# CHAPTER 5: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

## 5.1 SUMMARY DESCRIPTION

The reactor coolant system includes those systems and components that contain or transport fluids to or from the reactor core. These systems form a major portion of the nuclear system process barrier. This chapter provides information regarding the reactor coolant system and pressure-containing appendages out to and including isolation valving. This group of components is defined as the reactor coolant pressure boundary (RCPB), in Section 50.2(v) of 10 CFR 50 as follows:

Reactor coolant pressure boundary means all those pressure- containing components of boiling and pressurized water- cooled nuclear power reactors, such as pressure vessels, piping, pumps, and valves, which are

a. Part of the reactor coolant system

b. Connected to the reactor coolant system, up to and including all of the following:

    1. The outermost containment isolation valve in system piping which penetrates primary reactor containment

    2. The second of the two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment

    3. The reactor coolant system safety/relief valves.

Section 5.5 of this chapter also deals with various subsystems of the RCPB that are closely allied to it. These are briefly reviewed below.

The nuclear pressure relief system (NPRS) protects the RCPB from damage due to overpressure. To protect against overpressure, pressure-operated safety/relief valves are provided to discharge steam from the nuclear steam supply system (NSSS) to the suppression pool. The NPRS also acts to automatically depressurize the NSSS in the event of a LOCA in which the high pressure coolant injection (HPCI) system fails to maintain reactor pressure vessel (RPV) water level. Depressurization of the NSSS allows the low- pressure core cooling systems to supply enough cooling water to cool the fuel adequately.

The RCPB leak detection system, described in Subsection 5.2.7, detects system leakage inside the primary containment so that appropriate action can be taken before the integrity of the nuclear system process barrier is impaired.

The RPV and appurtenances are described in Section 5.4. The major safety functions of the RPV are to maintain water over the core and to act as a radioactive material barrier. The RPV meets the requirements of applicable codes and criteria. The possibility of brittle fracture is considered, and suitable design and operational limits are established that avoid conditions where brittle fracture is possible.

The reactor recirculation system (RRS) provides coolant flow through the core. Adjustment of the core coolant flow rate changes reactor power output, thus providing a means of following plant load demand without adjusting control rods. The arrangement of the RRS routing is such that a piping failure cannot compromise the integrity of the floodable inner volume of the RPV, thereby ensuring adequate core cooling following a LOCA.

The main steam line flow restrictors are venturi-type flow devices. One restrictor is installed in each main steam line inside the primary containment. The restrictors are designed to limit the loss of coolant resulting from a main steam line break outside the primary containment. The coolant loss is limited so that RPV water level remains above the top of the core during the time required for the main steam line isolation valves (MSIVs) to close. This action maintains the integrity of the fuel cladding (fuel barrier).

The MSIVs automatically isolate the nuclear system process barrier in the event a pipe break occurs, thereby limiting the loss of coolant and the release of radioactive materials from the NSSS. Two MSIVs are installed on each main steam line, one inside and the other outside the primary containment. Closure of either of the two MSIVs acts to seal the primary containment in the event that a main steam line break occurs there. A third stop valve (third MSIV) is in each steam line downstream of the outboard MSIVs.

The reactor core isolation cooling (RCIC) system provides makeup water to the core during a reactor shutdown in which feedwater flow is not available. The system is started either automatically upon receipt of a low reactor water level signal or manually by the operator. Water is pumped to the core by a turbine pump driven by reactor steam.

The residual heat removal (RHR) system includes a number of pumps and heat exchangers that can be used to cool the NSSS under a variety of situations. During normal shutdown and reactor servicing, the RHR system removes residual and decay heat. The RHR system allows decay heat to be removed whenever the main heat sink (main condenser) is not available. Another operational mode of the RHR system is low pressure coolant injection (LPCI). Low pressure coolant injection operation is an engineered safety feature (ESF) system for use during a LOCA. This operation is described in Subsection 6.3.2.2.4.

The reactor water cleanup (RWCU) system functions to maintain the required purity of reactor coolant by circulating coolant through a system of filter-demineralizers.

### 5.1.1 Schematic Flow Diagram

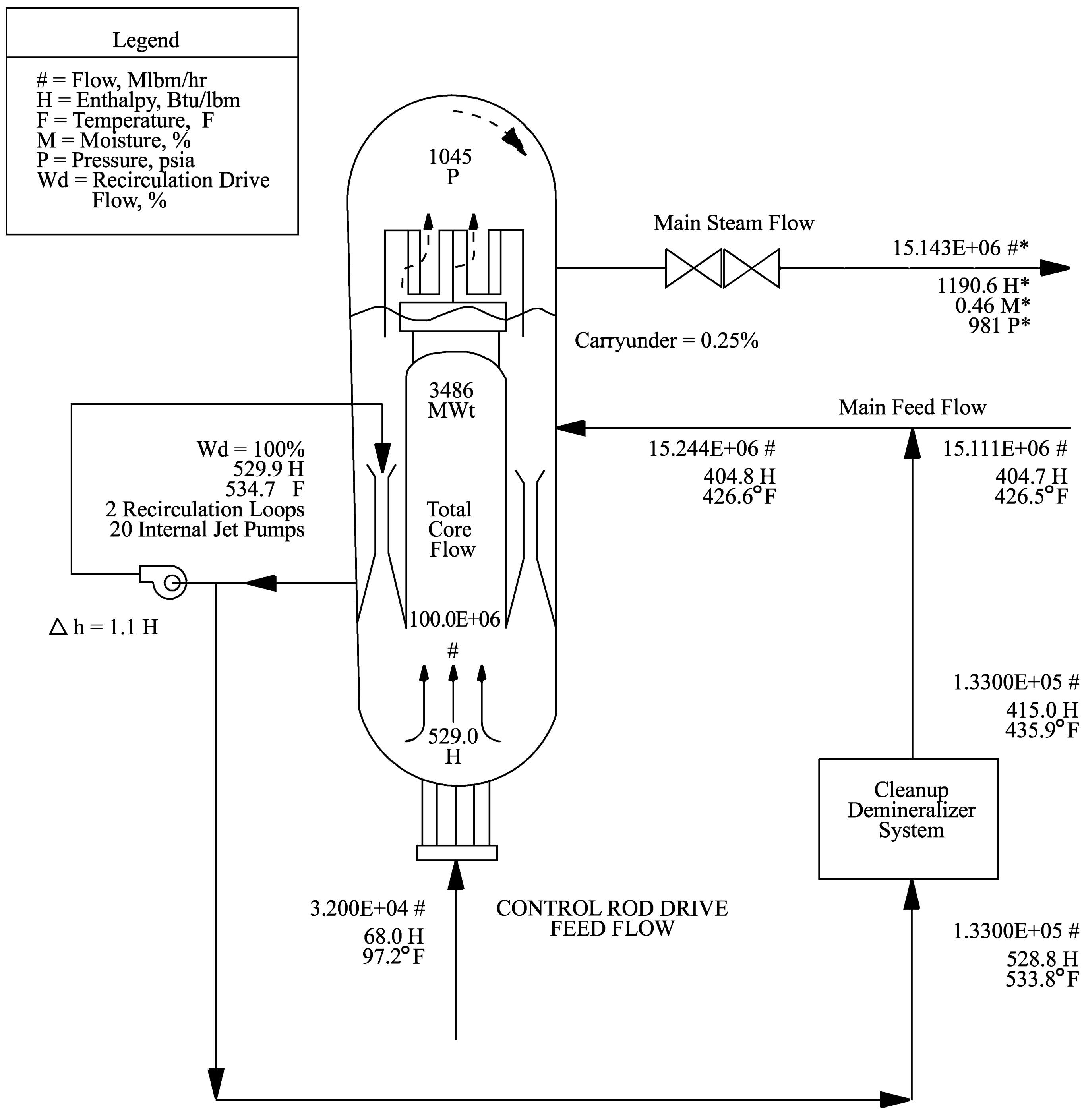
A schematic flow diagram of the reactor coolant system denoting all major components, principal pressures, temperatures, flow rates, and coolant volumes under normal steady-state full-power operating conditions is presented in Figures 5.1-1 and 5.1-2.

### 5.1.2 Piping and Instrumentation Diagram

A piping and instrumentation diagram for the NSSS is presented in Figure 5.1-3.

### 5.1.3 Elevation Drawing

Elevation drawings showing the containment system perspective and the principal dimensions of the reactor coolant system in relation to the containment are shown in Figures 5.1-4 and 5.1-5.




*Conditions at upstream side of TSV

| | |
|---|---|
| Core Thermal Power | 3486.0 |
| Pump Heating | 10.6 |
| Cleanup Losses | -4.4 |
| Other System Losses | -1.2 |
| Turbine Cycle Use | 3491.0 MWt |

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FIGURE 5.1-1A

REACTOR COOLANT SYSTEM FLOW DIAGRAM-
100% POWER / 100% CORE FLOW

| | | VOLUME OF FLUID (ft$^3$) |
|---|---|---|
| A | LOWER PLENUM | 3918 |
| B | CORE | 2054 |
| C | UPPER PLENUM & SEPARATORS | 1320 |
| D | DOME (ABOVE NORMAL WATER LEVEL) | 7553 |
| E | DOWNCOMER REGION | 5795 |
| F | RECIRC LOOPS & JET PUMPS | 1394 |

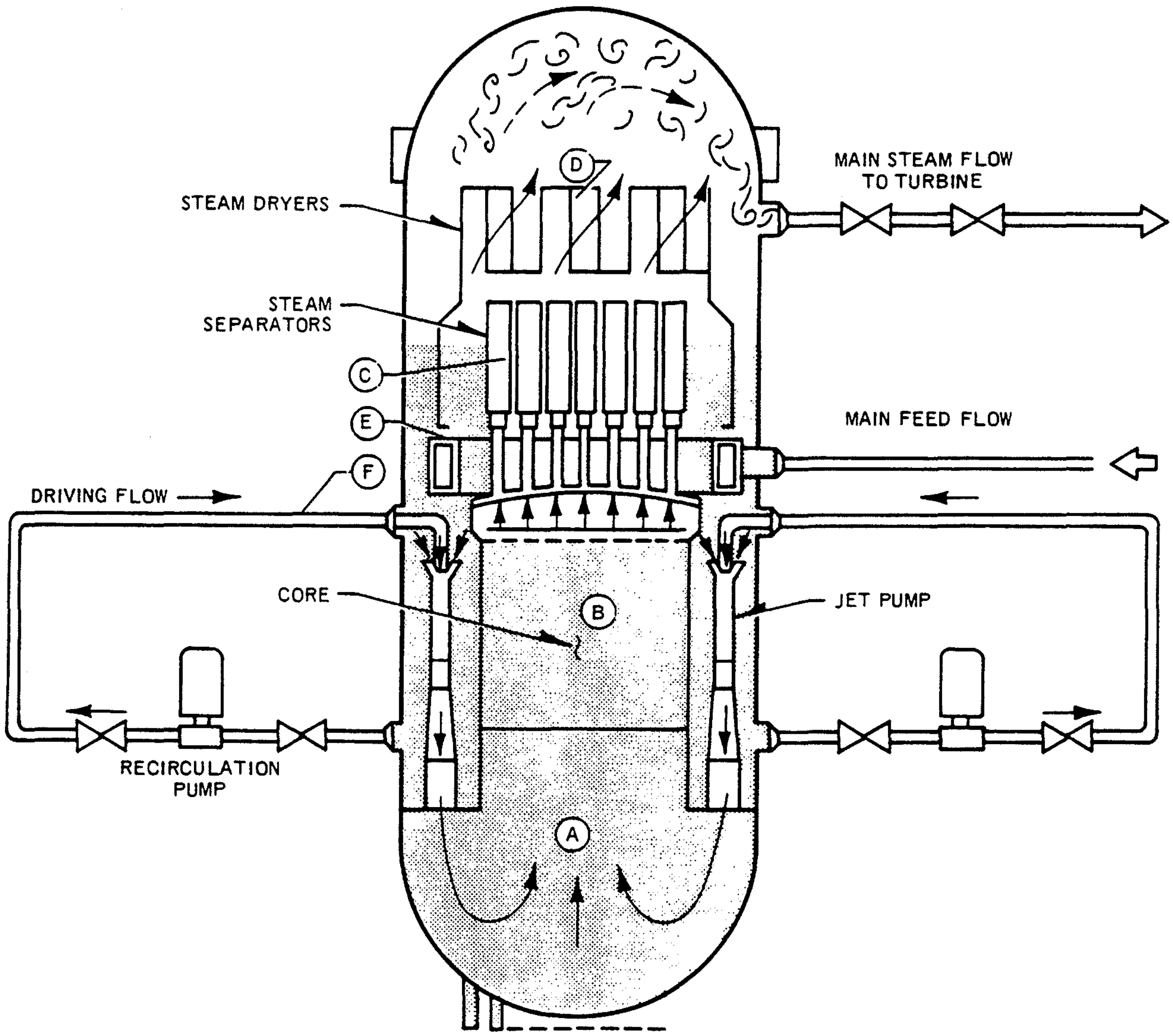


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FIGURE 5.1-2

COOLANT VOLUMES

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Refer to Plant Drawing M-2089

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FIGURE 5.1-3, SHEET 1

NUCLEAR STEAM SUPPLY SYSTEM
P&ID

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Refer to Plant Drawing M-2090

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FIGURE 5.1-3 SHEET 2
NUCLEAR STEAM SUPPLY SYSTEM P&ID

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Refer to Plant Drawing M-5538

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**FIGURE 5.1-3, SHEET 3**

**NUCLEAR STEAM SUPPLY SYSTEM P&ID**

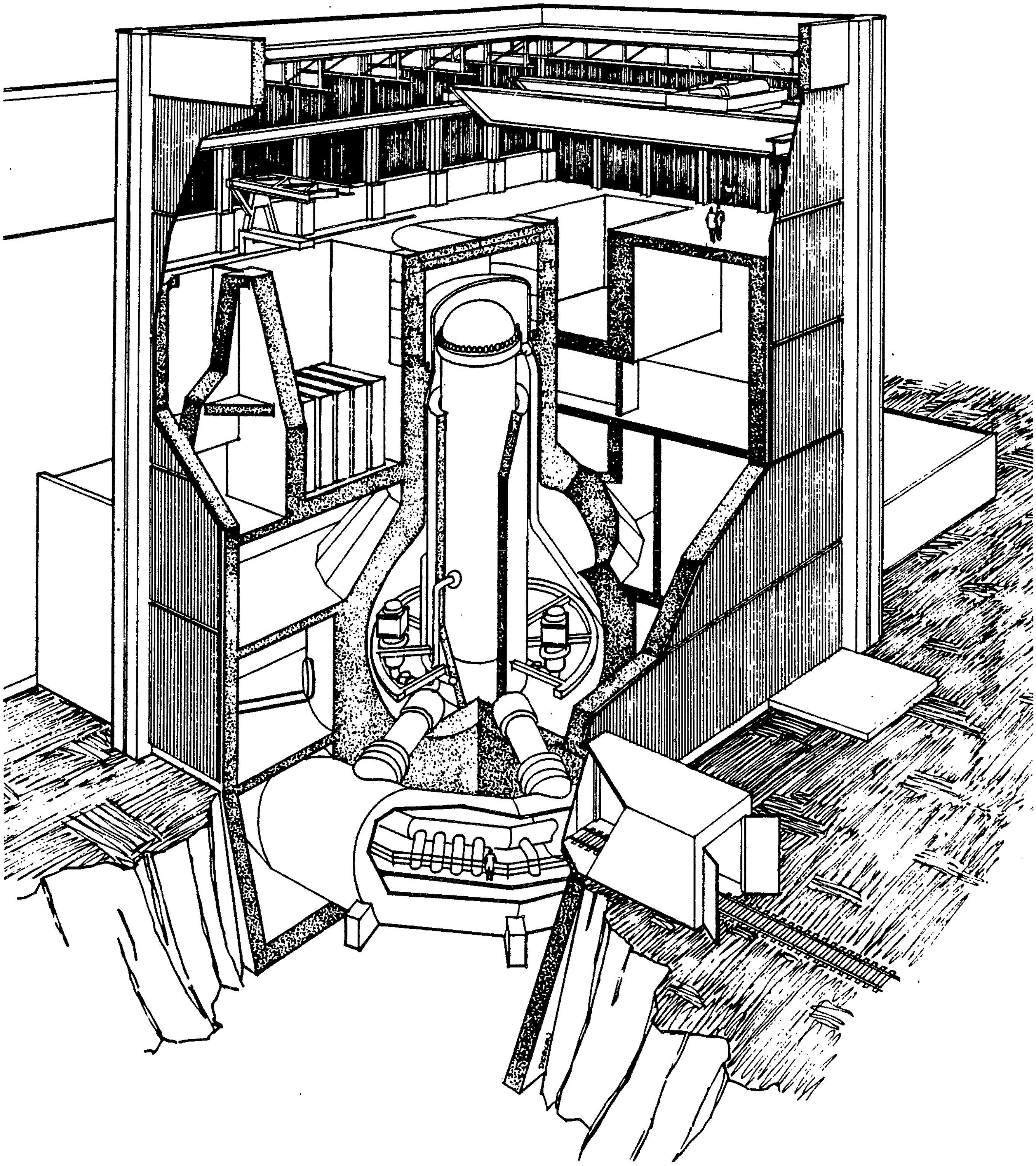

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FIGURE 5.1-4

CONTAINMENT SYSTEM PERSPECTIVE

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Plant Drawing GENERAL PC SKETCH

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FIGURE 5.1-5

GENERAL ARRANGEMENT, PRIMARY
CONTAINMENT SECTION

# 5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

## 5.2.1 Design of Reactor Coolant Pressure Boundary Components

### 5.2.1.1 Performance Objectives

#### 5.2.1.1.1 Reactor Pressure Vessel and Appurtenances

The function of the reactor pressure vessel (RPV) design is to provide a volume in which the core can be submerged in coolant, thereby allowing power operation of the reactor. Design of the RPV and appurtenances provides the means for attaching pipelines to the RPV and for installing RPV internal components. All or portions of each of the following support systems interface with the RPV and form part of the reactor coolant pressure boundary.

#### 5.2.1.1.2 Reactor Pressure Vessel Vent

The function of the RPV vent is to remove noncondensibles from the top dome of the reactor during power operation and to provide a vent path for floodup of the vessel prior to vessel head removal during refueling outages.

#### 5.2.1.1.3 Nuclear Pressure Relief System

The function of the nuclear pressure relief system (NPRS) is to limit any overpressure that occurs during abnormal operational transients.

#### 5.2.1.1.4 Main Steam Line Flow Restrictors

The function of the main steam line flow restrictors is to protect the fuel barrier by not allowing the core to be uncovered. The restrictors limit the loss of coolant from the RPV to a value that will ensure the core will remain covered with water before the main steam isolation valve closure, should rupture occur in a main steam line outside the primary containment. Additionally, the restrictors limit the depressurization rate of the reactor to a value which ensures that the steam dryer and other reactor internal structures will remain in place. This is to prevent fragments from the dryer to be blown down the steam lines that may prevent tight closure of the main steam isolation valves.

#### 5.2.1.1.5 Main Steam Line Isolation Valves

The function of the MSIVs, one of which is on the drywell side while the other is just outside the primary containment, is to prevent damage to the fuel barrier by limiting loss of reactor coolant for a major steam piping leak outside the primary containment. Main steam isolation valves also limit radioactive releases to the plant environs.

#### 5.2.1.1.6 Deleted

5.2.1.1.7 Feedwater System

The function of the feedwater system is to provide normal feed flow to the reactor pressure vessel during plant power operation. The feedwater inboard isolation valves also serve as the first isolation valve for return lines from the HPCI, RCIC and RWCU systems described below.

5.2.1.1.8 Reactor Recirculation System

The function of the reactor recirculation system (RRS) is to provide a variable moderator (coolant) flow to the reactor core for adjusting reactor power level.

5.2.1.1.9 Standby Liquid Control System

The function of the standby liquid control system (SLCS) is to provide backup reactivity control in the event the control rods do not completely shutdown the core following a scram initiation.

An additional function of the SLCS is to provide suppression pool pH control in the event of a loss-of-coolant accident in order to prevent iodine re-evolution.

5.2.1.1.10 Residual Heat Removal System

The function of the residual heat removal (RHR) system is as follows.

a. To remove decay heat and residual heat from the nuclear steam supply system (NSSS) so that refueling and NSSS servicing can be performed

b. To supplement the fuel pool cooling and cleanup system (FPCCS) capacity, when necessary, with additional cooling capacity

c. To provide containment (suppression pool) cooling and containment spray

d. To provide low-pressure coolant injection (LPCI) flow for RPV reflood and core cooling in the event of a DBA-LOCA

5.2.1.1.11 Core Spray System

The function of the core spray (CS) system is to provide low pressure coolant flow directly to the core fuel elements in the event of a loss-of-coolant accident.

5.2.1.1.12 High Pressure Coolant Injection System

The function of the high pressure coolant injection (HPCI) system is to provide high-pressure makeup to the RPV in the event of a small-break loss-of-coolant-accident.

5.2.1.1.13 Reactor Core Isolation Cooling System

The function of the reactor core isolation cooling (RCIC) system is to provide makeup water to the RPV during shutdown and isolation to ensure adequate core cooling.

5.2.1.1.14 Reactor Water Cleanup System

The function of the reactor water cleanup (RWCU) system is to maintain high reactor water purity to limit chemical and corrosive action, thereby limiting fouling and deposition on heat-transfer surfaces. It also removes excess reactor coolant during shutdown, startup, and hot standby conditions.

5.2.1.1.15 Nuclear System Leak Detection System

The function of the NSSS leak detection system (LDS) is to detect leakage from the nuclear system process barrier before predetermined limits are exceeded.

5.2.1.2 Design Parameters

Table 5.2-1 lists design temperature, pressure, and maximum test pressure for the reactor coolant pressure boundary (RCPB) structures and components. The specified operating transients used for the design of components within the RCPB are given in Table 5.2-2. A discussion of the input criteria for seismic design is contained in Subsection 3.7.1.

The design requirements established to protect the principal components of the reactor coolant system against environmental effects are discussed in Subsection 3.11.2.

5.2.1.3 Compliance With 10 CFR 50, Section 50.55a

Compliance with the guidelines of 10 CFR 50, Section 50.55a, "Code and Standards," is included in Tables 3.2-1 and 3.2-3 and in Section 3.2.

5.2.1.4 Applicable Code Cases

The RPV is designed in accordance with the 1968 ASME Boiler and Pressure Vessel (B&PV) Code Section III, Class A, with addenda through summer 1969. The steam and recirculation piping is designed in accordance with the 1969 ANSI B31.7 Nuclear Power Piping Code, Class I, including addenda ANSI B31.7b-1971. The recirculation system, motor-operated valves and pumps, and MSIVs are designed in accordance with the 1968 Draft ASME Code for Pumps and Valves for Nuclear Power, Class I. Main steam safety/relief valves comply with 1968 ASME B&PV Code, Secton III, 1969 Summer Addenda, Paragraph N911.4 for pilot-activated valves. Applicable code cases used in various aspects of the design are given in Table 5.2-3.

5.2.1.5 Design Transients

5.2.1.5.1 Loading and Stress Criteria for Reactor Coolant Pressure Boundary Components Designed by Rational Stress Analysis

The loading conditions may be divided into four categories: normal, upset, emergency, and faulted conditions. These categories are generically described in the ASME B&PV Code Section III, 1968 Edition, N-412. Representative loading combinations, design procedures, and acceptability criteria are listed in Tables 3.9-17 and 3.9-18. These tables apply only to the pressure-containing components of the RCPB. The seismic criteria for the RCPB are discussed in Subsection 3.7.2.

#### 5.2.1.5.2 Components Designed Primarily by Empirical Methods

There are some structural and electrical nonpressure-containing parts of equipment that are not normally designed or sized directly by stress analysis techniques.

Simple stress analyses are sometimes used to augment the design of these components, but the primary design work does not depend on detailed stress analysis. These components are usually designed from tests and empirical experience. Field experience and testing are used to support the design. Where the structural or mechanical integrity of components is essential to safety, the components referred to in these criteria have been designed to accommodate the events of the safe-shutdown earthquake (SSE), a design-basis pipe rupture, or a combination of these events where appropriate. The reliability requirements of such components cannot be quantitatively described in a general criterion because of the varied nature of each component and its specific function in the system.

#### 5.2.1.5.3 Detailed Analyses of Reactor Coolant Pressure Boundary Pressure Parts of the Reactor Pressure Vessel

The RPV is designed in accordance with the ASME B&PV Code (1968) Section III, its interpretations and applicable requirements for Class A vessels as defined therein, as of the summer 1969 addenda.

Both elastic and inelastic stress analysis techniques were used in the design of the RPV core support and reactor internal structures to show that stress limits were not exceeded, as described in Subsection 3.9.1.6.

Stress analysis requirements and load combinations for the RPV are evaluated as described in Tables 3.9-13 through 3.9-15. The RPV was designed for an operational life of 40 years. (Refer to Appendix B for evaluation of 60 years.)

### 5.2.1.6 Identification of Active Pumps and Valves

#### 5.2.1.6.1 Classification of Pumps and Valves

Pumps and valves (NPS > 1-1/4 in.) within the RCPB are listed in Table 5.2-4. These components may be classified as either active or inactive.

Active components are those whose operability (e.g., valve opening or closure, pump operation or trip) is relied on to perform a safety function and/or reactor-shutdown function during or following the transient or event under consideration. Inactive components are those whose operability is not relied on to perform safety or shutdown functions during or following the transient or event under consideration.

There are no active pumps within the RCPB. The RCPB valves are generally assumed to be active during normal operating and seismic events and system functional evaluations performed only for accident conditions.

Leaktightness capability requirements for all RCPB valves are included in the applicable valve specifications. Valve parts forming the RCPB were pressure tested in accordance with the requirements of Nuclear Pump and Valve Code or ASME B&PV Code Section III. The maximum allowable leakage past valve seats is 2 $cm^3$/hr/in. of seat diameter for gate and globe type valves and 10 $cm^3$/hr/in. of seat diameter for check valves under the system design pressure during manufacturer's shop test.

#### 5.2.1.6.2 Design Methods and Procedures for Pipe Rupture

The design objectives used to ensure that active RCPB components function as designed in the event of a pipe rupture are described in Section 3.6.

### 5.2.1.7 Design of Active Pumps and Valves

To ensure the functional performance of active valves of the RCPB, stringent design requirements were applied. Operability is ensured in the following manner.

All active valves were qualified for operability assurance by first being subjected to the following tests:

a. Shop tests, which include hydrostatic tests and seal leakage tests, were performed as specified in the applicable code

b. The valves are required to open and close within specified time limits when subjected to design or environmental conditions as required by applicable codes and regulatory guides. These valves were also subjected to cold hydrostatic tests and functional tests as part of the Preoperational Test Program.

Valves are designed to withstand the accelerations and/or loads predicted by the piping stress analysis. Assurance is therefore provided that the components will function as required when subjected to design loadings.

Finally, active valves are also required to be operated periodically, as required in the Technical Specifications. This repeated operability requirement throughout the life of the specified valve further provides assurance of reliable valve operation.

The representative combination of loads and analysis to ensure valve operability are summarized in Tables 3.9-17 and 3.9-18.

### 5.2.1.8 Inadvertent Operation of Valves

A discussion of the design-basis events and appropriate limits for this plant is given in Subsections 15.1.4, 15.2.2, 15.2.4, and 15.2.7. The events in Chapter 15 have been selected to envelop the most severe change in critical parameters from events that have been postulated to occur during planned operation.

### 5.2.1.9 Stress and Pressure Limits

Paragraphs NB-3655 and NB-3656 of ASME B&PV Code Section III are not directly applicable to pumps and valves. On the basis of the method of establishing design pressure, however, it can be stated that the requirements of Paragraph NB-3655.1 and NB-3656.1 of the above code are met for these components.

The allowable stress limits and design loads for NSSS components are summarized in Tables 3.9-8, 3.9-14, 3.9-17 through 3.9-26, 3.9-28 through 3.9-39, and 3.9-43.

### 5.2.1.10 Stress Analysis for Structural Adequacy

Stress analysis is used to determine structural adequacy of pressure components of the RCPB under various operating conditions and earthquakes. Significant discontinuities such as nozzles and flanges are considered. In addition to the design calculations required by the ASME Codes, stress analysis is performed by methods outlined in the code appendixes or by other methods applicable to the design condition through reference to analogous codes or other published literature.

Results of areas with potentially significant stress concerns are given for major components in Tables 3.9.17 through 3.9-26.

### 5.2.1.11 Analysis Method for Faulted Condition

Elastic stress analysis methods in conjunction with elastic system analysis were generally used for RCPB components. In the event that an inelastic stress analysis was performed, the analysis methods conform to the requirements of ASME B&PV Code Section III, Appendix F.

### 5.2.1.12 Protection Against Environmental Factors

The protection of the principal components of the reactor coolant system against environmental effects is discussed in Section 3.11. Missile protection is discussed in Section 3.5, and fire protection is discussed in Subsection 9.5.1.

### 5.2.1.13 Compliance With Code Requirements

For components that were constructed in accordance with Section III of the ASME B&PV Code Subsection NB, the analytical calculations or experimental testing was performed to demonstrate compliance with the code. Brief descriptions of the mathematical or test models and the methods of calculation or testing, including any simplifying assumptions with summary of results, are provided in Subsection 3.9.1 and in Table 3.9-13 and in Tables 3.9-18 through 3.9-24.

### 5.2.1.14 Stress Analysis for Emergency- and Faulted-Condition Loadings

The types of stress analysis that were used for the emergency and faulted conditions are given in tables in Section 3.9.

### 5.2.1.15 Stress Levels in Category I Systems

A representative list of Category I RCPB systems and associated stress levels is provided in Tables 3.9-13 through 3.9-24. Piping isometrics for the major systems are shown in Figures 3.9-6 through 3.9-15.

### 5.2.1.16 Analytical Methods for Stresses in Pumps and Valves

The methods and criteria for analysis of stresses and deformations in the pressure boundary portions of Class 1 pumps are as described in the ASME B&PV Code Section III and the Nuclear Pump and Valve Code.

The methods and criteria for design and acceptability of stresses and deformations, as determined for the pressure boundary portions of Class 1 line valves and safety/relief valves (SRVs), are those described in the applicable portions of the ASME B&PV Code Section III, and the Nuclear Pump and Valve Code.

Pumps, line valves, and safety/relief valves purchased for this project were constructed and designed in accordance with the categories explicitly addressed by the ASME Nuclear Pump and Valve Code and ANSI B-31.7 Nuclear Power Piping Code. In the event that components supplied with geometries or design conditions for which code limits had not been developed, a complete description of the analytical methods and criteria used for evaluation of stresses and deformations was submitted by the manufacturer.

The summary of the detailed analyses for selected RCPB components (analytical models, method of calculation, and a summary of results) is shown in Tables 3.9-17 and 3.9-18.

### 5.2.1.17 Analytical Methods for Evaluation of Pump Speed and Bearing Integrity

The Rayleigh's approximation method is used to calculate the combined pump and motor shaft critical speed. This procedure, which equates the inertial forces of the rotating masses to the elastic restoring forces in the shafts, yields the lowest possible frequency of resonant shaft excitation. The lowest vibration frequency thus calculated must be at least 130 percent of the maximum expected pump speed. The hydrodynamic bearings in the motor or pump are designed using "A Solution for Finite Journal Bearings and Its Application to Analysis and Design," by A. A. Raimondi and J. Boyd, ASME Transactions, Volume I, No. 1, April 1958 or by an equivalent method. If the pump has a hydrostatic bearing, the motor bearings are analyzed as above while the pump bearing is analyzed by use of a computer code which is the proprietary information of one of our pump vendors.

### 5.2.1.18 Operation of Active Valves Under Transient Loadings

The qualification test program to verify that active valves within the RCPB whose operability is relied upon to perform a safety function or to shut down the reactor operate under the transient loadings experienced during service life is described in the following subsections.

5.2.1.18.1 Motor-Operated Gate Valve

A motor operator built to the same design as the RRS gate valves has been tested to demonstrate its performance capability under expected operating conditions, including the containment environment after the LOCA. Performance was tested under maximum moisture, pressure, and temperature conditions after exposure to lifetime radiation dose and under design-basis seismic conditions.

5.2.1.18.2 Main Steam Line Isolation Valves

Components of the MSIVs that are required to operate during transient conditions and whose functional capabilities are sensitive to the abnormal ambient pressure and temperature associated with the transient were subjected to a test sequence that simulates the abnormal ambient conditions. Functional requirements were verified throughout the test sequence. Components prototypical of Fermi 2 valve components were tested.

5.2.1.18.3 Safety/Relief Valves

The SRVs were subjected to tests described in Subsection 5.2.2.6 that simulate conditions similar to those experienced during service life.

5.2.1.19 Field-Run Piping

All piping 2 in. in diameter and smaller was designed by Edison but was fabricated and installed in the field by the piping erection subcontractor. This includes all small process piping, instrument piping, and branches from large piping (2-1/2 in. and larger). Small piping exists in all Category I piping systems.

Design, materials procurement, fabrication, erection, and testing of field-run piping are done in accordance with documented process control procedures. Review and approval, particularly for Category I pipe routings, location, and identification of all shop and field welds, are required by these procedures.

Small RCPB piping is generally analyzed using the computerized elastic stress analysis techniques described in Section 3.9.

Hydrostatic testing, prior to erection, is required for any pipe spool that is embedded in concrete or installed in an inaccessible location.

5.2.1.20 Feedwater Sparger and Thermal Sleeve

Several distinct problems have been experienced with the feedwater nozzle and spargers of the design originally planned for Fermi 2. These problems resulted in sparger arm cracks, flow hole cracks, thermal sleeve cracks, and cracks in the feedwater nozzle itself. The causes for these problems were identified, solutions were investigated, and a new design for the feedwater thermal sleeve and sparger was developed.

General Electric prepared a detailed report on the problems, description of solutions, verification of solutions, and safety considerations. This report, "Boiling Water Reactor

Feedwater Nozzle/Sparger," NEDE-21821, March 1978, was submitted to the NRC. The report gives a considerable body of data to show the acceptability of the GE design.

Fermi 2 incorporates all elements of the new design.

The new sparger/thermal sleeve design meets the following objectives:

a. Protects the feedwater nozzle against the high-frequency thermal cycles that initiate nozzle cracks

b. Is immune to the vibration that causes sparger arm cracks

c. Eliminates low-flow stratification

d. Eliminates the nozzle flow separation that causes flow hole cracks.

In the spargers, top-mounted elbows, each with a converging discharge nozzle, replace the front discharge holes. These features solve two problems. The top-mounted elbows ensure that the sparger/thermal sleeve remains full of cold feedwater during low-flow conditions, thereby eliminating low-flow stratifica-tion. The converging discharge orifices eliminate the flow separation that was the cause of flow hole cracking.

The junction between the thermal sleeve and sparger arms uses a forged tee, which improves resistance to vibration-induced cracking.

The thermal sleeve configuration is drastically different from previous designs. The inner thermal sleeve is the feed pipe for the sparger and is sealed against the safe-end with a piston ring. The inner thermal sleeve is welded to the forged tee.

Since leakage will eventually occur past the primary seal, a means must be provided to protect the nozzle against this leakage. To provide the required protection, a second seal is provided downstream of the primary seal. This secondary seal is attached to an intermediate thermal sleeve, which is open to the reactor at its downstream end. The annulus between the inter-mediate and inner thermal sleeves has a low hydraulic resistance and serves to channel leakage to the reactor without impinging on the feedwater nozzle. As a further impediment to leakage and to provide damping against vibration, an interference fit is provided between the ring, which contains the secondary seal, and the nozzle safe end.

The two seal members are joined by a slotted member. This slotted member provides a structural tie between the two seal members, which allows radial thermal expansion while providing rigidity against the translational motion of vibration. The slots also provide a flow path for the primary leakage flow to enter the inner annulus.

Since the second seal is exposed to a very small pressure differential, its tendency to leak is very small.

Primary leakage flowing between the inner and intermediate sleeves would cool the intermediate sleeve and thereby produce a cold boundary layer on the outside of the intermediate sleeve. This boundary layer might then shed and produce nozzle thermal cycles. To preclude this, an outer sleeve is provided to isolate the nozzle against such shedding.

Thermal cycling is the cause for blend radii cracking. The presence of cladding increases thermal stresses by approximately a factor of 2. Most plants have elected to machine off the cladding in this region. The design and fabrication of the Fermi 2 vessel did not clad the feedwater nozzle and blend radii. Therefore, incorporation of the new design will not involve the task of removing cladding. Calculations and test program results show the potential for crack initiation is essentially zero for extended high-frequency, low differential-temperature thermal cycling, expected with the new design and unclad nozzles.

Preservice ultrasonic (UT) examinations of the blend radii were conducted by SWRI, and magnetic-particle examination was conducted by the RPV manufacturer. No recordable indications were found by either technique. The Fermi 2 feedwater sparger and thermal sleeve design is in conformance with NUREG-0619.

