allow-ables (7,15)
Eq. 10a NC-3652.3 ND-3652.3 (11)														X				Normal, upset, emergency, faulted	3.0 S_c
Eq. 11 NC-3652.3 ND-3652.3	X	X		X									X					Normal & upset	$S_h + S_A$

TABLE 3.9A-2 (Cont'd.)

NOTES:

(1) The SRV, SBA, IBA, and DBA load cases (the columns between Fluid Trans and Jet Imp) occur in the reactor building only. Except for AP, they are suppression pool-related dynamic events. Piping inside or attached to the reactor building is affected by the reactor building vibrations up to the first anchor outside the building.

(2) References used to develop table:
a. ASME III issued 1974 dated July 1, 1974, and Code Case 1606-1.
b. Mark II Containment Dynamic Forcing Functions Information Report (DFFR) Rev. 4, November 1981.

(3) The amplitudes of events characterized by random vibration are determined with conservatism in such a way that the load combinations are in accordance with NRC NUREG-0484, Rev. 1.

Only OBEI and OBEA are combined by absolute summation. All other combinations of dynamic loads are by the SRSS method. Both OBEI and (OBEI + OBEA) are considered dynamic loads.

(4) In addition to inertia loads, piping submerged in the suppression pool experiences: Water drag loads during SRV and RHR relief valve discharge, CO, bubble formation, DBA pool swell and fallback, and acoustic pressure loads during chugging. Piping above the water level of the suppression pool experiences pool swell impact loads or froth loads, and fallback loads, in the DBA case.

(5) The damping values used for dynamic analysis are based on Regulatory Guide 1.61 (Table 3.7A-1), with OBE values applicable to normal and upset loads and SSE values applicable to emergency and faulted loads. Alternate damping values for seismic analysis will be those described in ASME Code Case N-411.

(6) Jet impingement from water or steam jets emanating from postulated breaks of piping of other systems applies only to areas identified as targets.

(7) Allowables given are based on the ASME Section III, 1974 edition. In addition, all ASME III 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 of NEDO-21985.

(8) For systems that require hydrotesting, repeat Equation 9 without occasional loads, using test pressure and deadweight for water-filled pipe. The allowables for the hydrotest case are given in NB-3226.

(9) OBEA load can be included in Equation 9 instead of Equation 10 in Class 2 and 3 systems (in accordance with ASME III, Paragraphs NC-3652 and ND-3652, respectively) and in Equation 12 instead of Equation 13 in Class 4 systems (in accordance with ANSI B31.1). Option is used in a few cases only.

(10) Due to the nature of dynamic cyclic loads, the inertia effects of dynamic loads including OBE displacements are considered in fatigue evaluation of ASME Code Class 1 piping. The cycles are consumed in a manner consistent with the example given in ASME Section III, Subsection NB, Subparagraph NB-3653.1.

(11) Equation 10a is adapted from the 1977 edition of ASME Section III, Subsections NC and ND.

(12) For systems that require hydrotesting, repeat Equation 8, using test pressure and deadweight for water-filled pipe. The allowable for the hydrotest case is 90 percent of the yield stress at temperature.

(13) Reference used to develop table: ANSI B31.1 dated 1973 and all addenda thereto up to and including Addendum C (issued December 31, 1973).

(14) A T-quencher of the same design as the main steam SRVDL has been installed to the RHR relief valve discharge piping to mitigate the relief valve actuation loads. As a result, the RHR Chugging and RHR CO loads are negligible.

(15) The faulted stress limits and analysis techniques specified in ASME Section III, Appendix F, can be applied. Inelastic methods can be used as allowed by the Code.

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) Section 3.7A.3.8.2 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 (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.	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. With the exception of the RCIC pressure control valve (2ICS*PCV115), 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. Piping design acceptance criteria ensure that actual dynamic loading is to be within the qualified levels for each valve. In the case where the TRS does not fully envelop the RRS, the general requirement for a retest may be exempted if the following criteria are met: 1) A point of the TRS may fall below the RRS by 10 percent or less, provided the adjacent 1/6 octave points are at least equal to the RRS and the adjacent 1/3 octave points are at least

TABLE 3.9A-4 (Cont'd.)

Equipment	Methods	Results
		10 percent above; 2) A maximum of 5 of the 1/6 octave analysis points, as in 1) above, may be below the RRS, provided they are at least one octave apart. For valve 2ICS*PCV115, an entire valve assembly with a similar actuator was qualified. (The tested assembly did not include the CONOFLOW I/P converter and filter regulator. Those are addressed in Table 3.10A-1). The assembly was subjected to biaxial, random multifrequency tests for each of the five OBE and SSE conditions. The tests were repeated in the second horizontal and vertical orientation. The valve assembly functioned as required before, during and after the tests. For valves affected by combined seismic and hydrodynamic 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 Valcor) 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-10

SUMMARY OF SEISMIC STRESS ANALYSIS RESULTS[1]

Components		Service Water Pumps (2SWP*P1A,B,C,D,E,F)		Spent Fuel Pumps (2SFC*P1A,B)	
		Actual	Allowable	Actual	Allowable
Motor holddown bolt stress	- shear, psi - tensile, psi	2,503 5,567	15,000 40,000	6,558 7,815	10,000 17,507
Pump holddown bolt stress	- shear, psi - tensile, psi	14,101 30,711	15,390[2] 40,500[2]	12,271 27,675	12,320 30,366
Anchor bolt stress	- shear, psi - tensile, psi	6,792 7,065	12,500 25,000	18,551 17,758	18,900 45,150
Shaft stress, psi		20,153	26,250	16,789	26,250
Frame stress, psi		19,125	21,600		
Thrust retainer bolt stress	- tensile, psi	3,251	21,960	2,284	20,000
Pump bearing bolt stress	- shear, psi - tensile, psi	2,450 6,332	10,800 21,960	2,267 4,894	10,000 20,000
Pump pedestal stress, psi		10,838	21,600	3,479	21,600
Nozzle stress	- discharge, psi - suction, psi	15,209 2,180	26,250 26,250	8,634 4,291	20,400 20,400
Nozzle flange stress	- discharge, psi - suction, psi	21,841 17,095	26,250 26,250	18,493 20,296	20,400 20,400
Pedestal weld stress, psi		3,989	10,800	4,870	10,800
Pump bearing load	- inboard, lb - outboard, lb	54 11,410	200 38,123	104 7,930	200 25,943
Flexible coupling misalignment, radians		0.0007	0.0178	0.00043	0.017
Impeller key stress	- shear, psi	4,042	10,500	2,624	10,500
Impeller relative deflection, in		0.005	0.007	0.005	0.0105

TABLE 3.9A-10 (Cont'd.)

Components		Chilled Water Pumps (2HVK*P1A,B)		Condensing Water Pumps (2SWP*P2A,B)	
		Actual	Allowable	Actual	Allowable
Motor holddown bolts stress	- shear, psi - tensile, psi	1,726[5] 3,447	10,000 20,000	5,093 7,821	10,000 19,851
Pump holddown bolts stress	- shear, psi - tensile, psi	12,336 24,551	16,800 45,150	12,058 17,570	16,800 25,000
Anchor bolt stress	- shear, psi - tensile, psi	6,475 14,224	10,000 17,640	2,279 3,323	10,000 20,000
Shaft stress, psi		6,760	17,500	5,551	17,500
Frame stress, psi		5,868	21,600	8,232	21,600
Thrust retainer bolt stress, psi		390	20,000	816	20,000
Upper pump frame bolt stress	- shear, psi - tensile, psi	14,714 81	22,800 55,200	7,508 2,258	10,000 15,987
Lower pump frame bolt stress	- shear, psi - tensile, psi	61 9,505	10,000 20,000	1,656 8,402	10,000 20,000
Frame adapter bolt stress	- tensile, psi	8,636	25,000	14,468	15,000
Frame adapter flange stress, psi		19,915	21,000	13,964	21,000
Maximum nozzle stress	- discharge, psi - suction, psi	6,456 15,586	21,000 21,000	3,728 12,480	20,760 20,760
Nozzle flange stress	- discharge, psi - suction, psi	14,926 18,493	21,000 21,000	8,974 17,194	20,760 20,760
Adapter/frame bolting stress	- tensile, psi	18,528	46,200	11,236	20,000
Pump bearing loads	- inboard, lb - outboard, lb	450 1,342	10,688 17,865	465 1,800	9,104 15,037

TABLE 3.9A-10 (Cont'd.)

Components	Chilled Water Pumps (2HVK*P1A,B)		Condensing Water Pumps (2SWP*P2A,B)		Standby Diesel Generator Fuel Oil Transfer Pump (2EGF*P1A,B,C,D, 2EGF*P2A,B)	
	Actual	Allowable	Actual	Allowable	Actual	Allowable
Flexible coupling misalignment, radians	0.0008	0.017	0.009	0.017		
Impeller connection stress - tensile, psi - shear, psi	863 1,933	17,500 8,750	521 1,079	17,500 8,750		
Impeller relative deflection, in	0.0052	0.010	0.004	0.012		
Maximum column stress, psi					19,250	22,500
Maximum column flange stress, psi bolt stress, psi					17,295 29,946	26,250 45,000
Maximum pump casing stress, psi					11,061	21,000
Maximum shaft stress, psi					16,515	17,500
Motor holddown bolt stress, psi - tensile - shear					2,427 782.5	20,000 10,000
Shaft key stress, psi					705.6	10,000
Shaft deflection, in					0.0279	0.05
Impeller deflection (clearance), in					0.000093	0.015
Nozzle stress, psi					17,257	21,000
Nozzle flange stress, psi bolt stress, psi					25,671 33,319	26,250 62,500
Support plate/cover bolt stress, psi - tensile - shear					2,531 2,397	22,500 9,300
Discharge head holddown bolts, psi - tensile - shear					25,587 1,607	40,000 12,320
Discharge head stress, psi					4,711	21,000

TABLE 3.9A-10 (Cont'd.)

Components	Chilled Water Pumps (2HVK*P1A,B)		Condensing Water Pumps (2SWP*P2A,B)		Standby Diesel Generator Fuel Oil Transfer Pump (2EGF*P1A,B,C,D, 2EGF*P2A,B)	
	Actual	Allowable	Actual	Allowable	Actual	Allowable
Motor adapter bolts, psi - tensile					5,696	20,000
- shear					1,211	10,000

(1) Values given represent the SSE + maximum nozzle + normal actual values compared to the normal (OBE) allowables.
(2) Both OBE and SSE allowables were used when calculating the pump holddown bolt interaction factors.
(3) Not applicable for this pump.
(4) Deleted.
(5) The installation of shear block to motor skid removes shear loading from the bolt. The value shown does not consider the addition of shear block and envelopes both the with and without shear block conditions.

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*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
	2CCP*V996, 997, 998, 999	4	Check	150	3	17	None	84
	2CCP*RV1019A, 1020A, 1021A, 1022A	3/4 x 1	SRV	150/150	2	8	None	56,9
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)	9
	2CPS*SOV133	1	Globe	1500	2	6	76P-035(7)	9
	2CPS*V50	1 1/2	Check	600	2	1	None	9
	2CPS*V51	1 1/2	Check	600	2	1	None	9
High Pressure Core Spray (CSH)	2CSH*V7	4	Check	150	2	1	None	78
	2CSH*V16	20	Check	150	2	1	None	77,78,30
	2CSH*AOV108	12	Check	900	1	3	Series 2A(6)	9,16
	2CSH*RV113	3/4 x 1	SRV	150/150	2	8	None	4,56
	2CSH*RV114	3/4 x 1	SRV	150/150	2	8	None	4,56
	2CSH*V9	16	Check	900	2	3	None	23
	2CSH*V17	3	Check	900	2	1	None	30
	2CSH*V55	3	Check	900	2	1	None	30
	2CSH*V59	14	Check	150	2	1	None	22
	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 2	SRV	300/150	2	8	None	4,9,56
	2CSL*RV123	3/4 x 1	SRV	150/150	2	8	None	4,9,56
	2CSL*V4	16	Check	300	2	1	None	74
	2CSL*V14	2	Check	600	2	1	None	30
	2CSL*EFV1	3/4	Check	1250	2	13	None	16
	2CSL*V9	12	Check	150	2	1	None	22,78
	2CSL*V21	2	Check	600	2	1	None	22

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
	2DER*RV344	3/4 x 1	SRV	150/150	2	8	None	56,9
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
	2DFR*RV228	3/4 x 1	SRV	150/150	2	8	None	56,9
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	22,88,78
	2EGF*V13	1	Check	600	3	1	None	22,88,78
	2EGF*V32	1	Check	600	3	1	None	22,88,78
	2EGF*V33	1	Check	600	3	1	None	22,88,78
	2EGF*V52	1	Check	600	3	1	None	22,88,78
	2EGF*V53	1	Check	600	3	1	None	22,88,78
Nitrogen Tanks (GSN)	2GSN*SOV166	1	Globe	1500	2	6	76P-035(7)	9
	2GSN*V70A,B	1	Check	600	3	18	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	20	Butterfly	150	2	9	85430(3)	11
	2GTS*MOV3A	20	Butterfly	150	2	9	85430(3)	11
	2GTS*MOV4A,B	8	Gate	150	2	1	SMB-00-15(1)	11
	2GTS*MOV28A	8	Butterfly	150	2	9	86040(3)	12
	2GTS*PV5A	14	Butterfly	150	2	9	86060(3)	13
	2GTS*AOV2B	20	Butterfly	150	2	9	NT420-SR3(2)	11
	2GTS*AOV3B	20	Butterfly	150	2	9	NT420-SR3(2)	11
	2GTS*AOV28B	8	Butterfly	150	2	9	NHD732-SR60(2)	12
	2GTS*PV5B	14	Butterfly	150	2	9	NHD732(2)	13
	2GTS*V68B	3/4	Check	2735	1	3	None	22
	2GTS*V74B	1	Check	2735	1	3	None	78
	2GTS*RV78B	1x2	Relief	600x150	3	8	None	4
Hydrogen Recombiner (HCS)	2HCS*MOV1A,B	3	Gate	150	2	15	SMB-00-10(1)	3,9
	2HCS*MOV2A,B	3	Globe	150	2	1	SMB-000-5(1)	3,9
	2HCS*MOV3A,B	3	Gate	150	2	15	SMB-00-10(1)	3,9
	2HCS*MOV4A,B	3	Gate	150	2	15	SMB-000-5(1)	3,9

TABLE 3.9A-12 (Cont'd.)

System Name	Mark Number	Size	Valve Type	Pressure Rating(*)	ASME Class	Mfg.	Valve Operator Model (Mfg.)	Active Function
Hydrogen Recombiner (HCS) (cont'd.)	2HCS*MOV5A,B	3	Globe	150	2	1	SMB-000-5(1)	3,9
	2HCS*MOV6A,B	3	Gate	150	2	15	SMB-000-5(1)	3,9
	2HCS*MOV25A,B	3	Globe	1500	2	18	SMB-000-2(1)	3
	2HCS*MOV26A,B	3/4	Globe	1500	2	18	SMB-000-2(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
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*RV1,2	3/4 x 1	SRV	150/150	4	8	None	4,97
	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*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	22,59,78
	2IAS*V471	1 1/4	Check	600	3	1	None	22,59,78
	2IAS*V421	1 1/4	Check	600	3	1	None	22,59,78
	2IAS*V431	1 1/4	Check	600	3	1	None	22,59,78
	2IAS*V526	1 1/4	Check	600	3	1	None	22,59,78
	2IAS*V546	1 1/4	Check	600	3	1	None	22,59,78
	2IAS*V581	1 1/4	Check	600	3	1	None	22,59,78
	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	3	SMB-0-25(1)	43
	2ICS*MOV121	10	Gate	900	1	1	SB-2-60(1)	17,94
	2ICS*MOV122	12	Gate	150	2	1	SMB-0-25(1)	44
	2ICS*MOV124	4	Gate	900	2	1	SB-00-10(1)	41
	2ICS*MOV126	6	Gate	900	1	1	SMB-1-60(1)	9,87
	2ICS*MOV128	10	Gate	900	1	1	SB-2-60(1)	17,94
	2ICS*MOV129	6	Gate	150	2	1	SMB-00-10(1)	46
	2ICS*MOV136	6	Gate	150	2	1	SMB-00-10(1)	47,16
	2ICS*MOV143	2	Globe	1500	2	1	SMB-00-5(1)	31
	2ICS*MOV148	1 1/2	Globe	1500	2	1	SMB-000-2(1)	16,48
	2ICS*MOV164	1 1/2	Globe	1500	2	1	SMB-000-2(1)	16,48
	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*V28	6	Check	150	2	1	None	82,78
	2ICS*V29	12	Check	150	2	1	None	80,78
	2ICS*V38	2	Check	1500	2	1	None	78,79
	2ICS*V39	1 1/2	Check	600	2	1	None	22,48,78
	2ICS*V40	1 1/2	Check	600	2	1	None	22,48,78
	2ICS*PCV115	2	Globe	900	2	5/19	(10)	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
	2ICS*EFV5	3/4	Check	1250	2	13	None	9
Reactor Vessel Instrumentation (ISC)	2ISC*RV33A,B	24	Vac Brkr	150	2	10	(4)	1
	2ISC*RV34A,B	24	Vac Brkr	150	2	10	(4)	1
	2ISC*RV35A,B	24	Vac Brkr	150	2	10	(4)	1
	2ISC*RV36A,B	24	Vac Brkr	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
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	20	V526-5688-19(11)	9
	2RSC*SOV105	3/4	Globe	1500	2	20	V526-5688-19(11)	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*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)	16,49,52, 53,55
	2RHS*MOV1C	24	Butterfly	300	2	9	SMB-0-25(1)	16,49
	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, 55,78
	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,B	14	Globe	300	2	5	SMB-00-5(1)	52,53
	2RHS*FV38C	14	Globe	300	2	5	SMB-00-5(1)	52
	2RHS*LV17A,B	4	Globe	300	2	5	None	52
	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	50,78
	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, 95
	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)	16,96
	2RHS*MOV113	20	Gate	900	1	1	SB-3-150(1)	16,96
	2RHS*MOV115	16	Gate	300	2	1	SMB-0-25(1)	51,89
	2RHS*MOV116	16	Gate	150	3	1	SMB-0-25(1)	51,89
	2RHS*MOV142	3	Globe	300	2	1	SMB-000-5(1)	52,96
	2RHS*MOV149	3	Gate	300	2	1	SMB-000-5(1)	52,96
	2RHS*PV21A,B	8	Globe	900	2	5	None	52
	2RHS*RV20A-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
	2RHS*RV152	3/4 x 1	SRV	900/150	1	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*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*SOV72A,B	1	Globe	1500	2	6	76P-051(7)	52
	2RHS*SOV73A,B	1	Globe	1500	2	6	76P-051(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	18	Butterfly	300	2	2	SMB-00-10(1)	50,53,55
	2RHS*MOV12B	18	Butterfly	300	2	2	SMB-00-10(1)	50,51,53, 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*RV57A,B	3/4 x 1	SRV	300/300	2	8	None	4
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)	36
	2SFC*HV18A,B	8	Butterfly	300	3	2	N721C-SR80-M3HW(2)	35,36
	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	22,78
	2SFC*V9	8	Check	150	3	1	None	78
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
	2SLS*V10	2	Check	1500	1	1	None	16,22,78
	2SLS*V12	1 1/2	Check	1500	2	1	None	66,78,22

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) (cont'd.)	2SLS*V14	1 1/2	Check	1500	2	1	None	66,78,22
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

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*RVV305	2 1/2	Check	150	3	10	None	21
	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)	71
	2SWP*MOV1A-F	4	Ball	150	3	11	SMB-000-2/H1BC(1)	2
	2SWP*MOV3A,B	30	Butterfly	150	3	9	SMB-2-60/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)	71
	2SWP*MOV50A,B	36	Butterfly	150	3	9	SMB-3-150/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*MOV17A,B	12	Gate	150	3	1	SMB-0-25(1)	91
	2SWP*MOV18A,B	12	Gate	150	3	1	SMB-0-25(1)	91
	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)	92
	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	Rotary Gte	150	3	12	SMB-0-15/H4BC(1)	72
	2SWP*MOV77A,B	54 x 54 (overall) 48 x 48 (nominal)	Rotary Gte	150	3	12	SMB-00-10/H3BC(1)	73
	2SWP*RV34A,B	4 x 6	SRV	300/150	3	8	None	56
	2SWP*TV35A,B	4	Globe	150	3	5	(3)	93
	2SWP*V1A-F	18	Check	150	3	9	None	22,78
	2SWP*V202A	30	Check	150	3	9	None	22,78
	2SWP*V202B	30	Check	150	3	9	None	22
	2SWP*V219A,B	4	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
Service Water (SWP) (cont'd.)	2SWP*V240A	4	Check	150	3	16	None	22,78
	2SWP*V240B	4	Check	150	3	16	None	22,78
	2SWP*V259	8	Check	150	3	1	None	22,78
	2SWP*V260	8	Check	150	3	1	None	22,78
	2SWP*V1002A,B	3	Check	150	3	1	None	78,90
	2SWP*V1027	30	Check	150	3	9	None	22,78
	2SWP*V1024	6	Check	150	3	1	None	22,78
	2SWP*V1025	6	Check	150	3	1	None	22,78
	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
	2SWP*V1194,1195, 1196,1197	6	Check	150	3	17	None	85
Reactor Water Cleanup (WCS)	2WCS*MOV102	8	Globe	600	1	1	SB-2-60(1)	16
	2WCS*MOV112	8	Globe	600	1	1	SB-2-60(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
Alternate Decay Heat Removal (ADH)	2ADH*V21A	10	Check	150	4	3	None	86
	2ADH*V21B	10	Check	150	4	3	None	86
	2ADH*V22A	10	Check	150	4	3	None	86
	2ADH*V22B	10	Check	150	4	3	None	86

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
17 = BNL Industries, Inc.
18 = Edward
19 = Control Components, Inc.
20 = Valcor

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
10 = Air/Control Components
11 = Valcor

(*) Pressure Rating - Inlet/Outlet

TABLE 3.9A-12 (Cont'd.)

Key to Active Functions

1 = Relieve pressure from suppression chamber to drywell.
2 = Pressure control (open and close) for self-cleaning strainers for service water pumps.
3 = Hydrogen recombiner isolation (active function to open 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.
10 = Open to provide 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. (Following design basis accidents and transients.)
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 post-accident 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 on LOCA signal to ensure maximum flow is injected into the vessel.
28 = Open for LPCS system injection and close for 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. Close on discharge line flow signal. A vent hole was drilled in the disc of 2CSL*MOV107 to prevent pressure locking.
32 = Containment isolation on low suppression pool level, and open to provide suction flow path.
33 = Reactor vessel drain line isolation and class break (ASME III to ANSI B31.1).
34 = Sample isolation.
35 = SFC loop isolation.
36 = Open to provide makeup for spent fuel pool.
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 isolation.
46 = Isolates on low CST level.
47 = Opens on low CST level.
48 = Turbine exhaust vacuum relief.
49 = LPCI injection.
50 = Shutdown cooling.

TABLE 3.9A-12 (Cont'd.)

Key to Active Functions (cont'd.)

51 = RHR/SWP cross-connect isolation for post-LOCA containment flooding; they are modified to meet the requirements of Generic Letter 95-07.
52 = System boundary isolation.
53 = Suppression pool cooling.
54 = RHR pump miniflow.
55 = Containment spray.
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 isolation 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. Open and close with pump start and stop.
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.
77 = HPCS system suppression pool supply line.
78 = Allow flow in the forward direction.
79 = Not used.
80 = RCIC turbine exhaust.
81 = Not used.
82 = RCIC system suppression pool supply line.
83 = CMS atmosphere sample valves.
84 = Maintain secondary containment integrity during operation of alternate drywell cooling system.
85 = Maintain secondary containment integrity during SWP chemical cleaning. Note: During normal plant operation, valve/piping assemblies (2SWP*V1194/V1195 and 2SWP*V1196/V1197) will be removed and penetrations will be secured with blind flanges.
86 = Provides secondary containment integrity.
87 = Open for RCIC injection and close for containment/reactor isolation.
88 = Fuel oil day tank transfer line.
89 = Opens for containment flooding; they are modified to meet the requirements of Generic Letter 95-07.
90 = Emergency makeup to spent fuel pool.
91 = SWP/CCP crosstie for SFC heat exchanger. Open to provide service water flow to SFC heat exchanger.
92 = Close on low header pressure to isolate SWP supply line to diesel generator.
93 = Flow/temperature modulation only.
94 = Close to isolate steam line break.
95 = Open for "pseudo LPCI" mode of RHR.
96 = Normally de-energized in the closed position and the power source removed for main control room fire concerns.
97 = The pressure relief valve is not required to maintain pressure boundary. It provides overpressure protection for the HVK system. It meets the requirements of ASME Section VIII.

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.

pc-Crack Program The pc-Crack program performs linear elastic fracture mechanics, elastic-plastic fracture mechanics, and limit load, in accordance with applicable codes and standards, to establish crack growth and weld overlay design. Verification of this program has been performed in accordance with the requirements of 10CFR50 Appendix B.

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.

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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

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. Under the combination of primary membrane (general or local) plus bending stresses, the faulted limit is 3.6 S_m or 57,600 psi at the design temperature in accordance with ASME Section III, Table F 1322.2-1. According to ASME Section III, Appendix I, Table I-1.2, 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.

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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], NEDE-21175-3-P[13], and for GE11 fuel, NEDE-31917P[17]. 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

<u>Preoperational and Startup Vibration Testing of Recirculation Piping</u>

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.

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], NEDE-21175-3-P[13], and for GE11 fuel, NEDE-31917P[17]. 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 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.

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 preservice tests, functional tests, and the required periodic inservice inspections and inservice tests. 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.

Analysis of Loading, Stress, and Acceleration Conditions

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

14. Design and Performance of General Electric Boiling Water Reactor Main Steam Isolation Valves, APED-5750, General Electric Company, March 1969.

15. BWRVIP-01 Rev. 1 (GENE-523-113-0894), "BWR Core Shroud Inspection and Flaw Evaluation Guidelines," March 1995.

16. BWRVIP-07, EPRI TR-105747, "BWR Vessel and Internals Project - Guidelines for Reinspection of BWR Core Shrouds," February 1996.

17. GE11 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR II), NEDE-31917P, General Electric Company, April 1991.

TABLE 3.9B-1

PLANT EVENTS

Normal, Upset, and Testing Condition	No. of Cycles
1. Boltup[1]	123
2. Design hydrostatic test[5]	130
3. Startup (100°F/hr heatup rate)[2] and cooldown (100°F/hr cooldown rate)[2][5]	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)[5]	80
8. 50% SSE event at rated operating conditions (OBE)	10/50[3]
9. Scram:[5]	
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.
(5) Monitor these events per Technical Specifications Section 5.5.5. Item 2 has a design cycle pressurization of ≥930 psig and ≤1250 psig; item 3 has a design cycle of 70°F to 565°F to 70°F; and item 9 has a transient of 100% to 0% of rated thermal power.

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]	$S_H = \frac{fMo}{Lg_1^2 B} + \frac{PB}{4g_o} < 1.5 S_m$ $S_R = \frac{(4te/3+1)Mo}{Lt^2 B} < 1.5 S_m$ $S_T = \frac{YMo}{t^2 B} - ZS_R < 1.5 S_m$ 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 = 0.25 S_m = 0.17 (allowable) S_T = 1.08 S_m = 0.72 (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 in^2 Outlet: $Am_1 (>Am_2)$ = 4.65 in^2	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: $t_{min} < t$ (3) Where: t_{min} = Minimum calculated thickness requirement, including corrosion allowance t_A = Actual nozzle wall thickness	Section near nozzle base: $t_{me} < t_{me}(actual)$ Nozzle midsection: $t_{mc} < t_{mc}(actual)$ Thin section near valve seat: $t_{mb} < t_{mb}(actual)$ Thinnest section at nozzle tip - just below valve seat: $t_{ma} < t_{ma}(actual)$ Nozzle Material: ASME SA350 LF2	$t_{me} = 0.84\,inch$ $t_{mc} = 0.81\,inch$ $t_{mb} = 0.79\,inch$ $t_{ma} = 0.206\,inch$ Actual thickness greater than t_m at the section under consideration	$t_{me}(actual) = 1.58\,t_{me}$ $t_{mc}(actual) = 1.54\,t_{mc}$ $t_{mb}(actual) = 1.012\,t_{mb}$ $t_{ma}(actual) = 1.68\,t_{ma}$

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: $It = \Sigma \frac{Nri}{Ni}$	$It = \Sigma \frac{Nri}{Ni} (i = 1,2,3,4,5)$	It (max) ≤ 1.0	It=0.0014
	Fatigue: $N_a \geq 2{,}000$ cycles, as based on Sa, where Sa is defined as the larger of: $Sp_1 = (2/3)Qp + \frac{Peb}{2} + Q_{T3} + 1.3 Q_{T1}$ or $Sp_2 = 0.4 Q_p + \frac{K}{2}(Peb + 2Q_{T3})$ Where: Sp_1 = Fatigue stress intensity at inside surface of crotch, psi Sp_2 = Fatigue stress intensity at inside surface of crotch, psi	$N_a \geq 2{,}000$ cycles, as based on Sa, where $Sa = Sp_1 (> Sp_2)$ [(Uses same notation as codes)]	$N_a \geq 2{,}000$ cycles	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: $S_R = \pm \frac{6MP}{t^2(3.14C - nD)}$ $S_T = \pm \frac{5.46MP}{BT^2}\left[0.318\left[\frac{C-B}{C+B} + \frac{2h_c}{C+A}\right] + r_B\right] - \frac{E\theta A t}{B}$ 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_H, S_R, and S_T), = 28,350 psi	S_R = .6 S_m = .4 (allowable) S_T = .14 S_m = .09 (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	$Am_1 = \frac{Wm_1}{Sb}$ $Am_2 = \frac{Wm_2}{Sa}$ Where: Am (required minimum) is the greater of Am_1 and Am_2; and	$Am_1 (>Am_2)$ = 7.995 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
	Am_2 = Total required bolt (stud) area for gasket seating	Ab (actual bolt area) must exceed Am. Body to bonnet bolting material: ASME SA-193 Gr. B7		
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: $t = 1.2\left[W\left(\frac{1-2r_o}{\frac{3R}{S}}\right)\right]^{1/2}$ W is based on p=1,375 psi under the disc	w = 27,432 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-2o

FUEL ASSEMBLY (INCLUDING CHANNEL)[2][3]

Acceptance Criteria	Loading	Primary Load Type	Calculated Peak Acceleration	Evaluation Basis Acceleration
Acceleration envelope	Horizontal direction Peak pressure Safe shutdown earthquake Annulus pressurization	Horizontal acceleration	2.6 G	[1]
	Vertical direction Peak pressure Safe shutdown earthquake Safety relief valve Chugging	Vertical accelerations	2.8 G[4]	[1]

[1] Evaluation basis accelerations and evaluations are contained in NEDE-21175-3-P-A, and for GE11 fuel, NEDE-31917P.
[2] The calculated maximum fuel assembly gap opening for the most limiting load combination is 0.01 in based on the methodology contained in NEDE-21175-3-P. This is much less than the gap required to start the disengagement of the lower tie-plate from the fuel support casting.
[3] The fatigue analysis indicates that the fuel assembly has adequate fatigue capability to withstand loadings resulting from multiple SRV actuations and the OBE + SRV event.
[4] These values are determined using methodology contained in NEDE-21175-3-P-A.

TABLE 3.9B-2r

RESIDUAL HEAT REMOVAL HEAT EXCHANGER

Loading	Criteria/Location	Allowable Stress or Minimum Thickness Required	Calculated Stress or Actual Thickness
1. Closure Bolting	Bolting loads and stresses calculated in accordance with Rules for Bolted Flange Connections, ASME Section III, Appendix XI		
Loads: Normal			
Design pressure and temperature			
Design gasket load	a. Shell-to-tube sheet bolts	25,000 psi	22,009 psi
	b. Channel cover bolts	25,000 psi	18,626 psi
2. Wall Thickness	Shell side ASME Section III, Safety Class 2 and TEMA, Class C		
Loads: Normal			
Design pressure and temperature	Tube side ASME Section III, Safety Class 3 and TEMA, Class C		
	a. Shell	0.78 in	0.875 in
	b. Shell cover	0.77 in	0.81 in min
	c. Channel	0.808 in	0.875 in
	d. Tubes	0.0515 in	0.054 in min
	e. Channel cover	6.77 in	6.82 in
	f. Tube sheet	6.22 in	6.25 in

TABLE 3.9B-2r (Cont'd.)

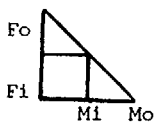
Loading	Criteria	Allowable Nozzle Forces and Moments	Calculated Nozzle Loads
3. Nozzle Loads: Faulted Design pressure and temperature Deadweight SSE SRV LOCA	The maximum moments due to pipe reaction and the maximum forces shall not exceed the allowable limits. Primary stress smaller of 0.7 Su or 2.4 S in accordance with ASME Section III allowable.	(1,2)	(3)
4. Support Brackets and Attachment Welds Loads: Faulted Design pressure and temperature Deadweight Nozzle loads SSE SRV LOCA	Stress allowables in accordance with ASME Section III, Subsection NT (upset condition). a. Lower bracket welds Bending stress Shear stress b. Upper bracket welds Bending stress Shear stress	 14,438 21,000 14,438 21,000	 2,169 3,646 1,983 853
5. Anchor Bolts Loads: Faulted Design pressure and temperature Deadweight Nozzle loads SSE SRV LOCA	Stress allowable in accordance with ASME Section III, Appendix XVII Lower support bolting Tension Shear	 29,000 11,990	 6,919 3,771

TABLE 3.9B-2r (Cont'd.)

Loading	Criteria/Location	Allowable Stress (psi) or Minimum Thickness Required	Calculated Stress or Actual Thickness
6. Shell Adjacent to Support Brackets Loads: Faulted Design pressure and temperature Deadweight Nozzle loads SSE SRV LOCA	Shell stress allowables in accordance with ASME Section III Subsection NC (upset condition) a. Maximum principal stress adjacent to upper support b. Maximum principal stress adjacent to lower support	 28,875 28,875	 18,864 23,097
7. Shell Away from Discontinuities Loads: Faulted Design pressure and temperature Deadweight Nozzle loads SSE SRV LOCA	Stress allowable in accordance with ASME Section III Subsection NC (upset condition) Principal stress	 19,250	 17,838

(1) Maximum allowable piping load combinations for faulted conditions (including DBE) shall not exceed the following relationship for each nozzle:

$$\frac{|Fi|}{|Fo|} + \frac{|Mi|}{|Mo|} \leq 1$$



Where:

Fi = Largest of the three actual external orthogonal forces (Fx, Fy, and Fz)
Mi = Largest of the three actual external orthogonal moments (Mx, My, and Mz) for the same reference coordinates
Fo = Allowable value of Fi when all moments are zero
Mo = Allowable value of Mi when all forces are zero

One coordinate axis must be the nozzle centerline. Another coordinate axis must be parallel to the heat exchanger centerline except where the heat exchanger centerline is parallel to the nozzle centerline. In this case, the coordinate axis must be orthogonal to the nozzle centerline and at 0°-180° or 90°-270° azimuths.