The Fermi 2 ISI NDE program requires performance of periodic feedwater nozzle inner radius examination as required by ASME Section XI and NUREG-0619, or other NRC approved alternative program, to detect service induced degradation (cracking).

In-service penetration (PT) examination of the nozzle blend radii area will not be performed because of very limited access and the possibility of damage to the thermal sleeve and sparger assemblies in preparing the surface for PT examination.

### 5.2.2 Overpressurization Protection

#### 5.2.2.1 Location of Pressure-Relief Devices

Figure 5.1-3 shows the schematic location of all pressure-relieving devices for

a. The reactor coolant system
b. The primary side of the auxiliary or emergency systems interconnected with the primary containment system
c. All blowdown or heat dissipation systems connected to the discharge side of the pressure-relieving devices.

#### 5.2.2.2 Mounting of Pressure-Relief Devices

##### 5.2.2.2.1 Safety Design Bases

The NPRS is designed

a. To prevent overpressurization of the NSSS that could lead to the failure of the nuclear system process barrier
b. To provide automatic depressurization for small breaks in the NSSS so that the LPCI and the core spray systems can operate to protect the fuel barrier
c. To permit verification of its operability
d. To withstand adverse combinations of loadings and forces resulting from operation during abnormal, accident, or special-event conditions.

##### 5.2.2.2.2 Power Generation Design Bases

The NPRS SRVs have been designed

a. To maintain reactor pressure below the ASME B&PV Code Section III allowable maximum pressure during abnormal operational transients

b. To provide automatic depressurization for small breaks in the NSSS occurring with maloperation of high pressure coolant injection (HPCI) so that the low pressure coolant systems (LPCI and core spray) can operate to protect the fuel barrier

c. To discharge to the primary containment suppression pool

d. To correctly reclose following operation so that maximum operational continuity can be obtained.

5.2.2.2.3 Description

The NPRS consists of SRVs located on the main steam lines between the RPV and the first isolation valve within the drywell. These valves protect against overpressurization of the NSSS.

The SRVs provide four main protection functions:

a. Overpressure-relief operation - The valves open by application of external power to limit a pressure rise. In the relief valve mode, any of these valves can be operated by manual action from the control room. No particular setpoint applies to this method of operation, as the operator may open a valve at his discretion for blowdown or test over a wide pressure range

b. Overpressure-safety operation - The valves function as safety valves and open to prevent NSSS overpressurization. These valves are self-actuated at their spring setpoint if not already opened for relief operation

c. Depressurization operation - Five valves are opened by indirectly operated devices (pneumatic) as part of the emergency core cooling system (ECCS) for events involving small breaks in the nuclear system process barrier. These valves, which are selected for automatic depressurization, are activated automatically

d. Post fire depressurization operation – Selected valves are manually operated from the control room using their pneumatic controls to enable use of low pressure makeup for certain post fire shutdowns.

Figure 5.1-3 shows the schematic location of the valves and piping. The SRVs are constructed and marked with data in accordance with the 1968 Draft of the ASME Nuclear Pump and Valve Code and addenda through March 1970. The popping-point tolerance, the pressure at which valves open by high steam pressure, conforms with the ASME B&PV Code Section III.

The majority of events that lead to actuation of the primary system SRVs are those that initially or eventually produce a NSSS pressure increase. These pressure-increase events result from sudden reductions of steam flow while the reactor is operating at power.

Table 5.2-5 shows the set pressures of the safety/relief valves. Once any SRV opens, subsequent actuations are controlled by two SRVs that are armed with the low-low set relief logic. The duration of each relief discharge should in most cases be less than 15 sec.

The SRVs are designed to operate in the accident environments stated in Table 3.11-1.

These conditions envelop the predicted pressure and temperature response of the containment following the design-basis LOCA (Subsection 6.2.1).

Each SRV discharges steam through a discharge line to a quencher device located below the minimum water level in the primary containment suppression pool. The SRV discharge piping is designed to limit valve outlet pressure to 40 percent of maximum valve inlet pressure with the valve wide open. Water in the line more than a few feet above suppression pool water level would cause excessive pressure at the valve discharge when the valve is again opened. For this reason, a vacuum relief valve is provided on each SRV discharge line to prevent drawing an excessive amount of water up into the line as a result of steam condensation following termination of relief operation. In addition, the safety/relief blowdown control system ensures that subsequent SRV discharges will not occur during periods of elevated water legs in the discharge piping (see Fig. 7.3-12, Sheet 1).

The selection of size of safety/relief line vacuum breakers for Fermi 2 was based on the following parameters:

a. Instant condensation of steam is assumed following SRV closure

b. The vacuum created must be equalized in 2 sec

c. The volume to be relieved is based upon the longest safety/relief line

d. The drywell pressure was its minimum value, 14.2 psia

e. Conservative L/Ds and $C_v$'s were selected for the valve.

The Fermi 2 study selected 8-in. vacuum relief valves. The capacity, $C_v$, set pressure, and pressure drop at rated flow for these valves used in the study calculation were supplied by the vendor based on extrapolation of experimental data taken from smaller but similar valves. The calculations in the study showed that under the parameters selected above, the vacuum will be relieved in 1.5 sec (versus the recommended 2 sec), and that the water leg inside the line would rise less than 4.3 ft past the submerged end of the line in this time.

The SRVs are located on the main steam line piping, rather than on the RPV top head, primarily to simplify the discharge piping to the pool and to avoid the necessity of having to remove sections of this piping when the reactor head is removed for refueling. In addition, valves located on the steam lines are more accessible to correct possible valve malfunctions during a shutdown.

Each of the five SRVs provided for automatic depressurization system (ADS) is equipped with an accumulator and check valve arrangement. Each accumulator receives pneumatic pressure from the safety-grade Division I primary containment pneumatic supply lines, which also supplies pressure to the air operators of two non-ADS SRVs. Division I primary containment pneumatic supply is normally fed from the nitrogen supply system, with Division I noninterruptible control air (NIAS) available by operator action to be manually cross connected as a backup supply for the normal pneumatic supply. There is also a qualified connection located outside the secondary containment to permit bottled nitrogen to be supplied as an additional backup source for the Division I pneumatics. The sizing for the ADS accumulators allows about 17 hours for the recovery of a backup pneumatic supply under the most limiting postulated event conditions requiring ADS. Leakage from the accumulator assembly and the SRV air operator subassembly were considered in evaluating the accumulator sizing. Each accumulator has adequate storage capacity to allow five actuations of an SRV at the long-term drywell pressure of the design SBLOCA analysis (see Figure 6.2-15) without the recovery of backup pneumatic supply pressure. This provides adequate pneumatic storage to cover interruptions if the pneumatic supplies are switched from the normal to the emergency backup sources. There are also eight non-ADS SRVs supplied by Division II of the primary containment pneumatic supply system, which is a separate, fully qualified pneumatic subsystem, but does not include NIAS as a backup supply. Backup nitrogen is provided bottles located inside the secondary containment to allow the use of Division II SRVs for certain Appendix R post-fire shutdowns from the control room accompanied by a loss of offsite power. An additional separate qualified connection located outside the secondary containment is provided to permit bottled nitrogen to be supplied for a backup source of the Division II pneumatics. The backup pneumatic supplies of both divisions of primary containment pneumatic supply system, although no credit is taken for, would allow the ten non-ADS SRVs to be operated as a backup for reactor pressure relief. The ten non-ADS SRVs include two SRVs, one associated with each pneumatic division, which have accumulators for the Low-Low Set function (see Section 5.2.2.5.3). Refer to Figure 5.2-1 for a diagram of the primary containment pneumatic supply system. The drywell nitrogen pneumatic system is described in Section 9.3.6.

The NPRS automatically depressurizes the NSSS sufficiently to permit the LPCI and core spray systems to operate. Depressurization occurs when five of the SRVs are opened automatically (ADS).

Descriptions of the operation and features of the automatic depressurization system are found in Subsections 6.3.2 and 7.3.1.

The NSSS can be depressurized manually if the main condenser is not available as a heat sink after reactor shutdown. The SRVs are operated by remote manual controls from the main control room. Controls for two of the relief valves are located on the remote control panel, and can thus be operated outside the main control room.

### 5.2.2.3 Overpressure Protection Analysis

The vessel overpressure protection system is designed to satisfy the requirements of Section III, Nuclear Vessels, of the ASME Boiler and Pressure Vessel Code 1968. The general requirements for protection against overpressure, as given in Article 9 of Section III, recognize that reactor vessel overpressure protection is one function of the reactor protection system and allows the integration of pressure relief devices with the protection system of the nuclear reactor. Hence, credit is taken for the reactor protection system as a complementary pressure protection device. However, the vessel overprotection analysis for Fermi 2 takes credit only for reactor protection system signals which are indirectly derived.

Included in this subsection are the design bases for sizing of the SRVs, the overpressure protection analysis, and the effects on the vessel pressure transients of valve capacity. The overpressure protection analysis used the actual Fermi 2 scram characteristics (e.g., for BWR/4 scram and control rod drive (CRD) systems).

The head spray piping (Class 2 pipe in the drywell) is no longer connected with the RPV. Therefore, it is no longer protected by the RPV overpressure protection system. However, a blank flange is installed in the line, preventing any pressurization of the head spray pipe.

#### 5.2.2.3.1 Design Basis

##### 5.2.2.3.1.1 Safety/Relief Valve Sizing

The safety/relief valve capacity of the Fermi 2 plant is sized to limit the primary system pressure, including transients, to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, 1968, Nuclear Vessels. The essential ASME requirements, which are all met by this analysis, are stated below.

It is recognized that the protection of vessels in a nuclear power plant is dependent upon many protective systems to relieve or terminate pressure transients. Installation of pressure relieving devices may not independently provide complete protection.

The safety/relief valve sizing evaluation assumes credit for operation of the reactor protection system. A scram may be initiated by any one of three sources; i.e., steam system isolation (i.e., direct), neutron flux, or reactor vessel pressure signal. The system isolation scram signal is derived from position switches mounted on the main steamline isolation valves or the turbine stop valves or from pressure switches mounted on the dump valve of the turbine control valve hydraulic actuation system. The position switches are actuated when the respective valves are closing and following 10 percent travel of full stroke. The pressure switches are actuated when a fast closure of the control valves is initiated. However, according to General Electric methodology, the safety/relief valve sizing evaluation does not assume credit for direct scram, only for the indirect flux scram. Further, no credit is allowed for power operated pressure relieving devices. Credit is taken only for the dual purpose safety/relief valves in their ASME Code qualified mode of safety operation.

The above considerations in the vessel overpressure analysis methodology require multiple equipment failures to occur. The probability of this many multiple failures (loss of direct scram and no automatic power operated relief valve actuation) is sufficiently low that the event should be considered, as a minimum, an “emergency” condition. However, the analysis applies the more conservative “upset” code requirements rather than the “emergency” limits such that the rated capacity of the pressure relieving devices is required to be sufficient to prevent a rise in pressure within the protected vessel of more than 110 percent of the design pressure (1.10 x 1250 = 1375 psig). All combination safety/relief valves discharge into the suppression pool through a discharge pipe from each valve which is designed to achieve sonic flow conditions through the valve, thus providing flow independence to discharge piping losses.

The ASME Boiler and Pressure Vessel Code requires the nominal pressure setting of at least one safety/relief valve connected to any vessel or system to not be greater than a pressure at the safety/relief valves corresponding to the design pressure (1250 psig) anywhere in the protected vessel. Valves which are additional to the one(s) set at or below design pressure, may be set higher, but in no case are any of these settings to exceed a pressure at the safety/relief valves corresponding to 105 percent of the design pressure anywhere in the vessel.

#### 5.2.2.3.2 Method of Analysis

To design the pressure protection for the nuclear boiler system, a detailed analytical model representing all essential dynamic characteristics of the system is simulated on a computer. This model includes the hydrodynamics of the flow loop, the reactor kinetics, the thermal characteristics of the fuel and its transfer of heat to the coolant; and all the principal controller features, such as feedwater flow, recirculation flow, reactor water level, pressure, and load demand. These characteristics are represented with all their principal nonlinear features in a model that has evolved through extensive experience and favorable comparison of the analysis results with actual BWR test data. A detailed description of the model is documented in a General Electric licensing topical report.*

Typical capacity characteristics, as modeled, are represented in Figure 5.2-1(a) for the safety/relief valves. The associated bypass, turbine control valve, and main steam isolation valve (MSIV) characteristics are, of course, also represented fully in the model.

**NOTE:** * Report reference above is: General Electric Company, Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors, NEDO-24154, October 1978. (ODYN)

#### 5.2.2.3.3 System Design

A parametric study was conducted to determine the required steam flow capacity of the safety/relief valves which satisfies the ASME Code requirements. The parameters used in the study have been updated to evaluate the impact of the 105% steam flow power uprate.

##### 5.2.2.3.3.1 Analytic Conditions

| Parameter | Value |
|---|---|
| Power level, MWt | 3499 (102% of 3430) |
| Steam flow, lb/hr | 15,200,000 |
| Core flow, lb/hr | $105 \times 10^6$ |
| Vessel dome pressure, psig | 1048 |
| Doppler coefficient | (a) |
| Average fuel temperature, °F | 1330 |
| Dynamic void reactivity coefficient | (a) |
| Void fraction | (a) |
| Control rod scram speed | See Figure 5.2-1(b) |
| Scram reactivity curve | (a) |
| High neutron flux (APM) scram percent of initial power (3430 MWt) | 124.4 (b) |

| | |
|---|---|
| High vessel dome scram pressure, psig | 1126 [(b)] |
| High vessel dome pressure recirculation pump trip set point, psig | 1135 |

(a) This input is calculated by ODYN analysis.
(b) Maximum safety limit.

The ATWS recirculation pump has been simulated in the overpressure analysis performed with ODYN.

5.2.2.3.3.2 Transients

The overpressure protection system must accommodate the most severe pressurization transient. Both the closure of all main steam isolation valves and a turbine trip with bypass failure produce severe transients. The evaluation of transient behavior with final plant configuration has shown that the isolation valve closure is slightly more severe when credit is taken only for indirect derived scrams, therefore, it is used as the overpressure protection basis event.

5.2.2.3.3.3 Scram

a. Direct reactor scram - Not credited (failure assumed)

b. Scram reactivity curve - This input is calculated by ODYN analysis

c. Control rod drive scram motion - See Figure 5.2-1(b)

5.2.2.3.3.4 Safety/Relief Valve Characteristics

| | |
|---|---|
| Type | Target Rock |
| Number | 15[a] |
| SRV capacity, steam flow | $87 \times 10^4$ lb/hr at 1090 psig[b] |
| First safety relief analytical setpoint, psig | 1169 |
| Number of safety relief groups simulated | 3 |
| Increment in SRV setpoint between groups, psi | 10 |
| Valve response characteristics | See Figure 5.2-1(a) |

a 11 SRVs were used in the overpressure protection analyses for power uprate.
b See Table 5.2-5 for SRV capacities and setpoints.

5.2.2.3.3.5 Safety/Relief Valve Sizing

The safety/relief valve capacity required for overpressure protection is determined from the minimum capacity that will provide an adequate margin between the peak vessel pressure and the vessel code limit (1375 psig) in response to the MSIV closure-flux SCRAM event. The number of safety/relief valves which provide a total capacity equal to or greater than the minimum required capacity constitutes the minimum safety valve requirement for overpressure protection.

The MSIV closure-pressure SCRAM event is evaluated as confirmation of the safety/relief valve capacity determined from the safety/relief valve sizing criteria and to demonstrate the overpressure protection capability of the safety/relief valve system at the highest level of indirect SCRAM.

#### 5.2.2.3.4 Evaluation of Results

##### 5.2.2.3.4.1 Safety/Relief Valve Capacity

The required SRV capacity is determined by analyzing the pressure rise from an MSIV closure with a flux scram transient. The plant is assumed to be operating at turbine-generator design conditions at a maximum vessel dome pressure of 1048 psig. The analysis hypothetically assumes the failure of the direct isolation valve position scram. The reactor is shut down by the backup, indirect, high-neutron flux scram. For the analysis, relief setpoints of the SRVs are assumed to be in the range of approximately 1169 to 1190 psig.

Under the general requirements for protection against overpressure as given in Section III of the ASME Code, credit can be allowed for a scram from the reactor protection system. As discussed in Section 5.2.2.3.1.1, the backup reactor high-neutron-flux scram is conservatively applied as a design basis for determining the required capacity of the pressure-relieving dual-purpose SRVs. The direct position scrams are not used in the design basis but could be since they qualify as acceptable pressure protection devices when determining the required SRV capacity of nuclear vessels under the provisions of the ASME Code.

The cycle specific overpressure protection analysis is included with the supplemental reload licensing report and Figure 5.2-1(c) shows the analytical results from TRACG, with only 11 of the 15 SRVs operating. Beginning with Cycle 16, the cycle specific overpressure analysis is performed with TRACG (References 23 and 24). The sequence of events assumed in this analysis was investigated to ensure that the ASME Code requirements were met and to evaluate the pressure relief system exclusively. The peak vessel (bottom) pressure for the MSIV transient with high-flux scram is less than the 1375 psig allowed by the ASME Code.

#### 5.2.2.3.5 Safety/Relief Valve Characteristics

##### 5.2.2.3.5.1 Schematic Arrangement

The schematic arrangements of the safety/relief valves are shown in Figures 5.2-1(d) and 5.2-1(e).

##### 5.2.2.3.5.2 Pressure Drop in Inlet and Discharge

Pressure drop on the piping from the reactor vessel to the valves is taken into account in calculating the maximum vessel pressure reported above.

Pressure drop in the discharge piping to the suppression pool is limited by proper discharge line sizing to prevent back pressure on each safety/relief valve from exceeding 40 percent of the valve inlet pressure, thus assuring choked flow in the valve orifice and no reduction of valve capacity due to the discharge piping. Each safety/relief valve has its own separate discharge line.

5.2.2.3.5.3 Safety/Relief Valve Description

These valves were manufactured by Target Rock Corporation to ASME Section III, 1968 with 1970 Summer Addenda. They comply with ASME Section III, 1969 Summer Addenda, Paragraph N911.4 for pilot activated valves.

Valve quantities and Technical Specification set pressures are as follows: Note: These values are based on actual vendor test data, not analytical values.

| Quantity | Set Pressure (psig) | ASME Rated Capacity at 103 Percent of Set Pressure (lb/hr minimum) |
|---|---|---|
| 5 | 1135 | 904,400 |
| 5 | 1145 | 912,200 |
| 5 | 1155 | 920,100 |

### 5.2.2.3.6 Conclusions

Safety requirements have long demanded very high reliability in the reactor scram functions. Recognition of this reliability as being completely adequate justification for these functions to contribute to vessel pressure protection is reflected in the ASME Section III Code provisions. As discussed in subsection 5.2.2.3.1.1, actual General Electric design practice very conservatively applies the code provisions through use of margins even beyond those necessary to satisfy code limits. This further enhances the reliability of vessel pressure protection.

This design basis for sizing safety valves with indirect scram credit is technically sound and a most realistic approach. It is allowed under Section III of the ASME Boiler and Pressure Vessel Code, and has been adopted by the General Electric Company in the design of the Fermi 2 boiling-water reactor.

### 5.2.2.4 Main Steam Safety/Relief Valves

The Fermi 2 valves are the Target Rock Corporation Model 7567F, two-stage, pilot-operated SRVs. The pilot stage is designed for stable setpoint performance and high tolerance of pilot seat leakage.

### 5.2.2.4.1 Description

Figure 5.2-2 shows the top works for the two-stage valve. Reactor pressure is communicated through port (5) around the stabilizer disk (7) to the pilot disk (6). With the pilot disk (6) seated, pressure is supplied through the connecting port (10) to volume (3) against the main piston (4) which holds the main disk closed. When the reactor pressure reaches the pilot setpoint, the pilot lifts and the stabilizer disk seats. The stabilizer holds the pilot disk open as long as the stabilizer is against its own seat. The open pilot valve forms part of the path that releases the steam in volume (3) through ports (8), (9), and (10)  The pressure in (3) drops quickly, and differential pressure across the main piston (4) opens the main stage valve.

SRV actuation is indicated by tailpipe pressure switches.  The operations are displayed and recorded in the control room.

The principal features of the Target Rock two-stage design, and how they relate to improved performance, are described below:

a. The pilot valve is connected directly to the main piston chamber (3).  If there is leakage past the pilot disk (6), it comes from the inlet pressure port (5) and through leakage passages around the main piston that maintain the pressure in chamber (3); leakage goes to the valve discharge line through port (9).  Tests have shown that, even with leakage at 200 lb/hr, there is no appreciable effect on setpoint performance, and leakage will not cause the valve to open and blow down the reactor.  Calculations show that the pilot leakage could reach a level greater than 1000 lb/hr without pilot lift or main-stage operation.

b. The 2-stage design has a direct-acting pilot with no pressure-sensing bellows and no need for a pressure switch.  This feature resolves three problems that have occurred in earlier designs which used a leakage containing pilot bellows.

    1. Bellows leak

    2. Switch failures

    3. Short circuits in switch wiring.

c. The air actuator (11) is an integral part of the bonnet and has improved diaphragm-sealing characteristics.  This change eliminates the need for grease or gaskets to effect an adequate seal.  Tests and operational experience have shown delamination failures of the diaphragm in earlier designs.  Tests under the same environmental conditions showed that the 2-stage pilot air operator diaphragm does not delaminate.

#### 5.2.2.4.2 Materials

The topworks body is made of ASME-SA-105 as a forging.  This combination of material and fabrication is code acceptable for this service.

### 5.2.2.5 Safety Evaluation

#### 5.2.2.5.1 Introduction

The ASME B&PV Code requires that each RPV designed to meet Section III be protected from overpressure.  The code allows a peak allowable pressure of 110 percent of RPV design pressure.  The SRVs are set to open as a safety function in the range of 1135 to 1155 psig.

There are two major transients that represent the most severe abnormal operational transients resulting in an NSSS pressure rise. They are the closure of all MSIVs, and a turbine trip with coincident loss of condenser vacuum.

The transient produced by the closure of all MSIVs and the failure of direct scram represents the most severe operational pressure rise. The required relief valve capacity is determined by analyzing the pressure rise from such a transient. The plant is assumed to be operating at the turbine-generator design conditions at a maximum vessel dome pressure of 1048 psig. The analysis hypothetically assumes the failure of the direct isolation valve position scram. The reactor is shut down by the backup, indirect, high neutron flux scram. For the analysis, the self-actuated setpoints (safety function) of the SRVs are assumed to be in the range of approximately 1169 to 1190 psig. The analysis indicates that the design valve capacity is capable of maintaining adequate margin (at least 50 psi at the bottom of the RPV) below the peak ASME B&PV Code allowable pressure in the NSSS (1375 psig). The sequence of events assumed in this analysis was investigated to confirm conformance to code requirements and to evaluate the adequacy of the NPRS.

Under the general requirements for protection against overpressure as given in Paragraph NB-7000 of the ASME B&PV Code Section III, credit can be allowed for a scram from the reactor protection system (RPS). When determining the required SRV capacity, credit is also taken for the protection signals, which are indirectly derived. The backup reactor high neutron flux scram is conservatively applied as a design basis in determining the required capacity of the pressure-relieving dual-purpose SRVs.

Studies have been made on the loadings that the SRVs impose on the main steam line. The loadings considered include

a. Thermal expansion effects of the SRVs discharge piping

b. Dynamic effects of the SRVs and discharge piping due to earthquakes

c. The dynamic and jet force exerted on the SRVs during the first millisecond after the valve is opened and prior to the time that steady-state flow has been established. With steady-state flow, the dynamic flow reaction forces are self-equilibrated by the valve discharge piping. For the analysis and forcing function, refer to Subsection 3.9.2.5

d. Deleted

Thermal expansion analyses were made for several cases including the relief valve piping, both cold and hot, and jet forces.

The critical effect is the stress at the branch connection below the valve. In no case does the stress at this point exceed code specifications.

The analysis that forms the basis for the evaluation of the pressure relief function of the NPRS appears in Subsections 15.2.2, 15.2.3, and 15.2.4.

The setpoints of the relief valves are adjusted to operate in the range from 1135 to 1155, by self-actuation (i.e., overpressure relief function). The reactor is shut down by the normal trip scram (turbine stop valve closure scram).

System malfunctions that pose threats to the radioactive material containment barriers are presented in Chapter 15.

5.2.2.5.2 Two-Stage Target Rock Safety/Relief Valves

The special test programs have shown that the 2-stage pilot operated Target Rock SRV design has potential for improved reactor safety, plant availability, and capacity factor as compared to an earlier Target Rock design from the following considerations:

a. The probability of spontaneous valve opening because of pilot valve leakage has been made essentially zero. This problem has had a significant effect on availability and capacity factor

b. The possibility of setpoint changes because of bellows leakage has been eliminated completely. Actual setpoint changes caused by bellows leakage have been rare; however, leakage past bellows seals, switch failures, and related problems have reduced availability and capacity factors. (Note that the function of the seal bellows on the stem of the air operator of the two-stage valve is in no way related to the function of the pilot bellows of the three-stage valve).

c. The probability of air operator diaphragm failures has been reduced. This item has been of lesser concern than the first two, but it is a significant improvement

d. The integral air actuator has improved the pressure boundary, and reduced the probability of bending and/or sticking of the actuator shaft.

5.2.2.5.3 Reducing Stuck-Open Relief Valve Events

In response to NUREG-0737, Item II.K.3.16, GE, on behalf of the BWR Owners Group, has performed a study of the feasibility contraindications of reducing challenges to the SRVs by various methods. This study reviews the potential methods of reducing the likelihood of stuck-open relief-valve (SORV) events in BWRs and estimates the reduction in such events that can be achieved by implementing these methods. The results of this study have been provided to the NRC.

Although the NUREG-0737 position deals primarily with the reduction of challenges to SRVs, its clear intent is to reduce the incidence of SORV events. Reducing challenges is only one of three approaches to reducing SORV events. The other two are reducing the causes of spurious blowdowns and reducing the probability of SRVs to stick open when challenged. All three of these approaches present feasible and effective opportunities for reducing the incidence of uncontrolled blowdowns via SRV.

The following proposed modifications by the BWR Owners Group exist at Fermi 2:

a. Two-Stage Target Rock Valves

The use of two-stage Target Rock valves at Fermi 2, as compared to the plants with 3-stage Target Rock valves, reduces the spurious blowdown events by 40 to 60 percent

b. Low-Low Set Relief

Fermi 2 is equipped with a "low-low set" design feature that changes the setpoints of selected SRVs following the initial opening of a number of SRVs.

This ensures that following the initial pressurization, the pressure will be relieved by the low-low set valve alone, and the remaining SRVs will not experience any subsequent actuations. The purpose of low-low set at Fermi is to mitigate postulated loads caused by a second (after initial) opening of an SRV. However, the low-low set will also serve to reduce the frequency of SORV events.

According to the BWR Owners Group evaluation, these existing modifications at Fermi 2 are equivalent to a reduction in SRV challenges by a factor of almost 10 (Table 5.1, Reference 14).

In addition to these proposed modifications, Edison further reduced the SORV frequency by lowering the RPV water level isolation setpoint for MSIV closure from Level 2 to Level 1 and lowering the pressure setpoint for MSIV closure. This results in reduced SRV challenges by eliminating isolation cycling of the SRVs resulting from transients such as feedwater controller failure, trip of both recirculation pumps, and loss of feedwater flow.

The two-stage Target Rock valves and low-low set relief feature plus lowering the RPV water level isolation setpoint for MSIV closure from Level 2 to Level 1 and lowering the pressure setpoint for MSIV closure reduce the SORV frequency by a factor of more than 10 and meet the requirement of NUREG-0737, Item II.K.3.16.

#### 5.2.2.6 Qualification Tests

##### 5.2.2.6.1 Inspection and Testing

In November and December of 1976, GE performed a qualification/ life-cycle test program on one valve of the 2-stage pilot operated design. The program consisted of 300 valve cycles (150 manual and 150 pressure-induced operations). The objective of the life-cycle test was to verify the ability of the design to meet the requirements for (1) set pressure, (2) opening and closing response time, (3) blowdown, (4) seat tightness, and (5) achievement of flow-rated capacity lift (ASME). These tests were performed at reactor conditions, using a test facility that had the capability of providing full steam flow through the SRV when it opened. During the course of the test program, it was noted that the delay time on opening was erratic, and the pressure difference between the setpoint and reclosure was not large enough. All other performance parameters were acceptable, even at the extremes of low and high pressurization rates and the extremes of ambient temperatures. The same valve was operated another 150 cycles to identify the causes of the observed anomalies.

Minor design improvements were made to the 2-stage pilot operated valve design as a result of these tests, although the valve was functionally acceptable.

Because there had been design changes, a new qualification test program was begun in late 1977 by Target Rock Corporation. The program consisted of 300 cycles on one valve, and 60 cycles on each of three additional valves. These tests were completed satisfactorily. The tests showed that the valves produced consistently repeatable setpoint pressure operation, consistent delay times of less than, or equal to, 400 msec, and consistent reclosure $\Delta P$'s for a given back pressure.

Additional tests were performed to provide the data necessary for final selection of the seal bellows area. Note that these tests and the final selection comprise a "fine-tuning" improvement of a thoroughly tested and qualified valve.

The electric-pneumatic actuator assembly was subjected to a qualification aging test that consisted of (1) a reference frame test to determine leakage, response time, and solenoid electrical characteristics for subsequent comparison; (2) radiation aging to a cumulative radiation dose of 19.6 x $10^6$ rads; (3) a reference frame test for the postradiation condition; (4) mechanical aging of 8000 cycles under normal ambient conditions of 150°F at 100 percent relative humidity; (5) thermal aging to 285°F at 100 percent relative humidity for 480 hr in air; (6) a reference frame test for the post-aging condition; (7) a simulated LOCA environment; (8) a reference frame test for the post-LOCA condition; (9) an accident radiation exposure of 13 x $10^6$ rads; and (10) a final reference frame test. The qualification aging test established that the actuator assembly was compatible with the service environments.

In parallel with the latter part of the above testing, a seismic qualification test was performed consisting of a valve mounted on a shake table subjected to biaxial vibration, with statically applied moment loads at the valve flanges. The test program consisted of (1) resonant frequency determination, (2) nozzle loading, (3) a simulated operating-basis earthquake (OBE), (4) an SSE, and (5) reference frame tests. The valve was operated under reactor conditions using a restricted steam flow arrangement.

The qualification test results: (1) verified that the SRV design will be operable and is structurally sound under the various normal and abnormal environmental and dynamic conditions to which the valve may be subjected in service; (2) established the basis for confirming the installed and qualified life of the valve; and (3) provided information necessary to enhance the established quality assurance program to ensure that new valves are equivalent to the qualified design.

The vessel overpressure protection analysis in Subsection 5.2.2.3 shows that the peak vessel (bottom) pressure for the limiting MSIV transient with high-flux scram and position trip scram is less than the 1375 psig allowed by the ASME code. The cycle-specific results of the vessel overpressure protection analysis are reported in the cycle-specific Supplemental Reload Licensing Report. The deviation of setpoints by a common-mode failure after installation is highly unlikely because of the qualification and the established quality assurance program previously discussed. However, even considering the possibility of setpoint drift, the peak pressure for the limiting operational transient will still be less than the ASME Code limit.

In addition, in response to comments from the NRC on operation of relief valves during abnormal transients, Edison, together with the BWR Owners Group, undertook a special SRV testing program reported in Reference 1. The results of the BWR Owners Group evaluation indicated that there is one event and single-failure combination that would lead to the discharge of liquid from the SRVs. This event and single-failure combination leads to the alternative shutdown mode of operation that uses the SRVs as a return flow path for low-pressure liquid to the suppression pool. The evaluation demonstrated that all other events postulated to produce liquid or two-phase SRV flow, including events under high-pressure conditions, are either of sufficiently low probability or that consequences are concluded to be acceptable. As such, no testing is needed for these events.

The BWR Owners Group testing program included the testing of typical SRVs for BWR/2 through BWR/6 plants to demonstrate the ability to perform satisfactorily under the condition in which low-pressure (i.e., up to 250 ± 20 psig) water passes through the valve instead of saturated steam. This corresponds to conditions expected during the alternate shutdown cooling mode; that is, the mode in which low-pressure pumps are injecting cold water into the reactor vessel and this water is vented through the SRVs back to the suppression pool. A plant-specific evaluation (Reference 2) of the test data correlated the generic program test conditions to the alternate shutdown cooling mode conditions for Fermi 2.

For Fermi 2, the alternate shutdown cooling mode of passing water through the SRVs to the suppression pool is not an anticipated operating condition. The Fermi 2 design includes a parallel flow path (see section 5.5.7.3) inside containment for shutdown cooling employing a normally closed, remote manual isolation valve powered from the alternate division emergency power supply. In any case, the test results demonstrated that the Fermi 2 SRVs would be available and can accommodate adequate water passage for shutdown cooling in the extremely unlikely event that the normal shutdown cooling path and its backup are unavailable.

Also, Edison participates in a utility-sponsored performance evaluation program for SRVs.

5.2.2.6.2 Inservice Inspection and Testing

The following inservice test program is applied.

a. Fifty percent of the valves are to be removed from service and tested at least once per 18 months

b. The remaining 50 percent are to be tested at least once per 40 months.

The program for the in-place monitoring of valve performance is conducted by monitoring the discharge pipe thermocouples. Thermocouples, with continuous readouts, provide the signals that establish the leaktightness of the valve. In addition, a position monitoring system has been provided that meets the requirements of NUREG-0578.

The SRV inspection and overhaul program is developed from the manufacturer's recommendations to ensure the operability of these valves. The frequency of visual inspection and overhaul is in accordance with applicable ASME operating and maintenance standards for SRVs.

This testing and inspection will provide added confidence that the valves will operate reliably, and that there are no deficiencies that could cause them to function, in service, in an unsafe manner.

The SRVs are tested in accordance with Quality Control (QC) procedures to detect defects and to prove operability prior to installation. The following tests are conducted:

a. Hydrostatic test at ANSI-specified test conditions

b. Pneumatic seat leakage test at 90 percent of set pressure, with maximum permitted leakage of 30 bubbles per minute emitting from a 0.250-in.-diameter tube submerged 0.5 in. below a water surface, or an equivalent test using an approved test medium

c. Set pressure test with valve pressurized with saturated steam, or other approved test medium with the pressure rising to the valve set pressure

d. Response time test with each SRV tested to demonstrate acceptable response time.

The valves are installed as received from the factory. The setpoints are adjusted, verified, and indicated on the valves by the vendor. Specified manual and automatic actuation relief mode of each SRV pilot is verified during the Preoperational Test Program.

It is not feasible to test the SRV setpoints while the valves are in place or during normal plant operation. The valves are mounted on 6-in.-diameter, 1500-lb primary service rating flanges. They are removed for maintenance or bench checks and reinstalled during inspection periods.

The external surface and seating surface of all SRVs are 100 percent visually inspected when the valves are removed for maintenance or bench checks.

The SRV inspection and overhaul program is developed from the manufacturer's recommendations to ensure the operability of these valves. The frequency of visual inspection and overhaul will be in accordance with applicable ASME operating and maintenance standards for SRVs.

The improbable failure of the relief mode function of this valve will not cause failure of the safety mode function of the valve, and vice versa.

The automatic depressurization capability of the ADS is evaluated in Subsections 6.3.2 and 7.3.1.

### 5.2.2.7 Routing of Nuclear Pressure Relief System Valves to Torus

The NPRS valves could discharge to the drywell without exceeding drywell design conditions. However, such a discharge would cause undesirable high temperature and high moisture transients on drywell equipment. Consequently, all valves are routed to the torus with discharge below the water.

A separate discharge line is provided for each of the 15 valves. The isometric of one typical line is shown in Figure 5.2-3. The lines do not penetrate containment; they are routed to the torus through the drywell-to-torus vent lines. Inside the torus, they penetrate the vent line and terminate in a T-quencher. Details of a typical line inside the torus are shown in Figures 5.2-4 through 5.2-7.

The portions of the lines inside the drywell and the torus are designed and classified as Quality Group B,* Category I, QA Level I. The discharge lines are made of Schedule 80, seamless carbon steel pipe; joints are butt welded with a backing ring. Each line is equipped with an 8-in. vacuum breaker. The T-quenchers are designed and classified as Quality Group C, Category I, QA Level I.

The lines have been sized to be nonlimiting on flow; i.e., the back pressure at the relief valve is well below that which restricts the capacity of the valve. The lines are l0-in. nominal size in the drywell, 12-in. in the vents and torus. The discharge line supports are designed to handle the maximum reaction load. In addition, the supports in the torus are designed to accommodate the hydrodynamic loading conditions that occur during accident events. The evaluation is documented in Reference 2.

NOTE: * The portions of the lines in the vent line were originally installed as Quality Group D. These portions of the lines have been upgraded to include the requirements of Quality Group B components and are classified as Quality Group D+, Category I, QA Level I.

#### 5.2.2.8 Pressure Isolation Valves

There are several safety systems connected to the RCPB that have design pressures below the rated reactor coolant system (RCS) pressure. There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suctions below RCS pressure. To protect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high-pressure RCS and the low-pressure systems. The leaktight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low-pressure systems, thus causing an inter-system LOCA.

### 5.2.3 General Material Considerations

#### 5.2.3.1 Material Specifications

The principal pressure-retaining materials and the appropriate material specifications for the RCPB components are listed in Table 5.2-6.

#### 5.2.3.2 Compatibility With Reactor Coolant

The construction materials exposed to the reactor coolant are

a. Solution-annealed austenitic stainless steels (both wrought and cast) types 304, 304L, 316, and 316L

b. Nickel base alloys, Inconel 600 and Inconel X750

c. Carbon steel and low alloy pressure vessel steel

d. Some 400 series martensitic stainless steel, all tempered at a minimum of 1100°F

e. Colmonoy and Stellite hardfacing materials.

f. Precipitation hardenable stainless steel material, XM-13.

All of these construction materials are resistant to stress corrosion in the BWR coolant. General corrosion on all materials except carbon and low alloy steel is negligible. Conservative corrosion allowances are provided for all exposed surfaces of carbon or low alloy steels.

Contaminants in the reactor coolant are controlled to very low limits by the reactor water quality specifications. No detrimental effects will occur on any of the materials from allowable contaminant levels in the high purity reactor coolant.

#### 5.2.3.2.1 Steps To Minimize Stress Corrosion Cracking

In September 1974, cracking was experienced in the stainless steel piping at Dresden Nuclear Power Station Unit 2. This was the first of a series of incidents of intergranular stress corrosion cracking (IGSCC) that occurred in BWRs. The cracking occurred in weld heat-affected zones in type 304 stainless steel recirculation bypass piping systems and core spray lines.

In May 1984, during a recirculation piping system replacement at the Pilgrim Station, IGSCC was discovered and confirmed in the Inconel 182 butter welds for recirculation piping RPV nozzles. This was the first of several instances documenting IGSCC in Inconel buttering which was not directly attributed to resin intrusions or other causes. With the issuance of NUREG 0313, Revision 2, and NRC Generic Letter 88-01, Inconel 182 has been removed from the list of materials which were considered resistant to IGSCC (NUREG 0313, Revision 2. par. 2.1.1). Since most reactor pressure vessel (RPV) nozzles were "buttered" with Inconel 182 prior to welding the "safe-ends" to the nozzles and the nozzles to safe-end welds were made using Inconel 182 filler metal, these welds were reclassified as "susceptible" to IGSCC.

As a result of these incidents, studies were undertaken by the NRC, GE, and Electric Power Research Institute (EPRI). These studies have shown that such cracking is caused by a combination of the presence of significant amounts of oxygen in the coolant, high stresses, and some sensitization of metal adjacent to welds. Such cracks are not expected to occur outside the heat-affected zones adjacent to welds, provided that the pipe material is annealed properly.

Pipe runs containing stagnant or low-velocity fluids have been observed to be more susceptible during plant operation to stress corrosion cracking than pipes containing a continuously flowing fluid. Historically, these cracks have been identified either by volumetric examination, by leak detection systems, or by visual inspection. Because of the inherent high material toughness of austenitic stainless steel piping, stress corrosion cracking is unlikely to cause a rapidly propagating failure resulting in a design-basis LOCA.

Although the probability is extremely low that these stress corrosion cracks will propagate far enough to create a significant safety hazard to the public, the presence of such cracks is undesirable. Steps have been taken to minimize stress corrosion cracking in Fermi 2 piping systems, to eliminate this condition, and to improve overall plant reliability. The various mitigating programs used at Fermi 2 to minimize the potential for IGSCC fall into three major categories: (a) induction heating stress improvement (IHSI), (b) solution annealing, and (c) Mechanical Stress Improvement Process (MSIP).

The countermeasures using solution annealing are expected to remain effective for the life of the plant, since no sensitized material will be exposed to reactor water at these welds.

The IHSI treatment is also expected to remain effective for the life of the plant, since it was implemented prior to operation. Plants in Japan have been operating for approximately 5 years after having performed IHSI. Edison will monitor the applicable performance of these plants and will make adjustments accordingly.

The MSIP treatment results in the stress reversal at the weld root and is a permanent "life-of-plant" mitigation method.

The specific actions taken to minimize the potential for IGSCC are addressed in Subsections 5.2.3.2.1.1 through 5.2.3.2.1.5.

5.2.3.2.1.1 Piping Modifications

Operating experience has shown that the line most susceptible to IGSCC is the recirculation pump discharge valve bypass line. General Electric has developed operating procedures that do not require the use of this line, thereby enabling the line to be removed from the system. The 4-in. sweepolets in the 28-in. recirculation pipe are closed with caps clad with type 308L stainless steel. The design and installation of the caps include incorporation of geometries necessary for inservice UT examinations.

The other line susceptible to IGSCC cracking is the reactor core spray line. The initial design of this line for Fermi 2 specified carbon steel with a short stainless steel transition piece connected to the RPV stainless steel safe-end. This transition piece has been changed to carbon steel; the safe-end has been changed to Inconel with a carbon steel extension piece. Much of the IGSCC research done by GE concerned the recirculation system. This system is exposed to reactor coolant and is fabricated of type 304 stainless steel. Much of this system is 28-in. and 22-in. pipe. On the basis of GE studies, residual stress levels in welds in this pipe were thought to be below the threshold to develop IGSCC. To further reduce residual stress levels at field welds, special welding procedures were adopted that reduced the weld heat input to 50,000 joules per inch and which prohibited weld bead straightening. In addition, special restrictions were placed on internal grinding. To minimize susceptibility of the weld metal to IGSCC, the weld metal should contain at least 8 percent ferrite.

The GE studies show welds in 12-in. pipe in the recirculation system risers to be much closer to the IGSCC threshold. To minimize IGSCC susceptibility of these pieces, they were returned to the shop for solution annealing and for application of a nonsusceptible inlay to the ends. The inlay extends beyond the heat-affected zone from field welds. Thus, no sensitized 12-in. pipe is exposed to reactor coolant.

5.2.3.2.1.2 Recirculation Inlet Nozzles

The recirculation inlet nozzle configuration for Fermi 2 is shown in Figure 5.2-8. The thermal sleeve is type 304 stainless steel; the weld buildup pad on the nozzle is type 308.

This configuration is different from the ones which have developed IGSCC.

a. The thickness of the pressure retaining boundary at the attachment is 4.751 in. on Fermi 2 versus 0.5 in.; therefore, stresses are very much lower

b. The pad material on Fermi 2 is type 308 stainless steel versus Inconel. Type 308 is basically not susceptible to IGSCC.

Compared to the configuration that developed IGSCC, the lower stress and decreased vulnerability of the Fermi 2 configuration will greatly increase the time to IGSCC initiation (if any occurs at all) and slow the rate of growth if IGSCC is initiated.

The configuration of Fermi 2 recirculation line vessel nozzles is essentially the same as that on five other operating plants: Millstone, Pilgrim, Cooper, FitzPatrick, and Hatch 1.

The safe-end welds are scheduled to be examined as part of the ASME Section XI Inservice Inspection Program. In addition, welds selected in accordance with the rules of Section XI will receive an increased frequency of examination commensurate with the requirements of NUREG-0313 (Revision 2) and Generic Letter 88-01, or other NRC approved alternative program.

5.2.3.2.1.3 Induction Heating Stress Improvement

Operating experience has shown that many BWR plants have had problems with IGSCC in large-diameter recirculation system piping. To minimize the likelihood of IGSCC in portions of the recirculation system piping that had not received IGSCC remedies, IHSI was performed during July 1983. Induction heating stress improvement is recommended by both GE and EPRI as an effective IGSCC countermeasure, especially for plants under construction.

On completion of IHSI, only four welds in the recirculation system piping did not receive some IGSCC countermeasure. These welds have been included in the inservice inspection program and will be inspected on the inspection cycle detailed in NUREG-0313, Revision 2, and Generic Letter 88-01, or other NRC approved alternative program.

5.2.3.2.1.4 Mechanical Stress Improvement Process

During the first refueling outage, the Mechanical Stress Improvement Process (MSIP) was applied to twenty-one (21) reclassified RPV nozzle and safe-end welds, four (4) welds not treated by IHSI, and two (2) bi-metallic welds in the reactor water clean-up system, which, due to changes in the NUREG 0313, Revision 2, susceptibility criteria, were re-evaluated as IGSCC susceptible. On completion of the MSIP treatment of these twenty-seven welds, all ASME Section III welds which were evaluated as IGSCC susceptible have had an IGSCC mitigation method applied. All of the IGSCC susceptible welds have been included in the inservice inspection program and will be inspected on the inspection cycle detailed in NUREG 0313, Revision 2, and Generic Letter 88-01, or other NRC approved alternative program.

5.2.3.2.1.5 Control Rod Drive System Modifications

Some BWR plants have experienced IGSCC in the collet retainer tube in their CRDs. General Electric has attributed this cracking to thermal cycles during hot scrams, followed by exposure to oxygenated CRD cooling water that is aggressive to sensitized material.

The program adopted by Fermi 2 is consistent with GE recommendations. It consists of the following three parts:

a. An augmented surveillance and inspection program

b. Modification of CRD operations to eliminate unnecessary thermal cycling

c. Modification of the CRD water supply to provide high-purity deaerated water to the CRD system during plant operation.

Specifically, the Fermi 2 program consists of the following actions:

a. Each rod not fully inserted will be tested to confirm operability by inserting one or more notches in accordance with the frequency specified in the Technical Specifications.

b. All CRDs removed for maintenance will have a dye penetrant examination of the outer surface of the collet retainer tube. The criteria established by GE in Service Information Letter (SIL) 139 will be used to decide rejection. The term collet retainer tube refers to a portion of the outer tube, and replacement of a rejected collet retainer tube requires a new cylinder, tube, and flange subassembly

c. A CRD with a high-temperature alarm will not be cooled by giving it repeated drive signals

d. The source of water for the CRD system has been changed to the condensate treatment system effluent with the condensate storage tank as backup. The new water source is very pure and of very low oxygen content. (See torus water management system, Subsection 9.2.8.)

e. A flowing sample line downstream of the drivewater filter has been installed to provide for conductivity and oxygen grab sample measurement.

The use of high-purity deaerated water affects a significant increase in the time to crack formation. General Electric believes the time to crack initiation in current CRD collet retainer tubes may be increased by a factor of 100 with this reduction in dissolved oxygen content.

5.2.3.2.1.6 <u>Inservice Inspection and Leak Detection</u>

NUREG-0313, Revision 2, and Generic Letter 88-01, January 1988, present the technical bases for the NRC staff positions on materials, processes, and primary coolant chemistry to minimize and control IGSCC problems. Inspection schedules are comparable to those specified in Section XI of the ASME Boiler and Pressure Vessel Code in cases where the piping material is IGSCC resistant.

The modifications discussed in the previous subsections significantly reduce susceptibility to IGSCC. As detailed in Generic Letter 88-01, inspection schedules and inspection sample sizes are based on the susceptibility of weldments to initiation and propagation of IGSCC. Varying amounts of augmented inspections are specified for piping, with a greater susceptibility to cracking.

All applicable welds at Fermi 2 have been evaluated and classified according to the requirements of NUREG 0313, Revision 2, and Generic Letter 88-01. As required selected welds are included in the ASME Section XI Inservice Inspection Program.

The leak detection capability on Fermi 2 discussed in Subsection 5.2.7.3 is consistent with the 5 gpm rate discussed in NUREG-0313, Revision 2. As stated in Subsection 5.2.7.3, the unidentified leakage rate limit is established to allow time for corrective action before the nuclear system process barrier can be significantly compromised.

#### 5.2.3.2.2 Steps To Maintain Occupational Exposure As Low As Reasonably Achievable

Steps taken in the selection of material to minimize and control the buildup, transport, and deposition of activated corrosion products in the reactor coolant and auxiliary systems follow:

The primary coolant system consists primarily of carbon steel (very low nickel and cobalt content), except for the use of austenitic stainless steel (in the recirculation loops) and low alloy steel. The nickel content of these materials is low and is controlled in accordance with the applicable ASME material specifications. Because the cobalt in steel usually appears as a small-percentage component of the nickel (usually, 2 percent of the nickel), the amount of cobalt in the primary system components is also very low.

A small amount of nickel base material (Inconel 600) is used in the RPV internals. Inconel 600 is required where components are attached to the reactor vessel shell, and the coefficient of expansion must match the thermal expansion characteristics of the low alloy vessel steel. Inconel 600 was selected because it provides the proper thermal expansion characteristics and adequate corrosion resistance, and can be fabricated and welded readily.

Hardfaced and wear-resistant materials having a high percentage of cobalt were restricted to applications in which no satisfactory alternative materials were available at the time of construction.

### 5.2.3.3 Compatibility With External Insulation and Environmental Atmosphere

The RCPB is insulated with an all-metal (stainless steel and aluminum) reflective-type insulation in compliance with Regulatory Guide l.36. This type of insulation does not contain any silica, fluorides, or chlorides. It does not contribute to surface contamination, and it has no effect on the stainless steel components of the RCPB. The insulation is designed to perform its intended function throughout the expected life of Fermi 2.

### 5.2.3.4 Chemistry of Reactor Coolant

The coolant chemistry requirements discussed in this subsection are consistent with the requirements of Regulatory Guide 1.56.

Reactor water chemistry limits are established to provide an environment favorable to materials in contact with the water. Limits are placed on conductivity and chloride concentrations. Conductivity is limited because it can be continuously and reliably measured by an in-line conductivity cell and gives an indication of abnormal conditions and the presence of potentially detrimental constituents in the coolant. Chloride limits are specified to minimize the potential of stress corrosion cracking of stainless steel. The accuracy of the conductivity cell is verified once per week by radiation chemistry personnel.

Several investigations have shown that in neutral solutions some oxygen is required to cause stress corrosion cracking of stainless steel, while in the absence of oxygen no cracking occurs. One of these is the chloride-oxygen relationship of Williams (Reference 3), where it is shown that at high chloride concentration little oxygen is required to cause stress corrosion cracking of stainless steel and at high oxygen concentration little chloride is required to cause cracking. These measurements were determined in a wetting and drying situation using alkaline-phosphate-treated boiler water and, therefore, are of limited significance to BWR conditions. They are, however, a qualitative indication of trends.

The water quality requirements are further supported by GE stress corrosion test data, summarized as follows:

a. Type 304 stainless steel specimens were exposed in a flowing loop operating at 537°F. The water contained 1.5 ppm chloride and 1.2 ppm oxygen at pH 7. Test specimens were bent beam strips stressed over their yield strength. After 2100 hr exposure, no cracking or failures occurred

b. Welded type 304 stainless steel specimens were exposed in a refreshed autoclave operating at 550°F. The water contained 0.5 ppm chloride and 1.5 ppm oxygen at pH 7. Uniaxial tensile test specimens were stressed at 125 percent of their 550°F yield strength. No cracking or failures occurred at 15,000 hr exposure.

Zirconium alloys and Inconel alloys are highly resistant to chloride stress corrosion cracking failure.

When conductivity is in its normal range, pH, chloride and other impurities affecting conductivity will also be within their normal range. When conductivity becomes abnormal, chloride measurements are made to determine whether or not they are also out of their normal operating values. This would not necessarily be the case. Conductivity could be high due to the presence of a neutral salt which would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are high because of the purposeful use of additives. In BWRs, however, where few additives are used and where near-neutral pH is maintained, conductivity provides a good and prompt measure of the quality of the reactor water. Significant changes in conductivity provide the operator with a warning mechanism so he can investigate and remedy the condition before reactor water limits are reached. Methods available to the operator for correcting the off-standard condition include operation of the reactor cleanup system in the blowdown mode, reducing the input of impurities, and placing the reactor in the cold-shutdown condition. The major benefit of cold shutdown is to reduce the temperature-dependent corrosion rates and to provide time for the RWCS to reestablish the purity of the reactor coolant.

Zinc is added to the reactor water, via the feedwater system, to control radiation buildup on out-of-core primary coolant piping. The amount of zinc that will be added to the reactor water will increase the conductivity of the reactor water. This will not impact the use of conductivity as a good and prompt measure of the quality of the reactor water. The increases above the new equilibrium conductivity value can still be used as an indicator of impurities entering the reactor. The zinc added can be accounted for in overall conductivity of the reactor water.

The conductivity and dissolved oxygen levels of the reactor coolant are continuously monitored. The samples of the coolant which are taken periodically serve as a reference for calibration of these monitors and are considered adequate to ensure accurate readings of the monitors. If conductivity is within its normal range, chlorides and other impurities will also be within their normal ranges.

The relationship of chloride concentration to specific conductance measured at 25°C for chloride compounds such as sodium chloride and hydrochloric acid can be calculated (Reference 4). Values for these compounds essentially bracket values of other common chloride salts or mixtures at the same chloride concentration. Surveillance requirements are based on these relationships. The sampling frequency when reactor water has a low specific conductance is adequate for calibration and routine audit purposes. When specific conductance increases, and higher chloride concentrations are possible, or when continuous conductivity monitoring is unavailable, increased sampling is provided.

Chloride analysis of the reactor coolant is performed as required or at least daily on grab samples. Approved radiation chemistry section procedures, using methods such as specific ion electrode or titration, are used to determine the chloride concentration.

The reactor water quality for plant design and operational control when operating at rated power is:

a. Conductivity - ≤1.0 μmho/cm at 25°C

b. Chlorides (as $Cl^-$) - ≤200 ppb

c. pH - 5.6 to 8.6 at 25°C.

Reactor water quality in excess of the limits specified above is limited to 72 hrs for any instance. Exceeding the maximum limits specified below shall be cause for shutdown and cool down to ambient temperatures until the water is within the quality limits specified above:

a. Conductivity - 10 μmho/cm at 25°C

b. Chlorides (as $Cl^-$) - 0.5 ppm

Reactor water quality is also limited based on time in excess of operational limits on conductivity and chlorides.

a. Time above 1 umho/cm - 2 weeks per 12-month period

b. Time above 200 ppb ($Cl^-$) - 2 weeks per 12-month period

The addition of zinc will add to the dissolved metals, total metals, and conductivity in the reactor water. The zinc will provide the beneficial outcome of controlling radiation build-up on out-of-core surfaces; however, overall metals concentration will still be maintained within the fuel warranty limits to ensure no impact on fuel performance. The amount of conductivity of the added zinc is much less than the 1 μS/Cm operating conductivity limits.

See Subsection 10.4.6 for further details.

## 5.2.4 Fracture Toughness

### 5.2.4.1 Compliance With Code Requirements

The ferritic pressure boundary material of the RPVs was qualified by impact testing in accordance with the 1968 edition of Section III of the ASME Code, with addenda to and including summer 1969 addenda. From an operational standpoint, the minimum temperature limits for pressurization are used as the basis for compliance with the 1968 Edition of the ASME Code Section III. (The minimum temperature limits for pressurization are defined by the summer 1972 addenda, Appendix G, Protection Against Nonductile Failure.)

### 5.2.4.2 Compliance With 10 CFR 50, Appendix G

#### 5.2.4.2.1 Introduction

Versions of 10 CFR 50, Appendix G, prior to the 1983 edition had specific requirements for the preparation and testing of all reactor coolant pressure boundary materials. In lieu of these specific requirements, the present version of Appendix G requires that for a reactor vessel which was constructed in conformance with an ASME Code Section III earlier than the summer 1972 addenda of the 1971 edition, the fracture toughness data and data analyses must be supplemented in a manner approved by the Director, Office of Nuclear Reactor Regulation, to demonstrate equivalence with the present fracture requirements of Appendix G. The Fermi 2 reactor vessel was constructed in compliance with an ASME Code earlier than the summer 1972 addenda of the 1971 edition. The NRC has stated in Supplement 1 to NUREG 0798, the Fermi 2 Safety Evaluation Report, that the alternative methods proposed by Fermi 2 to demonstrate compliance with Appendix G has been reviewed, evaluated, and found to provide the safety margin required by Appendix G. Accordingly, Fermi 2 has supplied sufficient information to demonstrate equivalency with the fracture toughness requirements of the present version of 10 CFR 50, Appendix G, (1983 as amended November 1986 and October 1988).

A major condition necessary for full compliance with 10 CFR 50, Appendix G prior to the 1983 edition is satisfying the requirements of the summer 1972 addenda to Section III of the ASME Code. This is not possible with components that were purchased in accordance with earlier Code requirements.

Ferritic material complying with 10 CFR 50, Appendix G, must have both drop-weight tests and Charpy V-Notch (CVN) tests with the CVN specimens oriented transversely to the maximum material working direction to establish the $RT_{NDT}$. The CVN tests must be evaluated against both absorbed-energy and lateral-expansion criteria. The maximum acceptable $RT_{NDT}$ must be determined in accordance with the analytical procedures of ASME Code Section III, Appendix G. Appendix G of 10 CFR 50 requires a minimum of 75-ft-lb upper shelf CVN energy for beltline material. It also requires at least 45-ft-lb CVN energy and 25 mils lateral expansion for bolting material at either the preload or lowest service temperature, whichever is lower.

By comparison, material for the Fermi 2 reactor vessel was qualified by either drop-weight tests or longitudinally oriented CVN tests (both not required), confirming that the material nil-ductility transition temperature (NDTT) is at least 60°F below the lowest service temperature. When the CVN test was applied, a 30-ft-lb energy level was used in defining the NDTT. There was no upper shelf CVN energy requirement of the Fermi unit beltline material. The bolting material was qualified to a 30 ft-lb CVN energy requirement at 60°F below the minimum preload temperature.

From the previous comparison, it can be seen that the fracture toughness testing performed on the reactor vessel material cannot be shown to comply with 10 CFR 50, Appendix G; however, to determine operating limits in accordance with 10 CFR 50, Appendix G, estimates of the beltline material $RT_{NDT}$ and the highest $RT_{NDT}$ of all other material were made, as explained in Subsection 5.2.4.2.3. The method for developing these operating limits is also described therein.

#### 5.2.4.2.2 Method of Compliance

A detailed description of compliance with 10 CFR 50 Appendix G is included in General Electric Report 004N8586, Reference 21. The 1998 Edition of the ASME Boiler and Pressure Vessel code, including 2000 Addenda, was used in this evaluation. The P-T curve methodology includes the following: 1) the use of $K_{IC}$ from Figure A-4200-1 of Appendix A to determine T-$RT_{NDT}$, and 2) the use of the $M_m$ calculation in the ASME Code paragraph G-2214.1 for a postulated defect normal to the direction of maximum stress. NRC approved methodology was utilized as detailed in NEDC-33178P-A, Reference 26.

The pressure-temperature (P-T) curves are established to the requirements of 10 CFR 50, Appendix G to assure that brittle fracture of the reactor vessel is prevented. Part of the analysis involved in developing the P-T curves is to account for irradiation embrittlement effects in the core region, or beltline. The method used to account for irradiation embrittlement is described in Regulatory Guide 1.99, Rev. 2.

The beltline region in the Fermi Unit 2 vessel includes a thickness discontinuity between the lower and lower-intermediate shells. In addition to beltline considerations, there are non-beltline discontinuity limits such as nozzles, penetrations, and flanges that influence the construction of P-T curves. The non-beltline limits are based on generic analyses that are adjusted to the maximum reference temperature of nil ductility transition ($RT_{NDT}$) or the applicable Fermi 2 vessel components.

#### 5.2.4.2.3 Method of Obtaining Operating Limits Based on Fracture Toughness

Operating limits that define minimum reactor-vessel metal temperatures versus reactor pressure during normal heatup, cooldown, inservice hydrostatic testing, and anticipated operational occurrences were initially established using the methods of Appendix G of Section III of the ASME B&PV Code, 1971 Edition.

Updated Operating limits that define minimum reactor-vessel metal temperatures versus reactor pressure during normal heatup, cooldown, inservice hydrostatic testing, and anticipated operational occurrences were established using the methods of Appendix G of Section III of the ASME B&PV Code, 1998 Edition (including 2000 Addenda). This later edition of the Code is discussed in section 5.2.4.2.2.

Weld material toughness test coupons were made with the exact same weld filler metal and procedure as for the actual vessel weld. However, these weld deposits were not necessarily made on the exact same heat of baseplate as in the vessel. Baseplate of the same specification was used for this purpose. This small difference in baseplate would not affect the testing of the weld metal since the Charpy specimen would be in the weld metal. Toughness testing of the exact baseplates in the vessel was done separately. As part of the BWRVIP Integrated Surveillance Program (ISP), materials irradiated in other vessels were utilized to provide verification of material properties as detailed in section 5.2.4.4. This information was utilized in the development of the pressure temperature curves per General Electric Report 004N8586 (Reference 21), and as shown in figures for 52 EFPY contained in the Pressure and Temperature Limits Report (PTLR) (Reference 25).

5.2.4.2.4 Temperature Limits for Inservice Inspection Hydrostatic or Leak Pressure Tests

The fracture toughness analysis for system pressure tests resulted in the curve labeled A shown in the figures contained in the PTLR (Reference 25). The beltline materials are less limiting even at end-of-service fluence levels, based on evaluation according to Regulatory Guide 1.99, Revision 2, where the predicted shift in the $RT_{NDT}$ (based on the neutron fluence at 1/4 of the vessel wall thickness) has been added to the beltline curve to account for the effect of neutron embrittlement as detailed in Reference 21.

5.2.4.2.5 Temperature Limits for Boltup

The flanges and adjacent shell are required to be warmed to minimum temperatures of 72°F before they are stressed by the full intended bolt preload as shown on the figures contained in the PTLR (Reference 25).

5.2.4.3 Operating Limits During Heatup, Cooldown, and Core Operation

The fracture toughness analysis was done for the normal heatup or cooldown rate of 100°F/hr. The temperature gradients and thermal stress effects corresponding to this rate were included. The results of the analyses are a set of operating limits for non-nuclear heatup or cooldown shown as Curves labelled B in the PTLR (Reference 25). Curves labelled C in the PTLR (Reference 25), apply whenever the core is critical. The basis for curves labelled C is described in 10 CFR 50, Appendix G, January 1990 Edition, Paragraph IV.A.3.

5.2.4.4 Surveillance Programs for the Reactor Pressure Vessel

A surveillance program will be carried out to monitor the neutron radiation effects on the RPV base metal, the weld HAZ metal, and the weld metal from a steel joint that simulates a welded joint in the RPV beltline. Versions of 10 CFR 50, Appendix H, prior to the 1983 edition required that the surveillance program conducted prior to the first capsule withdrawal comply with the 1973 edition of ASTM E185. The present version of Appendix H requires that the surveillance program conducted prior to the first capsule withdrawal comply with the requirements of the edition of ASTM E185 that was current with respect to the ASME Code to which the reactor vessel was purchased.

The Fermi 2 surveillance program was shown to comply with the revised requirements of 10 CFR 50, Appendix H (1983 as amended November 1986 and October 1988).

Subsequent to development of the Fermi plant specific surveillance program, the BWR Vessel and Internals Project (BWRVIP) developed an integrated surveillance program to comply with the requirements of 10 CFR 50 Appendix H, Paragraph III.C. "Requirements for an Integrated Surveillance Program." No capsules from the Fermi 2 vessel are currently required to be withdrawn or tested as part of the BWRVIP Integrated Surveillance Program (ISP). Capsules from other plants have been removed, and specimens were tested in accordance with the ISP implementation plan. The results from these tests have provided the necessary data to monitor embrittlement of the Fermi 2 vessel as documented in Reference 21. A description of the BWRVIP ISP and its application to Fermi is contained in Section 5.2.4.4.3.

5.2.4.4.1 Original Program Content

The original Fermi program consisted of three baskets, each containing tensile and CVN specimens hermetically sealed in an inert gas environment in thin-wall austenitic stainless steel capsules. The capsules are not buoyant and thus present no handling problems. The three baskets have been placed near core midplane adjacent to the RPV wall where the neutron flux and temperature will simulate that of the RPV wall. The three baskets contain test specimens made from the original RPV beltline material in accordance with the requirements of ASTM E185-73. In total, the program consists of 108 impact and 22 tensile specimens. In addition, there are 51 impact and 18 tensile baseline and spare specimens. The specimens include the following.

a. Base metal impact, transverse and longitudinal

b. Weld metal impact

c. HAZ impact

d. Base metal tensile

e. Weld metal tensile

f. HAZ tensile.

The following general statements apply to these specimens:

a. Base metal impact and tensile specimens are taken from the 1/4 T planes of the specimen plate

b. HAZ impact and tensile specimens are all oriented parallel to the rolling direction

c. Weld metal impact specimens are all transverse to the axis of the weld; tensile specimens are parallel. The fracture areas consist of all weld metal.

Details of the manufacture of these specimens are given in Reference 7.

The specimens were taken from two plates trimmed from the lower intermediate shell section of the reactor vessel. The plate sections for the base material specimens were given a simulated stress relief for 40 hr at 1150°F to ensure that they represent the metallurgical condition of the lower intermediate shell plates of the reactor vessel after final fabrication. The plate sections for the weld and HAZ specimens were joined with a continuous central weld identical to the reactor vessel longitudinal weld. The welded plate was then given a simulated stress relief for 40 hr at 1150°F, similar to the base material plate. The weld was X-rayed to ensure quality; no repair to the weld was allowed by the specifications.

The surveillance specimens were not taken from alongside the ASME NB-2300 specimens. This was not considered critical, since they are just as representative of the material in the vessel as the NB-2300 specimens. The actual specimens in each capsule and capsule locations are the following.

| | Tensile | Charpy V-Notch |
|---|---|---|
| Capsule 3<br>(azimuth 300°) | 2 BM, long.<br>2 WM<br>2 HAZ | 12 BM, longitude<br>12 WM<br>12 HAZ |
| Capsule 2<br>(azimuth 120°) | 3 BM, long.<br>3 WM<br>2 HAZ | 12 BM, longitude<br>12 WM<br>12 HAZ |
| Capsule 1<br>(azimuth 30°) | 3 BM, long.<br>2 WM<br>3 HAZ | 12 BM, transverse<br>12 WM<br>12 HAZ |

Each capsule includes an iron, nickel, and copper flux wire. A separate neutron dosimeter was attached at azimuth 30° and contains three copper and three iron flux wires at the Capsule 1 location. The separate capsule was removed from the reactor during the first refueling outage and tested in 1990.

Capsule 3 was removed from the vessel at 8.1 Effective Full Power Years. Testing of this capsule was deferred due to the ongoing development of the BWRVIP Integrated Surveillance Program.

The attachment method of the capsules is in accordance with GE drawing 922D218. The assembly is attached to mounting brackets (upper and lower), and a bolt at approximately the center of the assembly can be adjusted to secure the holder firmly against the top and bottom brackets.

The lead factor is the relationship between the measured flux/ fluence at the surveillance sample and the peak flux/fluence at the inside surface of the vessel wall. This relationship has two variations. One variation is the axial variation from the elevation of the surveillance sample to the elevation peak flux.

The second variation is the variation of the flux as a function of angle from a position adjacent to the surveillance sample to the position of the peak flux.

The lead factor for the capsule calculated with respect to the inside surface location is the ratio of the flux greater than 1 MeV at the surveillance sample, divided by the flux greater than 1 MeV at the point of greatest flux in the vessel. For Fermi 2 this value is 0.90 as detailed in Table 4.3-2.

The peak fluence at one-quarter thickness was calculated from the peak inside surface fluence using the methods of Regulatory Guide 1.99, Revision 2. The peak inside surface fluence was predicted by an ‘absolute’ fluence calculation compliant with Regulatory Guide 1.190.

5.2.4.4.2 Withdrawal Schedule

The withdrawal schedule of the three sets of specimens in the reactor is planned as follows.

a. The first set was withdrawn at 8.1 EFPY which was approximately 25 percent of the original licensed reactor service life (i.e., 40 years) and remains onsite untested.

b. The second set will be a standby.

c. The third set will be a standby.

5.2.4.4.3 Description of BWRVIP Integrated Surveillance Program

A 1997 NRC review of a surveillance capsule report identified that a licensee lacked adequate unirradiated baseline Charpy V-notch (CVN) data for materials in their RPV surveillance program. This lack of baseline data could inhibit the ability to effectively monitor changes in the RPV fracture toughness properties as required per 10 CFR 50 Appendix G. Subsequent discussions between the NRC and the BWRVIP identified several plants (including Fermi 2) that potentially lacked adequate unirradiated baseline CVN data for materials in their plant specific RPV surveillance programs.

Subsequent to this concern, the BWRVIP developed a BWR RPV Integrated Surveillance Program (ISP) to meet the requirements of 10 CFR 50 Appendix H Paragraph III.C. This effort resulted in development of reports BWRVIP-78 and BWRVIP-86 (as amended by responses to NRC RAIs), that were submitted to the NRC for review and approval (References 15 through 18). The NRC approved these reports by issuing NRC Safety Evaluation as an attachment to NRC letter to Carl Terry dated February 1, 2002 (Reference 19).

BWRVIP-78 describes the technical basis related to material selection and testing for the ISP. The report defines the methodology utilized to identify existing plant specific surveillance capsules and surveillance capsules from the Supplemental Surveillance Program (SSP) required for the ISP. Required surveillance materials are those that best represent the actual limiting plate and weld materials from which BWR RPVs are fabricated. BWRVIP-78 establishes the connection between the required surveillance materials and the specific BWR RPV plate or weld materials which they represent and provide a test matrix for the ISP.

BWRVIP-86 establishes specific guidelines for ISP implementation. It addresses surveillance capsule withdrawal and testing dates, information dealing with ISP project administration, information on neutron fluence determination, information on data utilization and sharing, and information on licensing aspects of ISP implementation. The BWRVIP issued BWRVIP-86-A (Reference 20) to incorporate NRC Requests for Additional Information (RAI), industry responses to RAIs and to include a copy of the NRC Safety Evaluation accepting the ISP Program.

BWRVIP Report BWRVIP-135, Reference 22 provides a detailed discussion of the analysis performed by the BWRVIP (ISP) of irradiated material samples representative of the Fermi 2 reactor pressure vessel assembly. This information was utilized in the development of the new pressure-temperature curves that are detailed in Section 5.2.4.

The NRC has approved use of the BWRVIP ISP as an acceptable alternative to a plant specific RPV surveillance program; with two conditions. First, that licensees submit a license amendment requesting NRC approval of their participation in the ISP. Second, that BWRs commit to utilizing an acceptable neutron fluence calculation methodology. Section 4.3.2.8 provides information dealing with Fermi 2 neutron fluence calculation methodology. The NRC has approved the Fermi 2 participation in the ISP per License Amendment No. 152.

#### 5.2.4.5 Reactor Vessel Annealing

In-place annealing of the reactor vessel because of radiation embrittlement should not become necessary because the predicted EOL value of adjusted reference temperature will not exceed 200°F and the EOL upper shelf energy should remain above 50 ft-lb.

### 5.2.5 Austenitic Stainless Steel

#### 5.2.5.1 Cleaning and Contamination Protection Procedures

During fabrication, the stainless steel surfaces were cleaned by mechanical methods (grinding, brushing with stainless steel brushes, machining), solvent cleaners, or chemical cleaning agents. Caustic cleaners and other solvents and cleaners containing halogens, sulfides, or other harmful constituents were not used for cleaning parts that contain crevices or entrapment areas.

Stainless steel materials were not pickled unless they were in the solution heat-treated condition. Stainless steel components were suitably packaged and protected during shipment, storage, and construction, to prevent contamination from potentially corrosive agents.

Immediately prior to hydrostatic testing of the reactor vessel, all interior surfaces that would contact water during the hydrostatic test, all nozzle fixtures, all piping to be used to fill the vessel, and all external surfaces of stainless and nickel-chrome-iron components were cleaned of all halogen-bearing soils, grease, oil, penetrant materials, inks, chalk or crayon marks, and all dirt and debris. Testing and operation of components and systems were performed using either inhibited water or high-purity demineralized water to avoid exposure to detrimental contaminants.

All loose dirt and other foreign materials were removed by sweeping or vacuuming. Deposits of grease and oil were removed with an approved solvent. Tightly adhering soils were removed with the aid of stainless steel brushes or by grinding. The vessel interior was then cleaned with high-pressure water containing corrosion inhibiting additives. The vessel and water temperatures were less than 180°F during the cleaning step. The water pressure was a minimum of 6000 psi. Water was potable, containing less than 25 ppm chlorides, 10 ppm fluorides, and l ppm sulfides.

The cleanliness of the vessel was checked visually and with the aid of an ultraviolet light to ensure that the vessel is clean. The ultraviolet examination was conducted under darkened conditions with a lamp providing a minimum intensity of 100 foot candles. All fluorescent materials were removed from the surface.

All plumbing, welding, or testing work was performed prior to cleaning. During any entry of personnel into the vessel after cleaning was completed, shoe covers were worn and clean conditions were maintained in the reactor vessel.

#### 5.2.5.2 Solution Heat Treatment Requirements

Solution heat treatment of austenitic stainless steel consisted of heating the material to 1950 +/- 50ºF, holding for 1/2 hr per inch of thickness (minimum 1/2 hr), and quenching in water to below 800°F. Stainless steel castings may have been heated to 2050°F maximum prior to quenching. Nickel-chrome-iron alloys that may have been subjected to temperatures in excess of 1700°F exclusive of welding were rechecked for grain size for information and specified mechanical properties for acceptance and reported to the buyer.