TABLE 3.9B-2r (Cont'd.)

(2) Allowable limits (design basis):

		N1	N2	N3	N4
Fx	=	13,000 lb	13,000 lb	15,500 lb	15,500 lb
Fy	=	13,000 lb	13,000 lb	15,500 lb	15,500 lb
Fz	=	13,000 lb	13,000 lb	15,500 lb	15,500 lb
Mx	=	46,000 ft-lb	46,000 ft-lb	60,000 ft-lb	60,000 ft-lb
My	=	46,000 ft-lb	46,000 ft-lb	60,000 ft-lb	60,000 ft-lb
Mz	=	46,000 ft-lb	46,000 ft-lb	60,000 ft-lb	60,000 ft-lb

(3) See as shown:

Faulted

Heat Exch	Nozzle No.	Fi	Mi	Fo	Mo	$\frac{\lvert Fi \rvert}{\lvert Fo \rvert} + \frac{\lvert Mi \rvert}{\lvert Mo \rvert}$
2RMS	N1	6,520	20,611	13,000	46,000	0.95
*E1A	N2	3,751	10,622	13,000	46,000	0.52
	N3	NA	NA	NA	NA	NA
	N4	NA	NA	NA	NA	NA
2RMS	N1	4,444	10,976	13,000	46,000	0.58
*E1B	N2	3,517	13,437	13,000	46,000	0.563
	N3	7,179	31,363	15,500	60,000	0.986
	N4	3,987	18,051	15,500	60,000	0.56

TABLE 3.9B-2s

RCIC TURBINE

Criteria/Loading	Component	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
The highest stressed sections of the various components of the RCIC turbine assembly are identified. Allowable stresses are based on ASME Section III for normal conditions: Pressure boundary castings SA-216-WCB: S = 17,500 psi Pressure boundary boltings, SA-193-B7: S = 25,000 psi Alignment dowel pins: AISI4037, Rc28-35 τ_a = 61,000 psi S_y = 106,000 psi				
Normal Condition Loads: 1. Design pressure 2. Design temperature 3. Inlet nozzle loads 4. Exhaust nozzle loads	Castings: a. Stop valve b. Governor valve c. Turbine inlet d. Turbine case Pressure-containing bolts Structure alignment pins	General membrane General membrane Local bending Local bending Tensile Shear	17,500 17,500 21,000 21,000 25,000 61,000	(1)
Upset, Emergency, or Faulted Condition(2): 1. Design pressure 2. Design temperature 3. SSE or OBE (horizontal and vertical, see Figures 3.9-1 and 3.9-2) 4. Inlet nozzle loads 5. Exhaust nozzle loads 6. SRV 7. LOCA	Castings: a. Stop valve b. Governor valve c. Turbine inlet d. Turbine case Pressure-containing bolts Structure alignment pins	General membrane General membrane Local bending Local bending Tensile Shear	19,250 19,250 25,200 25,200 25,000 61,000	13,860 15,300 15,300 18,000 20,100 53,080

TABLE 3.9B-2s (Cont'd.)

Criteria/Loading	Component	Limiting Stress Type	Allowable Load Criteria	Load Calculated
Nozzle Load Definition: Turbine vendor has defined allowable nozzle loads for the turbine assembly. The above calculated stresses assume these allowable nozzle loads have been satisfied.			Inlet: $F \leq \frac{(2,620-M)}{3}$ Exhaust: $F \leq \frac{(6,000-M)}{3}$	 F_R = 590.0 lb M_R = 837.0 ft-lb F_R = 1085.0 lb M_R = 2503.0 ft-lb
Normal Condition Loads: 1. Design pressure 2. Design temperature 3. Weight of structure 4. Thermal expansion			F = Resultant force (lb) M = Resultant moment (ft-lb)	
Upset, Emergency, and Faulted Condition Loads: 1. Design pressure 2. Design temperature 3. Weight of structure 4. Thermal expansion 5. SSE or OBE 6. SRV 7. LOCA			Inlet: $F \leq \frac{(7,000-M)}{7}$ Exhaust: $F \leq \frac{(8,500-M)}{0.34}$ but <7,000 F = resultant force (lb) M = resultant moment (ft-lb)	 F_R = 605.0 lb M_R = 893.0 ft-lb F_R = 1361.0 lb M_R = 3114.0 ft=lb

(1) Calculated stresses for the upset, emergency, or faulted condition are lower than the allowable stresses for the normal condition; therefore, normal condition does not need to be evaluated.

(2) Analysis indicates that shaft deflection with faulted loads is 0.014 in, which is fully acceptable, and maximum bearing load with faulted condition is 80% of allowable.

TABLE 3.9B-2t

RCIC PUMP

Criteria/Loading	Component	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Pressure boundary stress limits of the various components for the RCIC pump assembly are based on ASME Section III for pressure boundary parts at 140°F.				
1. Forged barrel SA-105 Gr. II — S_y = 36,000 psi				
2. End cover plates SA-105 Gr. II — S_y = 36,000 psi				
3. Nozzle connections SA-105 Gr. II — S_y = 36,000 psi				
4. Aligning pin SA-515 Gr. 60 — S_y = 36,000 psi				
5. Closure bolting SA-193-87 — S_y = 105,000 psi				
6. Pump holddown bolting SA-325 — S_y = 77,000 psi				
7. Taper pins SA-108 Gr. B1112 — S_y = 75,000 psi				
Normal and Upset Condition Loads:				
1. Design pressure	1. Forged barrel	General membrane	17,500	
2. Design temperature	2. Nozzle reinforcement	General membrane	17,500	
3. OBE	3. Alignment pin	Shear	15,000	
4. Suction nozzle loads	4. Taper pins	Shear	15,000	
5. Discharge nozzle loads	5. Pump holddown bolts	Tensile	40,000	
Emergency or Faulted Condition Loads[1]:				
1. Design pressure	1. Forged barrel	General membrane	17,500	7,052
2. Design temperature	2. Nozzle reinforcement at barrel discharge	General membrane	26,250	7,855
3. SSE	3. Alignment pin	Shear	18,000	2,230
4. Suction nozzle loads	4. Taper pins (bearing housing)		15,000	2,280
5. Discharge nozzle loads	5. Pump holddown bolts	Tension	48,000	33,662

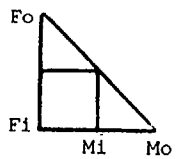
TABLE 3.9B-2t (Cont'd.)

Criteria/Loading	Component	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)

Nozzle Load Definition:

Units: Forces - lb
Moments - ft-lb

The allowable combinations of forces and moments are as follows:



$$\frac{Fi}{Fo} + \frac{Mi}{Mo} \leq 1$$

Where:

Fi = Largest absolute value of the three actual external orthogonal forces (Fx, Fy, Fz) that may be imposed by the interface pipe.
Mi = Largest absolute value of the three actual external orthogonal moments (Mx, My, Mz) permitted from the interface pipe when they are combined simultaneously for a specific condition.

TABLE 3.9B-2t (Cont'd.)

Criteria/Loading	Component	Limiting Stress Type	Allowable Loads (Ft-ft/lb)	Calculated Loads (Ft-ft/lb)
Normal and Upset Condition Loads: 1. Design pressure 2. Design temperature 3. Weight of structure 4. Thermal expansion 5. OBE		Fo - Allowable value of Fi when all moments are zero Mo - Allowable value of Mi when all forces are zero	Suction: Fo ≤ 1,940 Mo ≤ 2,460 Discharge: Fo ≤ 3,715 Mo ≤ 4,330	 Fi = 842 Mi = 1,224 Fi = 1,767 Mi = 2,012
Emergency or Faulted Condition Loads: 1. Design pressure 2. Design temperature 3. Weight of structure 4. Thermal expansion 5. SSE			Suction: Fo ≤ 2,325 Mo ≤ 2,950 Discharge: Fo ≤ 4,450 Mo ≤ 5,200	 Fi = 860 Mi = 1,248 (emergency) Fi = 1,799 Mi = 2,126 (faulted)

(1) Operability: state analysis for emergency or faulted condition shows that the maximum shaft deflection is 0.004 in (with 0.0055 in allowable), shaft stresses are 5,975 psi with 32,000 psi allowable, and bearing loads of, drive end 376 lb, with 7,670 lb allowable and thrust end 1,323 lb with 17,200 lb allowable.

TABLE 3.9B-2u

ECCS PUMPS

Location	Loading Condition	Criterion	Calculated Stress or Actual Thickness	Allowable Stress or Min Thickness
(i) Residual Heat Removal Pump				
Discharge head shell	Design pressure, Nozzle loads, SRV, Seismic loads, LOCA	ASME Section VIII, Division 1, Para. UG-27	24,253 psi	34,650 psi
Discharge head cover	Design pressure	ASME Section VIII, Division 1, Para. UG-34, UG-39, UG-40	3.00 in	2.63 in
Nozzle shell intersection	Faulted Condition Design pressure, Nozzle loads, SRV, Seismic load, LOCA	ASME Section VIII, Division 1, Para. UG-37	34,518 psi (suction) 28,905 psi (discharge)	34,650 psi
Discharge pipe	Faulted Condition Design pressure, Nozzle loads	ASME Section VIII, Division 1, Para. UG-27	14,791 psi	18,000 psi
Discharge head bolting	Faulted Condition Design pressure, Nozzle loads, Seismic, LOCA, SRV	Bolting loads and stresses in Rules for Bolted Flange Connections, ASME Section VIII, App. II	35,030 psi	45,000 psi
Motor bolting	Faulted Condition Seismic load, SRV, LOCA	Bolting loads and stresses in Rules for Bolted Flange Connections, ASME Section VIII, App. II	10,741 psi	25,000 psi
(ii) Low-Pressure Core Spray Pump				
Discharge head shell	Faulted Condition Design pressure, Nozzle loads, SRV, Seismic, LOCA	ASME Section VIII, Division 1, Para. UG-27	9,173 psi	34,650 psi
Discharge head cover	Design pressure	ASME Section VIII, Division 1, Para. UG-34, UG-39, UG-40	3.0 in	2.37 in
Nozzle shell intersection	Faulted Condition Design pressure, Nozzle loads, SRV, Seismic, LOCA	ASME Section VIII, Division 1, Para. UG-37	14,794 psi (suction) 16,929 psi (discharge)	34,650 psi
Discharge pipe	Faulted Condition Design pressure, Nozzle loads	ASME Section VIII, Division 1, Para. UG-27	17,920 psi	18,000 psi
Discharge head bolting	Faulted Condition Design pressure, Nozzle loads, SRV, Seismic, LOCA	Bolting loads and stresses in Rules for Bolted Flange Connections, ASME Section VIII, App. II	17,281 psi	45,000 psi

TABLE 3.9B-2u (Cont'd.)

Location	Loading Condition	Criterion	Calculated Stress or Actual Thickness	Allowable Stress or Min Thickness
Motor bolting	Faulted Condition Seismic load, SRV, LOCA	Bolting loads and stresses in Rules for Bolted Flange Connections, ASME Section VIII, App. II	3,159 psi	25,000 psi
(iii) High-Pressure Core Spray Pump				
Discharge head shell	Faulted Condition Design pressure, Nozzle loads, Seismic load	ASME Section VIII, Division 1, Para. UG-27	8,904 psi	34,650 psi
Discharge head cover	Design pressure	ASME Section VIII, Division 1, Para. UG-34, UG-39, UG-40	3.25 in	2.75 in
Nozzle shell intersection	Faulted Condition Design pressure, Nozzle loads, Seismic load	ASME Section VIII, Division 1, Para. UG-37	10,534 psi (suction) 15,611 psi (discharge)	34,650 psi
Discharge pipe	Faulted Condition Design pressure, Nozzle loads	ASME Section VIII, Division 1, Para. UG-27	10,175 psi	21,000 psi
Discharge head bolting	Faulted Condition Design pressure, Nozzle loads, Seismic load	Bolting loads and stresses in Rules for Bolted Flange Connections, ASME Section VIII, App. II	18,796 psi	45,000 psi
Motor bolting	Faulted Condition Seismic load	Bolting loads and stresses in Rules for Bolted Flange Connections, ASME Section VIII, App. II	5,878 psi	25,000 psi

TABLE 3.9B-2v

STANDBY LIQUID CONTROL PUMP

Criteria/Loading		Component	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Pressure boundary parts[1]:					
1. Fluid cylinder - SA-182-F304	S_y = 30,000 psi				
2. Discharge valve stop and cylinder head extension SA-479-304	S_y = 30,000 psi				
3. Discharge valve cover, cylinder head and stuffing box flange plate, SA-240-304	S_y = 30,000 psi				
4. Stuffing box gland, SA-564-630	S_y = 115,000 psi				
5. Studs, SA-193B7	S_y = 115,000 psi				
6. Dowel pins[2] alignment, SAE-4140	S_A = 23,400 psi				
7. Studs, cylinder tie, SA-193-B7	S_A = 25,000 psi				
8. Pump holddown bolts, SAE Gr. 8	T_A = 30,000 psi				
	Q_A = 37,500 psi				
9. Power frame, foot area, cast iron	S_A = 15,000 psi				
10. Motor holddown bolts, SAE Gr. 1	T_A = 12,000 psi				
	Q_A = 15,000 psi				
11. Motor frame foot area, cast iron	S_A = 7,500 psi				
Normal and Upset Condition Loads:					
1. Design pressure		1. Fluid cylinder	General membrane	17,800	
2. Design temperature		2. Discharge valve stop	General membrane	17,800	(4)
3. OBE		3. Cylinder head extension	General membrane	17,800	
4. Nozzle loads[3]		4. Discharge valve cover	General membrane	17,800	
5. SRV discharge		5. Cylinder head	General membrane	17,800	
6. Deadweight		6. Stuffing box flange plate	General membrane	17,800	
7. Thermal expansion		7. Stuffing box gland	General membrane	35,000	

TABLE 3.9B-2v (Cont'd.)

Criteria/Loading	Component	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Emergency Condition:				
1. Design pressure	1. Fluid cylinder	General membrane	21,360	4,450
2. Design temperature	2. Discharge valve stop	General membrane	21,360	13,600
3. Deadweight	3. Cylinder head extension	General membrane	21,360	13,600
4. Thermal expansion	4. Discharge valve cover	General membrane	21,360	8,150
5. Nozzle loads	5. Cylinder head	General membrane	21,360	8,150
6. Safety relief valve discharge	6. Stuffing box flange plate	General membrane	21,360	10,390
7. LOCA	7. Stuffing box gland	General membrane	42,000	11,420
Faulted Condition:				
1. Design pressure	1. Cylinder head studs	Tensile	25,000	18,820
2. Design temperature	2. Stuffing box studs	Tensile	25,000	24,750
3. Nozzle loads	3. Dowel pins[2]	Shear only[2]	23,400	19,430
4. Safety relief valve discharge	4. Studs, cylinder tie	Tensile[2]	25,000	8,685
5. LOCA	5. Pump holddown bolts	Shear	30,000	11,350
6. SSE	6. Pump holddown bolts	Tensile	37,500	17,680
	7. Power frame-foot area	Shear	15,000	1,850
	8. Power frame-foot area	Tensile	15,000	11,390
	9. Motor holddown bolts	Shear	12,000	3,470
	10. Motor holddown bolts	Tensile	15,000	5,660
	11. Motor frame-foot area	Shear	7,500	2,550
	12. Motor frame-foot area	Tensile	7,500	5,100

TABLE 3.9B-2v (Cont'd.)

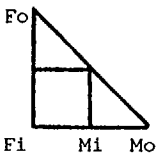
Criteria/Loading	Component	Limiting Stress Type	Allowable Stress (psi)	Actual Loads
Nozzle Load Definition: Units: Forces - lb Moments - ft-lb Allowable combination of forces and moments is as follows:  $\frac{Fi}{Fo} + \frac{Mi}{Mo}$ Where: Fi = The largest absolute value of the three actual external orthogonal forces (Fx, Fy, Fz) that may be imposed by the interface pipe, and M = The largest absolute value of the three actual internal orthogonal moments (Mx, My, Mz) permitted from the pipe when they are combined simultaneously for a specific condition.				
Normal and Upset Condition Loads: 1. Design pressure 2. Design temperature 3. OBE 4. Nozzle loads 5. SPV discharge 6. Deadweight 7. Thermal expansion		Fo = Allowable value of Fi when all moments are zero. Mo = Allowable value of Mi when all forces are zero.	Suction: Fo = 770 Mo = 490 Discharge: Fo = 370 Mo = 110	Suction: 2SLS*P1A Fi = 350 Mi = 182 2SLS*P1B Fi = 390 Mi - 231 Discharge: 2SLS*P1A Fi = 95 Mi = 70 2SLS*P1B Fi = 95 Mi = 70

TABLE 3.9B-2v (Cont'd.)