#### 5.2.5.3 Material Inspection Program

The raw material inspection program used to verify that the unstabilized austenitic stainless steels were properly solution heat-treated and not susceptible to intergranular attack is as follows.

a. No testing was required if valid documentation was furnished proving that the stainless steel had been given a suitable water quench from a temperature above l800°F, and that no subsequent heating had been employed

b. If documentation to verify adequate water quenching was not available, the material was required to be tested in accordance with ASTM A-262 Practice E.

#### 5.2.5.4 Unstabilized Austenitic Stainless Steels

The nonstabilized grades of austenitic stainless steels with a carbon content greater than 0.03 percent used for RCPBs are types 304 and 316.

#### 5.2.5.5 Avoidance of Sensitization

##### 5.2.5.5.1 Base Metal

Wrought and cast austenitic stainless steels used for the RPV system (except for RPV cladding) were supplied in the solution heat-treated condition and thereafter were not subjected to any heating above 800°F except for welding, IHSI, or re-solution heat treatment.

Sensitization of wrought austenitic stainless steel was avoided for piping and RCPB pumps and valves. Austenitic stainless steel was considered to be furnace-sensitized if it had been heated by means other than welding within the range of 800°F to 1800°F, regardless of subsequent cooling rate. Such stainless steel was required to either pass the requirements of ASTM A-262 Practice E or be re-solution heat-treated. When heated above 1800°F, the austenitic stainless steel was required to be rapidly cooled through the range 1800°F to below 800°F by agitated water quench to produce an acceptable grain structure. Since severe sensitization of austenitic stainless steel was to be avoided, AISI type 304 and type 316 (0.08 percent maximum carbon) materials were used. Where severe sensitization could not be avoided, such as for parts that were required to be hard surfaced, low carbon AISI type 304 cast material was used.

5.2.5.5.2 Welding Controls

During stainless steel welding, the interpass temperature is controlled to a maximum of 350°F. Weld layers are built up uniformly along the joint and across the width of the joint. Block welding is not permitted and weld stops and starts are staggered. Welds are cleaned free of slag, flux, and other foreign material prior to depositing subsequent beads.

Austenitic weld materials are selected and controlled to produce welds that contain a minimum of 3 percent ferrite. Ferrite content is determined by one of the following methods.

a. Actual chemical analysis compared to the Schaeffler and Schoefer

b. Magne-gage

c. Metallography

d. Severn-gage.

The stainless steel components and systems for which stainless steel welding was controlled by GE or Dravo, Inc., include the following.

a. RRS

b. CRD hydraulic return

c. CRD housing to flange

d. RCIC system (suction from condensate storage).

The GE equipment was ordered, fabricated, and, in most cases, delivered prior to the issuance of Regulatory Guide 1.31. Therefore, there was no test program specifically directed toward the inspection of welds for delta ferrite. However, the welds were made by long-established procedures that included control of ferrite content of filler materials and had proved adequate for consistently producing satisfactory welds without evidence of fissuring. General Electric BWR 4/5/6 Standard Safety Analysis Report, Subsection 5.2.3.4.2.1, as amended in May 1978, provides an acceptable testing program for control of ferrite. The indicated testing program of welds on five BWRs was produced under the same procedures as the Fermi 2 equipment and fully demonstrated the presence of a minimum of 3 percent delta ferrite in the welds.

Similarly, stainless steel welds fabricated by Dravo were made with weld material having 5 to 15 percent delta ferrite. Inspection of welds made since the Fermi 2 piping was fabricated, but using the same procedures, has also consistently demonstrated the presence of a minimum of 3 percent delta ferrite.

The field pipe erection contractors were required to incorporate the requirements of Regulatory Guide 1.31 into their stainless steel weld procedures, including procedures for inspection of fabricated welds. See Subsection A.1.31 for conformance by Edison's and piping contractor's welding procedures with Regulatory Guide 1.31.

#### 5.2.5.6 Retesting Unstabilized Austenitic Stainless Steels Exposed to Sensitizing Temperatures

Welding procedures require control of heat input to avoid severe sensitization and susceptibility to intergranular attack. No retesting of "as-welded" unstabilized austenitic stainless steel is required or planned.

Unstabilized austenitic stainless steel subjected to heat in the range of 800°F to 1500°F by any means other than welding or IHSI is required to be retested in accordance with ASTM A-262, Practice E.

#### 5.2.5.7 Control of Delta Ferrite

The procedures and requirements that are used for the control of delta ferrite in austenitic stainless steel welds are discussed in Subsection 5.2.5.5.1. Additional information on delta ferrite in austenitic stainless steel weldments may be found in Subsections 5.2.3.2.1.1, 5.2.5.5.1, and A.1.31.

### 5.2.6 Pump Flywheels

Pumps with flywheels are not used in Fermi 2.

### 5.2.7 Reactor Coolant Pressure Boundary Leak Detection System

#### 5.2.7.1 Leak Detection Methods

##### 5.2.7.1.1 General

The RCPB leak detection system consists of temperature, pressure, flow, and fission product sensors with associated instrumentation and alarms. This system detects and annunciates abnormal leakage in the following systems:

a. Main steam lines

b. RWCU system

c. RHR system

d. RCIC system

e. Reactor feedwater system

f. HPCI system

g. Reactor recirculation system.

A summary of isolation and/or alarm of affected systems and the methods used appear in Table 5.2-11.

Small leaks are generally detected by temperature and pressure changes, fillup rate of drain sumps, and fission product concentration inside the primary containment. Large leaks are also detected by changes in reactor water level and changes in flow rates in process lines.

Leakage into systems that are directly or indirectly connected to the RCPB is detected by the leak detection system (LDS). The RHR system service water, general service water, and reactor building closed cooling water (RBCCW) have been provided with process radiation monitors for the detection of intersystem leaks.

Leakage into systems that are normally connected to the RCPB through closed isolation valves is detected by pressure and temperature indications. The core spray, RCIC, and HPCI systems are in this category. Leakage into the RWCU system is detected by differential flow and temperature devices. The standby liquid control system (SLCS) is monitored for intersystem leakage by the system pressure and tank level indicators provided.

5.2.7.1.2 Detection of Abnormal Leakage Within the Primary Containment

Leaks within the primary containment are detected by monitoring for

a. Abnormally high pressure and temperature within the primary containment
b. Sump pump frequency of operation on floor and equipment drains
c. A decrease in the RPV water level
d. Hydrogen and oxygen concentration
e. High flow rate in process lines
f. High gaseous radiation levels in the primary containment atmosphere
g. Floor drain sump level rate of change.

Temperatures within the primary containment are monitored at various elevations. Excessive temperature in the primary containment, increased drain sump flow, and increased fission product radiation level are annunciated by alarms in the main control room. Low RPV water level and high drywell pressure are annunciated by alarm in the main control room and cause automatic isolation of the containment. In addition, low RPV water level isolates the main steam lines.

The systems within the drywell share a common area; therefore, their LDSs are common. Each LDS inside the drywell is designed with a capability of detecting leakage less than established leakage rate limits.

5.2.7.1.3 Leak Detection

5.2.7.1.3.1 General

The drywell floor drain sump measurement system monitors the normal design leakage collected in the floor drain sump consisting of leakage from the CRDs, valve flange leakage, closed cooling water, air cooler drains, and any leakage not connected to the equipment drain sump.

The design includes a supplementary drywell floor drain level monitor to enhance the leak detection capability of the drywell floor drain sump system. A continuous analog level measurement of the drywell floor drain level is provided to meet the sensitivity requirement of Regulatory Guide 1.45. This sump level monitor provides a rate-of-change measurement, which is qualified seismically and has the sensitivity to detect a 1-gpm leak integrated over a 1-hr interval.

The drywell equipment drain sump level monitors identify leakage collected in the equipment drain sump. The sump receives condensate drainage from pump seal leakoff. Collection in excess of background leakage would indicate reactor coolant leakage. The equipment drain sump temperature is also monitored. High temperature would indicate leakage of high temperature water.

Four basic leak detection methods are used to determine sump collection rates. As the water in each of the floor or equipment drain sumps is pumped out, the flow is metered by a flow integrator. Level switches are used to set fill time and pump-out time periods using adjustable reset timing devices. If the nominal pumping out or filling time for the particular sump is exceeded, an alarm is generated in the control room. In addition, if both pumps are started to handle the flow into the sump, an alarm is generated. The drywell sump sensitivity is 21 gal/in. of level. The sumps are located at the lowest elevation of the drywell area, and there are no areas that can act as a temporary reservoir.

The level switches can be functionally checked during plant operation by manually controlling the pumps. The operators use careful monitoring of the flow integrators and the actual pumping times to verify the operating condition of the level switches.

The primary containment is maintained at a slightly positive pressure during reactor operation. The pressure fluctuates slightly as a result of barometric pressure changes and outleakage. A pressure rise above the normally indicated values indicates the presence of a leak within the drywell.

The primary containment cooling system recirculates the primary containment atmosphere through heat exchangers (air coolers) to maintain the primary containment at its average operating temperature as given in the Technical Specifications. The RBCCW system provides cooling water to the air coolers. An increase in primary containment atmosphere temperature would increase the temperature rise in the cooling water passing through the coils of the air coolers. Thus, the RBCCW temperature difference increase between inlet and outlet to the air coolers indicates the presence of a reactor coolant or steam leakage. Also, a drywell ambient temperature rise above normal indicates the presence of reactor coolant or steam leakage.

The drywell cooler units have been provided to maintain the ambient drywell temperature at a relatively low value, and steam leaks will be condensed by contact with the relatively cold surfaces in the drywell. If the steam finds its way to the cooler units, condensation will definitely occur. The drains from the coolers are collected in the drywell sumps and can then be detected via the leak detection scheme. It is expected that the normal operating humidity will be at or near saturation, which will promote rapid condensation and subsequent detection. In addition, the airborne gaseous sampling system monitors the airspace and detects leaks in a very timely manner.

Radiation monitoring of the primary containment is provided as required by Criterion 30 of 10 CFR 50, Appendix A, and Regulatory Guide 1.45. The primary containment radiation monitoring system is part of the redundant LDS. The primary containment radiation monitoring system information is used in conjunction with the drywell floor drain sump level indicating system. It is provided to improve the total drywell LDS diversity and sensitivity.

However, since the supplementary drywell floor drain level monitor is seismically qualified, and meets the sensitivity requirement of Regulatory Guide 1.45, the particulate channel of the containment radiation monitor is not required as a leak detection system and has been removed from the leak detection system, but the gaseous monitor was retained to meet diversity requirements.

The design basis for the primary containment radiation monitoring system, and the associated instrumentation are presented in Subsections 7.1.2.1.22 and 7.6.1.12.1.

Additional components monitored are discussed below.

5.2.7.1.3.2 Reactor Pressure Vessel Head Seal

Leakage past the first of two RPV head closure seals is detected by monitoring the drain line connected to the region between the seals. Leakage is collected in a small-volume, normally closed system that can be drained to the equipment drain sump. When the pressure in this volume increases, an alarm in the main control room is actuated.

5.2.7.1.3.3 Reactor Recirculation System Pump Seal

Reactor recirculation system pump seal leaks are detected by monitoring the drain line. Leakage is indicated by high-flow alarms in the main control room. Leakage is piped to the equipment drain sump, as shown in Figure 5.5-2.

5.2.7.1.3.4 Safety/Relief Valves

Safety/relief valve leakage is detected by monitoring the discharge path. High temperature is alarmed in the main control room.

5.2.7.1.4 Detection of Abnormal Leakage Outside the Primary Containment

Outside the primary containment, the piping within each system monitored for leakage is in compartments or rooms, separate from other systems where feasible, so that leakage may be detected by area temperature indications. Each LDS discussed below is designed to detect leak rates that are less than the established leakage limits. The method used to monitor for leakage for each RCPB component is shown in Table 5.2-11.

5.2.7.1.4.1 Room Ventilation or Standby Cooler Temperature

A differential temperature-sensing system is installed in each area containing equipment that is part of the nuclear system process barrier. These areas are the RCIC, HPCI, RHR, and RWCU systems equipment rooms, as well as the suppression chamber room and main steam line tunnel.

Temperature sensors are placed in the inlet and outlet ventilation ducts or ventilation air flow paths. Other sensors are installed in the equipment areas to monitor ambient temperature. A differential temperature switch between each set of sensors and/or ambient temperature switch initiates an alarm in the main control room when the temperature reaches a preset value. Remote readouts from temperature sensors are indicated in the relay room.

Due to the design characteristics of the reactor building ventilation design, the differential temperature isolation provides an alarm function only on the RCIC and HPCI areas. Similarly, the temperature sensors for the torus subbasement area provide an alarm function only and do not trip either the RCIC or HPCI systems. The HPCI and RCIC trip function is provided by the (redundant) HPCI and RCIC area ambient sensors.

5.2.7.1.4.2 Reactor Building Sump Flow Measurement

Monitors indicate the amount of leakage into the reactor building floor drainage system. The normal design leakage collected in the system consists of leakage from the RWCU, FPCCS, RCIC, HPCI, core spray, CRD, RHR, feedwater, and main steam systems and from other miscellaneous vents and drains.

5.2.7.1.4.3 Visual and Audible Inspection

Accessible areas are inspected periodically. The temperature and flow indicators discussed above are monitored regularly. Any instrument indication of abnormal leakage is investigated.

5.2.7.1.4.4 Differential Flow Measurement for Reactor Water Cleanup System Only

Because of the RWCU system arrangement, differential flow measurement provides an accurate leak detection method. The flow from the RPV is compared with the flow back to the RPV. An alarm in the main control room and an isolation signal are initiated when higher flow out of the RPV indicates that a leak greater than the established leak rate limit may exist.

5.2.7.2 Indication in Main Control Room

Details of the LDS indications are included in Subsection 7.6.1.8.

5.2.7.3 Limits for Reactor Coolant Leakage

5.2.7.3.1 Total Leakage Rate

The total leakage rate consists of all leakage, identified and unidentified, that flows to the drywell floor drain and equipment drain sumps. The total leakage rate limit is well within the makeup capability of the RCIC system. The total leakage rate limit is established at 25 gpm.

The total leakage rate limit is also set low enough to prevent overflow of the drywell sumps. The equipment drain sump and the floor drain sump, which collect all leakage, are each drained by two 50-gpm pumps. The total leakage rate limit for each sump of 25 gpm is set below the removal capacity of one pump in each sump because of the possibility that most of the total leakage could flow into one sump.

#### 5.2.7.3.2 Identified Leakage

The pump packing glands, valve stems, and other seals in systems that are part of the nuclear system process barrier and from which a normal design leakage of 20 gpm is expected are provided with drains or auxiliary sealing systems. Nuclear steam supply system valves and pumps inside the drywell are equipped with double seals and packings.

Leakage from the primary RRS pump seals is piped to the equipment drain sump. Leakage from the main steam line SRVs is identified by temperature sensors that transmit to the main control room. Any temperature increase above the drywell ambient temperature detected by these sensors indicates valve leakage. Leakage from the RPV head flange gasket is detected by a pressure switch, as described in Subsection 5.2.7.1.3.2.

Thus, the leakage rates from pumps and valve seals are measurable during plant operation. These leakage rates, plus any other leakage rates measured while the drywell is open, are defined as identified leakage rates.

### 5.2.7.4 Unidentified Leakage

#### 5.2.7.4.1 Unidentified-Leakage Rate

The unidentified-leakage rate is the portion of the total leak-age rate received in the drywell sumps that is not identified as previously described. A threat of significant compromise to the nuclear system process barrier exists if the barrier contains a crack that is large enough to propagate rapidly. The unidentified-leakage rate limit must be low because the unidentified leakage might be emitted from a single crack in the nuclear system process barrier.

An allowance is made for normal plant operation leakage that does not compromise barrier integrity and is not identifiable. The unidentified-leakage rate limit is established at a 5-gpm rate to allow time for corrective action before the nuclear system process barrier could be significantly compromised. This proposed limit is based on a calculated flow from a critical crack in a primary containment system pipe.

#### 5.2.7.4.2 Sensitivity and Response Times

Sensitivity, including sensitivity tests and response time of the LDS, is presented in Subsection 7.6.1.8.

#### 5.2.7.4.3 Length of Through-Wall Flaw

Experiments conducted by GE and Battelle Memorial Institute (BMI) permit an analysis of critical crack size and crack opening displacement. This analysis relates to axially oriented through-wall cracks.

5.2.7.4.3.1 Critical Crack Length

Both the GE and the BMI test results indicate that theoretical fracture mechanics formulas do not predict critical crack length. However, satisfactory empirical expressions may be developed to fit test results. A simple equation that fits the data in the range of normal design stresses for carbon steel pipe is:

$$\ell_C = \frac{15000D}{\sigma h} \tag{5.2-1}$$

where

| | | |
|---|---|---|
| $\ell_C$ | = | critical crack length (inches) |
| D | = | mean pipe diameter (inches) |
| $\sigma h$ | = | nominal hoop stress (psi) |

Data correlation for Equation 5.2-1 is shown in Figure 5.2-11.

5.2.7.4.3.2 Crack Opening Displacement

The elasticity theory predicts a crack opening displacement of

$$W = \frac{2\ell\sigma}{E} \tag{5.2-2}$$

where

| | | |
|---|---|---|
| $\ell$ | = | crack length |
| $\sigma$ | = | applied nominal stress |
| E | = | Young's modulus |

Measurements of crack opening displacement made by BMI show that local yielding greatly increases the crack opening displacement as the applied stress approaches the failure stress $\sigma_f$. A suitable correction factor for elasticity effects is:

$$C = \frac{\pi}{2}\frac{\sigma}{\sigma_f} \tag{5.2-3}$$

The crack opening area is given by

$$A = C\frac{\pi}{4}W = \frac{\pi}{2}\frac{\sigma}{\sigma_f}\frac{\pi\ell\sigma}{2E} \tag{5.2-4}$$

For a given crack length $\ell$, $\sigma_f = 15{,}000\,{}^{D}/_{\ell}$

5.2.7.4.3.3 Leakage Flow Rate

The maximum flow rate for blowdown of saturated water at 1000 psi is 55 lb/sec/in.$^2$, and for saturated steam the rate is l4.6 lb/sec/in.$^2$. Friction in the flow passage reduces this rate, but for cracks leaking at 5 gpm (0.7 lb/sec), the effect of friction is small. The required leak size for a 5-gpm flow is:

| | | | |
|---|---|---|---|
| a. | A | = | 0.0126 in.$^2$ (saturated water) |
| b. | A | = | 0.0475 in.$^2$ (saturated steam). |

From this mathematical model, the critical crack length and the 5-gpm crack length have been calculated for representative BWR pipe sizes (Schedule 80) and pressure (1050 psi). Results are tabulated as follows.

| Normal Pipe Size (Sch. 80) (in.) | Average Wall Thickness (in.) | Steam Line Crack Length $\ell$ (in.) | Water Line Crack Length $\ell$ (in.) |
|---|---|---|---|
| 4 | 0.337 | 7.2 | 4.9 |
| 12 | 0.687 | 8.5 | 4.8 |
| 24 | 1.218 | 8.6 | 4.6 |

The ratios of crack length ($\ell$) to the critical crack length ($\ell_c$) as a function of nominal pipe size are

| Nominal Pipe Size (Sch. 80) (in.) | Ratio $\ell/\ell$C | |
|---|---|---|
| | Steam Line | Water Line |
| 4 | 0.745 | 0.510 |
| 12 | 0.432 | 0.243 |
| 24 | 0.247 | 0.132 |

It is important to recognize that the failure of ductile piping with a long through-wall crack is characterized by large crack opening displacements that precede unstable rupture. Judging from observed crack behavior in the GE and BMI experimental programs involving both circumferential and axial cracks, it is estimated that leak rates of hundreds of gallons per minute will precede crack instability. Measured crack opening displacements for the BMI experiments were in the range of 0.1 to 0.2 in. at the time of incipient rupture, corresponding to leaks of the order of 1 in.$^2$ in size for plain carbon steel piping. For austenitic stainless steel piping, even larger leaks are expected to precede crack instability, although there are insufficient data to permit quantitative prediction.

The results given are for a longitudinally oriented flaw at normal operating hoop stress. A circumferentially oriented flaw could be subjected to stress as high as the 550°F yield stress, assuming high thermal expansion stresses exist. A good mathematical model that is supported by test data is not available for the circumferential crack. Therefore, it is assumed that the longitudinal crack, subject to a stress as high as 30,000 psi, approaches "worst case" with regard to leak rate versus critical size relationships. Given the same stress level, differences between the circumferential and longitudinal orientations are not expected to be significant in this comparison.

Figure 5.2-12 shows general relationships among crack length, leak rate, stress, and line size, using the mathematical model described above. The asterisks denote conditions at which the crack opening displacement is 0.1 in., at which time instability is imminent. This provides a realistic estimate of the leak rate to be expected from a crack of critical size. In every case, the leak rate from a crack of critical size is greater than the 5-gpm criterion.

5.2.7.4.4 Margins of Safety

The margins of safety for a detectable flow to assume critical size are presented in Subsection 5.2.7.4.3. Figure 5.2-12 shows general relationships among crack length, leak rate, stress, and line size obtained using the mathematical model.

5.2.7.4.5 Criteria To Evaluate the Adequacy and Margin of the Leak Detection System

For process lines that are normally open, there are at least two different methods of detecting abnormal leakage from each system within the nuclear system process barrier located in the primary containment and reactor building (Table 5.2-11). The instrumentation can be set to provide alarms at established leakage rate limits and isolate an affected system when necessary. The alarm points are determined analytically or, where appropriate, are based on measurements of appropriate parameters made during startup and preoperational tests.

The unidentified leakage rate limit is based, with an adequate margin for contingencies, on the crack size large enough to propagate rapidly. The established limit is sufficiently low so that, even if the entire unidentified leakage rate were coming from a single crack in the nuclear system process barrier, corrective action could be taken before the integrity of the barrier would be threatened.

The LDS is able to satisfactorily detect unidentified leakage of 5 gpm.

Sensitivity, including sensitivity tests and response time of the LDS, is included in Subsection 7.6.1.8. Subsection 7.1.2 presents the criteria for shutdown when the leakage limits are exceeded.

5.2.7.5 Maximum Allowable Total Leakage

The total leakage rate is presented in Subsection 5.2.7.3.1.

5.2.7.6 Differentiation Between Identified and Unidentified Leaks

Subsection 5.2.7.1 describes the systems that are monitored by the LDS. The ability of the LDS to differentiate between identified and unidentified leakage is discussed in Subsection 7.6.1.8 and Subsections 5.2.7.1 through 5.2.7.4.

5.2.7.7 Sensitivity and Operability Tests

Testability of the LDS is discussed in Subsection 7.6.1.8.

5.2.7.8 Leakage Reduction Program

Edison has developed a leakage reduction program to reduce and maintain leakage to as-low-as-practical levels from systems outside the primary containment that could or would contain highly radioactive fluids during and after a serious transient or accident. In addition, the program is designed to reduce potential paths due to design and/or operation deficiencies. This program is based on Requirement 2.1.6a of NUREG-0578 (Reference 8) and the requirements of Item III.D.1.1 of NUREG-0660, NUREG-0694, and NUREG-0737, "Primary Coolant Sources Outside Containment" (References 9, 10, and 11, respectively).

Table 5.2-12 identifies systems included in the leakage reduction program. Table 5.2-13 lists systems to which the leakage reduction program is not applicable and further provides the justification for their exclusion. Only the systems listed in Table 5.2-12 are included in the program.

#### 5.2.7.8.1 Program Description

The Edison leakage reduction program includes the following features.

a. A combination of periodic visual inspections of accessible portions of the systems and detailed system walkdowns to identify leakage into the secondary containment out of components such as valve stems, pump seals, fittings, relief valve discharge lines, drains, vents, and instrument loops. When possible, these inspections are performed with the systems at approximately operating pressure in a normal or test condition

b. An aggressive maintenance program to correct identified leakage problems and assign a high priority to leakage-related work requests for systems in this program. Essentially all leakage of concerned (i.e., those identified in Table 5.2-12) systems will be addressed. These preventive and corrective maintenance measures ensure minimum leakage on a continuing basis

c. Periodic leak-rate testing of systems listed in Table 5.2-12 and system components such as valves at intervals not to exceed each refueling outage. The general test methods used to determine leakage from systems within the scope of this leakage reduction program are provided in Subsection 5.2.7.8.2

d. Maintenance of records on inspections and tests to identify chronic or generic leakage problems to implement modifications and/or corrective maintenance measures. A summary report on program effectiveness will be provided to plant management within 90 days of the conclusion of each reactor refueling.

In addition to the testing program, system leakage tests will be performed on many of these systems as part of the 10 CFR 50, Appendix J, leakage testing program. The systems and components subject to this testing and that form part of the containment boundary are identified in Table 6.2-2. This leakage reduction program will be completed by the time Fermi 2 reaches full-power operation.

Prior to the start of the second fuel cycle, Edison will revise the general criteria to the extent necessary according to the experience gained during the first operating cycle of Fermi 2. These revised criteria will be used as the basis for the long-term leakage reduction/monitoring program for Fermi 2.

#### 5.2.7.8.2 Test Methods

The following methods are used to test systems identified in Table 5.2-12 for leakage:

a. Liquid systems - Systems or portions of systems that could contain radioactive liquids during or after an accident are periodically placed into normal operation or a testing mode. During these test conditions, the systems are visually inspected for leakage with all results being recorded. All leakage detected during the periodic visual inspections, or the less frequent integrated leak-rate test, will be measured where possible and recorded. Techniques used for leakage measurement will include collection into a graduated container and estimation by equating drops per unit of time to a standard volume

b. Gaseous systems - For systems or portions of systems that may contain radioactive gases during or after an accident, a pressure drop or makeup gas rate test is used. Clean air or nitrogen is used for these tests. When leakage is indicated by a pressure drop or excessive makeup, visual inspection techniques are applied to components during pressurization.

   Gaseous systems are tested by pressurizing the system with air or nitrogen to a specified pressure (usually accident pressure of 56.5 psig or the system relief valve set pressure) and measuring to within 20 standard $cm^3$ per minute the flow required to maintain test pressure using a local leak-rate test panel. The makeup flow is equivalent to the system leakage rate. This method of leak testing is similar to that required by 10 CFR 50, Appendix J, for leak-rate testing of the primary containment. If flow is detected, each system component will be tested with a soapy liquid in accordance with the procedure to identify sources of leakage. Corrective action will be taken as warranted to reduce the leakage from each source, and the system will be retested to yield a quantitative indication of the leakage reduction achieved. This measuring methodology, leakage source identification procedure, and corrective action will ensure that leakage is reduced to the lowest practical level, as dictated by system hardware limitations. The application of the helium leak detection method of inspection may be considered for some gaseous systems.

5.2.7.8.3 Test Procedures

Each system identified in Table 5.2-12 has surveillance testing procedures. These test procedures contain the following elements as applicable:

a. A description of system and plant operating conditions necessary to conduct each leak test. Test boundaries are identified and include only those portions of the system that could contain radioactive fluids during or after an accident. For example, the core spray suction piping from the condensate storage tank would not be inspected as this suction line is used for test purposes only and would not contain radioactive fluid during or after an accident

b. Elaboration of special test methods necessary to supplement general test methods

c. Data sheets listing the specific areas to be inspected. The data sheets will identify isometric drawing numbers and provide spaces to record inspection results.

## 5.2.8 Inservice Inspection Program

### 5.2.8.1 Inservice Inspection Program for Class 1, 2, and 3 Components

The inservice inspection (ISI) program for Class 1, 2, and 3 components complies to the extent practicable with the requirements of the ASME Code Section XI. The program for the initial inspection interval complies with the requirements of the 1980 edition of the Code including the winter 1981 addenda except that the extent of examination for Class 2 piping welds will be determined by the 1974 edition, summer 1975 addenda. The initial 10-year inspection interval commenced with the start of commercial operation. When compliance with ASME Code Section XI was impracticable, relief was requested from the NRC in compliance with 10 CFR 50.55a(g)(5)(iii). The Fermi 2 inservice inspection program plan for the initial 10-year inspection interval was submitted to the NRC for review and was found to be acceptable and in compliance with the provisions of 10 CFR 50.55a(g)(4). The first ten year interval was completed February 16, 2000. Upon completion of the first inspection interval, the inservice inspection program was updated to include later Editions and Addenda of ASME Section XI as required by 10 CFR 50.55.a. Successive ten year updates will be similarly processed.

### 5.2.8.2 Provisions for Access to Reactor Coolant Pressure Boundary

Fermi 2 uses reflective metal insulation typical of that used by GE for this series of RCPB. The RCPB design has been reviewed in detail to ensure adequate access for inspection according to ASME B&PV Code Section XI, Articles IS 141 and 142. The insulation design has considered access for inservice inspection.

#### 5.2.8.2.1 Reactor Pressure Vessel Access Provisions

In the region of the sacrificial shield, there is a nominal 12-in. annulus between the insulation and the outside surface of the RPV. Access to this annulus can be gained from the bottom at locations adjacent to the support skirt to the lower head weld, and from two 3 x 3-1/2-ft openings, 8 ft from the top of the shield, at azimuths 180° and 351°. Inservice inspection of longitudinal and circumferential welds in the RPV will be performed using mechanized equipment.

Vessel nozzles are accessible for inservice examination through openings in the sacrificial shield. Automatic scanning devices enable complete inspection while minimizing personnel exposure.

The bottom head contains the penetrations for the CRD system and in-core flux monitoring system. The spacing between these penetrations makes volumetric inspection impractical. These nozzles are partial-penetration welds, and, typically, have not been included in normal ISI schedules as they meet exception criteria under a postulated CRD ejection accident.

#### 5.2.8.2.2 Piping Access Provisions

Insulation on Class 1 piping inside the primary containment is of the removable reflective-metal type. Removable nonmetallic insulation is used on the portion of Class 1 lines outside the primary containment. Welds requiring an ISI inspection have identification tags attached to the insulation covering each weld.

The preservice baseline examination of the ASME Class 1 piping has been performed in accordance with the ASME Code Section XI, 1974 edition, through summer 1975 addenda to the extent possible. The scope and extent of examination of Class 1 piping is in accordance with Table 5.2-14. The preservice inspection program identified all welds that have access limitations for examination. For all welds that cannot be examined ultrasonically, alternate means of examination were used (such as radiography, liquid penetrant or magnetic particle, supplemented by visual examination during hydrostatic testing). The preservice inspection program exempted from volumetric and surface examination certain portions of Class 1 piping in accordance with the provisions of IWB-1220(b)(1) and (2), "Component Connections, Piping and Associated Valves (and their supports) One Inch Nominal Pipe Size and Smaller." The exempt components were examined in accordance with IWA-5000 during the system hydrostatic pressure test required by IWB-5000.

In addition, limited space between the process and guard pipes in the primary containment penetrations makes it impractical to perform an ultrasonic examination of the process pipe-to-flued head weld.

The ASME Code incorporated inservice inspection requirements for Class 2 and 3 systems after most of the design and manufacture of these systems had been completed. In September 1976, Edison engaged SWRI to analyze the extent to which Fermi 2 could comply with these new sections of the ASME Code. The study was based on the latest edition of the ASME Code available which was the 1974 Edition, including addenda through summer 1976 and reported in Reference 12.

Reference 12 shows that the layout of these systems and the design of the system supports are such that welds and components requiring examination by the ASME Code are accessible with, basically, no exceptions. The examination of some welds is limited partially by the close proximity of fittings or lugs; a few welds have limited accessibility and can be inspected from only one side. All limitations were identified in the preservice inspection report.

#### 5.2.8.3 Equipment for Inservice Inspection

All equipment used for inservice inspection of the RPV and piping has been proven reliable on other preservice and inservice inspections. Included in this equipment are mechanized and manual inspection devices.

Pipe butt welds will be inspected using conventional ultrasonic inspection equipment. Basically, this is a light-weight, portable UT flaw detection equipment package with manually held search units.

#### 5.2.8.4 Mechanized Inspection Equipment

In general, the RPV will be ultrasonically examined by automated equipment. Typically, the data acquisition system contains a multichannel recorder, cathode ray tube (CRT) displays, a TV video camera, and a minicomputer.

The computer is the primary data recording/comparison system, and the other systems are intended for backup to be used when required.

Computer programs have been written to allow comparison of the data obtained on the subsequent inservice examinations.

#### 5.2.8.5 Reactor Pressure Vessel Acceptance Standards

The acceptance standards that were used to establish acceptability of the RPV for service during preservice mapping of the RPV by ultrasonic examination were those standards required by the ASME B&PV Code.

#### 5.2.8.6 Coordination of Inspection Equipment With Access Provisions

The access provisions are designed to accommodate currently available examination equipment. This equipment has been used successfully on other preservice and inservice inspections.

#### 5.2.8.7 Inservice Testing Program for Pumps and Valves

The testing program for pumps and valves complies to the extent practicable with the requirements of the Code and Addenda identified in 10 CFR 50.55a at the time the program is updated to the next 10 year interval. The scope of the program encompasses those pumps and valves necessary to safely shut down the plant or mitigate the consequences of an accident. The scope also includes those valves that perform an isolation function between high-pressure and low-pressure portions of systems connected to the reactor coolant system.

When compliance with Code requirements is impractical, relief is requested from the NRC in compliance with 10 CFR 50.55a.

Table 5.2-15 lists the valves that perform an isolation function between high-pressure and low-pressure portions of systems connected to the RCS. These pressure isolation valves are categorized as A or AC and are tested in compliance with Technical Specifications and the ASME Inservice Testing Code and Addenda applicable to the current ten year interval. The testing program for the valves, which is referenced in the Technical Specifications, consists of the following methods.

a. Exercise the valve and verify the position in accordance with the IST Program.

b. Exercise the valve (full stroke) and measure stroke time (as applicable) in accordance with the IST Program.

c. Leak test the valve seat before reaching power operation during refueling and after valve maintenance before the return to service, in accordance with the IST Program.

These valves shall not be routinely exercised every 3 months during plant operation (except E4100F005 and E5100F014, which are exercised during quarterly surveillance and then verified closed) as required by ASME Code because of the following:

a. Such tests remove one of the two barriers protecting the low-pressure portion of the emergency core cooling system (ECCS)

b. The operators on testable check valves cannot overcome the force on the valve with reactor pressure on one side.

Instead, the valves will be exercised during cold-shutdown periods as time permits (but not more frequently than once every 3 months).

A routine surveillance test every 3 months to exercise the valve presupposes that the test can be done with the plant operating at full power (and pressure). The purpose of dual barriers is to provide pressure isolation and protection even if one of the barriers should be faulty. Should one of the barriers be faulty by being inoperable, the core cooling systems have sufficient redundancy to perform their function. In addition, an inoperable barrier would be found during the proposed tests made at cold shutdown.

However, should one of the barriers be faulty by having excessive leakage, the core cooling system connected to that barrier could be severely damaged. Therefore, the test could cause a significant loss of primary coolant. On the other hand, had the test not been performed for this latter case, the core cooling system would have performed its function normally.

The full closure of these valves, except for the HPCI and RCIC check valves, is verified in the control room by direct monitoring position indicators. In addition, these lines are equipped with overpressure detection and protection devices should pressure isolation valves leak; these are summarized in Table 5.2-16, which shows that every line is protected by a relief valve and has pressure monitoring.

For the HPCI and RCIC system, pressure isolation is provided by normally closed gate valves, E4150F006 and E5150F013, and check valves E4100F005 and E5100F014, which are leak tested. E4150F007 and E5150F012 are normally open and not credited for pressure isolation.

If there is excessive leakage through the normally closed gate and check valves, the operator will be alerted by the high pump suction pressure alarm indicated in Table 5.2-16. The operator will then be directed to close the normally open gate valve per the Alarm Response Procedure.

The inservice testing program (IST) for pumps and valves for Fermi 2 commenced March 20, 1985. The first 10-year interval commenced following the initial start of Fermi 2 commercial operation in accordance with ASME Section XI, Paragraph IWA-2420, 1980 Edition including winter 1980 Addenda. The second and subsequent ten-year intervals will be updated to include later editions of the Code as required by 10 CFR 50.55a.

#### 5.2.8.8 Preservice Inspection Program

A preservice inspection program was performed on all Class 1 components (except the RPV) and other components noted in Table 5.2-14 in accordance with the requirements of the 1974 Edition of ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," with addenda through the summer of 1975 (74/S75). The preservice inspection program for some Class 1 components was performed in accordance with the requirements of ASME Section XI, 1980 Edition, winter 1981 addenda, for compatibility with the ISI program. These components are identified in the remarks column of Table 5.2-14. Preservice inspection of the reactor vessel was performed in accordance with the 1971 Edition of Section XI (reference Subsection 5.4.2).

Southwest Research Institute was engaged by Edison to be the inspection agent to perform preservice examinations of welds. Southwest Research Institute supplied inspectors, equipment, and procedures. The Hartford Steam Boiler Inspection and Insurance Company was engaged by Edison to be the Authorized Inspector.