Criteria/Loading	Component	Limiting Stress Type	Allowable Stress (psi)	Actual Loads
Emergency or Faulted Condition Loads:			Suction:	Suction:
				2SLS*P1A
1. Design pressure			Fo = 920	Fi = 479
2. Design temperature			Mo = 590	Mi = 249
3. Nozzle loads				
4. SRV discharge				2SLS*P1B
5. LOCA				
6. SSE				Fi = 491
				Mi = 212
			Discharge:	Discharge:
				2SLS*P1A
			Fo = 440	Fi = 97
			Mo = 130	Mi = 75
				2SLS*P1B
				Fi = 97
				Mi = 75

(1) Based on ASME Boiler & Pressure Vessel Code Section III.
(2) Dowel pins take all shear.
(3) Nozzle loads produce shear loads only.
(4) Calculated stresses for emergency or faulted condition are less than the allowable stresses for the normal and upset condition loads; therefore, the normal and upset condition is not evaluated.
(5) Will be provided in a future amendment.

NOTE: Operability: The sum of the plunges and rod assembly (pounds mass times 1.75) acceleration is much less than the thrust loads encountered during normal operation conditions. Therefore, the loads during the faulted condition have no significant effect on pump operability.

TABLE 3.9B-2y

CRD HOUSING SUPPORTS

Criteria	Loading	Allowable Stress (psi)	Calculated Stress (psi)
<u>Beams</u>			
Allowable stresses based on AISC specification for the design, fabrication and erection of structural steel for buildings.			
F_y @ 150°F = 36,000 psi[1]			
For normal and upset conditions:	Normal and upset loads:[2]		
F_a = 0.60 F_y (tension)	(Negligible)		
F_b = 0.66 F_y (bending)			
F_v = 0.40 F_y (shear)			
For emergency condition:	Emergency loads:[2]		
	(Negligible)		
For faulted condition:	Faulted loads:		
F_a = 1.50 x 0.60 x F_y (tension)	1. Deadweight	F_b = 33,000 (top chord)	f_b = 28,700
F_b = 1.5 x 0.60 x F_y (bending)	2. Impact force from blow-out of CRD housing	F_b = 33,000 (bottom chord)	f_b = 22,000
F_v = 1.5 x 0.40 x F_y (shear)			
<u>Grid Structure</u>			
Allowable stresses based on AISC specification for the design, fabrication and erection of structural steel for buildings.			
F_y @ 150°F = 46,000 psi[1]			
For normal and upset conditions:	Normal and upset loads:[2]		
F_a = 0.60 F_y (tension)	(Negligible)		
F_b = 0.66 F_y (bending)			
F_v = 0.40 F_y (shear)			

TABLE 3.9B-2y (Cont'd.)

Criteria	Loading	Allowable Stress (psi)	Calculated Stress (psi)
For emergency condition:	Emergency loads:[2]		
	(Negligible)		
For faulted condition:	Faulted loads:		
F_a = 1.50 x 0.60 x F_y (tension)	1. Deadweight	F_b = 41,500 (top chord)	f_b = 40,700
F_b = 1.5 x 0.60 x F_y (bending)	2. Impact force from blow-out of CRD housing	F_b = 27,500 (bottom chord)	f_b = 12,500
F_v = 1.5 x 0.40 x F_y (shear)			

NOTE: Cumulative usage factor is not significant because only one loading cycle (blow-out of CRD housing) is applied in the design life of the equipment.

(1) F_y = Material yield strength.
(2) Deadweights and earthquake loads are very small compared to impact force.

TABLE 3.9B-2z

MAIN STEAM ISOLATION VALVES

Item No.	Component/Load/Stress Type	Design Procedure	Allowable Limit	Design/ Calculated Value
1.0	Body and Bonnet			
1.1	Loads: Design pressure Design temperature Pipe reaction Thermal effects	 System requirement System requirement Not specified Not specified	 1,375 psig 586°F N/A N/A	 N/A N/A N/A N/A
1.2	Pressure rating	ASME Section III[1] Paragraph NB-3543c	P_r = 575 psig	P_r = 575 psig
1.3	Minimum wall thickness	ASME Section III[1] Paragraph NB-3542.1	t (nominal) = 2.05 in	t_m = 1.76 min, in
1.4	Primary membrane stress	ASME Section III[1] Paragraph NB-3545.1	$P_m \leq S_m$ (500°F) = 21,600 psi	P_m = 7,840 psi
1.5	Secondary stress due to pipe reaction	ASME Section III[1] Paragraph NB-3545.2	P_e = greatest value of P_{ed} P_{eb} and $P_{et} \leq 1.5\ S_m$ (500°F) 1.5 (21,600) = 32,400 psi	P_{ed} = 4,150 psi P_{eb} = 9,570 psi P_{et} = 7,040 psi $P_e = P_{eb}$ = 9,570 psi
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) = 64,800 psi	Q_p = 24,700 psi
1.7	Thermal secondary stress	ASME Section III[1] Paragraph NB-3545.2 (c)	$S_n \leq 3\ S_m$ (500°F) = 64,800 psi	Q_t = 4,130 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) = 64,800 psi	$S_n = Q_p + P_e\ 2Q_t$ = 42,500 psi
1.9	Fatigue requirements	ASME Section III[1] Paragraph NB-3545.3	$N_a \geq$ 2,000 cycles	N_δ = 30,000 cycles
1.10	Cyclic rating	ASME Section III[1] Paragraph NB-3550	$I_t \leq 1.0$	I_t = 0.0054

TABLE 3.9B-2z (Cont'd.)

Item No.	Component/Load/Stress Type	Design Procedure	Allowable Limit	Design/ Calculated Value
2.0	Body to Bonnet Bolting			
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	$A_b \leq 53.40$ in^2 $S \leq 34,700$ psi	$A_b = 53.40$ in^2 $S = S_b + S_{te} + S_{tv} = 22,700$ psi
2.3	Body Flange Stresses			
2.3.1	Operating conditions	ASME Section III[1] Paragraph NB-3647.1, XI-3240	$S_H \leq 1.5\ S_m$ (500°F) = 26,200 psi $S_R \leq S_m$ (500°F) = 17,500 psi $S_T \leq S_m$ (500°F) = 17,500 psi $(S_H + S_R)/2 \leq S_m = 17,500$ $(S_H + S_T)/2 \leq S_m = 17,500$	$S_H = 15,100$ psi $S_R = 9,800$ psi $S_T = 8,900$ psi $(S_H + S_R)/2 = 12,500$ psi $(S_H + S_T)/2 = 12,000$ psi
2.3.2	Gasket seating condition	ASME Section III[1] Paragraph NB-3647.1	$S_H \leq 1.5\ S_m$ (100°F) = 26,200 psi $S_R \leq S_m$ (100°F) = 17,500 psi $S_T \leq S_m$ (100°F) = 17,500 psi $(S_H + S_R)/2 \leq S_m$ (100°F) = 17,500 psi $(S_H + S_T)/2 \leq S_m$ (100°F) = 17,500 psi	$S_H = 17,600$ psi $S_R = 11,400$ psi $S_T = 10,300$ psi $(S_H + S_R)/2 = 14,500$ psi $(S_H + S_T)/2 = 14,000$ psi
3.0	Stresses in Stem			
3.1	Loads: operator thrust	--	--	--
3.2	Stem tensile stress	Calculate stress due to operator thrust in critical cross section	$S_t \leq S_m$ (500°F)[2] = 23,400 psi	$S_t = 19,200$ psi, max.
3.3	Stem thread stress	Calculate shear stress due to maximum stem load on thread stress area	$S \leq 0.6\ S_m$ (500°F)[2] = 14,000 psi	$S = 7,500$ psi

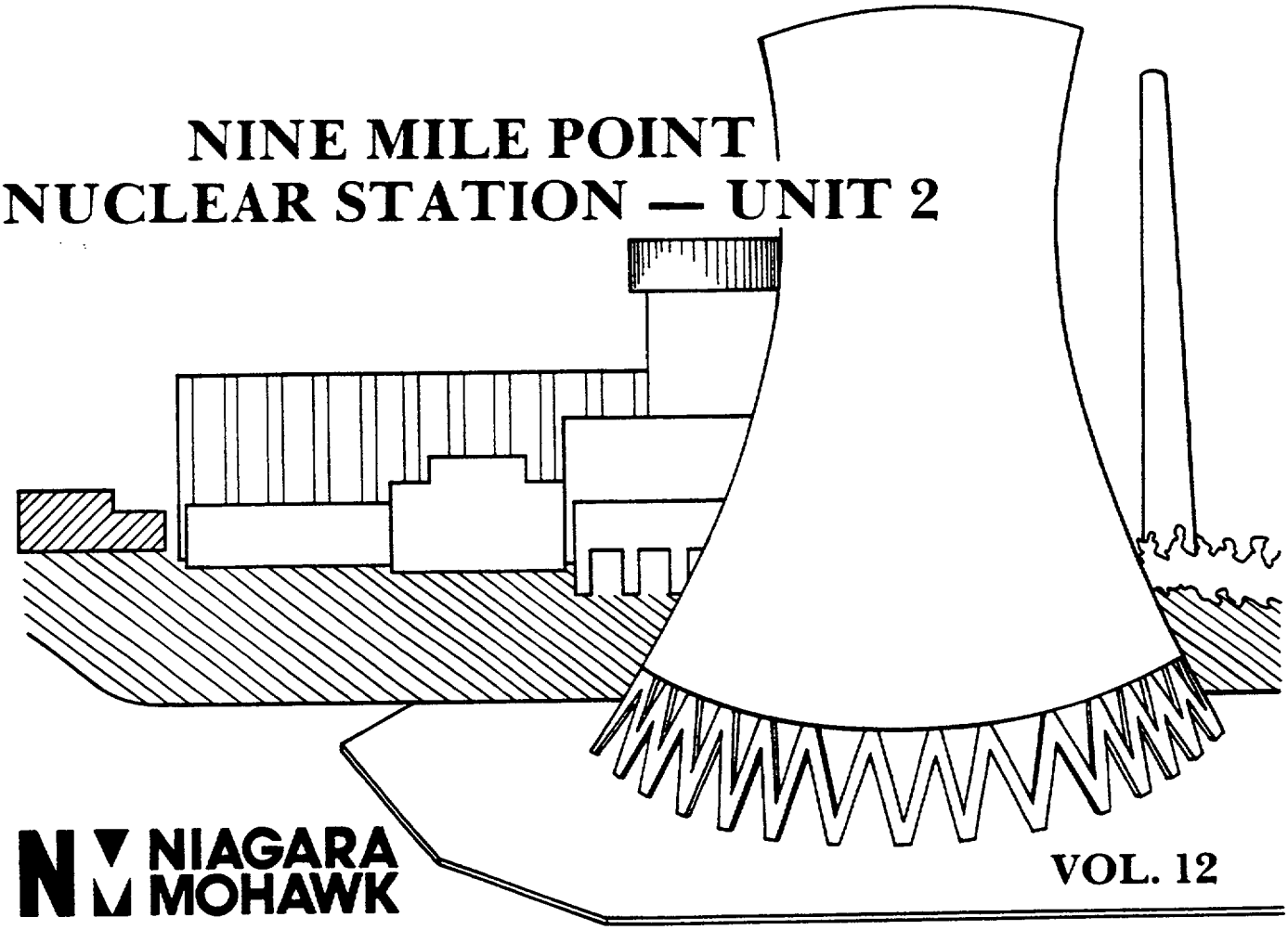
TABLE 3.9B-2z (Cont'd.)

Item No.	Component/Load/Stress Type	Design Procedure	Allowable Limit	Design/ Calculated Value
3.4	Buckling on stem	Calculate critical load, P_{cr} Actual stem load should be less than P_{cr}	P_{cr} = 99,136 lb	Actual stem load = 39,450 lb; therefore, no buckling
4.0	Disk Analysis			
4.1	Maximum stress in disk	The disk stress intensity is calculated using a finite element computer program, which iterates disk thickness until the stress in the plate is less than that allowed by code	S_m (500°F) = 19,400 psi	Max. stress = 13,000 psi

~~(1)~~ ASME Section III, 1977 Edition through Summer 1977 Addenda.
(2) Valve stem material ASTM A-182, Gr F6A, Cl 3 in accordance with ASME Section III, 1980 Edition through Summer 1981 Addenda.

UPDATED SAFETY ANALYSIS REPORT

NINE MILE POINT NUCLEAR STATION — UNIT 2



NIAGARA MOHAWK

VOL. 12

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 a 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 a 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 TRS closely envelop applicable portions of the RRS 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 ARS 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 ARS 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.

TABLE 3.10A-1 (Cont'd.)

Equipment	Methods	Results
Solenoid-Operated Valves - ASCO	The solenoid operators are qualified by dynamic testing. Applicable standards and guidelines are IEEE-323-1974, IEEE-344-1975, IEEE-382-1972, IEEE-627-1980, and RG 1.89 and 1.100.	Some of the solenoid valves are affected by seismic loads only. Others are affected by both seismic and hydrodynamic loads. For valves affected by seismic loads, the testing consists of a sinusoidal sweep resonance search in all three axes from 1 to 35 Hz followed by a series of single frequency, single axis sine beat tests at 38 test frequencies between 1 and 35 Hz. The required safety function of the valves to shift position is demonstrated during and after the testing. The tests establish the valve adequacy to a minimum input level of 7.2 g. For solenoid valves affected by hydrodynamic loads, the results of resonance searches performed on various models in the frequency range of 1-100 Hz are utilized to demonstrate the valve's adequacy in the entire range of dynamic loading. In addition, the durations and stress cycle requirements of the dynamic loading are met by a large number of single frequency, single axis, sine beat tests. The safety function of the valve is demonstrated during and after testing, and valve adequacy is established up to an input level of 6.2 g. Piping design acceptance criteria ensure actual dynamic loadings to be within the qualified levels for each valve.
Solenoid Valves - AVCO	Solenoid valves are qualified by dynamic testing. The applicable standards and guidelines are IEEE-323-1974, IEEE-344-1975, IEEE-382-1980, IEEE-382-1985, and RG 1.89 and 1.100.	The valves are affected by seismic loads only. The testing consists of two single-axis sine sweep tests, from 2 Hz to 35 Hz to 2 Hz, followed by single-axis sine beat tests at each one-third octave frequency from 2 Hz to 32 Hz. The required safety function of the valves to shift position is demonstrated during and after the testing. Piping design acceptance criteria ensure that actual dynamic loading is within the qualified levels for each valve.
Pressure Switches - Static O-Ring Switches	Pressure switches are qualified by dynamic testing. The applicable standards and guidelines are IEEE-323-1974, IEEE-344-1975, and RG 1.89 and 1.100.	The pressure switches are affected by seismic loads only. The pressure switches are qualified by biaxial, random multifrequency tests of 30-sec duration each for each of the five OBE and one SSE conditions. The tests were repeated in the second horizontal and vertical orientation. The pressure switches successfully completed the seismic testing by performing their intended safety functions during and after all tests. The TRS enveloped the RRS within the applicable frequency range with a margin of at least 10 percent.
Positioner - Wyle/Virginia Valve	Pressure switches are qualified by dynamic testing. The applicable standards and guidelines are IEEE-323-1974, IEEE-344-1975, and RG 1.89 and 1.100.	The positioner is affected by seismic loads only. The positioner is qualified by triaxial, random multifrequency tests of 30-sec duration each for each of the five OBE and one SSE conditions. The positioner successfully completed the seismic testing by performing its intended safety functions after all tests. The TRS enveloped the RRS within the applicable frequency range with a margin of at least 10 percent.

TABLE 3.10A-1 (Cont'd.)

Equipment	Methods	Results
Control Panels and Instrument Racks	Control panels, instrument racks, and mounted devices are qualified by a combination of dynamic testing and analysis. Applicable code, standards, and guidelines are the AISC Code, IEEE-323-1974, IEEE-344-1975, and NRC RG 1.61, 1.89, 1.92, and 1.100.	Control panels and racks are affected by seismic loads only. The panels and racks are qualified by a finite element analysis. The natural frequency results indicate that the panel and racks are rigid and that the floor motion was not amplified at the various component mounting locations. Results of the static analysis indicate that the stresses are well within the allowable limits of paragraph 3.9A.2.2.2, and the margin of safety is over 15 percent. All of the active components, including Foxboro racks, are qualified by dynamic testing. The dynamic test is performed as follows: The test items are mounted during the test to simulate plant installation and are instrumented to record accelerations and monitor operability. The test consists of a resonance search from 1 to 35 Hz in three axes, followed by a biaxial, random multifrequency series of five OBE and one SSE tests with amplitude controlled in one-third octave bandwidths over a frequency range of 1 to 40 Hz. These tests are repeated in the second horizontal orientation to consider all three axes of loading. The TRS envelops the RRS within the applicable frequency range with at least a 10 percent margin. All of the tested items successfully completed these tests and performed their intended functions.

Radiation Monitoring System
 	
Component
 	
• Liquid monitor assembly
• Gas monitor assembly with process monitor microcomputer
• Gas monitor assembly without microcomputer
• Particulate and gas monitor assembly
• Remotely mounted microcomputer
• Control room cabinet, including:
  • Interface modules
  • Safety isolation modules
  • Remote indication and control units
  • Analog and digital isolation modules
	The radiation monitoring system is qualified by a combination of analysis and dynamic testing. The following is a list of essential components and the corresponding methods of qualifications:
	
	Method
	
	Static analysis, dynamic test
	
	Static analysis, dynamic test
	
	Static analysis, dynamic test
	
	Static analysis, dynamic test
	
	Dynamic test
	
	Dynamic analysis
	
	Dynamic test
	Dynamic test
	
	Dynamic test
	
	Dynamic test
		All components are affected by seismic loads, with the ion chamber detectors additionally subjected to hydrodynamic loads. All devices are identical or similar to parent items, which are qualified by dynamic testing, with the exception of the isokinetic nozzles and the control room cabinets. In addition, all radiation monitor skids are analyzed for structural integrity using ANSYS modal finite element and static analysis options to determine frequencies, stresses, and deflections. The control room cabinets are qualified by use of the dynamic finite element analysis option of the same program. The isokinetic probes, being simple collector tubes, are analyzed by conventional manual methods, as are the pump/motor assemblies (with the exception of the liquid monitor sample pumps in 2SWP*CAB23A and 2SWP*CAB23B, and 2SWP*CAB146A and 2SWP*CAB146B), since these items have natural frequencies higher than 33 Hz.
		
		The dynamic testing is performed as follows: A resonance search in three axes is performed from 1 to at least 33 Hz. Five OBE and one SSE random multifrequency tests of 30-sec duration each, using biaxial/triaxial motions, with amplitude controlled in one-third octave bandwidths over a frequency range of 1 to at least 33 Hz, are performed and, in the case of biaxial motions, repeated in the alternate horizontal orientation to consider all three axes of loading. The TRS envelopes the RRS by at least a 10 percent margin in the frequency range of interest. The components are instrumented to record accelerations and monitor operability before, during, and after the vibration tests. Equipment remained functional, and no structural damage was noted.

TABLE 3.10A-1 (Cont'd.)

Equipment	Methods	Results
Component • Strip chart recorders • Ion chamber detector • Remote indicator and alarm module • Electric motor for gas monitor sample pumps • Isokinetic probes • Pump/motor for liquid monitor sample pumps	Method Dynamic test Dynamic test Dynamic test Static analysis Static analysis Static analysis (with exception of 2SWP*CAB23A and 2SWP*CAB23B, and 2SWP*CAB146A and 2SWP*CAB146B, pumps originally qualified by static analysis; later replaced and qualified by dynamic testing). Non-Class 1E components are qualified through their similarity to the Class 1E components or by static analysis. The applicable standards and guidelines are IEEE-323-1974, IEEE-344-1975, and NRC RG 1.89, 1.92, 1.97, and 1.100.	For ion chamber detectors affected by hydrodynamic loads, testing includes high frequency and long duration characteristics of the loading as follows: The resonance search is extended to a frequency of 200 Hz. The input motions for the random testing contain frequencies between 1 and 100 Hz. The TRS envelopes the RRS by at least a 10 percent margin in the applicable frequency range. The random testing consists of several additional tests with a frequency range to 200 Hz, at various amplitudes, in addition to the six tests of 30-sec duration and for 30-min duration in each axis. An analysis to compare the test equivalent stress cycles with those from the postulated dynamic loads is also performed. This analysis shows that the test motions contain equivalent stress cycles greater than those from the postulated dynamic loads.
Loose Parts Monitoring System (Category II)	The equipment was qualified by a combination of test and dynamic analysis, utilizing the response spectrum modal analysis technique. Applicable code, guidelines, and standards are the AISC Code, NRC RG 1.100, 1.133 and IEEE-344-1975.	The loose parts monitoring system equipment is affected by seismic loads and is qualified to withstand five OBE events only. A dynamic response spectrum modal analysis was performed on a model simulating the cabinet and masses of the large instruments. The lowest natural frequency of the cabinet was 24 Hz. The maximum stress margin of safety is greater than 75 percent. The devices in the cabinet were qualified by testing a generic unit. The test consisted of a resonance search followed by a biaxial, random multifrequency series of five OBE tests. The biaxial tests were repeated in a second orientation to consider all three axes of loading. The TRS enveloped the RRS within the applicable frequency range with at least a 10 percent margin.
Centrifugal Liquid Chillers	Centrifugal liquid chillers are qualified by a combination of analysis and tests. The control panel and Class 1E electrical components of the centrifugal liquid chillers are qualified by dynamic testing. The skid, main shell, and pipes are qualified by finite element analysis. The applicable standards, guidelines, and codes are IEEE-323-1974, IEEE-334-1974, IEEE-344-1975, RG 1.89 and 1.100, and ASME Section III, Subsections ND and NF, including Addenda, 1974 Edition.	Centrifugal liquid chillers are affected by seismic loads only. The control panel and Class 1E electrical components of the chillers are qualified by dynamic testing. Representative test specimens were mounted on the vibration test table such that in-service conditions were simulated. The specimens were instrumented to record accelerations and monitor operability. A resonance search from 1 to 35 Hz was performed in each of three orthogonal axes. The seismic simulation vibration tests consisted of biaxial, random multifrequency tests of five OBEs and one SSE in each of two test orientations 90 deg apart. The TRS enveloped the RRS within the applicable frequency range with a margin of at least 10 percent. The test specimens did not exhibit any malfunctions as a result of the seismic simulation tests. The skid, main shell, and pipes are qualified by finite element analysis. A local shell analysis of all

TABLE 3.10A-1 (Cont'd.)

Equipment	Methods	Results
		major pipe-to-shell attachments was also performed. The stresses and deflections were found to be within the acceptable limits of Table 3.9A-8.
Electric Heat Tracing Control Panels	Electric heat tracing control panels are qualified by dynamic testing. The applicable standards and guidelines are IEEE-323-1974, IEEE-344-1975, and RG 1.89 and 1.100.	The heat tracing control panels are affected by seismic loads only. A test specimen of identical construction was mounted on the vibration test table such that in-service conditions were simulated. The specimen was instrumented to record accelerations and monitor operability. A resonance search from 1 to 35 Hz was performed in each of three orthogonal axes. The seismic simulation tests consisted of biaxial, random multifrequency tests, five OBEs and one SSE in each of two test orientations 90 deg apart. The TRS enveloped the RRS within the applicable frequency range with a margin of at least 10 percent. The test specimen did not exhibit any malfunction as a result of the seismic simulation tests.
Electrical Air Duct Heaters	Qualification of air duct heaters is by analysis and test. Applicable guidelines, codes, and standards are NRC RG 1.61, 1.89, 1.92, 1.100, the AISC Code, and IEEE-323-1974 and IEEE-344-1975.	The air duct heaters are affected by seismic loads only. Analysis of the enclosures for the heater and remote control panel is performed using a finite element model. The natural frequencies are determined. Static and dynamic analysis is performed to calculate stresses and deflections. The results show that stresses are within the allowables of paragraph 3.9A.2.2.2 and that deflections are negligible. For qualification of devices, a test program consisting of resonance search and random multiaxial and multifrequency tests shows that the TRS envelopes the RRS in the applicable frequency range by a margin of at least 10 percent.
Diesel Generator System <u>Component/System</u> • Engine mounted systems (including combustion air manifold, exhaust manifold, shutdown butterfly valve, jacket water headers, governor linkages, fuel oil and lube oil systems, and starting air system • Air intake filter	This system is qualified by a combination of analysis and dynamic testing programs. The following is a list of essential components and systems and the corresponding methods of qualification. <u>Method</u> Static analysis Static analysis	The diesel generator system is affected by seismic loads only. The main unit is qualified seismically by response spectrum finite element modal analysis. The model consists of major mass and structural items, including the engine itself, the flywheel, generator rotor, outboard bearing, pedestal, and base. Items too small to affect the dynamic characteristics of the system are excluded. Results of the dynamic analysis indicate that the stresses in the analyzed components are well within the allowable limits of paragraph 3.9A.2.2.2 with substantial margin. Amplified response spectra are also generated (by analysis) at different points of the diesel generator where various equipment is mounted. The tested components are subjected to sine sweeps from 1 to 40 Hz in three axes, followed by a biaxial, random multifrequency series of at least five OBE and one SSE tests. The TRS envelopes the RRS within the frequency range of interest with at least a 10 percent margin. Each component is mounted during the test to simulate the plant installation and instrumented to record acceleration and monitor operability.

TABLE 3.10A-1 (Cont'd.)

Equipment	Methods	Results
Component/System	Method	The lube oil thermostatic valve is qualified through analysis and single frequency testing. It is subjected to a series of increasingly severe sine sweeps at the rate of two octaves/min from 1 to 50 Hz at a level of 2.0 g, increased to 16.0 g. Auxiliary skid piping, jacket water standpipe, fuel oil filter and strainer, turbocharger lube oil filter, jacket water cooler, lube oil heat exchanger, and generator stator and brush mounting structures are all analyzed by the modal response spectrum analysis method using floor response spectrum. Torsional frequencies of the crankshaft system are determined by analysis. Torsional stresses are also determined at several locations of the crankshaft due to the stimulation torques at the potentially significant critical speeds. There is no conceivable operating condition in which the system torsional vibration can damage or adversely affect operation of the unit. The remaining components are qualified through static analysis. The results of all analyses and tests show substantial margins compared to the maximum required acceleration level, ensuring the ability of the diesel generator to function under all operating and postulated loadings.
• Air intake silencer	Static analysis	
• Exhaust silencer	Static analysis	
• Engine-driven lube oil pump	Static analysis	
• Jacket water standpipe	Dynamic analysis	
• Jacket water circulating pump	Static analysis	
• Fuel oil filter and strainer	Dynamic analysis	
• Turbocharger lube oil filter	Dynamic analysis	
• 3-in check valve	Static analysis	
• 6-in check valve	Static analysis	
• Lube oil and jacket water heater	Static analysis	
• Auxiliary skid piping	Static & dynamic analysis	
• Exhaust expansion joint	Static analysis	
• Intake expansion joint	Static analysis	
• Governor actuator and overspeed governor	Dynamic analysis	
• Intercooler water piping	Static analysis	
• Starting air tank	Static analysis	
• Lube oil filter	Static analysis	
• Outboard bearing	Static analysis	
• Jacket water cooler	Dynamic analysis	
• Lube oil heat exchanger	Dynamic analysis	
• Generator stator and brush mounting structure	Dynamic analysis	
• Ac outlet box	Dynamic test	
• Starting air separator	Static analysis	
• Lube oil strainer	Static analysis	
• Jacket water thermostatic valve	Static analysis & dynamic test	
• Lube oil thermostatic valve	Static analysis & dynamic test	
• Standby fuel oil booster pump	Static analysis	
• Lube oil circulating pump	Static analysis	

TABLE 3.10A-1 (Cont'd.)

Equipment	Methods	Results
Component/System • Starting air relief valve on compressor • Air compressor • Engine-driven water pump • Intercooler • Engine-driven fuel oil booster pump • Jacket water thermo regulating valve • Various control system components: control valves, check valve, pressure switch, relay, temperature switch, solenoid valve, diaphragm valve, shuttle valve, etc. • Jacket water level switches • Lube oil relief valve • Fuel oil relief valve • Starting air relief valve • Fuel oil cooler • Entronic control panel • High-voltage panels	Method Static analysis Static analysis Static analysis Static analysis Static analysis Static analysis Dynamic test Dynamic test Static analysis Static analysis Static analysis Static analysis Dynamic test Dynamic test The applicable standards and guidelines are IEEE-323-1974, IEEE-344-1975, and NRC RG 1.61, 1.89, 1.92, and 1.100.	
Electrohydraulic Valve Operators - Borg-Warner and Paul Munroe	The Borg-Warner and Paul Munroe electrohydraulic valve operators and electronic controllers are qualified by dynamic testing. The applicable standards and guidelines are IEEE-323-1974, IEEE-344-1975, IEEE-382-1972, and NRC RG 1.89 and 1.100.	The Borg-Warner electrohydraulic operators and remote electronic controllers are affected by seismic loads only. The operators are line (valve) mounted, whereas the electronic controllers are wall mounted. Therefore, the operators and controllers are seismically qualified to different parameters. The Borg-Warner electrohydraulic operator seismic qualifications are based on dynamic testing. A representative test specimen was mounted to the vibration test table using a specially designed test fixture specimen was instrumented to record accelerations and monitor operability. The specimen was then subjected to three orthogonal,

TABLE 3.10A-1 (Cont'd.)

Equipment	Methods	Results
		single axis, vibration aging tests over the frequency range of 5 to 200 to 5 Hz at the rate of two octaves per minute with an acceleration level of 0.75 g. Each uniaxial test was for a 90-min duration, and the operator assembly was continuously functionally stroked at a rate of approximately five cycles per minute. A resonance search in each orthogonal axis was performed over the frequency range of 1 to 100 Hz using an input acceleration of 0.2 g. The specimen was then subjected to biaxial, random multifrequency vibration tests for five OBEs followed by one SSE in each of two orthogonal axes. The TRS enveloped the generic OBE (2.3 g zpa) and SSE (4.6 g zpa) spectra. The specimen was then subjected to required input motion (RIM) OBE tests that consisted of two sinusoidal sweeps from 2 to 35 to 2 Hz with an acceleration input of 3.3 g. The test was performed in each of three orthogonal axes. The first sweep was performed with the operator assembly in the open position and the second sweep with the operator in the closed position. Each OBE RIM test was followed by a SSE RIM test which consisted of a series of sine beats at each one-third octave frequency from 2 to 32 Hz with an input acceleration of 4.95 g. The input at each frequency was a continuous series of sine beats of 15 oscillations per beat for a duration of 15 sec. The operator assembly was cycled under load during each test. The operator assembly did not exhibit any malfunctions or loss of structural integrity. The remote electronic controller seismic qualifications are also based on dynamic testing. Representative test specimens were attached to the vibration test table such that the in-service mounting conditions were simulated. The specimens were instrumented to record acceleration levels and monitor operability. A resonance search from 1 to 100 Hz was performed in each of three orthogonal axes with an input acceleration of 0.2 g. The specimens were then subjected to biaxial, random multifrequency vibration tests for five OBEs followed by one SSE in each of two orthogonal axes. The TRS enveloped the RRS over the frequency range of interest by a minimum margin of 10 percent. The specimens did not exhibit any malfunction or loss of structural integrity. The Paul Munroe electrohydraulic valve operators are affected by seismic loads only. The operator seismic qualification is based on dynamic testing. The test specimen was mounted to the vibration test table using a specially designed test fixture such that the in-service mounting conditions were duplicated. The specimen was instrumented to record accelerations and monitor operability. A resonance search in each of the three orthogonal axes was performed over the frequency range of 1 to 40 Hz using an input acceleration of

TABLE 3.10A-1 (Cont'd.)

Equipment	Methods	Results
		0.2 g. The results of the resonance search test indicated no frequencies below 35 Hz. The specimen was then subjected to single-axis sine beat tests at 32 Hz in each of its orthogonal specimen axes to satisfy the OBE level requirements. Tests in each axis consisted of a string of beats (35 beats) of 15 oscillations (a sinusoid of the test frequency) per beat, with a 2-sec pause between beats for the total test time duration of 75 sec. The horizontal and vertical input accelerations were 1.6 g. These tests were performed with the valve actuator in the open position. After the OBE RIM test, the specimen was subjected to SSE level single-axis sine beat tests at 32 Hz along each of its orthogonal axes. Tests in each axis consisted of a string of beats (a minimum of five beats) and 15 oscillations (a sinusoid of the test frequency) per beat with a 2-sec pause between beats for a total duration of 15 sec. The horizontal and vertical input accelerations were 2.4 g. The specimen was operated under loaded conditions during each test and did not exhibit any malfunctions or loss of structural integrity. Piping as-built analyses ensure that the OBE and SSE acceleration levels are smaller than the qualified levels for each electrohydraulic valve actuator.
I/P Converters - CONOFLOW	The I/P converters are qualified by dynamic testing. Applicable standards and guidelines are IEEE-323-1974 and IEEE-344-1975.	The I/P converters are affected by seismic loads only. The testing consisted of a resonance search testing and sine beat testing. The resonance search testing consisted of a sinusoidal vibration at an acceleration of 0.5 g from 5 to 50 Hz. No resonant frequencies are identified below 33 Hz. The sine beat testing consisted of applying a series of continuous sine beats over a frequency range of 5 to 50 Hz. The accelerations used for excitation were 4.5 g horizontal and 3.0 g vertical. Due to attenuation of the output at higher g levels, the I/P converters are not considered to be operable during the seismic event. However, they operated satisfactorily after the seismic event.
Pressure (Filter) Regulators - CONOFLOW	The pressure (filter) regulators are qualified by dynamic testing. Applicable standards and guidelines are IEEE-323-1974 and IEEE-344-1975.	The pressure (filter) regulators are affected by seismic loads only. The testing consisted of a resonance search testing and sine beat testing. The resonance search testing consisted of a sinusoidal vibration at an acceleration of 0.5 g from 5 to 50 Hz. No structural resonant frequencies less than 33 Hz are found. The sine beat testing consisted of applying a series of continuous sine beats over a frequency range of 5 to 50 Hz. The accelerations used for excitation were 4.5 g horizontal and 3.0 g vertical. The units operated satisfactorily both during and after the tests.

TABLE 3.10B-1 (Cont'd.)