In general, Table 5.2-14 outlines the preservice examination requirements for Class 1 components in accordance with Tables IWB-2500 and IWB-2600 of Section XI (IS-251 through IS-261 for the RPV).

Class 2 systems within the scope of the Section XI preservice inspection program are the following:

a. Residual heat removal, Division I and Division II
    1. ECC function in LPCI mode
    2. RHR function in RHR mode
    3. RHR function in containment spray mode
b. Core spray, ECC function
c. HPCI, ECC function
d. SLCS, up to the Class 1 boundary valve
e. Main steam system between the second and third isolation valves.

In accordance with 10 CFR 50.55(a), the 1974 Edition of ASME Section XI through the summer 1975 addenda was used for determining the extent of examination (the number of welds required to be examined) for Class 2 pipe welds in RHR systems, ECCSs, and containment heat removal systems. For all other Class 2 systems, either the 1974 Edition of Section XI through the summer 1975 addenda or the latest NRC-approved edition may be used. For consistency, the 1974 Edition through the summer 1975 addenda was used for determining the extent of the examination for the preservice inspection program for these other systems. This includes the head spray system and SLCS added to the Class 2 preservice inspection program.

The selection of the individual welds to be examined on each Class 2 system was based on the inspection philosophy identified in the 1980 Edition, winter 1981 addenda, of Section XI. The selection philosophy contained in the 1975 summer addenda is based on a random selection of welds and results in examining a particular weld only once in the plant's 40-year operational life. No trending of data is possible under the 1975 summer rules. The 1981 winter addenda identifies a selection philosophy that concentrates the examinations on those welds that historically have a greater probability of failure: namely, high-stress welds, welds at terminal ends, and dissimilar metal welds. In addition, the 1981 winter addenda requires examinations of the same welds in each 10-year interval so that meaningful data trending can be accomplished. There is general agreement in the industry that the 1981 winter addenda philosophy is superior to the random-selection approach identified in the 1975 summer addenda.

The criteria used for the selection of specific welds to be examined for the preservice inspection program were based on the following.

a. All high-stress welds defined as loading stresses greater than $0.8(1.2S_h + S_a)$ as per the 1981 winter addenda

b. All moderately stressed welds defined as loading stresses greater than $0.7(1.2S_h + S_a)$ and less than or equal to $0.8(1.2S_h + S_a)$. Inclusion of these moderately stressed welds in the Fermi 2, Class 2, preservice inspection program, is an added conservatism that exceeds the requirements of the ASME Section XI, 1980 Edition, winter 1981 addenda

c. All dissimilar metal welds

d. One terminal end of each type of terminal end within a system. (Note: This is a modified version of the ASME Section XI, 1980 Edition, winter 1981 addenda, Table IWC-2500-1, Category C-F, Footnote [1][b].) Edison has taken this approach to prevent skewing the weld examination sample to this particular type of weld. For example, the core spray system has four pumps, each with a terminal end at the suction and discharge attachment welds. To examine all eight terminal ends would be redundant. Therefore, to enable a more representative sample to be taken, only one pump suction terminal end weld and one pump discharge terminal end weld would be selected for examination

e. Additional random selections such that the total number of welds examined meets the number required by paragraph IWC-2411 of ASME Section XI, 1974 Edition, summer 1975 addenda.

Based on the above, Edison requested relief from two of the 1975 summer requirements for all the Class 2 system welds included in the preservice and inservice inspection programs. The first request for relief is to allow Edison to select those types of welds that historically have a higher probability of failure in lieu of the random-selection approach required by the 1975 summer addenda. The second request for relief was to allow repeated examination of the same welds in subsequent 10-year intervals in lieu of the requirements that different welds be inspected in each 10-year interval. This second relief request is applicable to the ISI-NDE program only.

The preservice inspection program delineated all required examinations, methods, code allowable exemptions, and relief requests. The preservice inspection program has been completed and is available for review by the NRC staff.

#### 5.2.8.9 Snubber Program

The examination and testing program for snubbers complies to the extent practicable with the requirements of the Code and Addenda identified in 10 CFR 50.55a at the time the program is updated to the next 10 year interval. When compliance with Code requirements is impractical, relief is requested from the NRC in compliance with 10 CFR 50.55a. The examination and testing program for snubbers is described in the Snubber Program Plan, as required by Technical Requirements Manual (TRM) Section 5.1.1.

## 5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

## REFERENCES

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## 5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

### REFERENCES

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TABLE 5.2-1 DESIGN TEMPERATURE AND PRESSURE AND MAXIMUM TEST PRESSURE FOR REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS

| Component | Design Temperature (°F) | Design Pressure (psig) | Maximum Test Pressure[a] (psig) |
|---|---|---|---|
| Reactor pressure vessel | 575 | 1250 | 1563[a] |
| Reactor Recirculation System | | | |
| Pump discharge piping | 575 | 1500 | b |
| Pump suction piping | 575 | 1250 | b |
| Discharge valves | 575 | 1525 | f |
| Suction valves | 575 | 1250 | f |
| Pump[e] | 562 | 1525 | c |
| RPV vent line | 575 | 1250 | b |
| Main steam line | 575 | 1250 | b |
| Main steam line isolation valves | 575 | 1250 | f |
| Residual heat removal system | | | |
| Shutdown suction | | | |
| RRS header to second isolation valve | | | |
| Piping | 575 | 1250 | b |
| Valves | 575 | 1250 | c |
| Pump Discharge | | | |
| RHR return from RRS header to second isolation valve | | | |
| Piping | 575 | 1500 | b |
| Valves | 575 | 1500 | c |
| Core spray system | | | |
| Pump discharge | | | |
| RPV to second isolation valve | | | |
| Piping | 575 | 1250 | b |
| Valves | 575 | 1250 | c |
| Standby liquid control system | | | |
| Pump discharge to RPV | | | |
| RPV to second isolation valve | | | |
| Piping | 575 | 1250 | b |
| Valves | 575 | 1250 | c |

TABLE 5.2-1 DESIGN TEMPERATURE AND PRESSURE AND MAXIMUM TEST PRESSURE FOR REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS

| Component | Design Temperature (°F) | Design Pressure (psig) | Maximum Test Pressure[a] (psig) |
|---|---|---|---|
| Reactor water cleanup system | | | |
| Pump suction | | | |
| RRS piping to isolation valve outside drywell | | | |
| Piping | 575 | 1250 | b |
| Valves | 575 | 1250 | c |
| Pump discharge to feedwater inlet | | | |
| Piping | 575 | 1300 | b |
| Valves | 575 | 1300 | c |
| RPV drain line | 575 | 1250 | b |
| Reactor feedwater system | | | |
| RPV to outer most isolation valve | | | |
| Piping | 450 | 1275 | b |
| Valves | 450 | 1275 | c |
| Reactor core isolation cooling system | | | |
| Steam to RCIC pump turbine | | | |
| MS line to second isolation valve | | | |
| Piping | 575 | 1250 | b |
| Valves | 575 | 1250 | c |
| Pump discharge to reactor via feedwater | | | |
| Piping | 450 | 1275 | b |
| Valves | 450 | 1275 | c |
| High Pressure coolant injection system | | | |
| Pump discharge to reactor via feedwater | | | |
| Piping | 450 | 1275 | b |
| Valves | 450 | 1275 | c |
| Steam to HPCI pump turbine MS line to second isolation valve | | | |
| Piping | 575 | 1250 | b |

TABLE 5.2-1 DESIGN TEMPERATURE AND PRESSURE AND MAXIMUM TEST PRESSURE FOR REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS

| Component | Design Temperature (°F) | Design Pressure (psig) | Maximum Test Pressure[a] (psig) |
|---|---|---|---|
| Valves | 575 | 1250 | c |
| Main steam drains system | | | |
| MS lines to second isolation valve | | | |
| Piping | 575 | 1250 | b |
| Valves | 575 | 1250 | c |
| Instrument lines | | | |
| Piping | d | d | d |
| Valves | d | d | d |

[a] Excluding shell test for valves according to Sections NB-3531-8 and NB-3531-9 of ASME B&PV Code Section III. The stress intensity ratio is interpreted from Section NB-6221 of the Code to be the ratio of the allowable stress; Sm, at test temperature to the allowable stress; Sm, at design temperature.

[b] Test pressure is 1.25 x design pressure x lowest stress intensity ratio.

[c] Test pressure is 1.50 x design pressure x lowest stress intensity ratio.

[d] Design and test conditions for the RCPB instrument lines are consistent with the conditions for the main pipeline they emanate from.

[e] The reactor recirculation system pump design pressure and temperature conditions envelop the system discharge piping design requirements.

[f] The reactor recirculation loop suction and discharge valves and the main steam isolation valves are tested per the 1968 ASME Pump and Valve Code, Article 7.

TABLE 5.2-2 REACTOR COOLANT PRESSURE BOUNDARY OPERATING THERMAL CYCLES

Normal, Upset and Testing Conditions

| Event Description | Number of Cycles[a] | Analyzed Cycles for 60 Years[b] |
|---|---|---|
| Boltup | 39 | 58 |
| Design Hydrostatic Leak Test | 55 | 75 |
| Startup | 183 | 246 |
| Turbine Roll | 152 | 201 |
| Weekly Reduction to 50% Power | 208 | 317 |
| Loss of FW Heaters – Turbine Trip with 100% Bypass | 7 | 10 |
| Loss of FW Heaters – Partial FW Heater Bypass | 15 | 19 |
| SCRAM – Turbine Generator Trip | 9 | 12 |
| SCRAM – All Others | 30 | 33 |
| Control Rod Drive Isolation | 32 | 47 |
| Single Control Rod Drive Scram | 32 | 47 |
| Reduction to 0% Power | 149 | 197 |
| Hot Standby (Injections) | 880 | 1307 |
| SBFW Injection (Cold Injection into Hot Piping) | 36 | 46 |
| SBFW Injection (Cold Injection into Cold Piping) | 11 | 18 |
| RCIC Injection (Cold Injection into Hot Piping) | 17 | 24 |
| RCIC Injection (Cold Injection into Cold Piping) | 779 | 1172 |
| HPCI Injection (Cold Injection into Hot Piping) | 23 | 29 |
| HPCI Injection (Cold Injection into Cold Piping) | 6 | 9 |
| FW Injection (Cold Injection into Hot Piping) | 5 | 10 |
| FW Injection (Cold Injection into Cold Piping) | 5 | 10 |
| Shutdown | 183 | 246 |
| Hydrostatic Test (1563 psig) | 1 | 2 |
| Unbolt | 39 | 58 |
| Pre-Op Blowdown | 2 | 3 |
| SCRAM – Loss of FW Pumps | 10 | 13 |
| Loss of RWCU Flow | 207 | 270 |
| Core Spray Injection | 3 | 4 |
| Multiple SRV Actuation | 7 | 9 |
| Individual SRV Actuation (Sum) | 1232 | 1851 |
| RRS Pump Seal Injection On-Off-On | 29 | 37[c] |
| RRS Single Loop Operation (SLO) | 10 | 10/loop[d] |
| OBE (Operating Basis Earthquake) | 1 | 2 |

TABLE 5.2-2 REACTOR COOLANT PRESSURE BOUNDARY OPERATING THERMAL CYCLES

Emergency Conditions

| Event Description | Number of Cycles[a] |
|---|---|
| SCRAM – Single Safety/Relief Valve Blowdown | 8 |
| Reactor Overpressure with Delayed Scram, Feedwater Stays On, Isolation Valves Stay Open | 1 |
| Automatic Blowdown | 1 |
| Improper Start of Cold RRS loop | 1 |
| Sudden Start of Pump in Cold RRS loop | 1 |
| Improper Startup with Recirculation System Pumps Off and Drain Shut Off Followed by Turbine Roll and Increase to Rated Power | 1 |
| Natural Circulation Startup | 3 |
| Loss of AC Power, Natural Circulation Restart | 5 |

Faulted Conditions

| Event Description | Number of Cycles[a] |
|---|---|
| Pipe Rupture and Blowdown | 1 |

Other Events with a Cyclic Limit

| Event Description | Expected Duty[a] | Analyzed for 60 Years[b] |
|---|---|---|
| RRS Pump A Hot Standby (hours in SLO, idle with backflow) | 464 | 697 |
| RRS Pump B Hot Standby (hours in SLO, idle with backflow) | 337 | 507 |
| Main Steam Bypass Line – Time of Operation at 30-45% Valve Open Position (days) | 72 | 100 |

[a] Expected number of cycles for a 40 year plant design life based on conservative projections of Fermi 2 operating history

[b] Analyzed number of cycles for License Renewal

[c] The Recirculation pump coolers were replaced in 1998. Through December 2012, 3 cycles had been experienced. The analysis input value for the coolers was 12 cycles.

[d] Per NEDC-32313P (Subsection 6.3, Reference 14), the 10 cycles are per loop. This analysis was not updated for License Renewal. This analysis applies to an isolated loop.

TABLE 5.2-3 CODE CASE INTERPRETATIONS

| | | |
|---|---|---|
| 1. | 1141 | Foreign Produced Steel |
| 2. | 1332 | Requirements for Steel Forgings |
| 3. | 1334 | Requirements for Corrosion Resisting Steel Bars and Shaping |
| 4. | 1335 | Requirements for Bolting Materials, Section III |
| 5. | 1336 | Requirements for Nickel-Chrom-Inn Alloy |
| 6. | 1337 | Requirements for Special Type 403 Modified Forgings and Bars |
| 7. | 1344 | Requirements for Nickel-Chromium, Age-Hardenable Alloys, Section III |
| 8. | 1359 | Ultrasonic Examination of Forgings, Section III |
| 9. | 1384 | Requirements for Precipitation Hardening Alloy Bars and Forgings, Section III |
| 10. | 1388 | Requirements for Stainless Steel Precipitation Hardening, Section III |
| 11. | 1390 | Requirements for Nickel-Chromium Age-Hardenable Alloy for Bolting, Section III |
| 12. | 1401 | Welding Repairs to Cladding of Class I Section III Components After Heat Treating |
| 13. | 1420 | SB-167 Nickel-Chromium-Iron Alloy Pipe or Tube |
| 14. | 1423 | Wrought Type 304 and 316 Nitrogen Added |
| 15. | 1433 | Normalized and Tempered 2-1/4 and 3A Low Alloy Forgings |
| 16. | 1434 | Postweld Heat Treatment of SA-487 Class 8N Castings |
| 17. | 1441 | Waiving of 3.0 $S_m$ Limit for Section III Construction |
| 18. | 1456 | Substitution of U.T. Examination for Progressive PT or MT of Partial Penetration and Oblique Nozzle Attachment |
| 19. | 1459 | Welding Repairs to Base Metal of Class I Section III Components After Final PWHT |
| 20. | 1487 | Evaluation of Nuclear Piping for Faulted Conditions |
| 21. | 1492 | Postweld Heat Treatment, Sections I, III, and VIII, Div. 1 and 2 |
| 22. | 1495 | Stress Indices in Table NB-3683.2-1 |
| 23. | 1501 | Use of SA-453 Bolts in Service Below 800$^o$F Without Stress Rupture Tests, Section III |
| 24. | 1504 | Electrical and Mechanical Penetration Assemblies, Section III, Classes 1, 2, and 3 Components |
| 25. | 1516-1 | Welding of Seats in Valves for Section III Application |
| 26. | N-32-4 | Hydrostatic Testing of Embedded Piping, Class 2 and 3 Piping |

TABLE 5.2-3 CODE CASE INTERPRETATIONS

| | | |
|---|---|---|
| 27. | N-237-2 | Hydrostatic Testing of Internal Piping, Class 2 and 3 |
| 28. | N-240 | Hydrostatic Testing of Open-Ended Piping |
| 29. | N-252 | Low Energy Capacitive Discharge Welding Method for Temporary or Permanent Attachments to Components and Supports |
| 30. | N-315 | Repair of Bellows, Class 2, 3, and MC |
| 31. | N-316 | Alternative Rules for Fillet Weld Dimensions for Socket Welded Fittings, Class 1, 2, and 3 |
| 32. | N-274 | Alternate Rules for Examination of Weld Repairs for Section III, Division 1 Construction |
| 33. | N-275 | Repair of Welds, Section III, Division 1 |
| 34. | N-236 | Repair and Replacement of Class MC Vessels |
| 35. | N-192-2 | Use of Braided Flexible Connectors |
| 36. | N-362-1 | Pressure Testing of Containment Items |
| 37. | N-411-1 | Alternate Damping Values for Spectral Analysis of Piping Sections |

# TABLE 5.2-4 VALVE AND PUMP DESCRIPTION

## Part A - Valve description

| System/Location | | Valve Identification | Valve Type |
|---|---|---|---|
| NUCLEAR BOILER/RPV Vent | | B2100F001 | manual globe |
| | | B2100F002 | manual globe |
| | | B2100F005 | manual globe |
| | | B21F403 | De-Energized & Abandoned in Place |
| | | B21F404 | De-Energized & Abandoned in Place |
| Main Steam Safety/Relief (Nuclear Pressure Relief) | | B2104F013A | dual-function, 2-stage relief |
| | | B2104F013B | |
| | | B2104F013C | |
| | | B2104F013D | |
| | | B2104F013E | |
| | | B2104F013F | |
| | | B2104F013G | |
| | | B2104F013H | |
| | | B2104F013J | |
| | | B2104F013K | |
| | | B2104F013L | |
| | | B2104F013M | |
| | | B2104F013N | |
| | | B2104F013P | |
| | | B2104F013R | |
| Main Steam Drains | | B2103F016 | motor-operated gate |
| | | B2103F019 | motor-operated gate |
| Main Steam Isolation | | B2103F022A | air-operated, Y-pattern globe |
| | (Inboard) | B2103F022B | |
| | | B2103F022C | |
| | | B2103F022D | |
| | (Outboard) | B2103F028A | |
| | | B2103F028B | |
| | | B2103F028C | |
| | | B2103F028D | |
| Feedwater | (Inboard) | B2100F010A | swing check |
| | | B2100F010B | swing check |
| | | B2100F011A | manual gate |
| | | B2100F011B | manual gate |
| Feedwater | (Outboard) | B2100F032A | testable swing check |
| | | B2100F032B | testable swing check |
| | | B2100F076A | spring-to-close swing check |
| | | B2100F076B | spring-to-close swing check |
| REACTOR RECIRCULATION | | | |
| Suction | | B3105F023A | motor-operated gate |
| | | B3105F023B | motor-operated gate |

TABLE 5.2-4 VALVE AND PUMP DESCRIPTION

Part A - Valve description

| System/Location | Valve Identification | Valve Type |
|---|---|---|
| Discharge | B3105F031A | motor-operated gate |
| | B3105F031B | motor-operated gate |
| Drain/sample line | B3100F029 | manual globe |
| | B3100F030 | manual globe |
| STANDBY LIQUID CONTROL | C4100F006 | swing check |
| | C4100F007 | swing check |
| | C4100F008 | manual globe |
| RESIDUAL HEAT REMOVAL | | |
| RHR Return (LPCI) | E1150F015A | motor-operated gate |
| | E1150F015B | motor-operated gate |
| | E1100F050A | testable swing check |
| | E1100F050B | testable swing check |
| | E1100F060A | manual gate |
| | E1100F060B | manual gate |
| RHR Supply (SDC) | E1150F008 | motor-operated gate |
| | E1150F009 | motor-operated gate |
| | E1150F608 | motor-operated gate |
| | E1100F067 | manual gate |
| CORE SPRAY | E2150F005A | motor-operated gate |
| | E2150F005B | motor-operated gate |
| | E2100F006A | testable swing check |
| | E2100F006B | testable swing check |
| | E2100F007A | manual gate |
| | E2100F007B | manual gate |
| HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM | | |
| Steam to HPCI turbine | E4150F002 | motor-operated gate |
| | E4150F003 | motor-operated gate |
| | E4150F600 | motor-operated globe bypass |
| Return through feedwater | E4150F006 | motor-operated gate |
| REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM | | |
| Steam to RCIC turbine | E5150F007 | motor-operated gate |
| | E5150F008 | motor-operated gate |
| Return through feedwater | E5150F013 | motor-operated gate |
| REACTOR WATER CLEANUP | | |
| Supply to RWCU | G3352F001 | motor-operated gate |
| | G3352F004 | motor-operated gate |

TABLE 5.2-4 VALVE AND PUMP DESCRIPTION

Part A - Valve description

| System/Location | Valve Identification | Valve Type |
|---|---|---|
| | G3352F100 | motor-operated gate |
| | G3352F101 | motor-operated gate |
| | G3352F106 | motor-operated gate |
| | G3352F119 | motor-operated gate |
| | G3352F102 | motor-operated Y-globe throttle |
| Return through feedwater | G3300F120 | swingcheck |
| | G3300F121 | swingcheck |
| | G3352F220 | motor-operated gate |

Part B – Pump Description

| | | |
|---|---|---|
| REACTOR RECIRCULATION System Pumps | B3101C001A&B | 28 X 28 X 35 DVSS |

TABLE 5.2-5 NUCLEAR STEAM SUPPLY SYSTEM SAFETY/RELIEF VALVES

| Types of Valves | No. of Valves | Set Pressure (psig) | ASME Rated Capacity at 103 Percent Set Pressure (lb/hr each)[b] |
|---|---|---|---|
| Safety/relief | 5 | 1135 | 904,400 |
| Safety/relief | 5 | 1145 | 912,200 |
| Safety/relief | 5[a] | 1155 | 920,100 |

[a] Indicates the number of safety/relief valves actuated to provide automatic depressurization. This provides sufficient flow capacity to satisfy automatic depressurization requirements, assuming that one valve fails to open.

[b] Flow capacity = W = 51.5 x K x 0.9 x A x P

where

K = 0.8 (friction coefficient)

A = $\pi/4 \times 5.125^2$ (flow area)

P = set pressure with 103 percent accumulation

This information is obtained from the Safety and Safety Relief Valve Relieving Capacity Certification - Target Rock Corporation, the National Board of Boiler and Pressure Vessel Inspectors, June 6, 1975.

TABLE 5.2-6 REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

| Component | Form | Material | Specification (ASTM/ASME) |
|---|---|---|---|
| Reactor pressure vessel heads, shells | Rolled plate or forgings | Low alloy steel | SA-533 Grade B, Class 1 |
| | Welds | Low alloy steel | SFA-5.5 |
| Closure flange | Forged ring | Low alloy steel | SA-508 Class 2 |
| | Welds | Low alloy steel | SFA-5.5 |
| Nozzles | Forged shapes | Low alloy steel | SA-508 Class 2 |
| | Welds | Low alloy steel | SFA-5.5 |
| Cladding | Weld overlay | Austenitic stainless steel | SFA-5.9 and SFA-5.4 TP 308, 309 and 312 carbon content of final surface limited to 0.8 percent maximum |
| Control rod drive stub tubes | Pipe | Incone1 | SB-167 |
| | Welds | Incone1 | SFA-5.11 TP ENiCrFe-3 |
| Control rod drive housing | Pipe | Austenitic stainless steel | SA-312 TP 304 |
| | Welds | Stainless steel | SFA-5.9 TP 308 |
| In-core housing | Pipe | Austenitic stainless steel | SA-213 TP 304 |
| | Welds | Stainless steel | SFA-5.9 TP 308 |

Additional RCPB component materials and specifications used are specified below.

Depending on whether impact tests are required and depending on the lowest service metal temperature when impact tests are required, the following ferritic materials and specifications were used:

TABLE 5.2-6 REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

| | |
|---|---|
| Pipe | SA-106 Grade B; SA-333 Grade 6 and SA-155 Grade KCF-70 |
| Valves | SA-105 Grade II; SA-350 Grade LF1 and SA-216 Grade WCB |
| Fittings | SA-105 Grade II; SA-350 Grade LF1; SA-234 Grade B; and SA-420 Grade WPL1 or WPL6 |
| Bolting | SA-193 Grade B7; SA-194 Grades 7 and 2H; and SA-540 Grade B22, B23 and B24 |
| Welding material | SFA-5.1 (E-7015, E-7016, E-7018) SFA-5.5 (E-7010A1, E-7015, E-7016, E-7018) SFA-5.17, SFA-5.18 |
| Other material | SA-516, Grade 70 |

For those systems or portions of systems, such as the reactor recirculation system, which require austenitic stainless steel, the following materials and specifications were used:

| | |
|---|---|
| Pipe | SA-376 Type 304; SA-312 Type 304; SA-358 Type 304 |
| Valves | SA-182 Grade F-304; SA-351 Grades CF-8 and CF-8M |
| Pump | SA-182 Grade F-304; SA-351 Grades CF-8 and CF-8M |
| Flanges | SA-182 Grade F-316 |
| Bolting | SA-193 Grade B7; SA-194 Grades 7 and 2H; SA-540 Grades B22, B23 and B24 |
| Welding material | SFA-5.4 (E308-15, E308L-15, E316-15); SFA-5.9 (ER-308, ER-308L, ER-316) |

TABLE 5.2-7 REACTOR VESSEL TOUGHNESS CHARPY V-NOTCH CHEMISTRY

| RPV | | Material | Drop Wt | Charpy V-Notch 50 | Charpy V-Notch 35 MILS | | | Chemistry | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | TNDT | ft-lb | LE | USE | $RT_{NDT}$ | Cu | P | S | Ni(g) |
| Location | Pc. No. (a) | Type | °F | °F | °F | ft-1b | °F | % | % | % | % |
| Closure Head-Lower Torus | 319-03 | SA-533-65 Grade B Class 1 | -10 | -40 | -40 | 140 | -10 | 0.13 | 0.012 | 0.019 | |
| Closure Head Flange | 319-02 | A-508 Class 2 | (b) | -40 | -40 | 186 | 0 | 0.03 | 0.007 | 0.012 | |
| Upper Shell | 306-1 | SA-533-65 Grade B Class 1 | -10 | 35 | 30 | 125 | -10 | NA | 0.012 | 0.018 | |
| Vessel Flange | 308-2 | A-508 Class 2 | (b) | -40 | -40 | 145 | 10 | 0.15 | 0.003 | 0.019 | |
| Lower Intermediate Shell | 305-01E | SA-533-65(c) Grade B Class 1 | -20 | -10 | -10 | 130 | -20 | 0.12 | 0.010 | 0.015 | 0.61 |
| | 305-03 | | -30 | 40 | 10 | 119 | -12 | 0.12 | 0.012 | 0.016 | 0.61 |
| Lower Shell | 305-04 | SA-533-65(c) Grade B Class 1 | -10 | -10 | -20 | 130 | -10 | 0.12 | 0.011 | 0.017 | 0.56 |
| Lower Intermediate to Lower Shell (d) | | Weld I-313 | NA | < 10 | NA | >105 | -50 | 0.23 | 0.016 | 0.010 | 1.0(h) |
| Lower Intermediate Long Seams (e)(c) | | Weld 15-308A-D | NA | < 10 | NA | >90 | -50 | 0.32 | 0.016 | 0.011 | 0.5(h) |
| Lower Shell Long Seams (f) | | Weld 2.307A-C | NA | < 10 | NA | >57 | -44 | 0.26(i) | 0.013 | NA | 0.87(i) |

(a) The values listed are for the piece having the highest $T_{NDT}$ at the indicated location.
(b) $RT_{NDT}$ assumed to be 10°F
(c) Values included for both plates from lower intermediate shell and the weld used for materials surveillance program
(d) Weld Charpy V-Notch Impact Tests at 10°F-101, 108, 107 ft-lb.
(e) Weld Charpy V-Notch Impact Tests at 10°F-83, 94, 97 ft-lb.
(f) Weld Charpy V-Notch Impact Tests at 10°F-62, 47, 62 ft-lb.
(g) Listed for plates and welds in the beltline region only
(h) Assumed, based on maximum allowables of filler metal specification, see report referenced in (i) below
(i) Calculated values from General Electric Report SASR 90-73, DRF 137-0010, Revision 1, January 1991
NA Not Available

TABLE 5.2-8 HAS BEEN INTENTIONALLY DELETED

TABLE 5.2-9 HAS BEEN INTENTIONALLY DELETED

TABLE 5.2-10 HAS BEEN INTENTIONALLY DELETED

# TABLE 5.2-11 SUMMARY OF ISOLATION/ALARM OF SYSTEM MONITORED AND THE LEAK DETECTION METHODS USED

| Function | | A[a] | A | A | A | A/I[b] | A | A/I | A | A | A/I | A/I | A/I | A/I | A | A | A/I |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Source of Leakage | Location | High PC[c] Temperature | PC Sump High Flow Rate | High PC Air Cooler CCW[d] $\Delta T$ | Equipment Area High T & $\Delta T$ | Low Steam Line Pressure | RB Sump High Flow Rate | Equipment Area High T Time Delay | Suppression Pool Area High T & $\Delta T$ Time Delay | PC Pressure (High) | High Flow Rate[e] | High Turbine Exhaust Pressure (RCIC) | CU[f] $\Delta$ Flow (High) | Reactor Low Water Level | Radiation Level | High Flow in Drain Line | High Radiation Level |
| Main steam line | PC | X | X | X | | | | | | X | X | | | X | X | | X |
| | RB[g] | | | | X | | X | | | | X | | | | | | |
| RHR | PC | X | X | X | | | | | | X | | | | | X | | |
| | RB | | | | X | | X | | | | | | | | | | |
| RCIC steam | PC | X | X | X | | X | | | | X | X | | | | X | | |
| | RB | | | | | X | X | X[h] | X | | X | X | | | | | |
| RCIC Water | PC | | | | | | | | | | | | | | | | |
| | RB | | | | | | X | | | | | | | | | | |
| HPCI steam | PC | X | X | X | | X | | | | X | X | | | | X | | |
| | RB | | | | | X | X | X[h] | X | | X | X | | | | | |
| HPCI water | | | | | | | X | | | | | | | | | | |
| Cleanup Water | PC | X | X | X | | | | | | X | | | X | X | X | | |
| | RB | Hot | | | X | | X | X[h] | | | | | X | X | | | |
| | RB | Cold | | | X | | X | | | | | | X | X | | | |
| Feedwater | PC | X | X | X | | | | | | X | | | | | X | | |
| | RB | | | | | | X | | | | | | | | | | |
| ECCS suction line | RB | | X | | | | | | | | | | | | | | |
| Recirculation System | PC | | | | | | | | | | | | | | | X | |

[a] A – Alarm.
[b] A/I – Alarm/isolation.
[c] PC – Primary containment.
[d] CCW – Closed cooling Water.
[e] Break downstream of flow element isolates the steam line.
[f] CU – Cleanup.
[g] RB – Reactor building.
[h] No time delay.

TABLE 5.2-12 SYSTEMS OUTSIDE PRIMARY CONTAINMENT THAT COULD CONTAIN HIGHLY RADIOACTIVE FLUIDS

Reactor core isolation cooling

Residual heat removal

- Containment spray
- Suppression pool cooling
- Low-pressure coolant injection
- Shutdown cooling

Core spray

Reactor water sample

Reactor water cleanup

High-pressure coolant injection

Standby gas treatment

Control rod drive discharge headers

Containment sampling system

TABLE 5.2-13 SYSTEMS OUTSIDE PRIMARY CONTAINMENT THAT WOULD NOT CONTAIN HIGHLY RADIOACTIVE FLUIDS

| System | Comment |
|---|---|
| RHR fuel pool cooling | Not directly affected by accident. |
| Standby liquid control | Injects fluid and does not circulate reactor coolant. |
| General service water / emergency equipment service water | Do not circulate reactor coolant and could become contaminated only due to system leaks. |
| Reactor building closed cooling water / emergency equipment cooling water | Do not circulate reactor coolant and could become contaminated only due to system leaks. |
| Condensate storage | Could become contaminated only due to isolation valve leakage. |
| Demineralized water makeup | Could become contaminated only due to isolation valve leakage. |
| Torus water management | Isolated during LOCA and not required for accident mitigation. |
| Control air/station air | Would require system or interface required for accident mitigation. |
| Fuel-pool cooling and cleanup | Not directly affected by accident. |
| Main steam lines | Would require failure of MSIVs. |
| Feedwater lines | Would require failure of isolation valves. |
| Drywell cooling system | Uses RBCCW of EECW and is not needed for safe shutdown of plant. |
| Reactor building floor/equipment drains | Not required for accident mitigation. Minimizing leakage from systems in Table 5.2-12 minimizes input to the system. |
| Radwaste | Not required for accident mitigation. |
| Supplemental cooling chilled water | Does not circulate reactor coolant and could become contaminated only due to system leaks. |
| Combustible gas control system | Could become contaminated only due to isolation valve leakage. |

TABLE 5.2-14 CLASS 1 PRESERVICE EXAMINATION REQUIREMENTS

| Examination Category | | Component or Part To Be Examined | Required Exam Method | Remarks |
|---|---|---|---|---|
| | | Pump Pressure Boundary (IWB – 2500) | | |
| B-G-1 | Pressure-retaining bolting greater than 2 in. in diameter | Recirculation pumps | Volumetric and visual | Preservice inspection performed to ASME Section XI, 1980 Edition, winter 1980 addenda, for compatibility with ISI program. |
| B-K-1 | Integrally welded supports | | Volumetric or surface | Preservice inspection performed to ASME Section XI, 1980 Edition, winter 1980 addenda, for compatibility with ISI program. |
| B-L-1 | Pump casing welds | | Volumetric | |
| B-L-1 | Pump casings | | Visual | |
| | | Valve Pressure Boundary (IWB – 2500) | | |
| B-G-1 | Pressure-retaining bolting greater than 2 in. in diameter | Class 1 valves | Volumetric and visual | Preservice inspection performed to ASME Section XI, 1980 Edition, winter 1980 addenda, for compatibility with ISI program. |
| B-G-2 | Pressure-retaining bolting smaller than 2 in. in diameter | | Visual | Preservice inspection performed to ASME Section XI, 1980 Edition, winter 1980 addenda, for compatibility with ISI program. |
| B-K-1 | Integrally welded supports | | Volumetric or surface | Preservice inspection performed to ASME Section XI, 1980 Edition, winter 1980 addenda, for compatibility with ISI program. |
| B-M-X | Valve body welds | | Volumetric | |
| B-M-2 | Valve bodies | | Visual | |
| B-P | Except components | | Visual | |
| | | Piping Pressure Boundary (IWB – 2500) | | |

TABLE 5.2-14 CLASS 1 PRESERVICE EXAMINATION REQUIREMENTS

| Examination Category | | Component or Part To Be Examined | Required Exam Method | Remarks |
|---|---|---|---|---|
| B-F | Dissimilar metal safe-end to piping welds and safe-end to branch piping | Safe-end welds | Volumetric and surface | |
| B-G-2 | Pressure-retaining bolting smaller than 2 in. in diameter | Bolting less than 2 in. in diameter | Visual | Preservice inspection performed to ASME Section XI, 1980 Edition, winter 1980 addenda, for compatibility with ISI program. |
| B-J | Circumferential and longitudinal pipe welds | Piping welds | Volumetric | |
| | Branch pipe connection welds exceeding 6 in. in diameter | Piping welds | Volumetric | |
| | Branch pipe welds 6 in. in diameter and smaller | Piping welds | Surface | |
| | Socket welds | Socket welds | Surface | |
| B-K-1 | Integrally welded supports | Piping lugs | Surface | Preservice inspection performed to ASME Section XI, 1980 Edition, winter 1980 addenda, for compatibility with ISI program. |
| B-P | Exempted components | Exempted components | Visual | |
| | | Reactor Pressure Vessel (IS – 251) | | |
| A | Pressure-retaining welds in reactor bolting region | RPV longitudinal and circumferential weld in core region | UT | A manual UT examination was performed on the RPV longitudinal and circumferential welds in the combustion engineering fabrication shop. |
| B | Pressure-retaining welds in vessels | RPV closure head and meridional welds and bottom head meridional and circumferential welds | UT | See category A. |

TABLE 5.2-14 CLASS 1 PRESERVICE EXAMINATION REQUIREMENTS

| Examination Category | | Component or Part To Be Examined | Required Exam Method | Remarks |
|---|---|---|---|---|
| | | RPV longitudinal and circumferential welds above and below core region | UT | See category A. |
| C | Pressure-retaining welds: vessel-to-flange and head-to-flange | RPV closure head-to-flange weld<br>RPV shell-to-flange weld | UT | Vessel-to-flange examined manually from the seal surface. |
| D | Primary nozzle-to-vessel welds and nozzle inside radius section | Nozzle-to-shell welds and inner radius section on the following nozzles:<br>Recirculation inlet<br>Recirculation outlet<br>Main steam<br>Feedwater<br>Jet pump instrumentation<br>Core spray<br>Head spray and instrumentation spare<br>CRD hydraulic system return<br>RPV vent line | UT | Nozzle-to-shell welds examined manually in the fabrication shop. Inner radius examinations performed. |
| E-1 | Pressure-containing welds in vessel penetration | CRD penetration | Visual | UT not possible; visual examination for leakage substituted. |
| G-1 | Pressure-retaining bolting 2 in. and larger in diameter | RPV closure studs and nuts, washers, ligaments and bushings | UM/MT | |
| H | Vessel external skirts | RPV support skirt-to-vessel weld | UT | Examination completed. |
| I-1 | Interior clad surfaces of reactor vessels | RPV cladding | Visual | |
| N | Interior surfaces and interior components of reactor vessel | RPV internals | Visual | |