Unit 2 Ship Loose			
Temperature element	145C3224	A	California alloy
Temperature element	159C4520	P	California alloy
Temperature element	133D9679	P	California alloy
Temperature element	117C3485	P	California alloy
Temperature switch	157C4629	P	Weed
Temperature indicator	169C8974	P	Weed
Differential pressure transmitter	163C1560	P	Rosemount
Pressure transmitter	188C7360	A	Rosemount
Level transmitter	188C4775	A	Gould
Level transmitter	145C3156	P	Barton
Level switch	184C4776	A	Magnetrol
Level switch	159C4361	P	Magnetrol
Pressure indicator	163C1184	P	Robertshaw/Sycon
Insulated detectors	DD188C8915	A	GE
IRM detector	237X731	A	GE
Conductivity element	112C3144	P	Balsbaugh
Condensing chamber	163C1544	P	GE
	204B7269		

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LIST OF FIGURES (Cont'd.)

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3A.9-7	PERMANENT PIPE DEFORMATION FOR THE MIT EXPERIMENT
3A.10-1	LIMITA III ELASTIC SPACE FRAME
3A.10-2	LIMITA III DISPLACEMENT TIME HISTORY AT JOINT 6
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3A.10-4	STRUCTURE AND MODEL USED FOR LIMITA3 ANALYSES
3A.12-1	SSLAM SAMPLE PROBLEM (ANNULAR POOL)
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3A.15-1	MATHEMATICAL MODEL FOR FLEXIBILITY ANALYSIS VERIFICATION
3A.15-2	NUPIPE PROGRAM FORCE TIME HISTORY VERIFICATION
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LIST OF FIGURES (Cont'd.)

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3A.20-1	SUDDEN DISCHARGE OF GAS FROM A PIPELINE THROUGH A NOZZLE (CASE A)
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APPENDIX 3A

COMPUTER PROGRAMS FOR DYNAMIC AND STATIC ANALYSIS OF CATEGORY I STRUCTURES, EQUIPMENT, AND COMPONENTS

INTRODUCTION

This introduction details the principal use(s) of the most common computer programs for dynamic and static analysis of Category I structures, equipment and components within the balance of plant (BOP) scope of supply (originally supplied by Stone & Webster Engineering Corporation [SWEC]). See Section 3.9B for a listing and a description of the corresponding nuclear steam supply system (NSSS) computer programs (originally supplied by General Electric Company [GE]).

The following computer programs are used for the stress analysis of the containment system:

1. SHELL 1
2. ASAAS
3. TAC2D
4. CWL

The following computer programs are used in dynamic and static analysis of Category I structures:

1. MAT 6
2. STRUDL
3. TIME HISTORY (TIMHIS6) Program
4. SHELL 1
5. GHOSH-WILSON
6. TRANFUN AND INVATRAN

The following computer programs are used in the analysis of Category I equipment and components:

1. ME 121
2. DINASAW
3. LIMITA II
4. LIMITA III
5. STARDYNE
6. MISSILE
7. BIJLAARD
8. SLOSH
9. ANSYS
10. IMAGES

The following computer programs are used for the analysis of Category I piping systems, including pipe supports:

1. NUPIPE
2. HTLOAD

3. WATAIR
4. STEHAM
5. WATHAM
6. PITRUST
7. PILUG
8. ANSYS
9. LUGSTR
10. SNUFFE
11. PITRIFE
12. SUPERPIPE
13. PIPSYS

The following computer programs are used to calculate submerged structure loads due to various disturbances in the suppression pool:

1. SSLAM
2. SSLOAD

The following computer program is used for the analysis of the primary containment airlock:

1. NASTRAN

For each computer program, there is a brief description of the program's theoretical basis, the assumptions and references used in the program, and the extent of its application.

3A.15 NUPIPE

3A.15.1 General Description

NUPIPE was developed by the Nuclear Services Corporation and is fully documented. The SWEC version of NUPIPE[3] differs slightly from the public domain program NUPIPE in the postprocessing of the analytical results.

NUPIPE performs a linear elastic analysis of three-dimensional piping systems subjected to thermal, static, and dynamic loads. It utilizes the finite element method of analysis with special features incorporated to accommodate specific requirements in piping analysis. In addition, it checks analytical conformance to ASME Section III and ANSI B31.1. This program accepts the complete geometric and physical description of the piping system, provides a complete error and coordinate check for the inputs, and computes internal forces and moments, support and equipment reactions, and displacements and stress values for a variety of loading cases.

NUPIPE-SWPC is a personal computer (PC) version of NUPIPE. This PC version of NUPIPE includes the same features as the mainframe version referenced above, with some enhancements.

3A.15.2 Program Verification

NUPIPE has been verified with ADLPIPE[1] for thermal, weight, and response spectrum seismic analyses. The results from both programs are presented in Tables 3A.15-1 through 3A.15-7. The model used for this comparison is presented on Figure 3A.15-1.

The comparison is also made with ASME Benchmark solution for force time history dynamic response[2]. The model used for this comparison is shown on Figure 3A.15-2. The results for comparisons are presented in the form of plots on Figure 3A.15-2. The natural frequencies are given in Table 3A.15-8.

The Safety Class 1 piping stress computation conforms with the hand calculations. The model used is shown on Figure 3A.15-3. The results are tabulated in Tables 3A.15-9 and 3A.15-10.

NUPIPE-SWPC[4] is a proprietary PC version of NUPIPE and has been verified by demonstrating that similar results are obtained for several identical problems benchmarked on the mainframe version. These results are documented in the verification manual[5].

In addition, whenever a new version of NUPIPE (mainframe or PC) is installed or the hardware configuration is changed, appropriate verification problem(s) are rerun. The old and new results are compared to ensure that program integrity has been maintained.

3A.15.3 References

1. Arthur D. Little, Inc. ADLPIPE: Static, Dynamic, Thermal Pipe Stress Analysis.

2. American Society of Mechanical Engineers. Pressure Vessel and Piping; 1972 Computer Programs Verification. Problem No. 5.

3. NUPIPE-SW Program User's Manual, ME-110 Versions 03, 04, 05 and 06, Stone & Webster Engineering Corporation.

4. NUPIPE-SWPC Program User's Manual, ME-110.01U, Version 01, Stone & Webster Engineering Corporation.

5. NUPIPE-SWPC Program Verification Manual, ME-110.01, Version 01, Stone & Webster Engineering Corporation.

The peak pressures of Volumes 215-1 and 215-6 are based on the 8-in RWCU DER in Volume 215-1 on the suction side of the RWCU recirculation pumps. This blowdown is listed in Table 3B-11. Volume 215-2's peak pressure is based on the 10-in RWCU DER in the vertical pipe chase at Azimuth 29° (Volume 261-5). This break is described in Section 3B.3.4.

The peak pressures of Volumes 215-3 and 215-8 are based on the 10-in RCIC DER in Volume 215-1. Volumes 215-4, 215-5, and 215-7's peak pressures are calculated from an 8-in RWCU DER in Volume 215-5 on the suction side of the RWCU recirculation pumps. The peak pressures of Volume 215-9 and Volume 215-10 come from 4-in RWCU suction side DERs within the respective volumes. The blowdown listings for these breaks are given in Tables 3B-10 and 3B-12.

3B.3.3 Elevation 240 ft 0 in to Elevation 261 ft 0 in

Four cases were modeled on this elevation. The 8-in RWCU line DERs on the suction and discharge sides of the RWCU recirculation pumps were analyzed in the vertical pipe chase at Azimuth 180° (Volume 240-1). A 10-in RCIC steam line DER and an 8-in RWCU DER were postulated in the cubicle containing penetrations Z-11 and Z-13 (Volume 240-2).

RCIC valve isolation occurs at 32.85 sec based on a 2.85-sec delay between the time of the break and valve actuation and a 30-sec valve closure time. RWCU valve isolation occurs at 16.5 sec based on a 2.5-sec delay between the time of the break and valve actuation and a 14-sec valve closure time. Flow was maximized by assuming a constant reactor pressure until the valve was fully closed.

Vent curtains are attached to the wire-mesh door that leads from the TIP cubicle at Azimuth 45° (Volume 240-3) to the general volume (Volume 240-5). All four cases assumed that one of the blowout panels between Volume 175-2 and Volume 175-3 fails to open.

The peak calculated absolute pressures with their respective design values and margins are listed in Table 3B-3. Graphs showing the calculated pressures in Volumes 240-1 through 240-8 with respect to time are given on Figures 3B-24 through 3B-31.

The peak pressures of Volume 240-1 and Volume 240-2 are based on an 8-in RWCU DER in Volume 240-1 on the suction side of the RWCU recirculation pumps. The blowdown listing for this break is shown in Table 3B-13.

The peak pressures of Volume 240-3 and 240-4 are determined from a 10-in RWCU DER in the vertical pipe chase at Azimuth 29° (Volume 261-5). This break is reviewed in the section that studies breaks between el 261 ft 0 in and el 289 ft 0 in.

The peak pressures of Volume 240-5, Volume 240-7, and Volume 240-8 come from a 6-in RWCU DER in a RWCU filter/demineralizer cubicle (Volume SC328190). The peak pressure of Volume 240-6 is determined by the 10-in RWCU DER in the horizontal pipe tunnel at Azimuth 45° (Volume 306-4). These breaks are respectively reviewed in the section that studies breaks above el 328 ft 10 in and the section that studies breaks between el 306 ft 6 in and el 328 ft 10 in.

3B.3.4 Elevation 261 ft 0 in to Elevation 289 ft 0 in

Four cases were modeled on this elevation. A 10-in RCIC steam line DER and an 8-in RWCU line DER were postulated in the vertical pipe chase at Azimuth 180° (Volume 261-1). An 8-in RWCU line DER is also postulated in the vertical pipe chase directly above Volume 261-1 (Volume 261-2). A 10-in RWCU line DER is analyzed in the vertical pipe chase at Azimuth 29° (Volume 261-5).

RCIC valve isolation occurs at 32.85 sec based on a 2.85-sec delay between the time of the break and valve actuation and a 30-sec valve closure time. RWCU valve isolation occurs at 16.5 sec based on a 2.5-sec delay between the time of the break and valve actuation and a 14-sec valve closure time. Flow was maximized by assuming a constant reactor pressure until the valve was fully closed.

Vent curtains are across the door opening that connects Volume 261-1 to the general volume (Volume 261-4). All four cases assumed that one of the blowout panels between Volume 175-2 and Volume 175-3 fails to open.

The peak calculated absolute pressures with their respective design value and margins are listed in Table 3B-3. Graphs showing the calculated pressures in Volumes 261-1 through 261-5 with respect to time are given on Figures 3B-32 through 3B-36.

The peak pressure of Volume 261-1 is based on an 8-in RWCU DER in Volume 261-1. Volume 261-2's peak pressure is determined by the 8-in RWCU DER in Volume 261-2. Volume 261-3's and Volume 261-5's peak pressures are determined from a 10-in RWCU DER in the vertical pipe chase at Azimuth 29° (Volume 261-5). The blowdown listing for these breaks is given in Tables 3B-14 and 3B-15.

Volume 261-4's peak pressure is calculated from a 6-in RWCU steam line DER in a filter/demineralizer cubicle (Volume SC328190). This break is reviewed in the section that studies breaks above el 328 ft 10 in.

3B.3.5 Elevation 289 ft 0 in to Elevation 306 ft 6 in

Two cases were modeled on this elevation. An 8-in RWCU line DER was postulated in the vertical pipe chase at Azimuth 180° (Volume

system), a single failure is not postulated in the redundant safety-related train of that system or subsystem.

3C.4.4 Analytical Methods

As described in the spraying evaluation procedure (Section 3C.4.2), all targets in a given zone were assumed to be sprayed by any water sources in the zone. Analytical calculations of spraying distance were not utilized in reevaluating problem areas. In these instances, shielding, moving equipment, and other modifications were considered.

3C.4.5 Results of Evaluation - Spraying FMEA

The following subsections present, building-by-building, the results of the spraying evaluation using the procedures and guidelines discussed in Sections 3C.4.2 and 3C.4.3.

The evaluation verifies that the plant can be safely shut down in the event of pipe cracks in fluid systems. As noted below, protective measures have been implemented to ensure the required system functional capability is maintained. A list of moderate-energy piping systems and system parameters is provided in Tables 3C.4-1 through 3C.4-6 for those buildings housing equipment required for safe shutdown.

3C.4.5.1 Reactor Building (Including Auxiliary Bays)

In the reactor building, spray sources include both safety-related and nonsafety-related systems. Components susceptible to failure from spray are motors and MCCs for RCIC, HPCS, spent fuel pool cooling and cleanup (SFC), RHR, and low-pressure core spray (LPCS) system pumps. A single spray source will not affect more than one of these pump motors. Failure of a RCIC, HPCS, SFC, or LPCS motor is acceptable; sufficient redundancy exists to maintain spent fuel pool cooling and to safely shut down the plant when considering an additional single active failure as described in Section 3C.4.3.10. The RHR pump motors (2RHS*P1A and P1B) are protected from spray as required to ensure safe shutdown of the plant.

MCCs for these pumps are protected from spray.

Junction boxes required to maintain safe shutdown capability are sprayproofed if subject to spraying conditions.

The spray sources which would fail these components will not fail the redundant trains by flooding (Section 3C.5).

3C.4.5.2 Control Building

The spray sources in the control building include chilled, service, domestic, and fire protection water systems (Table

3C.4-2). The spray-susceptible targets are the control panels, switchgear, ventilation systems, and pump motors. The control and relay rooms contain equipment serving all four trains for safe shutdown. These components are shielded from potential spraying, as required, to ensure availability of the system safe shutdown function when considering an additional single active failure as described in Section 3C.4.3.10. Junction boxes required to maintain safe shutdown capability are sprayproofed if subject to spraying conditions.

3C.4.5.3 Diesel Generator Building

The spray sources in the diesel generator building are fire protection, service water, and diesel generator fuel oil systems (Table 3C.4-3).

The diesel fuel pumps and air compressors are spray-susceptible targets. Failure of these motors by spraying is acceptable since this would only result in a loss of one of the two redundant trains. However, failure would not require immediate plant shutdown. Potential flooding from the spray source would not result in loss of the redundant trains of emergency power (Section 3C.5.5.4). Additionally, since the crack would not result in a reactor or turbine trip, offsite power would still be available.

3C.4.5.4 Piping Tunnels

The moderate-energy systems in the piping tunnels are water treatment, service water, component cooling water, floor drains, and turbine building equipment drains. There are no spray-susceptible targets in the pipe tunnel (Table 3C.4-4) that are required for safe shutdown.

3C.4.5.5 Electrical Tunnels

The only water source in the electrical tunnels is fire protection, and the electrical tunnel equipment rooms have fire protection and service water. There are no spray-susceptible targets in any of these areas (Table 3C.4-5).

3C.4.5.6 Screenwell Building

There are ten moderate-energy systems in the screenwell (Table 3C.4-6). The spray-susceptible targets are the service water system MCCs, junction boxes, and pumps located in the service water pump bays. These components are protected as required to ensure availability of the system safe shutdown function when considering an additional single active failure as described in Section 3C.4.3.10. Flooding from the postulated cracks will not affect the redundant trains (Section 3C.5.5.7). Junction boxes required to maintain safe shutdown capability are sprayproofed if subject to spraying conditions.

TABLE 3C.4-6

LEAKAGE RATES IN SCREENWELL BUILDING

Moderate-Energy Systems

System	System Design Conditions		Nominal Line Size In/Sch	Line Wall Thickness In	Circular Area (In^2)	Maximum Flooding Leakage Rate (gpm)
	Pressure (psig)	Temperature (°F)				
Component Cooling (CCS)	150	150	2/80	0.218	0.109	45
Fire Protection (FPW)	175	120	12/std	0.375	1.125	465
Yard Structures Ventilation (HVY)	50	150	2/80	0.218	0.109	25
Lube Oil (LOS)	70	170	3/40	0.216	0.162	45
Liquid Waste (LWS)	256	150	3/40	0.216	0.162	80
Makeup Water (MWS)	160	104	2/40	0.154	0.077	30
Domestic Water (DWS)	135	104	4/tubing	0.130	0.130	50
Service Water (SWP)	150	130	36/spw	0.500	4.500	1720
Traveling Screens Service Water (SWT)	130	100	8/40	0.322	0.644	230
Water Treating (WTS)	120	104	4/40	0.237	0.237	80

4.3 NUCLEAR DESIGN

The information in Section 4.3 is provided in the Licensing Topical Report, General Electric Standard Application for Reactor Fuel, GESTAR II[1]. The subsection numbers in Section 4.3 directly correspond to the subsection numbers of Appendix A of GESTAR II[1]. Any additions or differences are given below for each applicable subsection.

4.3.1 Design Bases

See Section A.4.3.1 of GESTAR II[1].

4.3.2 Description

See Section A.4.3.2 of GESTAR II[1].

4.3.2.1 Nuclear Design Description

The information is given in Subsection A.4.3.2.1 of GESTAR II, except for the reference (initial) core loading pattern which is described and shown on Figure 3-1a of NEDE-20944-P-1[2]. The initial core uses three enrichments of fuel bundles, which are shown on Figures 2-2.45, 2-2.49a and b, and 2-2.106a and b of GESTAR II[1]. Cycle-specific nuclear design description is covered in Appendix A, Section A.4.3.2.1.

4.3.2.2 Power Distribution

4.3.2.2.1 Power Distribution Calculations

A full range of typical calculated power distributions, along with the resultant exposure shapes and the corresponding control rod patterns, are shown in Appendix A of NEDE-20944-P-1[2].