TABLE 5.2-15 PRESSURE ISOLATION VALVES

| System | P&ID | Valve Numbers | Type | Size (in.) | Function |
|---|---|---|---|---|---|
| RHR | 6M721-2083 | E1150-F015A, B | Gate | 24 | Discharge to recirculation system |
| | 6M721-2084 | E1100-F050A, B | Check | 24 | Discharge to recirculation system |
| | | E1150-F008 | Gate | 20 | Suction from recirculation system |
| | | E1150-F009 | Gate | 20 | Suction from recirculation system |
| | | E1150-F608 | Gate | 20 | Suction from recirculation system |
| Core spray | 6M721-2034 | E2150-F005A, B | Gate | 12 | Discharge to core spray sparger |
| | | E2100-F006A, B | Check | 12 | Discharge to core spray sparger |
| HPCI | 6M721-2035 | E4150-F006 | Gate | 14 | Discharge to feedwater line |
| | | E4100-F005 | Check | 14 | Discharge to feedwater line |
| RCIC | 6M721-2044 | E5150-F013 | Gate | 6 | Discharge to feedwater line |
| | | E5100-F014 | Check | 6 | Discharge to feedwater line |

TABLE 5.2-16 PRESSURE ISOLATION PROTECTION AND MONITORING

| System/Line Needing Protection | Relief Valve Overpressure Protection | Control Room Alarm | Control Room Indicator | Local Indicator |
|---|---|---|---|---|
| RHR discharge | E1100F025A, B, 1-1/2 in. | E11-N022A, B at 435 psig | | E11-R003A, B, C, D, 0-600 psig |
| RHR suction | E1100F030A, B, C, D, E1100F029, 1in. | -- | | E11-R002A, B, C, D. 30 in. Hg, 150 psig |
| Core spray discharge | E2100F012A (V22-2016), E2100F012B (V22-2017), E2100F011B (V22-2119), E2100F011A (V22-2120), 2 in. | E21-N007A, B at 440 psig | E21-R600A, B, 0-600 psig | -- |
| HPCI Booster Inlet | E4100-F020 (V22-2044), 1-1/2 in. | E41-N031 at 70 psig | E41 R609 30 in. Hg, 785 psig | E41-R004, 30 in. Hg to 100 psig |
| RCIC suction | E5100-F017 (V22-2002), 1 in. | E51-N030 at 70 psig | E51 R609 30 in. Hg, 85 psig | E51-R002, 30 in. Hg, 100 psig |

Figure Intentionally Removed
Refer to Plant Drawing M-5007

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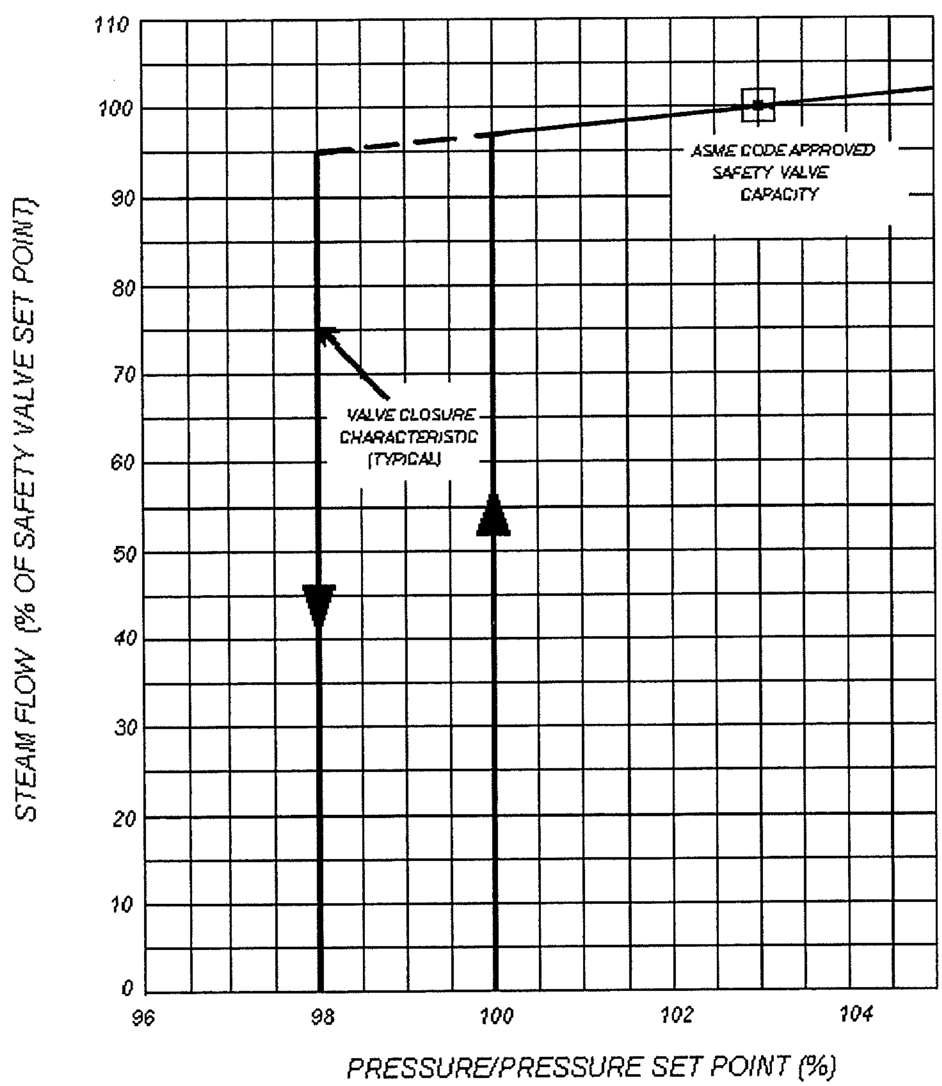
FIGURE 5.2-1

PRIMARY CONTAINMENT PNEUMATIC SUPPLY




SOURCE REFERENCE: GENERAL ELECTRIC DOCUMENT NO. 22A4070

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**FIGURE 5.2-1a**

**TYPICAL DUAL SAFETY/RELIEF VALVE CAPACITY CHARACTERISTICS**

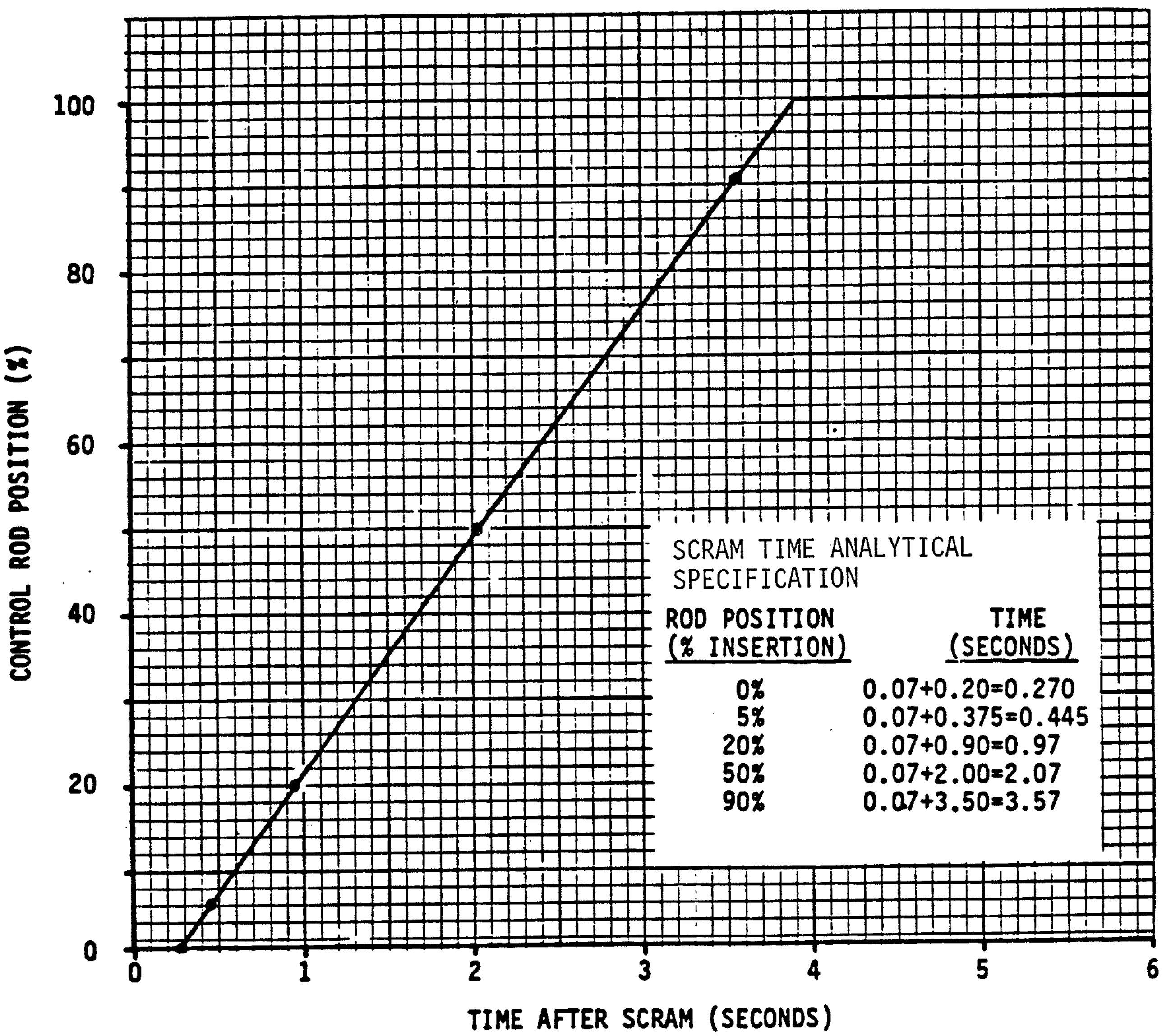


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FIGURE 5.2-1b

CONTROL ROD DRIVE VERSUS TIME

REFERENCE: GENERAL ELECTRIC
DOCUMENT 384 HA259, SH. 24, REV. 3

REV 6 3/93

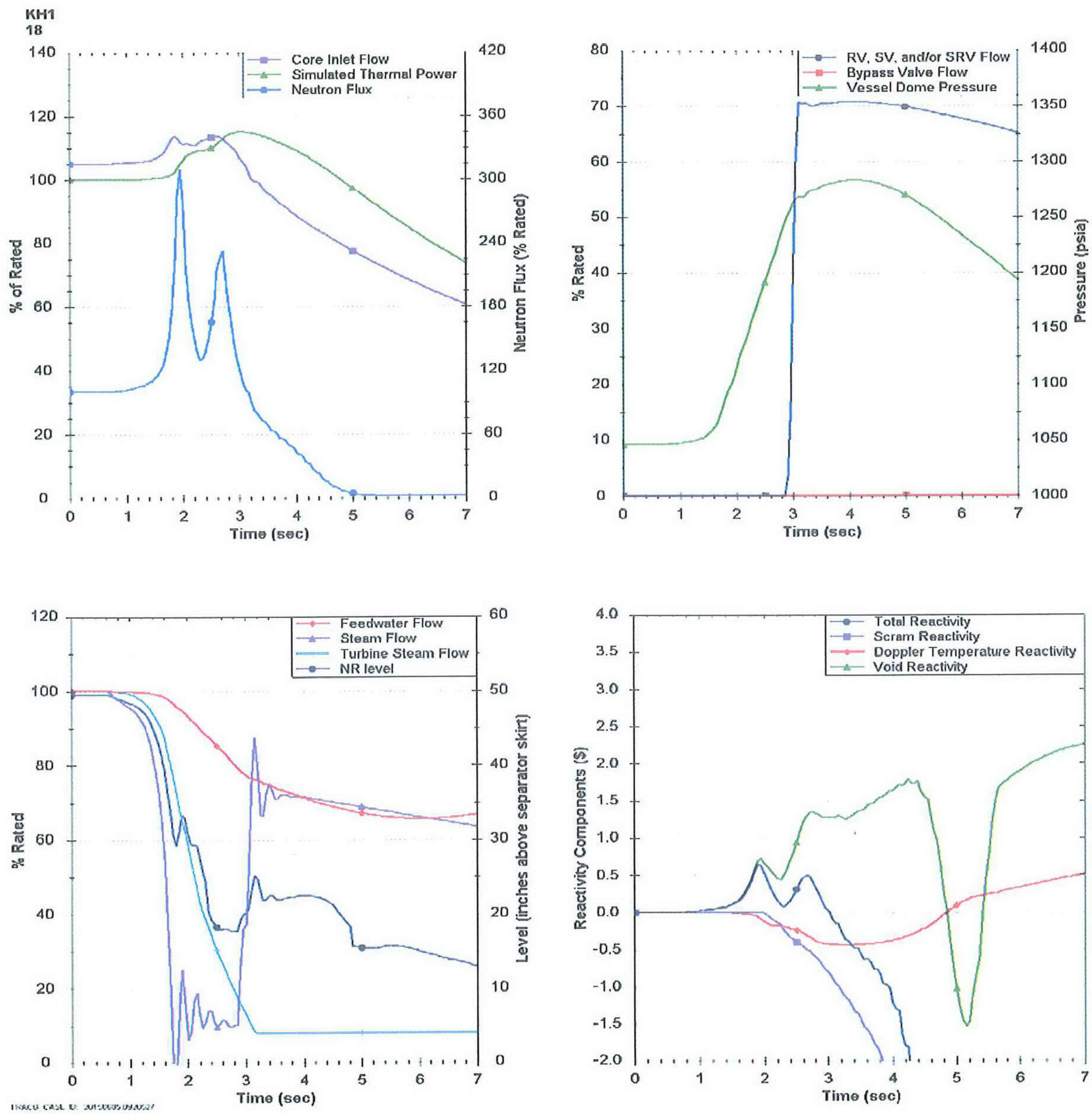


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FIGURE 5.2-1c

RESPONSE TO MSIV CLOSURE WITH FLUX SCRAM

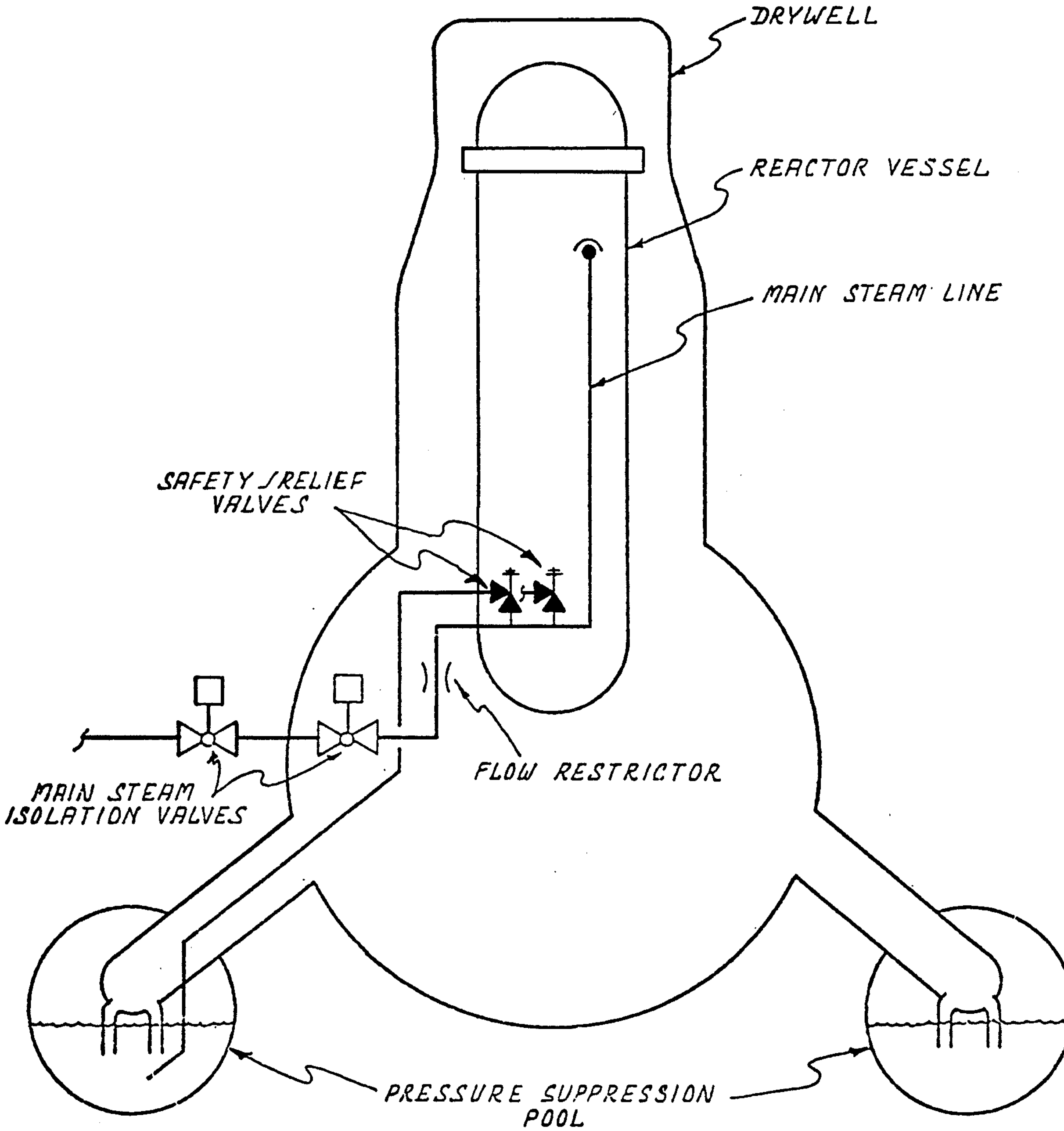


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**FIGURE 5.2-1d**

**SAFETY RELIEF VALVE SCHEMATIC ELEVATION**

SOURCE REFERENCE: GENERAL
ELECTRIC DOCUMENT NO. 22A4070

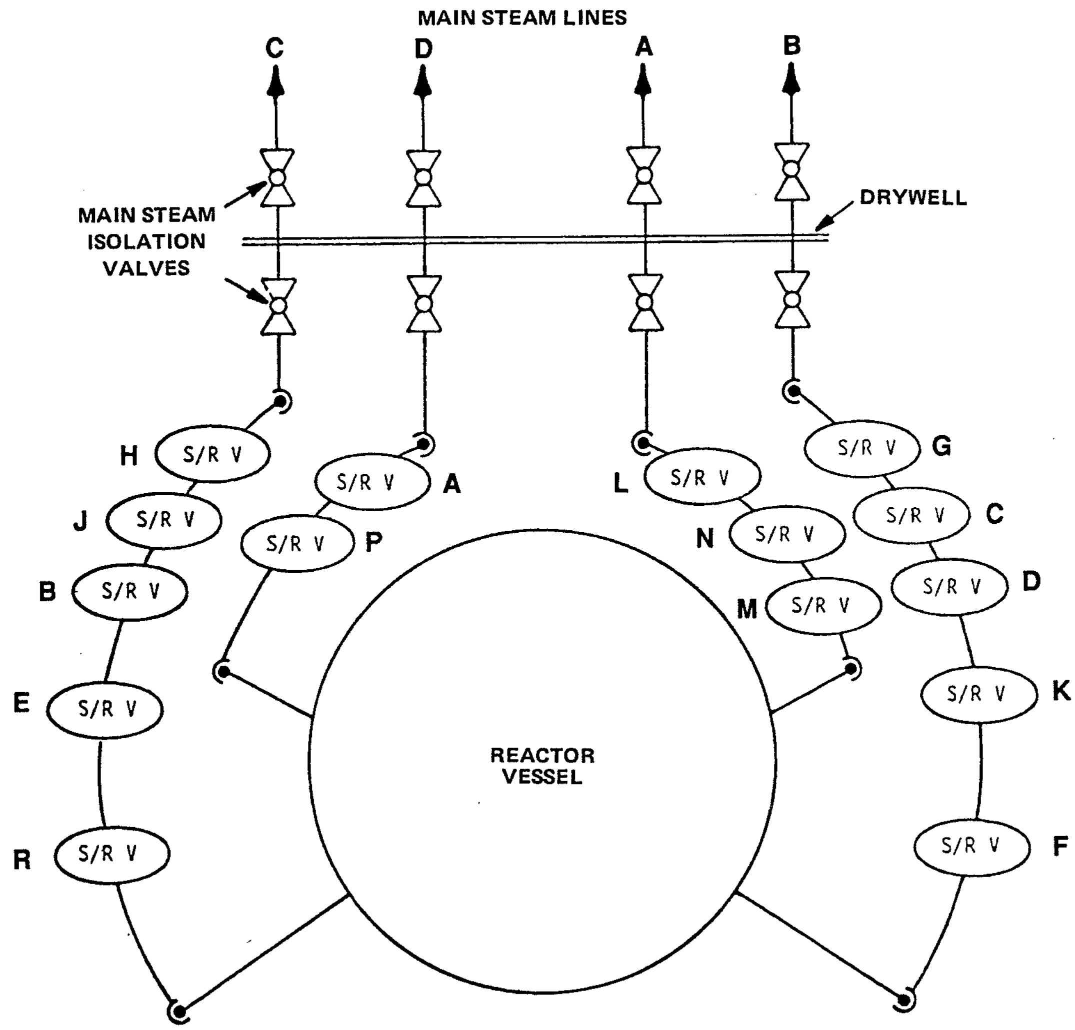


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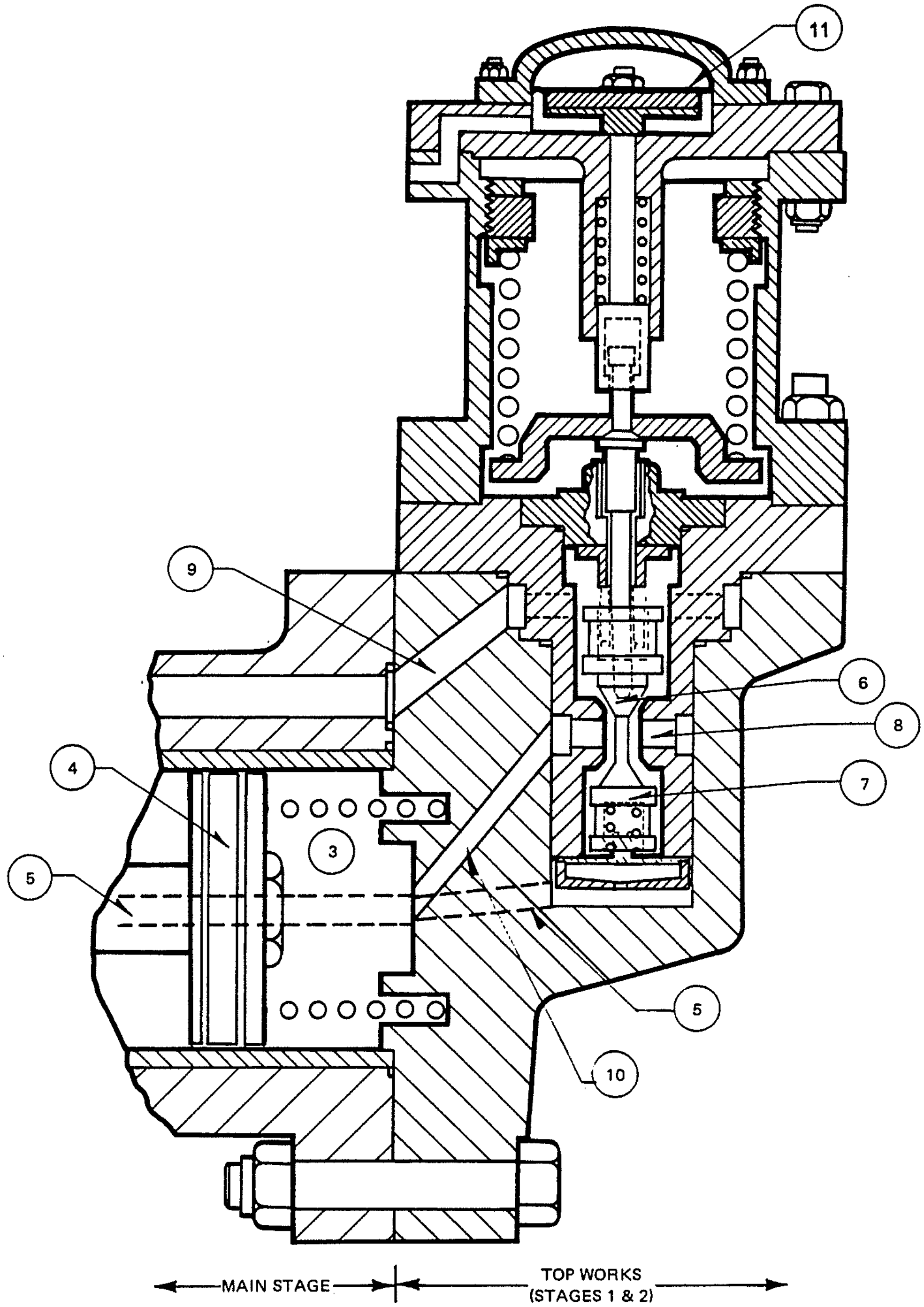
FIGURE 5.2-1e

SAFETY/RELIEF VALVE SCHEMATIC PLAN




NOTE: SUBSECTION 5.2.2.4.1 DESCRIBES THE NUMBERED PORTIONS OF THE VALVE.

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FIGURE 5.2-2

TARGET ROCK TWO-STAGE SAFETY/RELIEF VALVE

Figure Intentionally Removed
Refer to Plant Drawing M-4096-1

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FIGURE 5.2-3

ISOMETRIC OF NUCLEAR PRESSURE RELIEF SYSTEM VALVE DISCHARGE PIPING TO TORUS TYPICAL

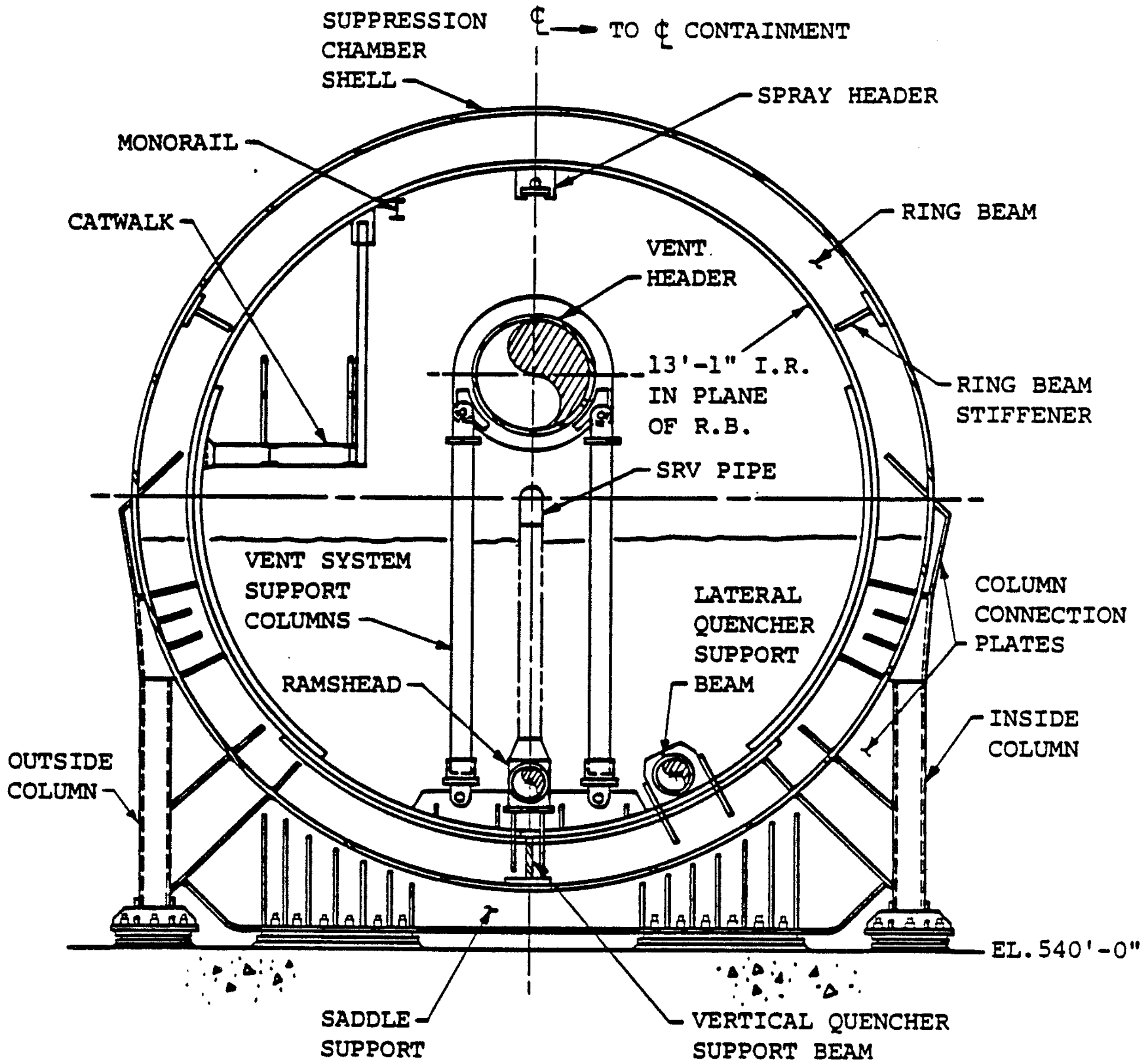


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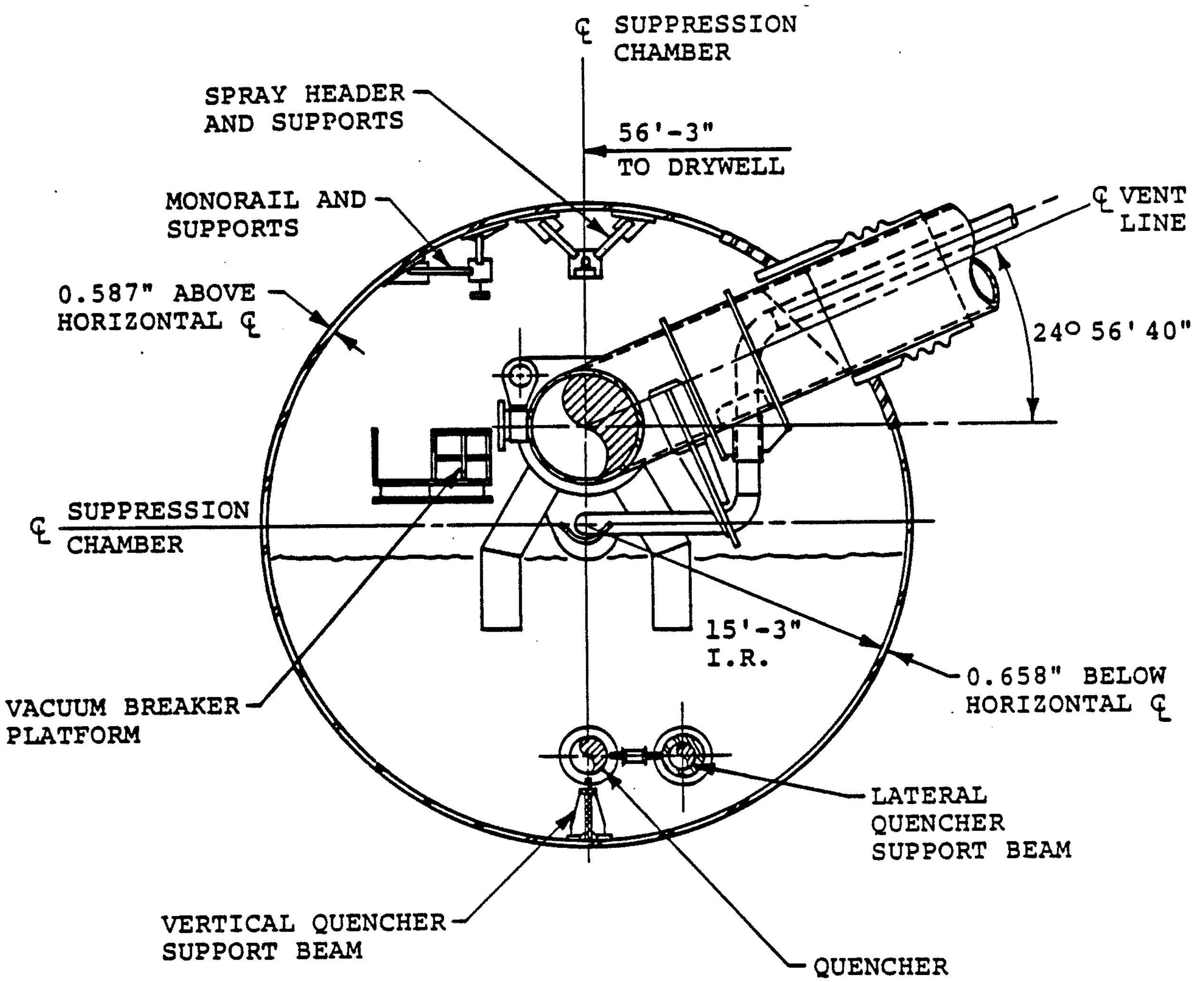
FIGURE 5.2-4

SUPPRESSION CHAMBER CROSS SECTION

NUTECH DRAWING NO. DET-04-028-1, REV. 0
PUAR FIGURE 1-2.1-4




NUTECH DRAWING NO. DET-04-028-1, REV. 0
PUAR FIGURE 1-2.1-3

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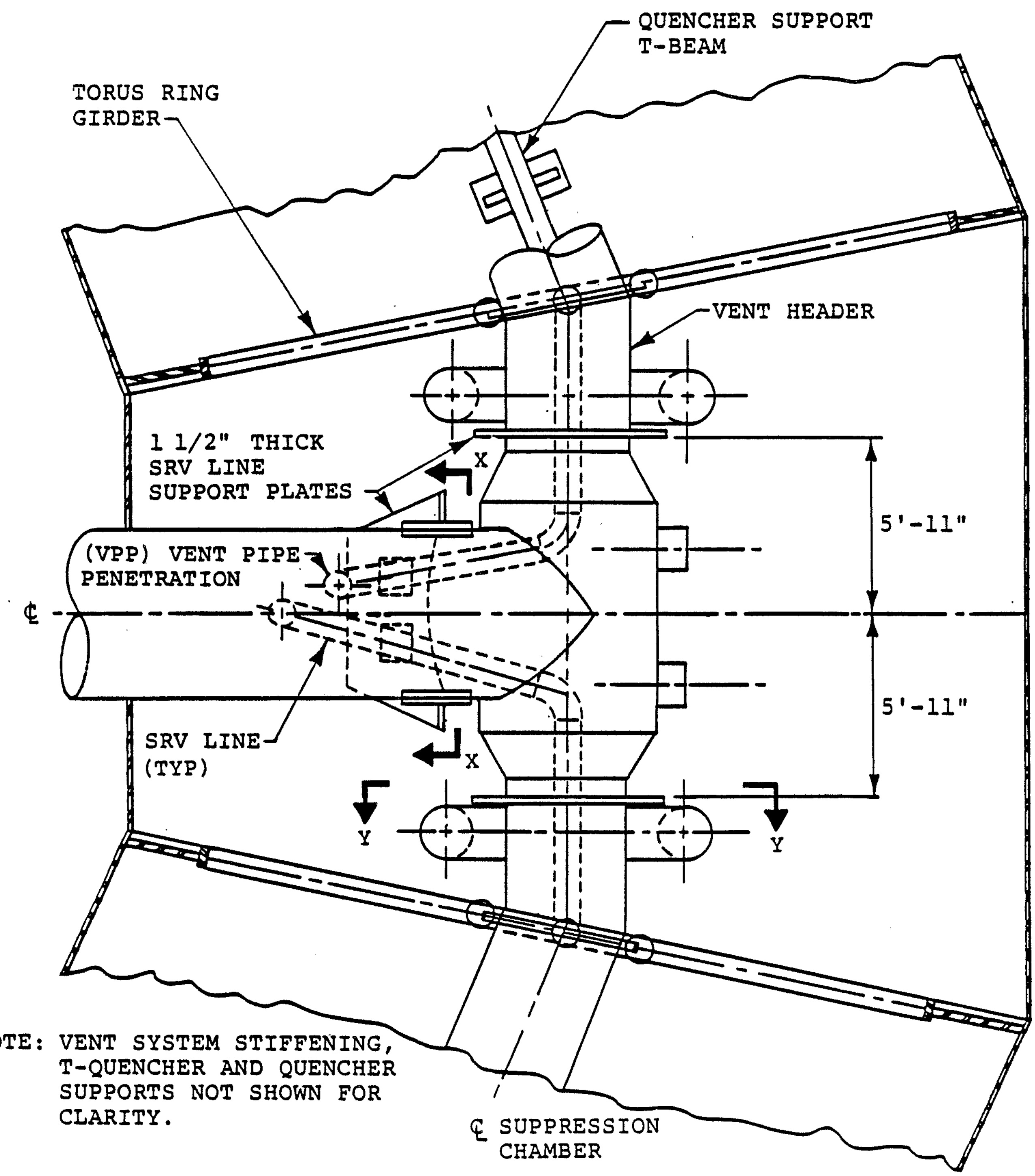
FIGURE 5.2-5

SUPPRESSION CHAMBER SECTION
MIDBAY VENT LINE BAY




NOTE: VENT SYSTEM STIFFENING, T-QUENCHER AND QUENCHER SUPPORTS NOT SHOWN FOR CLARITY.

NUTECH DRAWING NO. DET-20-015-5, REV. 0
PUAR FIGURE 5-2.1-5

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FIGURE 5.2-6

WETWELL SAFETY/RELIEF VALVE LINE ROUTING

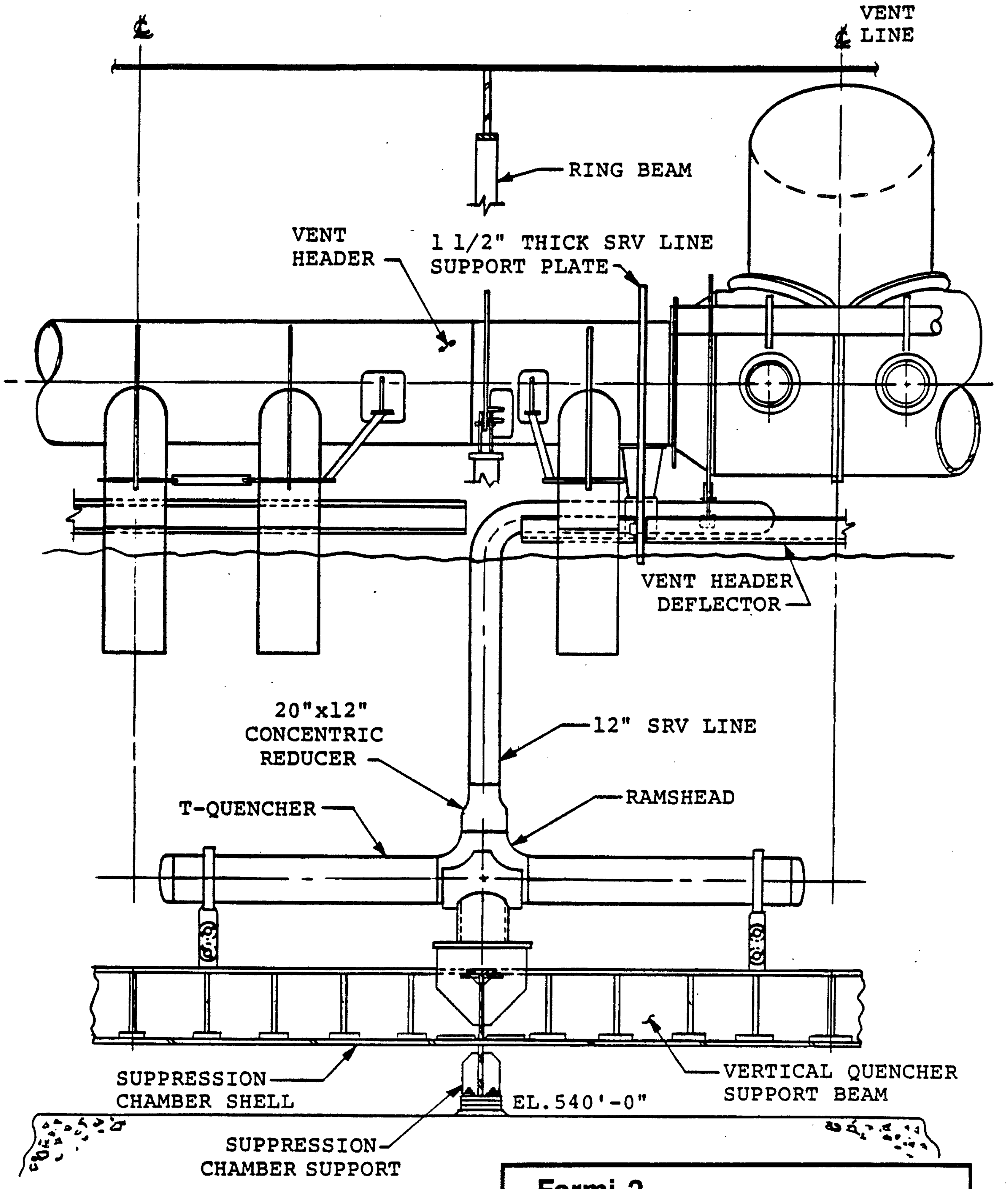


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FIGURE 5.2-7

SAFETY/RELIEF VALVE LINE IN THE SUPPRESSION CHAMBER

NUTECH DRAWING NO. DET-20-015-5, REV. 0
PUAR FIGURE 5-2.1-6

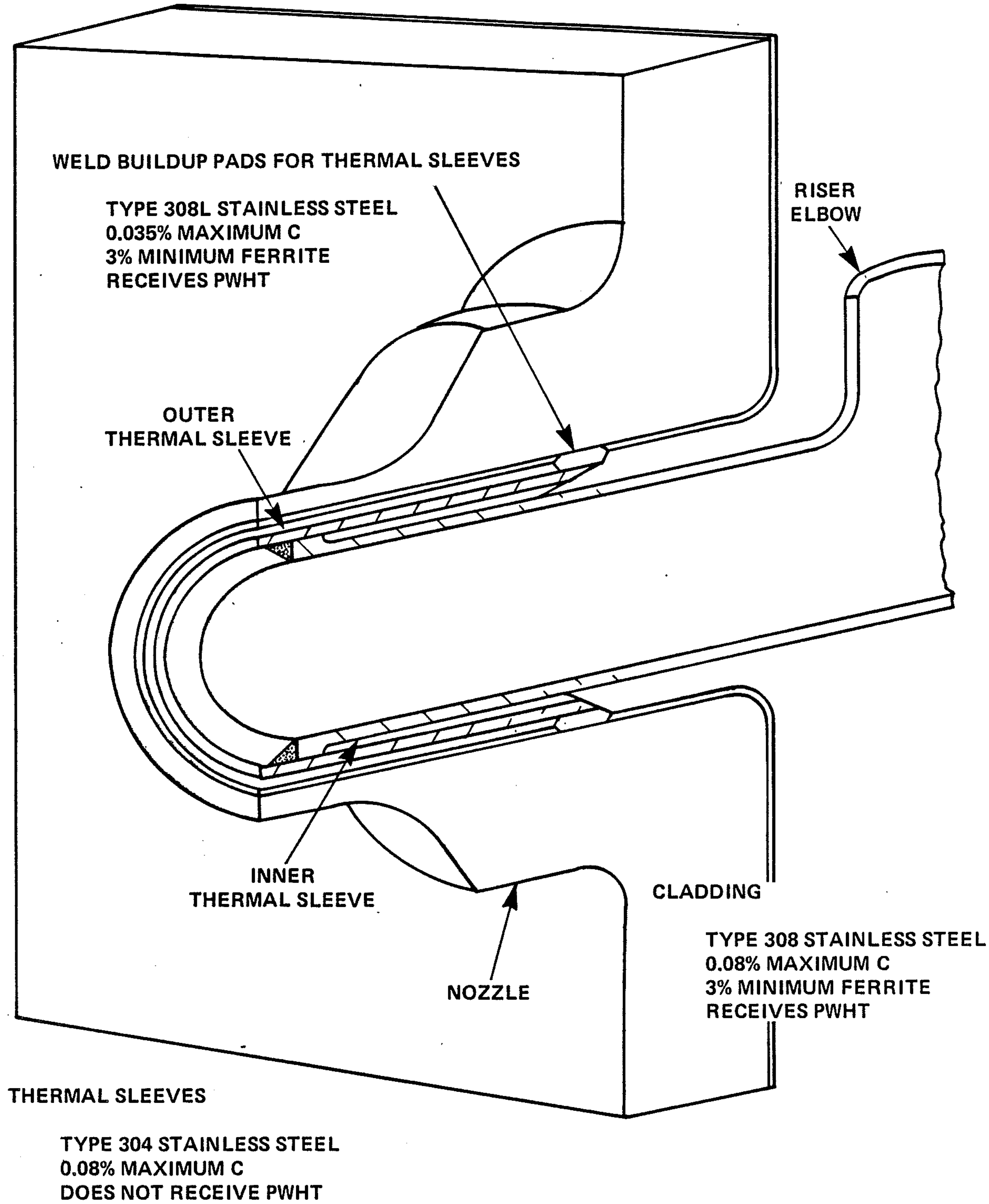


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FIGURE 5.2-8

RECIRCULATION INLET NOZZLE

FIGURE 5.2-9 HAS BEEN DELETED

FIGURE 5.2-10 HAS BEEN DELETED

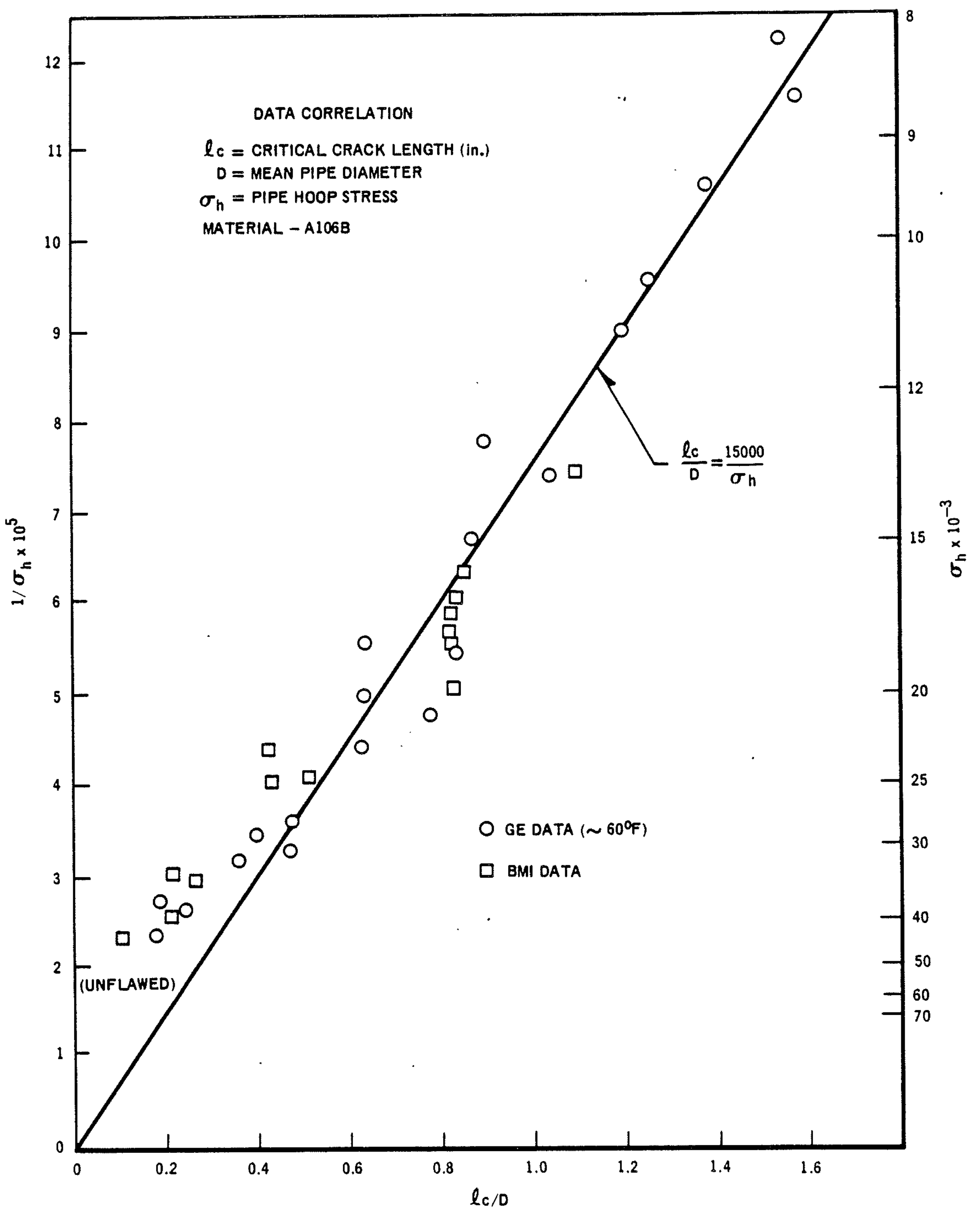


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FIGURE 5.2-11

AXIAL THROUGH-WALL CRACK

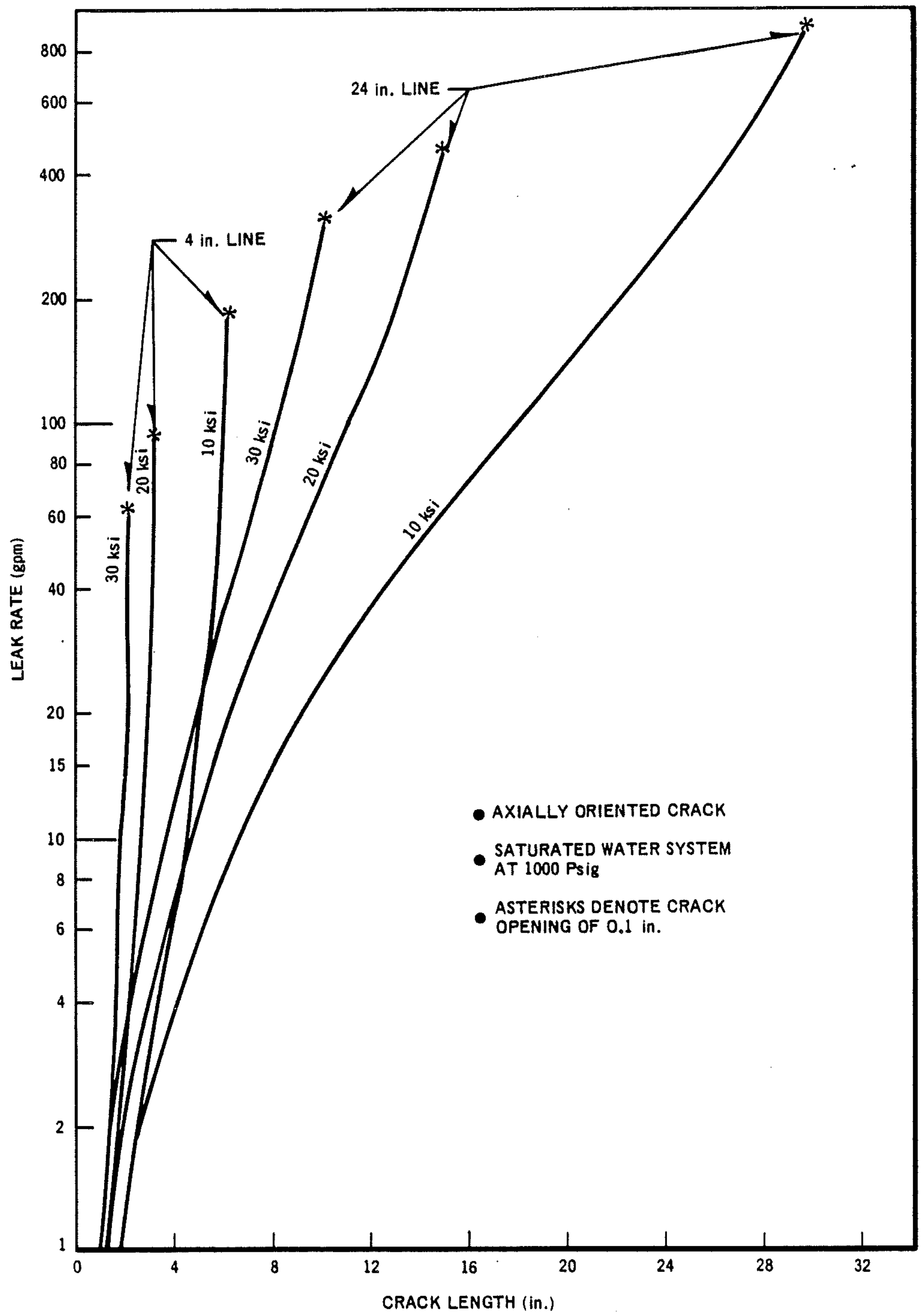


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FIGURE 5.2-12

CALCULATED LEAK RATE AS A FUNCTION OF CRACK LENGTH AND APPLIED HOOP STRESS

## 5.3 THERMAL HYDRAULIC SYSTEM DESIGN

### 5.3.1 Analytical Methods and Data

The analytical methods and thermodynamic and hydrodynamic data used to determine the thermal and hydraulic characteristics of the reactor coolant system are presented in Section 4.4.

### 5.3.2 Operating Restrictions on Pumps

The operating restrictions imposed on the coolant pumps to meet net positive suction head (NPSH) requirements are contained in Subsection 4.4.3.

### 5.3.3 Power-Flow Operating Map

A power-flow operating map that indicates the permissible operating range is contained in Subsection 4.4.3.

### 5.3.4 Load-Following Characteristics

The load-following characteristics are described in Subsection 4.4.3.

### 5.3.5 Transient Effects

The transient effects are presented in Chapter 15.

### 5.3.6 Thermal and Hydraulic Characteristics Summary Table

Thermal and hydraulic characteristics are summarized and compared in Table 4.4-1.

## 5.4 REACTOR PRESSURE VESSEL AND APPURTENANCES

### 5.4.1 Protection of Closure Studs

The Fermi 2 design and inspection procedures are in conformance with the requirements of Regulatory Guide 1.65 except those in regulatory positions 2b, 2e, and 3.

Studs were examined in accordance with the requirements of ASME Boiler and Pressure Vessel (B&PV) Code Section III, N-325 (1968 edition including Summer 1969 Addenda in effect at time of contract). Bored blank nuts were ultrasonically examined by both the longitudinal and shear wave methods. Shear wave examination on the nuts was performed in both the axial and circumferential directions.

Regulatory position 3 recommends provision for adequate corrosion protection during venting and filling of the vessel, and while the head is removed. General Electric supplies thread protectors that prevent stud damage, but stud holes are not plugged, and neither stud nor flange threads are protected from exposure to water. In practice this has been found to be adequate for studs complying with Regulatory Guide 1.65 Regulatory Position 1 & 2, as exposure to applied loads and operating and servicing environments has not required the replacement of any BWR studs (which were in compliance as stated above) or flange threads. No corrosion protection for studs is proposed.

### 5.4.2 Special Processes for Fabrication and Inspection

The product forms of the materials used to fabricate the reactor pressure vessel (RPV) are as follows.

| Vessel Part | Product Form |
|---|---|
| Cylindrical shell | Rolled plate |
| Heads | Rolled plate |
| Main closure flanges | Forged rings |
| Closure bolting | High-strength bolting |
| Nozzles | Forgings |
| Nozzle safe ends | Forgings (stainless steel) |
| Nozzle safe ends | Forgings (carbon steel) |

The rolled plate for vessel shells and head section was hot- formed, quenched, and tempered. These sections were welded into four rings for the vessel shell, and sections were welded to make up the top and bottom head. For a typical shell ring, the sequence of assembly is to weld the longitudinal seams, clad the inside diameter, and finally weld in the nozzles. The methods of fabrication used on the Fermi 2 reactor vessel are all allowed by the ASME B&PV Code and are not considered special or unusual.

From the standpoint of vessel inspection, normal radiographic techniques were used for the inspection of welds during fabrication. In addition, a preservice volumetric inspection using ultrasonic techniques was conducted in the fabrication shop.

This inspection was carried out in accordance with Section XI of the ASME Code, 1970 edition including winter 1971 addenda.

### 5.4.3 Features for Improved Reliability

No special features are incorporated in Fermi 2 that were not used before.

### 5.4.4 Quality Assurance Surveillance

The RPV was fabricated for GE by Combustion Engineering and was subject to Edison's QA audit.

Quality Assurance surveillance procedures were used to ensure that purchased material, equipment, and services associated with the RPV and appurtenances conformed to the requirements of the purchase documents. These procedures included provisions for source evaluation and selection, objective evidence of quality, inspection at the vendor source, and examination of the RPV upon delivery at the construction site.

### 5.4.5 Materials and Inspections

The materials that were used in the RPV are shown in Table 5.2-6.

The RPV was subject to the inspection requirements in accordance with the ASME B&PV Code Section III, 1968 edition with addenda through summer 1969, and the UT inspection discussed in Subsection 5.4.2.

The ASME Code Section XI baseline (preservice) inspection of the reactor vessel has been completed. One hundred percent of all RPV welds are included in the baseline. The Authorized Inspector has certified this inspection. The examination was conducted in the manufacturer's shop. It was completed on May 25, 1974.

At the site, during the preservice examination of piping, a new baseline was obtained on certain vessel welds because the inservice inspection program requires them to be examined from a surface different than the shop examination. These are

a. Top girth seam weld of head flange to reactor shell

b. Nozzle inter-radius areas.

At the time of fit and function of the mechanical equipment for the inservice inspection work, several typical areas were compared with the baseline data for assurance of baseline validity and reproducibility.

### 5.4.6 Reactor Pressure Vessel Design

#### 5.4.6.1 Safety Design Bases

Design of the RPV appurtenances meets the following safety design bases.

a. The RPV and appurtenances shall withstand adverse combinations of loading and forces resulting from operation under abnormal and accident conditions

b. To minimize the possibility of brittle fracture of the nuclear system process barrier, the following is required.

1. Maximum impact properties at temperatures related to RPV operation shall be specified for materials used in the RPV

2. Expected shifts in nil ductility transition temperature (NDTT) during design life as a result of environmental conditions, such as neutron flux, shall be considered in the design. Operational limitations ensure that NDTT shifts are accounted for in reactor operation

3. Operational margins to be observed with regard to the NDTT shall be specified for each mode of operation.

5.4.6.2 Power Generation Design Basis

Design of the RPV and appurtenances meets the following power generation design basis:

a. The RPV shall be designed for an operational life of 40 years (Refer to Appendix B for evaluations of 60 years)

b. External and internal supports that are integral parts of the RPV shall be located and designed so that stresses in the RPV and supports that result from reactions at these supports are within ASME Code limits

c. Design of the RPV and appurtenances shall allow for a suitable program of inspection and surveillance.

5.4.6.3 Description

5.4.6.3.1 Reactor Pressure Vessel

The RPV, shown in Figure 5.4-1, is a vertical, cylindrical pressure vessel with hemispherical heads of welded construction. The vessel design data are listed in Table 5.4-1. The RPV operating thermal cycles are listed in Table 5.2-2. The RPV is designed, fabricated, tested, inspected, and stamped in accordance with the ASME B&PV Code Section III, 1968, Class 1, up to and including summer 1969 addenda. Design of the RPV and its support system meets Category I equipment requirements.

The cylindrical shell and bottom head of the RPV are fabricated of low-alloy steel, the interior of which is clad with stainless- steel weld overlay. Internal surfaces of nozzles that connect to stainless-steel pipe are also clad.

Inplace annealing of the RPV because of radiation embrittlement is unnecessary, as described in Subsection 5.2.4.5.

Quality Assurance methods used during the fabrication and assembly of the RPV and appurtenances ensure that design specifications are met.

The RPV top head is secured to the RPV by studs and nuts. These nuts are tightened with a stud tensioner. The RPV flanges are sealed with two concentric metal seal rings. To detect seal failure, a vent tap is located between the two seal rings. A monitor line is attached to the tap to provide an indication of leakage from the inner seal ring.

Thermocouples are located on the exterior of the RPV. In some cases, thermocouples are attached to the RPV by magnets. At other thermocouple locations, two pads are provided. One is an end pad to hold the end of a 3/16-in. diameter thermocouple, and the other is a clamp pad equipped with a set screw to secure the thermocouple. These thermocouple locations provide a means of observing RPV temperature in response to changes in RPV coolant flow rate. Because RPV metal thickness and the thermal time constant cause the temperature of the RPV surface to lag the coolant temperature, measurements of surface temperature do not afford an effective means of monitoring thermal stresses in the RPV.

Procedural controls on plant operation are necessary to hold these thermal stresses within acceptable ranges. These restrictions on coolant temperature are

a. The average rate of change of reactor coolant temperature during normal heatup and cooldown shall not exceed 100°F during any 1-hr period

b. The RRS pumps shall not be operated unless the coolant temperatures in the upper and lower regions of the RPV are within 145°F of each other

c. The pump in an idle reactor recirculation system (RRS) loop shall not be started unless the coolant temperature in that loop is within 50°F of reactor coolant temperature.

The limit regarding the normal rate of heatup and cooldown described in Item a. ensures that the RPV closure, closure studs, RPV support skirt, control rod drive (CRD) housing, and stub tube stresses and usage remain within acceptable limits. The RPV temperature limit on RRS pump operation restriction described in Item b. augments the Item a. limit in further detail by ensuring that the RPV bottom head region will not be warmed at an excessive rate caused by rapid sweepout of cold coolant in the RPV lower head region by RRS pump operation. Cold coolant can accumulate as a result of CRD inleakage and/or low recirculation flow rate during startup or hot standby. The Item c. limit further restricts operation of the RRS pumps to avoid high thermal stress effects in pumps and piping, while also minimizing thermal stresses on the vessel nozzles.

#### 5.4.6.3.2 Shroud Support

The shroud support is a circular plate welded to the RPV wall. This support is designed to carry the weight of the shroud, shroud head, core support plate, top guide, steam separators, and jet pump system, and to laterally support the fuel assemblies. Design of the shroud support also accounts for pressure differentials across the shroud support plate, for the restraining effect of components attached to the support, and for earthquake loadings. The shroud support design is specified to meet appropriate ASME Code stress limits.

#### 5.4.6.3.3 Reactor Pressure Vessel Supports

5.4.6.3.3.1 Vessel Support Assembly

The RPV support assembly consists of a ring girder, sole plates, and the various bolts, shims, and set screws necessary to position and secure the assembly between the RPV support skirt and the support pedestal. The concrete and steel support pedestal is constructed integrally with the building foundation. Steel anchor bolts are set in the concrete with the threads extending above the surface. The sole plates are bolted to the underside of the RPV ring girder. The sole plate-ring girder assembly is set, leveled, and grouted to the top of the RPV pedestal.

The anchor bolts extend through both the sole plates and the ring girder bottom flange. High-strength bolts are used to bolt the flange of the RPV support skirt to the top flange of the ring girder. The ring girder is ASTM A-36 and the sole plates ASTM A-588 structural steel, both fabricated according to appropriate AISC Specifications.

The top of the pedestal is haunched slightly on the inside to accommodate the anchor bolts. The haunch size has been kept to a minimum to reduce stress concentrations. Reinforcing steel has been provided completely encircling the anchorage area of the bolts to transfer the bolt loads into the main part of the pedestal. The reinforcing details for the haunch have been reviewed and approved by the AEC (Reference 1).

5.4.6.3.3.2 Reactor Pressure Vessel Stabilizers

The RPV stabilizers are designed to permit radial and axial vessel expansion, to limit horizontal vibration, and to resist seismic and jet reaction forces. The stabilizers are connected between the RPV and the top of the shield wall surrounding the RPV to provide lateral stability for the upper part of the RPV. Eight stabilizer brackets are attached by full-penetration welds to the RPV at evenly spaced locations around the RPV below the flange. Each RPV stabilizer consists of a stabilizer rod threaded at the ends, springs, washers, a nut, a plate, and a bumper bracket with tapered shims. The stabilizers are attached to each bracket and apply tension in opposite directions. The stabilizers are evenly preloaded with tensioners to the values of the residual loads.

5.4.6.3.4 Control Rod Drive Housings

The CRD housings are inserted through the CRD penetrations in the RPV bottom head and are welded to stub tubes extending into the RPV. Each housing transmits a number of loads to the bottom head of the reactor. These loads include the weights of a control rod, a CRD, a control rod guide tube, a four-lobed fuel support piece, and the four fuel assemblies that rest on the fuel support piece. These loads are taken into account in designing the bottom head of the reactor. The housings are fabricated of type 304 austenitic stainless steel.

5.4.6.3.5 In-Core Neutron Flux Monitor Housings

Each in-core neutron flux monitor housing is inserted through the in-core penetrations in the bottom head of the RPV and is welded to the inner surface of the bottom head.

An in-core flux monitor guide tube is welded to the top of each housing, as described in Subsection 4.2.2. Either a source range monitor/intermediate range monitor (SRM/IRM) drive unit or a local power range monitor (LPRM) is bolted to the seal- ring flange at the bottom of the housing, as described in Subsection 4.2.2.

#### 5.4.6.3.6 Refueling Bellows

The refueling bellows forms a seal between the RPV and the surrounding primary containment drywell to permit flooding of the space (reactor well) above the RPV during refueling operations. The refueling bellows assembly consists of a type 304 stainless steel bellows, a backing plate, a spring seal, and a removable guard ring. The backing plate surrounds the outer circumference of the bellows to protect it and is equipped with a tap for testing and for monitoring leakage. The self energizing spring seal is located in the area between the bellows and the backing plate. This seal is designed to limit water loss in the event of a bellows rupture by yielding to make a tight fit to the backing plate. This seal is designed to limit water loss in the event of a bellows rupture by yielding to make a tight fit to the backing plate when subjected to full hydrostatic pressure. The guard ring attaches to the assembly and protects the inner circumference of the bellows. The guard ring can be removed from above to inspect the bellows. The assembly is welded to the reactor bellows support skirt and the reactor well seal bulkhead plate. The reactor bellows support skirt is welded to the RPV shell flange. The reactor well seal bulkhead plate bridges the distance to the primary containment drywell wall. This plate contains eight 12-in. holes for air circulation and two 30-in. holes for manways. Each hole is equipped with a watertight cover. For normal operation, the covers on the eight 12-in. holes are opened and removable air supply ducts are inserted into four of them. For refueling operations, all holes are covered.

IE Bulletin 84-03, Refueling Cavity Water Seal, was reviewed by Edison and deemed not to be applicable, since the design of the bellows described above differed markedly from the seal that failed and was reported in IE Bulletin 84-03.

#### 5.4.6.3.7 Reactor Pressure Vessel Insulation

The RPV insulation has an average maximum heat transfer rate of approximately 0.2-Btu/hr/ft$^2$/°F at the operating conditions of 550°F for the RPV and 134°F for the drywell air. The insulation panels for the cylindrical shell of the RPV are held in place by the sacrificial shield. The insulation is designed to be removable where inspection is required for inservice inspection. Shell course welds will be inspected remotely.

#### 5.4.6.3.8 Reactor Pressure Vessel Nozzles

All piping connected to the RPV nozzles, including instrument piping, has been designed so as not to exceed the allowable loads on any nozzle.

The RPV nozzles are low-alloy steel forgings made in accordance with the ASME Code SA-508, Class 2. Nozzles of nominal size larger than 2 in. are full-penetration welded to the vessel. Nozzles of 2 in. nominal size and under may be partial-penetration welded, as permitted by ASME B&PV Code Section III. Nozzles that are partial-penetration welded are low-alloy steel or carbon steel forgings made in accordance with ASME Code SA-508, SA-105, or SA-106.

The RPV top head nozzles are provided with flanges with small groove facing. The drain nozzle is of the full-penetration weld design and extends below the bottom outside surface of the RPV. The RRS inlet nozzles (Figure 5.4-1), the feedwater inlet nozzles, and the core spray inlet nozzles all have thermal sleeves. For more information on the feedwater sparger and thermal sleeve design, see Subsection 5.2.1.20.

Nozzles connected to stainless piping have safe ends made of stainless steel. These safe ends are normally welded to the nozzles after the RPV has been heat treated to avoid furnace sensitization of the stainless steel. The material used is compatible with the material of the mating pipe. For more information on the safe ends, see Subsection 5.2.3.2.

The nozzle for the core differential pressure and liquid control pipe is designed with a transition so that the stainless-steel outer pipe of the differential pressure and liquid control line can be socket welded to the inner end of the nozzle and so that the inner pipe passes through the nozzle. This design provides an annular region between the nozzle and the inner liquid control line to minimize thermal shock effects on the RPV in the event that use of the standby liquid control system (SLCS) is required.

#### 5.4.6.4 Safety Evaluation

The RPV design pressure of 1250 psig is based on an analysis of margins. The margins include additional allowances to accommodate transients above the operating pressure without initiating safety valve action. The RPV design temperature of 575°F is based on the saturation temperature of water that corresponds to the design pressure.

To withstand external and internal loadings while maintaining a high degree of corrosion resistance, a high-strength, low-alloy steel is used as the base metal, and an internal cladding of stainless steel is applied using weld overlay. Use of ASME B&PV Code Section III, Category I, RPV design criteria ensures that a vessel designed, built, and operated within its design limits has an extremely low probability of failure as a result of any known failure mechanism.

Stress analysis and load combinations for the RPV were evaluated for the cycles listed in Table 5.2-2, with the conclusion that ASME Code limits are satisfied.

### 5.4.7 Reactor Pressure Vessel Schematic

The RPV schematic is contained in Figure 5.4-2. The relation of the RPV to the biological shield is shown in Figure 5.1-4. Normal water level and high and low levels for alarm and trip are shown in Figure 7.3-12.

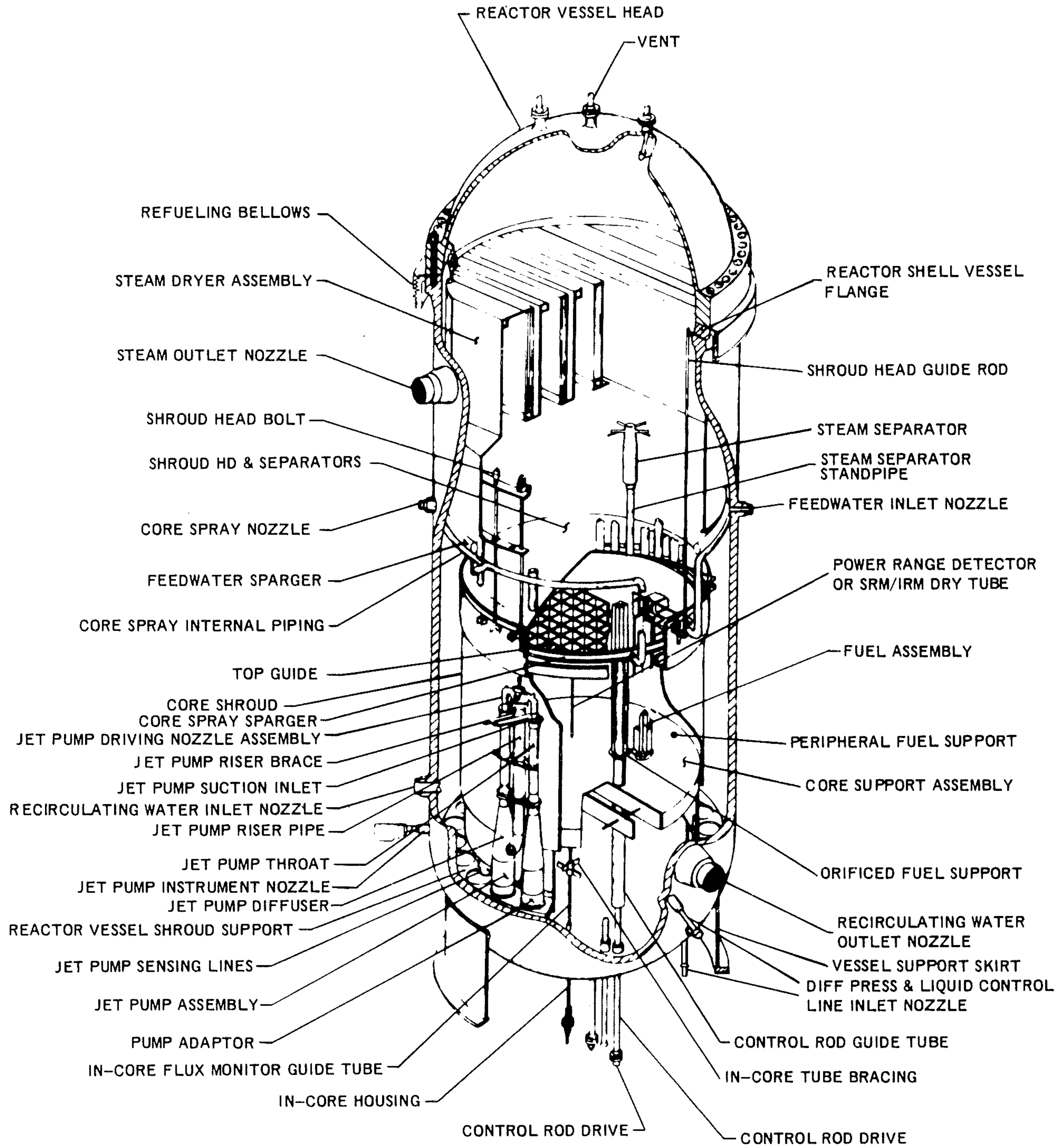
## 5.4 REACTOR PRESSURE VESSEL AND APPURTENANCES

## REFERENCES

1 Letter from V. M. Moore, AEC, to C. M. Heidel, Edison, Subject: Review of Design of Biological Shield (Fermi 2 Post Construction Permit Open Item No. 12), dated October 29, 1973.