4.3.2.2.2 Power Distribution Measurements

See Section A.4.3.2.2.2 of GESTAR II[1].

4.3.2.2.3 Power Distribution Accuracy

See Section A.4.3.2.2.3 of GESTAR II[1].

4.3.2.2.4 Power Distribution Anomalies

See Section A.4.3.2.2.4 of GESTAR II[1].

4.3.2.3 Reactivity Coefficient

See Section A.4.3.2.3 of GESTAR II[1].

4.3.2.4 Control Requirements

See Section A.4.3.2.4 of GESTAR II[1].

4.3.2.4.1 Shutdown Reactivity

The information is given in Subsection A.4.3.2.4.1 of GESTAR II[1] except for the cold shutdown margin for the reference initial core loading pattern which is given in Table 4.3-1.

4.3.2.4.2 Reactivity Variations

The information is given in Subsection A.4.3.2.4.2 of GESTAR II[1]. The combined effects of the individual constituents of reactivity are accounted for in each K_{eff} in Table 4.3-1.

4.3.2.5 Control Rod Patterns and Reactivity Worths

This information is discussed in Section 3.2.5 of NEDE-20944-P-1[2]. Control rod patterns and the associated power distributions for a typical BWR 5 are presented in Appendix A of NEDE-20944-P-1[2]. These control rod patterns are calculated with the BWR core simulator. Qualification for this model is discussed and referenced in Section 3.1 of GESTAR II[1].

4.3.2.5.1 Scram Reactivity

Scram reactivity is calculated as described in Section S.2 of GESTAR II[1] and discussed in Section 3.2.5.3 of NEDE-20944-P-1[2].

4.3.2.6 Criticality of Reactor During Refueling

See Section A.4.3.2.6 of GESTAR II[1].

4.3.2.7 Stability

See Section A.4.3.2.7 of GESTAR II[1].

4.3.2.7.1 Xenon Transients

See Section A.4.3.2.7.1 of GESTAR II[1].

4.3.2.7.2 Thermal-Hydraulic Stability

See Section 4.4.4.

4.3.2.8 Vessel Irradiations

The neutron fluxes at the vessel were originally calculated using the one-dimensional discrete ordinates transport code described in Section 4.1.4.5. The discrete ordinates code was used in a distributed source mode with cylindrical geometry. The geometry described six regions from the center of the core to a point beyond the vessel. The core region was modeled as a single homogenized cylindrical region. The coolant water region between the fuel channel and the shroud was described as containing saturated water at 550°F and 1,050 psi. The material

Nine Mile Point Unit 2 Compliance

Diverse and redundant level-sensing instrumentation on the SDIV is provided for the automatic scram function. SDIV water level is measured by utilization of both float-sensing and pressure-sensing devices. Instrument taps have been relocated from the vent and drain piping to the SDIV to protect the level-sensing instrumentation from the flow dynamics in the scram discharge system. Each SDIV has a redundant instrument loop. A one-out-of-two twice logic is employed for the automatic scram function. This instrumentation arrangement ensures the automatic scram function on high SDIV water level in the event of a single active or passive failure. These SDV modifications will be implemented at Unit 2.

d. Safety Criterion 4

System operating conditions which are required for scram shall be continuously monitored.

Nine Mile Point Unit 2 Compliance

See response to Safety Criterion 3.

e. Safety Criterion 5

Repair, replacement, adjustment, or surveillance of any system component shall not require the scram function to be bypassed.

Nine Mile Point Unit 2 Compliance

The SDIV scram level instrumentation arrangement and trip logic allows instrument adjustment or surveillance without bypassing the scram function or directly causing a scram. Each level instrument can be individually isolated without bypassing the scram function. A one-out-of-two twice trip logic is employed. Unit 2 plant Technical Specifications will ensure that the scram function is not bypassed during repair, replacement, adjustment, or surveillance of any system component.

3. Operational Criteria

a. Operational Criterion 1

Level instrumentation shall be designed to be maintained, tested, or calibrated during plant operation without causing a scram.

Nine Mile Point Unit 2 Compliance

See response to Safety Criterion 5.

b. Operational Criterion 2

The system shall include sufficient supervisory instrumentation and alarms to permit surveillance of system operation.

Nine Mile Point Unit 2 Compliance

Supervisory instrumentation and alarms such as accumulator trouble, scram valve air supply low pressure, and scram discharge instrument volume drain alarms, are adequate and permit surveillance of the scram system's readiness.

c. Operational Criterion 3

The system shall be designed to minimize the exposure of operating personnel to radiation.

Nine Mile Point Unit 2 Compliance

Minimizing the exposure of operating personnel to radiation is a consideration in equipment design and location.

d. Operational Criterion 4

Vent paths shall be provided to ensure adequate drainage in preparation for scram reset.

Nine Mile Point Unit 2 Compliance

A vent line is provided as part of the scram discharge system to ensure proper drainage in preparation for scram reset. The Unit 2 position is to provide a dedicated vent line with a nonsubmerged discharge during normal operating conditions. Furthermore, the vent line vacuum breaker provides additional vent capability under normal operating conditions and ensures adequate vent capability should the SDV vent line become submerged (i.e., following a LOOP). The vacuum

breaker is required to open at a differential pressure no greater than 5 in of water.

e. Operational Criterion 5

Vent and drain functions shall not be adversely affected by other system interfaces. The objective of this requirement is to preclude water backup in the scram instrument volume which could cause spurious scram.

Nine Mile Point Unit 2 Compliance

The SDV vent line and drain lines are dedicated lines. The SDV vent discharges into a reactor building equipment drain tank (2DER-TK2B). A vacuum breaker on the SDV vent line precludes water from siphoning back into the SDIV.

4. Design Criteria

a. Design Criterion 1

The scram discharge headers shall be sized in accordance with GE criteria and shall be hydraulically coupled to the instrumented volume(s) in a manner to permit operability of the scram level instrumentation prior to loss of system function. Each system shall be analyzed based on plant-specific maximum in-leakage to ensure that the system function is not lost prior to initiation of automatic scram. Maximum in-leakage is the maximum flow rate through the scram discharge line without control rod motion, summed over all control rods. The analysis should show no need for vents or drains.

Nine Mile Point Unit 2 Compliance

As discussed in response to Functional Criterion 1, a minimum SDV of 3.34 gal per drive is provided. Furthermore, there is no reduction in the pipe size of the header piping going from the HCUs to and including the SDIV. The SDIV is directly connected to the SDV at the low point of the scram discharge header piping. These requirements satisfy the NRC's acceptance criteria for Design Criterion 1.

b. Design Criterion 2

Level instrumentation shall be provided for automatic initiation while sufficient volume exists in the SDV.

Nine Mile Point Unit 2 Compliance

See response to Functional Criterion 1 and Design Criterion 1.

c. Design Criterion 3

Instrumentation taps shall be provided on the vertical instrument volume and not on the connected piping.

Nine Mile Point Unit 2 Compliance

See response to Safety Criterion 3.

d. Design Criterion 4

The scram instrumentation shall be capable of detecting water accumulation in the instrumented volume(s) assuming a single active failure in the instrumentation system or the plugging of an instrument line.

Nine Mile Point Unit 2 Compliance

See response to Safety Criterion 3.

e. Design Criterion 5

Structural and component design shall consider loads and conditions including those due to fluid dynamics, thermal expansion, internal pressure, seismic considerations, and adverse environments.

Nine Mile Point Unit 2 Compliance

The SDV and associated vent and drain piping is classified as important to safety and required to meet the ASME Section III, Class 2 and seismic Category I requirements.

f. Design Criterion 6

The power-operated vent and drain valves shall close under loss of air and/or electric power. Valve position indication shall be provided in the control room.

Nine Mile Point Unit 2 Compliance

The present vent and drain valve design operation meets this criterion.

3. Seismic Design Verification

The insert/withdrawal lines and SDV piping are designed in accordance with the requirements for ASME III Class 2 piping. Seismic Category I loads and hydrodynamic forces (i.e., water hammer) have been included in the design.

4. HCU-SDV Equipment Procedures Review

Plant operation, maintenance, and surveillance procedures will provide sufficient guidance to ensure the system integrity.

5. As-Built Inspection of SDV Piping and Supports

As-built inspection of SDV piping and its supports will be conducted prior to startup.

As stated previously, a break in the SDV is not postulated to occur based on the upgraded Unit 2 design. Additional means are also available to address other safety concerns of NUREG-0803 associated with an unlikely SDV system failure. First, the layout of essential emergency core cooling system (ECCS) equipment (i.e., pumps, motors) located in flood-tight areas in the reactor building would prevent the equipment from being sprayed with cascading water. Second, Unit 2 has implemented emergency operating procedures (EOPs), based upon the BWR Owners' Group (BWROG) emergency procedure guideline and severe accident guideline (EPG/SAG), that address potential leaks in secondary containment. In addition, reactor water iodine concentrations will be limited as described in the Technical Specification to ensure acceptable radiation levels in the reactor building.

Hydraulic Control Units Each HCU furnishes pressurized water, on signal, to a drive unit. The drive then positions its control rod as required. Operation of the electrical system that supplies scram and normal control rod positioning signals to the HCU is described in Section 7.7.1.1.

The basic components in each HCU are manual, pneumatic, and electrical valves; an accumulator; related piping; electrical connections; filters; and instrumentation (Figures 4.6-5 and 4.6-8). The components and their functions are described as follows:

Insert Drive Valve The insert drive valve is solenoid operated and opens on an insert signal. The valve supplies drive water to the bottom side of the main drive piston.

Insert Exhaust Valve The insert exhaust solenoid valve also opens on an insert signal. The valve discharges water from above the drive piston to the exhaust water header.

Withdraw Drive Valve The withdraw drive valve is solenoid operated and opens on a withdraw signal. The valve supplies drive water to the top of the drive piston.

Withdraw Exhaust Valve The solenoid-operated withdraw exhaust valve opens on a withdraw signal and discharges water from below the main drive piston to the exhaust header. It also serves as the settle valve which opens, following any normal drive movement (insert or withdraw), to allow the control rod and its drive to settle back into the nearest latch position.

Speed Control Units The insert drive valve and withdraw exhaust valve have a speed control unit. The speed control unit regulates the control rod insertion and withdrawal rates during normal operation. The manually-adjustable flow control unit is used to regulate the water flow to and from the volume beneath the main drive piston. A correctly-adjusted unit does not require readjustment except to compensate for changes in drive seal leakage.

Scram Pilot Valve Assembly The scram pilot valve assembly is operated from the RPS. The scram pilot valve assembly, with two solenoids, controls both the scram inlet valve and the scram exhaust valve. The scram pilot valve assembly is solenoid operated and is normally energized. On loss of electrical signal to the solenoids, the inlet port closes and the exhaust port opens. The pilot valve assembly (Figures 4.6-5 and 4.6-8) is designed so that both solenoids must be deenergized before air pressure can be discharged from the scram valve operators. This prevents inadvertent scram of a single drive in the event of a failure of one of the pilot valve solenoids.

Scram Inlet Valve The scram inlet valve opens to supply pressurized water to the bottom of the drive piston. This quick-opening globe valve is operated by an internal spring and system pressure. It is closed by air pressure applied to the top of its diaphragm operator. A position switch on this valve operates a light in the main control room as soon as the valve starts to open.

Scram Exhaust Valve The scram exhaust valve opens slightly before the scram inlet valve, exhausting water from above the drive piston. The exhaust valve opens faster than the inlet valve because of the higher air pressure spring setting in the valve operator. Scram exhaust valves are equipped with position indicators.

Scram Accumulator The scram accumulator stores sufficient energy to fully insert a control rod at lower reactor vessel pressures. At higher vessel pressures, the accumulator pressure is assisted or supplanted by reactor vessel pressure. The accumulator is a hydraulic cylinder with a free-floating piston. The piston separates the water on top from the nitrogen below. A

check valve in the accumulator charging line prevents loss of water pressure in the event supply pressure is lost.

During normal plant operation, the accumulator piston is seated at the bottom of its cylinder. Loss of nitrogen decreases the nitrogen pressure, which actuates a pressure switch and sounds an alarm in the control room.

To ensure that the accumulator is always able to produce a scram, it is continuously monitored for water leakage. A float-type level switch actuates an alarm if water leaks past the piston barrier and collects in the accumulator instrumentation block.

Control Rod Drive System Operation

The CRD system performs rod insertion, rod withdrawal, and scram. These operational functions are described in the following sections.

Rod Insertion Rod insertion is initiated by a signal from the Operator, which causes both insert solenoid valves to open. The insert drive valve applies reactor pressure plus approximately 90 psi to the bottom of the drive piston. The insert exhaust valve allows water from above the drive piston to discharge to the exhaust header.

As illustrated on Figure 4.6-3, the locking mechanism is a ratchet-type device and does not interfere with rod insertion. The speed at which the drive moves is determined by the flow through the insert speed control valve, which is set for approximately 4 gpm for a shim speed (nonscram operation) of 3 in/sec. During normal insertion, the pressure on the downstream side of the speed control valve is 90 to 100 psi above reactor vessel pressure. However, if the drive slows for any reason, the flow through and pressure drop across the insert speed control valve decreases. The full differential pressure (260 psi) is then available to cause continued insertion. With 260-psi differential pressure acting on the drive piston, the piston exerts an upward force of 1,040 lb.

Rod Withdrawal Rod withdrawal is, by design, more involved than insertion. The collet finger (latch) must be raised to reach the unlocked position (Figure 4.6-3). The notches in the index tube and the collet fingers are shaped so that the downward force on the index tube holds the collet fingers in place. The index tube must be lifted before the collet fingers can be released. This is done by opening the drive insert valves (in the manner described in the preceding section) for approximately 1 sec. The withdraw valves are then opened, applying driving pressure above the drive piston and opening the area below the piston to the exhaust header. Pressure is simultaneously applied to the collet piston. As the piston rises, the collet fingers are cammed outward, away from the index tube, by the guide cap.

The pressure required to release the latch is set and maintained at a level high enough to overcome the force of the latch return spring, plus the force of reactor pressure opposing movement of the collet piston. When this occurs, the index tube is unlatched and free to move in the withdrawal direction. Water displaced by the drive piston flows out through the withdrawal speed control valve, which is set to give the control rod a shim speed of 3 in/sec. The entire valving sequence is automatically controlled and is initiated by a single operation of the rod withdrawal switch.

For Cycle 7, a cycle-specific analysis has been completed for rod withdrawal rates up to 6.0 in per second. For all other cycles, a cycle-generic analysis has been completed for rod withdrawal rates up to 5.0 in per second.

Scram During a scram, the scram pilot valve assembly and scram valves are operated as previously described. With the scram valves open, accumulator pressure is admitted under the drive piston, and the area over the drive piston is vented to the SDV.

The large differential pressure (approximately 1,500 psi, initially, and always several hundred psi, depending on reactor vessel pressure) produces a large upward force on the index tube and control rod. This force gives the rod a high initial acceleration and provides a large margin of force to overcome friction. After the initial acceleration is achieved, the drive continues at a nearly constant velocity. This characteristic provides a high initial rod insertion rate. As the drive piston nears the top of its stroke, the piston seals close off the large passage (buffer orifices) in the stop piston tube, providing a hydraulic cushion at the end of travel.

Prior to a scram signal, the accumulator in the HCU has approximately 1,450 to 1,510 psig on the water side and 1,050 to 1,100 psig on the nitrogen side. As the inlet scram valve opens, the full water-side pressure is available at the CRD acting on a 4.1-sq in area. As CRD motion begins, this pressure drops to the gas-side pressure less line losses between the accumulator and the CRD. At low vessel pressures the accumulator completely discharges with a resulting gas-side pressure of approximately 575 psi. The CRD accumulators are required to scram the control rods when the reactor pressure is low, and the accumulators retain sufficient stored energy to ensure the complete insertion of the control rods in the required time.

The ball check valve in the drive flange allows reactor pressure to supply the scram force whenever reactor pressure exceeds the supply pressure at the drive. This occurs, due to accumulator pressure decay and inlet line losses, during all scrams at higher vessel pressures. When the reactor is close to, or at, full operating pressure, reactor pressure alone will insert the

control rod in the required time, although the accumulator does provide additional margin at the beginning of the stroke.

The average scram insertion time of all operable control rods from the fully withdrawn position, based on de-energization of

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the scram pilot valve solenoids at time zero, does not exceed any of the following:

Dropout of Reed Switch	Avg. of All Operable CRDs (Sec)	Avg. of the Fastest 3 of Each 2x2 Array of 4 CRDs (Sec)
45	0.43	0.45
39	0.86	0.92
25	1.93	2.05
05	3.49	3.70

Alternate Rod Insertion The alternate rod insertion (ARI) feature is designed to increase the reliability of the CRD system (Section 4.6.1). ARI provides for insertion of reactor control rods by depressurizing the scram discharge air header through valves which are redundant and diverse from the RPS scram function.

A signal to insert control rods results in energizing the eight ARI solenoid valves. Two solenoid valves in series with the backup scram valves also have parallel functioning check valves to assure venting of air from the air supply line in the event an ARI valve fails. Four solenoid valves provide for venting of the A and B HCU scram headers to the atmosphere to depressurize the headers and scram all rods. Two solenoid valves vent the air header to the SDV drain and vent lines, closing the vent and drain valves and isolating the SDV.

Instrumentation

The instrumentation for both the control rods and CRDs is defined by that given for the manual control system. The objective of the reactor manual control system (RMCS) is to provide the Operator with the means to make changes in nuclear reactivity so that reactor power level and power distribution can be controlled. The system allows the Operator to manipulate control rods. The instrumentation and controls for the RMCS are described in Section 7.7.1.1.

4.6.1.2 Control Rod Drive Housing Supports

4.6.1.2.1 Safety Objective

The CRD housing supports prevent any significant nuclear transient in the event a drive housing breaks or separates from the bottom of the reactor vessel.

4.6.1.2.2 Safety Design Bases

The CRD housing supports meet the following safety design bases:

1. Following a postulated CRD housing failure, control rod downward motion would be limited so that any resulting nuclear transient could not be sufficient to cause fuel damage.

2. The clearance between the CRD housings and the supports is sufficient to prevent vertical contact stresses caused by thermal expansion during plant operation.

4.6.1.2.3 Description

The CRD housing supports are shown on Figure 4.6-9. Horizontal beams are installed immediately below the bottom head of the reactor vessel, between the rows of CRD housings. The beams are welded to brackets which are welded to the steel form liner of the drive room in the reactor support pedestal.

Hanger rods, approximately 10 ft long and 1 3/4 in in diameter, are supported from the beams on stacks of disc springs. These springs compress approximately 2 in under the design load.

The support bars are bolted between the bottom ends of the hanger rods. The spring pivots at the top, and the beveled, loose-fitting ends on the support bars prevent substantial bending moment in the hanger rods if the support bars are overloaded.

Individual grids rest on the support bars between adjacent beams. Because a single-piece grid would be difficult to handle in the limited work space, and because it is necessary that CRDs, position indicators, and in-core instrumentation components be accessible for inspection and maintenance, each grid is designed for in-place assembly or disassembly. Each grid assembly is made from two grid plates, a clamp, and a bolt. The top part of the clamp guides the grid to its correct position directly below the CRD housing that it would support in the postulated accident.

When the support bars and grids are installed, a gap of 1 in (+0.50/-0.25 in, at a temperature of 110°F or less) is provided between the grid and the bottom contact surface of the CRD flange. During system heatup, this gap is reduced by a net downward expansion of the housings with respect to the supports. In the hot operating condition, the gap is approximately 1/2 in to 1 1/4 in.

In the postulated CRD housing failure, the CRD housing supports are loaded when the lower contact surface of the CRD flange contacts the grid. The resulting load is then carried by two grid plates, two support bars, four hanger rods, their disc springs, and two adjacent beams.

The American Institute of Steel Construction (AISC) Manual of Steel Construction, Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, was used as a guide

in designing the CRD housing support system. The CRD housing support system is a mechanical system; therefore, RG 1.94 does not apply. However, to provide a structure that absorbs as much energy as practical without yielding, the allowable tension and bending stresses used were 90 percent of yield and the shear stress used was 60 percent of yield. These design stresses are 1.5 times the AISC allowable stresses (60 and 40 percent of yield, respectively).

For purposes of mechanical design, the postulated failure resulting in the highest forces is an instantaneous circumferential separation of the CRD housing from the reactor vessel, with the reactor at an operating pressure of 1,086 psig (at the bottom of the vessel) acting on the area of the separated housing. The weight of the separated housing, CRD, and blade, plus the pressure of 1,086 psig acting on the area of the separated housing, gives a force of approximately 32,000 lb. This force is used to calculate the impact force, conservatively assuming that the housing travels through a 1 1/2-in gap before it contacts the supports. The impact force (124,000 lb for the maximum gap of 1 1/2 in) is then treated as a static load in design. The CRD housing supports are designed as Category I equipment in accordance with Section 3.2. Loading conditions and examples of stress analysis results and limits are shown in Table 3.9B-2. Safety evaluation is discussed in Section 4.6.2.3.3.

4.6.2 Evaluations of the CRDs

4.6.2.1 Failure Modes and Effects Analysis

The evaluation of CRD system failure is discussed under Nuclear Safety Operational Analysis (NSOA) in Appendix 15A (Section 15A.6.6.3) and on Figures 15A-51 through 15A-53.

4.6.2.2 Protection from Common Mode Failures

See Section 4.6.2.1.

4.6.2.3 Safety Evaluation

Safety evaluation of the control rods, CRDs, and CRD housing supports is described in the following sections. Further description of control rods is contained in Section 4.2.

4.6.2.3.1 Control Rods

<u>Materials Adequacy Throughout Design Lifetime</u>

The adequacy of the materials throughout the design life was evaluated in the mechanical design of the control rods. The primary materials, B_4C powder, hafnium and Type 304 austenitic stainless steel, have been found suitable in meeting the demands of the BWR environment.

Dimensional and Tolerance Analysis

Layout studies are done to assure that, given the worst combination of part tolerance ranges at assembly, no interference exists which will restrict the movement of the control rods. In addition, preoperational verification is made on each control blade system to show that the acceptable levels of operational performance are met.

Thermal Analysis of the Tendency to Warp

The various parts of the control rod assembly remain at approximately the same temperature during reactor operation, negating the problem of distortion or warpage. What little differential thermal growth could exist is allowed for in the mechanical design. A minimum axial gap is maintained between absorber rod tubes and the control rod frame assembly for this purpose. In addition, dissimilar metals are avoided.

Forces for Expulsion

An analysis has been performed which evaluates the maximum pressure forces that could tend to eject a control rod from the core. The results of this analysis are given in Section 4.6.2.3.2. In summary, if the collet were to remain open, which is unlikely, calculations indicate that the steady state control rod withdrawal velocity would be 2 ft/sec for a pressure-under line break, the limiting case for rod withdrawal.

Functional Failure of Critical Components

The consequences of a functional failure of critical components have been evaluated and the results are covered in Section 4.6.2.3.2.

Precluding Excessive Rates of Reactivity Addition

In order to preclude excessive rates of reactivity addition, an analysis has been performed both on the velocity limiter device and the effect of probable control rod failures (Section 4.6.2.3.2).

Effect of Fuel Rod Failure on Control Rod Channel Clearances

The CRD mechanical design ensures a sufficiently rapid insertion of control rods to preclude the occurrence of fuel rod failures which could hinder reactor shutdown by causing significant distortions in channel clearances.

Procedural controls are established during the core design process and during the process of determining the final core configuration, that mitigate the consequences of channel bowing.

differential pressure across the piston would result in an insignificant driving force of approximately 10 lb, tending to increase withdrawal velocity.

A pressure differential of 295 psi across the collet piston would hold the collet unlatched as long as the driving signal was maintained.

Flow resistance of the exhaust path from the drive would be normal because the ball check valve would be seated at the lower end of its travel by pressure under the drive piston.

Ball Check Valve Plug Blows Out As a means of access for machining the ball check valve cavity, a 1.25-in diameter hole has been drilled in the flange forging. This hole is sealed with a plug of 1.31-in diameter and 0.38-in thickness. A full-penetration weld, utilizing Type 308 stainless steel filler, holds the plug in place. The failure postulated is a circumferential crack in this weld leading to a blowout of the plug.

If the plug were to blow out while the drive was latched, there would be no control rod motion. No pressure differential would exist across the collet piston to unlatch the collet. As in the previous failure, reactor water would flow past the velocity limiter, down the annulus between the drive and thermal sleeve, through the vessel ports and drilled passage, through the ball check valve cage, and out the open plug hole to the drywell. The leakage calculations indicate the flow rate is less than 350 gpm. This calculation assumes liquid flow, but flashing of the hot reactor water to steam would reduce this rate to a lower value. Drive temperature would rapidly increase and initiate an alarm in the main control room.

If the plug failure were to occur during control rod withdrawal (it would not be possible to unlatch the drive after such a failure), the collet would relatch at the first locking groove. If the collet were to stick, calculations indicate the control rod withdrawal speed would be 11.8 ft/sec. There would be a large retarding force exerted by the velocity limiter due to a 35-psi pressure differential across the velocity limiter piston.

Drive/Cooling Water Pressure Control Valve Closure (Reactor Pressure, 0 psig) The pressure to move a drive is generated by the pressure drop of practically the full system flow through the drive/cooling water pressure control valve. This valve is either a motor-operated valve (MOV) or a standby manual valve; either one is adjusted to a fixed opening. The normal pressure drop across this valve develops a pressure 260 psi in excess of reactor pressure.

If the flow through the drive/cooling water pressure control valve were to be stopped, as by a valve closure or flow blockage, the drive pressure would increase to the shutoff pressure of the

supply pump. The occurrence of this condition during withdrawal of a drive at zero vessel pressure will result in a drive pressure increase from 260 psig to no more than 1,750 psig. Calculations indicate that the drive would accelerate from 3 in/sec to approximately 6.5 in/sec. A pressure differential of 1,670 psi across the collet piston would hold the collet unlatched. Flow would be upward, past the velocity limiter piston, but retarding force would be negligible. Rod movement would stop as soon as the driving signal was removed.

Ball Check Valve Fails to Close Passage to Vessel Ports Should the ball check valve sealing the passage to the vessel ports be dislodged and prevented from reseating following the insert portion of a drive withdrawal sequence, water below the drive piston would return to the reactor through the vessel ports and the annulus between the drive and the housing rather than through the speed control valve. Because the flow resistance of this return path would be lower than normal, the calculated withdrawal speed would be 2 ft/sec. During withdrawal, differential pressure across the collet piston would be approximately 40 psi. Therefore, the collet would tend to latch and would have to stick open before continuous withdrawal at 2 ft/sec could occur. Water would flow upward past the velocity limiter piston, generating a small retarding force of approximately 120 lb.

Hydraulic Control Unit Valve Failures Various failures of the valves in the HCU can be postulated, but none could produce differential pressures approaching those described in the preceding sections, and none alone could produce a high-velocity withdrawal. Leakage through either one or both of the scram valves produces a pressure that tends to insert the control rod rather than to withdraw it. If the pressure in the SDV should exceed reactor pressure following a scram, a check valve in the line to the scram discharge header prevents this pressure from operating the drive mechanisms.

Collet Fingers Fail to Latch This failure is presumed to occur when the drive withdraw signal is removed. If the collet fails to latch, the drive continues to withdraw at a fraction of the normal speed. This assumption is made because there is no known means for the collet fingers to become unlocked without some initiating signal. Because the collet fingers will not cam open under a load, accidental application of a down signal does not unlock them. (The drive must be given a short insert signal to unload the fingers and cam them open before the collet can be driven to the unlock position.) If the drive withdrawal valve fails to close following a rod withdrawal, the collet would remain open and the drive would continue to move at a reduced speed.

Withdrawal Speed Control Valve Failure Normal withdrawal speed is determined by differential pressures in the drive and is set for a nominal value of 3 in/sec. For Cycle 7, a cycle-specific analysis has been completed for rod withdrawal rates up to 6.0 in

per second. For all other cycles, a cycle-generic analysis has been completed for rod withdrawal rates up to 5.0 in per second. Withdrawal speed is maintained by the pressure regulating system and is independent of reactor

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vessel pressure. Tests have shown that accidental opening of the speed control valve to the full-open position produces a velocity of approximately 6 in/sec.

The CRD system prevents unplanned rod withdrawal and it has been shown previously that only multiple failures in a drive unit and in its control unit could cause an unplanned rod withdrawal.

Scram Reliability

High scram reliability is the result of a number of features of the CRD system. For example:

1. Two reliable sources of scram energy are used to insert each control rod: individual accumulators at low reactor pressure, and the reactor vessel pressure itself at power.

2. Each drive mechanism has its own scram valves and dual solenoid scram pilot valves; therefore, only one drive can be affected if a scram valve fails to open. Both pilot valve solenoids must be de-energized to initiate a scram.

3. The RPS and the HCUs are designed so that the scram signal and mode of operation override all others.

4. The collet assembly and index tube are designed so they will not restrain or prevent control rod insertion during scram.

5. The SDV is monitored for accumulated water and the reactor will scram before the volume is reduced to a point that could interfere with a scram.

Control Rod Support and Operation

As described in the preceding sections, each control rod is independently supported and controlled as required by safety design bases.

4.6.2.3.3 Control Rod Drive Housing Supports

Downward travel of the CRD housing and its control rod following the postulated housing failure equals the sum of these distances: 1) the compression of the disc springs under dynamic loading, and 2) the initial gap between the grid and the bottom contact surface of the CRD flange. If the reactor were cold and pressurized, the downward motion of the control rod would be limited to the spring compression (approximately 2 in) plus a gap of 1 in (+0.50/-0.25 in). If the reactor were hot and pressurized, the gap would be approximately 1/2 in to 1 1/4 in and the spring compression would be slightly less than in the cold condition. In either case, the control rod movement

following a housing failure is substantially limited below one drive "notch" movement (6 in). Sudden withdrawal of any control rod through a distance of one drive notch at any position in the core does not produce a transient sufficient to damage any radioactive material barrier.

The CRD housing supports are in place during power operation and when the nuclear system is pressurized. If a control rod is ejected during shutdown, the reactor remains subcritical because it is designed to remain subcritical with any one control rod fully withdrawn at any time.

At plant operating temperature, a gap of approximately 1/2 in to 1 1/4 in exists between the CRD housing and the supports. At lower temperatures the gap is greater. Because the supports do not contact any of the CRD housing except during the postulated accident condition, vertical contact stresses are prevented. Inspection and testing of CRD housing supports is discussed in Section 4.6.3.2.

4.6.3 Testing and Verification of the CRDs

4.6.3.1 Control Rod Drives

4.6.3.1.1 Testing and Inspection

Development Tests

The initial development drive (prototype of the standard locking piston design) testing included more than 5,000 scrams and approximately 100,000 latching cycles. One prototype was exposed to simulated operating conditions for 5,000 hr. These tests demonstrated the following:

1. The drive easily withstands the forces, pressures, and temperatures imposed.

2. Wear, abrasion, and corrosion of the nitrided stainless parts are negligible. Mechanical performance of the nitrided surface is superior to that of materials used in earlier operating reactors.

3. The basic scram speed of the drive has a satisfactory margin above minimum plant requirements at any reactor vessel pressure.

4. Usable seal lifetimes in excess of 1,000 scram cycles can be expected.

Factory Quality Control Tests

Quality control of welding, heat treatment, dimensional tolerances, material verification, and similar factors is maintained throughout the manufacturing process to assure

reliable performance of the mechanical reactivity control components. Some of the quality control tests performed on the control rods, CRD mechanisms, and HCUs are listed below:

1. CRD mechanism tests:

   a. Pressure welds on the drives are hydrostatically tested in accordance with ASME Codes.

   b. Electrical components are checked for electrical continuity and resistance to ground.

   c. Drive parts that cannot be visually inspected for dirt are flushed with filtered water at high velocity. No significant foreign material is permitted in effluent water.

   d. Seals are tested for leakage to demonstrate correct seal operation.

   e. Each drive is tested for shim motion, latching, and control rod position indication.

   f. Each drive is subjected to scram tests at various reactor pressures to verify correct scram performance.

2. HCU tests:

   a. Hydraulic systems are hydrostatically tested in accordance with the applicable code.

   b. Electrical components and systems are tested for electrical continuity and resistance to ground.

   c. Correct operation of the accumulator pressure and level switches is verified.

   d. The unit's ability to perform its part of a scram is demonstrated.

   e. Correct operation and adjustment of the insert and withdrawal valves is demonstrated.

Operational Tests

After installation, all rods and drive mechanisms can be tested through their full strokes for operability.

During normal operation, each time a control rod is withdrawn a notch, the Operator can observe the in-core monitor indications to verify that the control rod is following the drive mechanism. All control rods that are partially withdrawn from the core can be tested for rod-following by inserting or withdrawing the rod

one notch and returning it to its original position, while the Operator observes the in-core monitor indications.

To make a positive test of control rod to CRD coupling integrity, the Operator can withdraw a control rod to the end of its travel and then attempt to withdraw the drive to the overtravel position. Failure of the drive to overtravel demonstrates rod-to-drive coupling integrity.

Hydraulic supply subsystem pressures can be observed from instrumentation in the main control room. Scram accumulator pressures can be observed on the local nitrogen pressure gauges.

Acceptance Tests

Criteria for acceptance of the individual CRD mechanisms and the associated control and protection systems are incorporated in specifications and test procedures covering three distinct phases: 1) preinstallation, 2) after installation prior to startup, and 3) during startup testing.

The preinstallation specifications define criteria and acceptable ranges of such characteristics as seal leakage, friction, and scram performance under fixed test conditions which must be met before the component can be shipped.

The after-installation, prestartup tests (Chapter 14) include normal and scram motion and are primarily intended to verify that piping, valves, electrical components, and instrumentation are properly installed. The test specifications include criteria and acceptable ranges for drive speed, timer settings, scram valve response times, and control pressures. These tests are intended more to document system condition than system performance.

As fuel is placed in the reactor, the startup test procedure (Chapter 14) will be followed. The tests in this procedure are intended to demonstrate that the initial operational characteristics meet the limits of the specifications over a range of primary coolant temperatures and pressures. The detailed specifications and procedures follow the general pattern established for such specifications and procedures in BWRs presently under construction and in operation.

Surveillance Tests

The surveillance requirements for the CRD system are described as follows:

1. Sufficient control rods will be withdrawn one at a time, following a refueling outage when core alterations are performed, to demonstrate that the core can be made subcritical at any time in the subsequent fuel cycle with the strongest operable control rod