TABLE 5.4-1 REACTOR PRESSURE VESSEL DESIGN DATA

| | |
|---|---|
| Reactor pressure vessel | |
| Inside diameter, in. (minimum) | 251 |
| Inside length (including closure head), ft. | 72 |
| Design pressure and temperature, psig @°F | 1250 @ 575 |
| Reactor pressure vessel support | |
| Design mechanical loads shear, kips | 1300 |
| Design mechanical loads moment, in.-kips | 576,000 |
| Vessel nozzles | Number/Size (in.) |
| Recirculation outlet | 2/28 |
| Steam outlet | 4/26 |
| Recirculation inlet | 10/12 |
| Feedwater inlet | 6/12 |
| Core spray inlet | 2/10 |
| Instrument (spare) | 2/6 |
| Control rod drive | 185/6 |
| Jet pump instrumentation | 2/4 |
| Vent | 1/4 |
| Instrumentation | 6/2 |

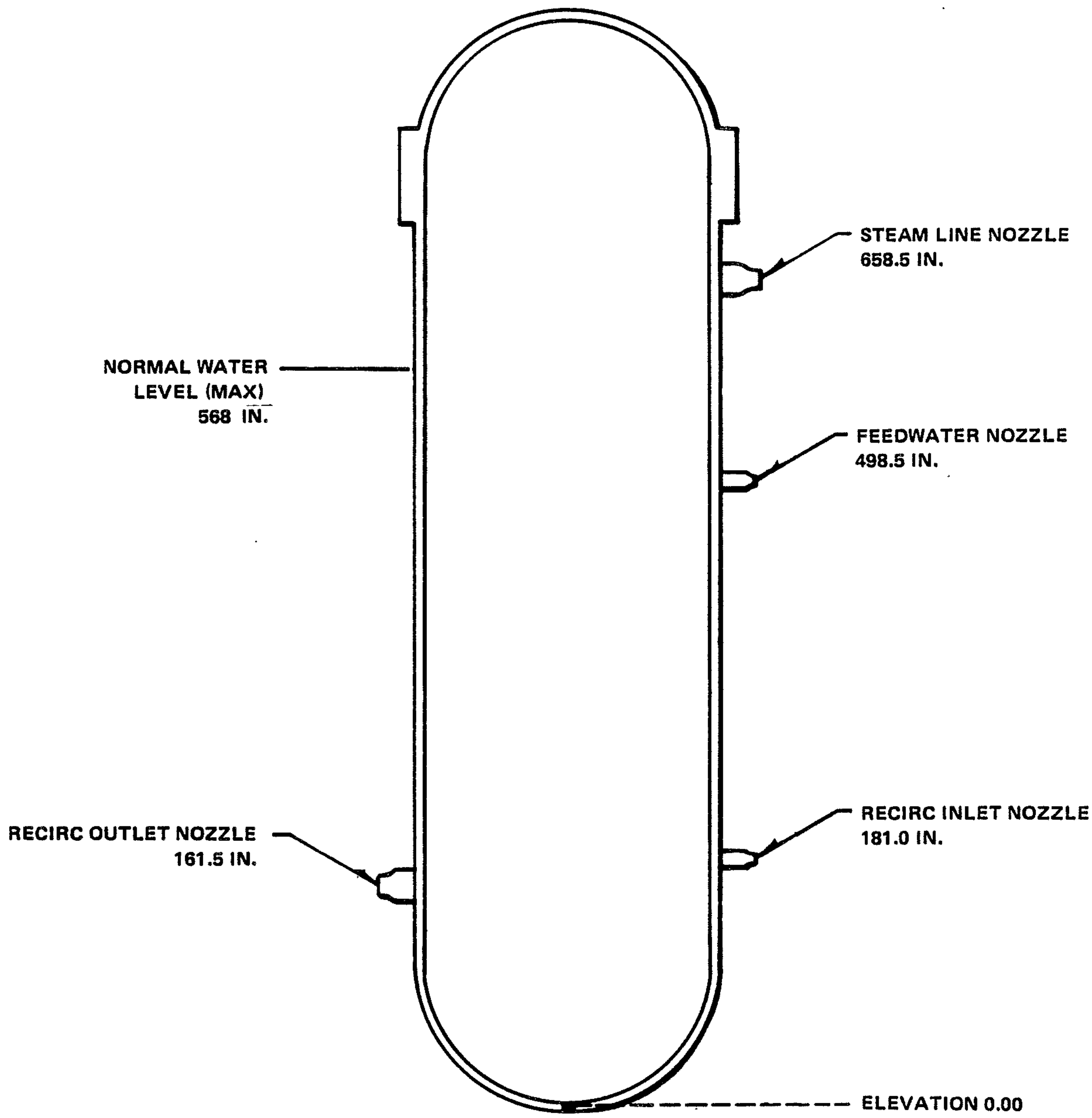



GENERAL ELECTRIC CO. DRAWING

**Fermi 2**
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 5.4-1

REACTOR VESSEL CUTAWAY




**Fermi 2**

UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 5.4-2

REACTOR PRESSURE VESSEL SCHEMATIC

## 5.5 COMPONENT AND SUBSYSTEM DESIGN

This section presents discussions of the performance requirements and design features to ensure overall safety of the various components within the reactor coolant pressure boundary (RCPB) and those subsystems closely allied with the reactor coolant system but not a portion of the RCPB. The subsystems and components discussed in this section are the reactor core isolation cooling (RCIC) system; residual heat removal (RHR) system; reactor water cleanup (RWCU) system; main steam lines, feedwater piping, and drains, valves, and component supports. The portions of these subsystems which are within the RCPB are discussed in Subsections 5.5.1 through 5.5.4.

### 5.5.1 Reactor Recirculation System and Pumps

#### 5.5.1.1 Safety Design Bases

The reactor recirculation system (RRS) is designed to meet the following safety design bases.

a. An adequate fuel barrier thermal margin shall be ensured during postulated transients

b. A failure of piping integrity shall not compromise the ability of the reactor pressure vessel (RPV) internals to provide a refloodable volume

c. The RRS shall maintain pressure integrity during adverse combinations of loadings and forces occurring during abnormal, accident, and special event conditions.

#### 5.5.1.2 Power Generation Design Bases

The RRS meets the following power generation design bases:

a. The RRS shall provide sufficient flow to remove heat from the fuel

b. System design shall minimize maintenance situations that would require core disassembly and fuel removal.

#### 5.5.1.3 Description

The RRS consists of the two RRS pump loops external to the RPV. These loops provide the piping path for the driving flow of water to the RPV jet pumps, as shown in Figures 5.5-1 and 5.5-2. Each external loop contains one variable-speed motor-driven RRS pump, two motor-operated gate valves, and a motor-generator set to control RRS pump speed. Each pump discharge line contains a venturi-type flow meter nozzle.

The RRS loops are part of the nuclear system process barrier and are located inside the primary containment structure. The jet pumps are RPV internals. Their location and mechanical design are discussed in Subsection 4.5.1.2.7. However, certain operational characteristics of the jet pumps are discussed in this subsection. Table 5.5-1 summarizes the design characteristics of the RRS.

The recirculated coolant consists of saturated water from the steam separators and dryers that has been subcooled by incoming feedwater. This water passes down the annulus between the RPV wall and the core shroud. A portion of the coolant flows from the RPV, through the two external RRS loops, and becomes the driving flow for the jet pumps. Each of the two external RRS loops discharges high pressure flow into an external manifold from which individual recirculation inlet lines are routed to the jet pump risers within the RPV. The remaining portion of the coolant mixture in the annulus becomes the driven flow for the jet pumps. This flow enters the jet pump at suction inlets and is accelerated by the driving flow. The flows, both driving and driven, are mixed in the jet pump throat section and result in partial pressure recovery. The balance of recovery is obtained in the jet pump diffusing section shown in Figure 5.5-3. The adequacy of the total flow to the core is discussed in Subsection 4.4.3. Documented tests show that the jet pump design is sound and that jet pump operation is stable and predictable.

The original design for Fermi 2 included a 4-in. bypass line around each pump discharge valve. The line was to be used during the startup of a loop to equalize the pressure across the discharge valve, to preheat the piping loop by reverse flow, and to prevent the pump from overheating prior to opening the discharge valve. Operating plants have found this line to be very susceptible to intergranular stress corrosion cracking. General Electric has developed a circuit for controlling the opening of the discharge valve that eliminates the need for the bypass line. Employment of this circuit enables the removal of the bypass line.

Based on this experience at other plants, the decision was made not to install the 4-in. bypass lines on Fermi 2 and to incorporate the controlled opening (jogging) circuit. Caps are welded onto the bypass line tees.

There is a very low probability that a RRS loop that has been allowed to cool would need to be placed in service again when the nuclear system is hot. The only valid reason for closing both the pump discharge valve and the suction valve is to prevent leakage out of that portion of the RRS loop between the valves; e.g., excessive leakage through the pump mechanical seal. A leak of this nature cannot be repaired without shutting the plant down to permit access to the drywell. The nuclear system would, in all probability, be cooled prior to repairing the leak.

Since the removal of RRS valve internals without alternate isolation capability requires unloading of the nuclear fuel, the valves are provided with high-quality back seats and a trim to facilitate stem-packing renewal and to provide adequate leaktightness. Alternative RRS loop isolation devices (plugs) have been approved for use only during Mode 5 to support maintenance activities without unloading the nuclear fuel.

The feedwater flowing into the RPV annulus during operation provides subcooling for the fluid passing to the RRS pumps, thus determining the additional net positive suction head (NPSH) available beyond that provided by the pump location below the RPV water level. If feedwater flow is below the minimum value that provides adequate NPSH for full speed RRS pump operation, the pump speed is automatically limited. This limit is chosen to prohibit pump cavitation. Operation with the suction pressure available only from the RPV provides adequate NPSH.

The RRS pumps can be operated during nuclear steam supply system (NSSS) heatup for hydrostatic tests. At this time, they act in conjunction with any contribution from reactor core decay heat to raise NSSS temperature above the limit imposed on the RPV by nil ductility transition temperature (NDTT) considerations so that the hydrostatic test can be conducted.

Each RRS pump is a single-stage, variable-speed, vertical, centrifugal pump equipped with mechanical shaft seal assemblies. In order to preclude shaft cracking due to thermal stress, the pumps have been upgraded to the 4$^{th}$ generation design.

The pump is capable of stable and satisfactory performance while operating continuously at any speed corresponding to a power supply frequency range from 11.5 to 57.5 Hz. For loop startup, each pump operates at a speed corresponding to a power supply frequency of 11.5 Hz.

Each RRS pump motor is a standard ac induction motor which is operated as a variable-speed pump driver by using a variable frequency power supply. The power supply is provided by a motor-generator set with a fluid coupler which allows continuous generator speed adjustment so that the output power frequency may be varied from 11.5 to 57.5 Hz. The pump motor design is capable of operating at any speed within the power supply frequency range corresponding to a pump speed control range from 20 percent to 102 percent rated pump speed. The electrical equipment is designed, constructed, and tested in accordance with the applicable sections of the NEMA Standards.

The variable-frequency ac motor-generator sets for both RRS pumps are located outside the drywell. The pump motors are electrically connected to the generators. Pump start begins when the generator excitation field breaker of the motor-generator set is closed.

The RRS pump shaft seal assembly consists of two individual seals built into a cartridge or cartridges, which can be readily replaced without removing the motor from the pump. The seal assembly is designed to require minimum maintenance over a long period of time, regardless of whether the pump is stopped or is operating at various speeds, with water at various pressures and temperatures. Each individual seal in the cartridge is capable of sealing against pump design pressure so that any one seal can adequately limit leakage in the event that the other seal should fail. Reduced clearances in the pump casing reduce leakage in the event of a gross failure of both shaft seals. Leakage due to massive seal failure will remain insignificant as compared to the available makeup supply. Provision is made for monitoring the pressure drop across each individual seal as well as the cavity temperature of each seal. Provision is also made for piping the seal leakage to a flow measuring device.

The effective inertias of the RRS pump and motor, motor-generator set, and variable speed coupling are specified in the following form, which takes into account the torque and speed conditions on each rotating shaft.

$$\sum_{\substack{\text{ALL}\\\text{SHAFTS}}} \frac{\text{Inertia}\left(\text{lb}-\text{ft}^2\right)\text{x Speed (radian/sec)}}{\text{g}\left(\frac{\text{ft}}{\text{sec}^2}\right)\text{xTorque (ft}-\text{lb)}}$$

The design objective for the RRS pump is to provide a unit that will not require removal from the system for rework or overhaul at intervals of less than one operating cycle. Pump casing overhaul and valve bodies are designed for a 40-year operational life. The pump drive motor, impeller, and wear rings are designed for as long a life as is practical. Pump mechanical seal parts are expected to have a life exceeding one operating cycle to afford convenient replacement during refueling outages.

The RRS piping is of all-welded construction and is designed and constructed to meet the requirements of the ANSI B31.7 Nuclear Power Piping Code-1969, Class 1.

The RRS is designed as Category I. The pump is assumed to be filled with water for the analysis. Vibration snubbers located at the top of the motor and at the bottom of the pump casing are designed to resist the horizontal reactions.

The RRS piping, valves, and pumps are supported by hangers to avoid the use of piping expansion loops that would be required if the pumps were anchored. In addition, the RRS loops are provided with a system of restraints designed so that reaction forces associated with any split or circumferential break do not jeopardize containment integrity. This restraint system provides adequate clearance for normal thermal expansion movement of the loop. Because possible pipe movement is limited to slightly more than the clearance required for thermal expansion movement, no impact loading on limit stops is considered.

The RRS piping, valves, and pump casings are covered with thermal insulation having a total average heat transfer rate of 65 Btu/hr/ft$^2$ with the system at rated operating conditions.

The insulation is the all-metal reflective type and is prefabricated into components for field installation. Removable insulation is provided at various locations to permit periodic inspection of equipment and inservice inspection access to components (Subsection 5.2.3.3).

#### 5.5.1.4 Safety Evaluation

RRS malfunctions that pose threats of damage to the fuel barrier are described and evaluated in Subsections 15.3.1 and 15.3.3. It is shown in Subsections 15.3.1 and 15.3.3 that none of the malfunctions, including pump trip or pump seizure, result in fuel damage.

The core flooding capability of the RRS and the core flooding capability of a jet pump design plant are discussed in detail in Reference 1.

Piping and pump design pressures for the RRS are based on peak steam pressure in the reactor dome, appropriate pump head allowances, and the elevation head above the lowest point in the RRS loop. Piping and related equipment pressure parts are chosen in accordance with applicable codes. Use of the applicable code design criteria listed in Tables 3.9-17, 3.9-18, 3.9-43, and 3.9-44 ensures that a system designed, built, and operated within design limits has an extremely low probability of failure caused by any known failure mechanism.

GE purchase specifications require that reactor coolant pressure boundary (RCPB) integrity of the pump case be maintained through all conditions. In addition, dynamic loads are transmitted by piping suspension system components attached to the motor. The parts of the pump and motor that withstand seismic loads as part of the piping suspension system are the pump lugs, pump case, bolting between the pump case and the motor stand, motor stand, bolts attaching the motor stand to the motor, motor frame and motor seismic lugs.

Analyses performed to determine if the RRS pump can become a missile indicate that, for the postulated full double-ended pipe break LOCA in the RRS pump suction line, destructive pump and motor overspeed could occur (Reference 2). In the event of motor failure, the motor stator and frame structure would prevent the release of any missiles. Given the postulated accident, RRS pump impeller destructive overspeed could occur. However, impeller missiles will not penetrate the pump case (Reference 3). Missiles could be ejected from the open end of the broken pipe. Analyses of the effects of missiles ejected from the broken pipe are contained in Reference 4. Additional piping restraints were added to prevent the potential missile exit points in the pipe from developing.

A comparison of break locations using the Fermi 2 recirculation piping stress report has confirmed that no unacceptable damage consequences can occur as a result of potential recirculation pump missiles.

The consequences of the loss-of-component cooling water to both the recirculation pumps have been evaluated. The cooling water is supplied from the reactor building closed cooling water (RBCCW) system during normal plant operation in all modes. Cooling water to the recirculation pump motors and seals is supplied through the divisional emergency equipment cooling water (EECW) system piping which is routed into the drywell from external supply and return flow tie-ins with RBCCW. Each pump is supplied through a different EECW piping division so that both pumps cannot simultaneously lose component cooling, except by closure of both divisions of the EECW supply line outboard isolation valves on an ECCS high drywell pressure signal. High drywell pressure would also cause a reactor protection system signal to initiate a reactor trip and to close the RRS pump seal purge supply flow drywell isolation valves.

If there were a gradual loss of cooling water to the pump motor, the following sequence of alarms would come into the control room.

a. Motor bearing oil cooling water discharge

b. Motor thrust bearing lower face

c. Motor thrust bearing upper face

d. Upper guide bearing

e. Motor windings

f. Lower guide bearing

A loss of RBCCW/EECW flow for pump seal cooling will also cause a low flow alarm to annunciate in the control room. Alarms would also come into the control room through the recirculation pump motor temperature recorder. As these alarms start to come in, the operator would respond by dropping the power level and changing the flow rates to minimize the transient in case it were to become necessary to trip the overheated pump. If the operator were to receive confirmation that the pump motor bearings or the pump seals were overheating, he would trip the pump.

On a sudden loss of cooling water to the pump motor, as could occur on high drywell pressure isolation of the EECW supply line, the motor bearings would begin to incur damage after 90 seconds of full speed operation. As the bearings fail, the pump motor trip would occur from an overcurrent protective relay opening when the loss of rotor stability causes the rotor to contact the stator. This would occur within 2 to 3 minutes from the loss of cooling.

The high drywell pressure isolation would also cause the secondary cooling supply to the pump shaft seals by the seal purge flow to be cut off. During the continued operation of the pump motor, the seals would be protected by the residual cooling capacity of the cooler. The bearing failure will not result in damage to the pump shaft seals due to the structural support of the motor and pump. Following the pump trip, seal cavity circulation would be lost and the seal cavity will gradually heat up. If cooling is restored within 10 to 15 minutes, the shaft seals will not be significantly damaged. However, the exposure to higher temperatures will shorten the operable life of the elastomereic components of the seals. If cooling cannot be restored, the resulting seal leakage rate would be 18 gpm loss of reactor coolant. This coolant loss rate is within the capacity of the normal operating and isolation mode plant makeup systems. The fuel thermal limits would not be exceeded and the seal leakage does not lead to further degradation of the RCPB barrier.

#### 5.5.1.5 Inspection and Testing

Quality control (QC) methods are used during fabrication and assembly of the RRS to ensure that design specifications are met. The reactor coolant system is thoroughly cleaned and flushed before fuel is loaded initially.

Prior to the Preoperational Test Program, the RRS was given a hydrostatic test at 125 percent of RPV design pressure. Preoperational tests on the RRS were performed as described in Chapter 14.

During the Startup Test Program, horizontal and vertical motions of the RRS piping and equipment are observed, and supports are adjusted, as necessary, to ensure that components are free to move as designed. The NSSS responses to RRS pump trips at rated temperatures and pressure are evaluated during the startup tests, and plant power response to recirculation flow control is determined.

Inservice inspection, in accordance with ASME Boiler and Pressure Vessel (B&PV) Code Section XI, 1971 edition including winter 1971 addenda was considered in the design of the RRS to ensure adequate working space and access for inspection of selected components. The criteria for selecting the components and locations to be inspected are based on the probability of a defect occurring or enlarging at a given location, including areas of known stress concentrations and locations where cyclic strain or thermal stress might occur. The RRS pump casings, valve bodies, and piping connection welds are visually inspected and given other nondestructive inspections from at least one side on a periodic basis. The inservice inspection program is described in Section 5.2.8.

### 5.5.2 Steam Generators

The steam generators are not applicable to the BWR.

### 5.5.3 Reactor Coolant Piping

The RRS loops are shown in Figures 5.5-1 and 5.5-2. The design characteristics are presented in Table 5.5-1.

### 5.5.4 Main Steam Line Flow Restrictors

5.5.4.1 Safety Design Bases

The main steam line flow restrictors are designed

a. To limit the loss of coolant from the RPV following a steam line rupture outside the primary containment to the extent that the RPV water level does not fall below the top of the core within the time required to close the main steam isolation valves (MSIVs)

b. To withstand the maximum pressure difference expected across the restrictor, following complete severance of a main steam line.

5.5.4.2 Description

A main steam line flow restrictor is provided for each of the four main steam lines, as shown in Figure 5.5-4. The restrictor is a complete assembly welded into the main steam line. It is located between the RPV and the first MSIVs and is downstream of the main steam line safety/relief valves. The restrictor limits the coolant blowdown rate from the RPV in the event a main steam line break occurs outside the primary containment to the maximum (choke) flow specified. The restrictor assembly consists of a venturi-type nozzle insert welded, in accordance with applicable code requirements, into the main steam line. The restrictor assembly is self-draining in that it contains low point pockets which are drained internally to the main steam line. The flow restrictor is designed and fabricated to ANSI B31.7.

The flow restrictor has no moving parts. Its mechanical structure can withstand the velocities and forces associated with a main steam line break. The maximum differential pressure is ASME Code limit pressure. The rated capacity of the RPV pressure relieving devices shall be sufficient to prevent a rise in pressure within the protected vessel of more than 110 percent of the design pressure (1.10 x 1250 = 1375 psig).

The ratio of venturi throat diameter to steam line diameter, approximately 0.55, results in a maximum pressure differential of 10 psi at rated flow. This design limits the steam flow in a severed line to approximately 200 percent rated flow, yet it results in a negligible increase in steam moisture content during normal operation. The restrictor is also used to measure steam flow and to initiate closure of the MSIVs when the steam flow exceeds preselected operational limits.

5.5.4.3 Safety Evaluation

In the event a main steam line should break outside the primary containment, the critical flow phenomenon would restrict the steam flow rate in the venturi throat to 200 percent of the rated value. Prior to isolation valve closure, the total coolant losses from the RPV are not sufficient to cause core uncovering. Thus, the core is adequately cooled at all times. Analysis of the steam line rupture accident shows that the core remains covered with water and that the amount of radioactive materials released to the environs through the main steam line break does not exceed the guideline values of 10 CFR 100. This accident analysis is described in Chapter 15.

Tests on a scale model determined final design and performance characteristics of the flow restrictor. The characteristics include maximum flow rate of the restrictor corresponding to the accident conditions, irreversible losses under normal plant operating conditions, and discharge moisture level. The tests showed that flow restriction at critical throat velocities is stable and predictable.

If moisture forms in the nozzle throat due to a momentary large static pressure reduction, the droplets of wet steam would have to be at saturation temperature corresponding to throat static pressure. When proceeding to the downstream region where vapor temperatures are higher, the droplets of wet steam vaporize somewhat and reach equilibrium with vapor at a lower pressure. The moisture is reduced and actually is negligible. It has negligible corrosion effect on the highly corrosion-resistant material (A-351 stainless steel) used for the inlet and throat sections. High velocity steam also has negligible erosion effect on this material.

The steam flow restrictor is exposed to steam of 1/10 to 2/10 percent moisture flowing at velocities of 150 ft/sec (steam piping inside diameter) to 600 ft/sec (steam restrictor throat). ASTM-A351 (type 304) cast stainless steel was selected for the steam flow restrictor material because it has excellent resistance to erosion-corrosion in this environment.

#### 5.5.4.4 Inspection and Testing

Because the flow restrictor forms a permanent part of the main steam line piping and has no moving components, no testing program is planned. Only very slow erosion will occur with time, and such a slight enlargement will have no safety significance. Stainless steel resistance to corrosion has been substantiated by turbine inspections at the Dresden Unit 1 facility, which have shown no noticeable effects from erosion on the stainless-steel nozzle partitions.

Calculations show that even if the erosion rates are as high as 0.004 in. per year, after 40 years of operation the increase in restrictor-choked flow rate will be no more than five percent (Refer to Appendix B for evaluation of 60 years). A five percent increase in the radiological dose calculated for the postulated main steam line break accident is not significant (Subsection 15.6.4).

### 5.5.5 Main Steam Line Isolation Valves

#### 5.5.5.1 Safety Design Bases

The MSIVs, individually or collectively, meet the following safety design bases.

a. The MSIVs shall close the main steam lines within the time established by design-basis accident analysis to limit the release of reactor coolant

b. The MSIVs shall close the main steam lines slowly enough that simultaneous (inadvertent) closure of all steam lines will not exceed the NSSS design limits

c. The MSIVs shall close the main steam line when required, despite single failure in either valve or in the associated controls, to provide a high level of reliability for the safety function

d. The MSIVs shall use separate energy sources as the motive force to independently close the redundant isolation valves in the individual steam lines

e. The MSIVs shall use local stored energy (compressed air and springs) to close at least one isolation valve in each steam pipeline without relying on the continuity of any variety of electrical power to furnish the motive force to achieve closure

f. The MSIVs shall be able to close the steam lines, either during or after seismic loadings, to ensure isolation if the nuclear primary system is breached

g. The MSIVs shall have the capability for being tested, during normal operating conditions, to demonstrate that the valves will function.

#### 5.5.5.2 Description

Two isolation valves are welded in a horizontal run of each of the four main steam pipes. One valve is as close as possible to the primary containment barrier and inside it, and the other is just outside the barrier. When closed, the valves form part of the nuclear system process barrier for openings outside the containment and part of the pressure barrier for nuclear system breaks inside the containment.

Figure 5.5-5 shows a typical MSIV, which does not necessarily reflect the actual detailed valve configuration utilized at Fermi 2. Each is a 26-in., Y-pattern, globe valve. Design steam flow rate through each valve is $3.72 \times 10^6$ lb/hr. The main disk or poppet is attached to the lower end of the stem. Normal steam flow tends to close the valve, and higher inlet pressure tends to hold the valve closed.

The bottom end of the valve stem closes a small pressure- balancing hole in the poppet. When the hole is open, it acts as a pilot valve to relieve differential pressure forces on the poppet. Valve stem travel is sufficient to give flow areas past the wide open poppet approximately equal to the seat port area. The poppet travels approximately 90 percent of the valve stem travel; approximately the last 10 percent of travel closes the pilot hole. The air cylinder can open the poppet with a maximum differential pressure of 200 psi across the isolation valve in a direction that tends to hold the valve closed.

A 45° angle permits the inlet and outlet passages to be stream-lined. This minimizes pressure drop during normal steam flow and helps prevent debris blockage. The pressure drop at rated flow is approximately 7 psi. The valve stem penetrates the valve bonnet through a stuffing box utilizing a live-loading configuration and graphite packing to help prevent leakage through the stem packing. The live-loading configuration consists of Belleville disc springs installed on the packing gland studs and the packing gland plate. This creates additional elasticity to the loading of the stuffing box packing. When the gland stud nuts are tightened to load the packing, the disc springs are compressed. As the packing consolidates inservice, the springs expand to maintain a relatively constant load on the packing providing a continual inservice adjustment.

Attached to the upper end of the stem is an air cylinder that opens and closes the valve and a hydraulic dashpot that controls its speed. The speed is adjusted by a valve in the hydraulic return line bypassing the dashpot piston. Valve closing time is adjustable to between 3 and 10 sec.

The air cylinder is supported on the valve bonnet by actuator support and spring guide shafts. Helical springs around the spring guide shafts with air pressure from either normal or accumulator sources are required together to close the valve.

The motion of the spring seat member actuates switches at fully open, 90 percent open and fully closed valve positions.

The valve is operated by pneumatic pressure and by the action of compressed springs. The control unit is attached to the air cylinder. This unit contains three types of control valves: pneumatic, ac, and dc. These control valves open and close the main valve and exercise it at slow and fast speed. Remote manual switches in the main control room enable the operator to operate the valves.

Operating air is supplied to the outboard valves from the plant interruptible control air system via accumulators protected by check valves. The accumulator tank between the control valve and the check valve provides a pneumatic reserve for the closing of each valve. Each valve is designed to accommodate saturated steam at 1250 psig and 575°F, with a moisture content of approximately 0.23 percent, an oxygen content of 30 ppm, and a hydrogen content of four ppm.

In the "worst case" condition of the main steam line rupturing downstream of the valve, steam flow would quickly increase to 200 percent of rated flow. Further increase is prevented by the venturi flow restrictor upstream of the valves.

During approximately the first 75 percent of closing, the valve has little effect on flow reduction because the flow is choked by the venturi restrictor upstream of the valves. After the valve is approximately 75 percent closed, flow is reduced as a function of the valve area versus travel characteristic.

The design objective for the valves is a minimum of 40 years of service at the specified operating conditions. Operating cycles are estimated to be 120 startup cycles, 120 shutdown cycles, and 180 scram cycles in the expected 40-year plant life. The valves shall be capable of actuating a minimum of 50 full cycles per year. The result of an updated evaluation for 60 years of projected cycles is contained in Reference 11.

In addition to minimum wall thickness required by applicable codes, a corrosion allowance of 0.120-in. minimum is added to provide for 40 years of service.

Design specification ambient conditions for normal plant operation are 135°F normal temperature, 150°F maximum temperature, 100 percent humidity, in a radiation field of 15 rads per hour due to radiation gamma and 25 rads per hour due to neutron plus gamma radiation, continuous for design life. The inside valves are not continuously exposed to maximum conditions, particularly during reactor shutdown, and valves outside the primary containment and shielding are in ambient conditions that are considerably less severe.

In addition, they are designed to close and remain closed under the post accident environment conditions listed in Table 3.11-1.

To sufficiently resist the response motion from the safe-shutdown earthquake (SSE), the MSIV installations are designed as Category I equipment. The valve assembly is manufactured to withstand the design-basis forces applied at the mass center, assuming the cylinder/spring operator is cantilevered from the valve body and the valve is located in a horizontal run of pipe. The stresses caused by horizontal and vertical forces are assumed to act simultaneously and are added directly. The stresses in the actuator supports caused by loads are combined with the stresses caused by other live and dead loads, including the operating loads. The allowable stress for this combination of loads is based on the ordinary allowable stress set forth in applicable codes. The parts of the MSIVs that constitute a process fluid pressure boundary are designed, fabricated, inspected, and tested as required by the ASME Nuclear Pump and Valve Code.

#### 5.5.5.3 Safety Evaluation

In a direct cycle nuclear power plant, the reactor steam goes to the turbine and to other equipment outside the reactor containments. Radioactive materials in the steam are released to the environs through process openings in the steam system or they escape from accidental openings. A large break in the steam system can drain the water from the reactor core faster than it is replaced by feedwater.

The analysis of a complete, sudden steam line break outside the primary containment is described in Subsection 15.6.4. The analysis shows that the fuel barrier is protected against loss of cooling if MSIV closure takes 10.5 sec or less. This 10.5 sec limitation includes as much as 0.5 sec for the instrumentation to initiate valve closure after the break. The calculated radiological time effects of the radioactive material assumed to be released with the steam are shown to be well within the guideline values for such an accident.

The shortest closing time, approximately 3 sec, of the MSIVs is also shown in Subsection 15.2.4 to be satisfactory. The switches on the valves initiate reactor scram when several valves are more than 10 percent closed. The pressure rise in the system from stored and decay heat may cause the NSSS relief valves to open briefly, but the rise in fuel cladding temperature will be insignificant. No fuel damage results.

The ability of this 45°, Y-design globe valve to close in a few seconds after a steam line break, under conditions of high pressure differentials and fluid flows with fluid mixtures ranging from mostly steam to mostly water, has been demonstrated in a series of tests in dynamic test facilities. Dynamic tests with a 1-in. valve show that the analytical method is valid. A fullsize, 20-in. valve was tested in a range of steam-water blowdown conditions simulating postulated accident conditions (Reference 5).

The following specified hydrostatic, leakage, and stroking tests, as a minimum, are performed by the valve manufacturer in shop tests:

a. To verify its capability to close between 3 and 10 sec, each valve is tested at pressure (1000 psig) and no flow. The valve is stroked several times, and the closing time is recorded. The valve test logic closes the valve by spring only then the combination of air cylinder and springs. Usually the closing time is slightly greater when closure is by springs only

b. Leakage is measured with the valve seated. The specified maximum seat leakage, using cold water at design pressure, is 2 $cm^3$/hr/in. of nominal valve size. In addition, an air seat leakage test is conducted using 50 psi pressure upstream. Maximum permissible leakage is 0.1 scfh per inch of nominal valve size. The valve stem is operated a minimum of three times from the closed position to the open position, and the packing leakage must still be zero by visual examination

c. Each valve is hydrostatically tested in accordance with the requirements of the ASME Nuclear Pump and Valve Code. During valve fabrication, extensive nondestructive tests and examinations are conducted. Tests include radiographic, liquid penetrant, or magnetic particle examinations of casting, forgings, welds, hardfacings, and bolts

d. The spring guides, the guiding of the spring seat member on the support shafts, and rigid attachment of the seat member ensure correct alignment of the actuating components. Binding of the valve poppet in the internal guides is prevented by making the poppet in the form of a cylinder longer than its diameter and by applying stem force near the bottom of the poppet.

After the valves are installed in the NSSS, each valve is tested several times in accordance with the Preoperational and Startup Test procedures. Two isolation valves provide redundancy in each steam line so that either can perform the isolation function, and either can be tested for leakage after the other is closed. The inside valve, the outside valve, and their respective control systems are separated physically.

The isolation valves and their installation are designed as Category I equipment. The design of the isolation valve has been analyzed for earthquake loading. These loads are small compared with the pressure and operating loads that the valve components are designed to withstand. The cantilevered support of the air cylinder, hydraulic cylinder, springs, and controls is the key area. The increase in loading caused by the specified earthquake loading is negligible at the joints between the support shafts and the valve bonnet.

Electrical equipment that is associated with the isolation valves and operates in an accident environment is limited to the wiring, solenoid valves, and position switches on the isolation valves. The expected containment pressure and temperature transient following an accident is discussed in Section 6.2.

#### 5.5.5.4 Inspection and Testing

Inspection and testing of the MSIVs will be conducted periodically in accordance with the Technical Specifications. Additional information on MSIV testing is contained in Subsection 6.2.6.4.

### 5.5.6 Reactor Core Isolation Cooling System

#### 5.5.6.1 Safety Design Bases

The RCIC system meets the following safety design bases.

a. The system shall ensure that adequate core cooling takes place to prevent the reactor fuel from overheating in the event the reactor isolation is accompanied by loss of flow from the reactor feedwater system

b. The system shall operate automatically in time to maintain sufficient coolant in the RPV so that the integrity of the radioactive material barrier is not compromised

c. Piping and equipment, including support structures, shall be designed to withstand the effects of an earthquake without a failure that could lead to a release of radioactivity in excess of the guideline values in published regulations.

### 5.5.6.2 Power Generation Design Bases

The RCIC system meets the following power generation design bases.

a. The system shall operate automatically in time to maintain sufficient coolant in the RPV so that the low-pressure core standby cooling systems (low-pressure coolant injection [LPCI] and core spray systems) are not actuated

b. Design shall provide for remote-manual operation of the system by an operator

c. To provide a high degree of assurance that the system shall operate when necessary

   1. The power supply for the system shall be from immediately available energy sources of high reliability

   2. Design shall provide for periodic testing during plant operation.

#### 5.5.6.2.1 Equipment and Component Description-Design Conditions

Operating parameters for the components of the RCIC system are shown in Figure 5.5-6. The RCIC components are the following.

a. One 100 percent-capacity turbine and accessories

b. One 100 percent-capacity pump assembly and accessories

c. Piping, valves, and instrumentation for

   1. Steam supply to the turbine

   2. Turbine exhaust to the suppression pool

   3. Makeup supply from the condensate storage tank to the pump suction

   4. Makeup supply from the suppression pool to the pump suction

   5. Pump discharge to the feedwater line, including a test line to the condensate storage tank, a minimum flow bypass line to the suppression pool, and a coolant water supply to accessory equipment.

The design conditions are from the ASME Section III, Nuclear Power Plant Components.

#### 5.5.6.2.2 Design Parameters

Design parameters for the RCIC system components are listed below. See Figure 5.5-7 for cross-reference of component numbers.

1. RCIC Pump Operation (C001)

| | |
|---|---|
| Flow rate | Injection flow, 600 gpm<br>Cooling water flow, 16 gpm<br>Total pump discharge, 616 gpm |
| Water temperature range | 40°F to 140°F |
| NPSH | 20 ft minimum |
| Developed head pressure | 2915 ft at 1184-psia reactor pressure<br>525 ft at 165-psia reactor pressure |
| BHP, not to exceed | 700 HP at 2915-ft developed head<br>100 HP at 525-ft developed head |
| Design pressure | 1515 psia |
| Design ambient | 148°F, maximum (Actual conditions to which this equipment is environmentally qualified under the Fermi 2 EQ program are documented in EQ0-EF2-018.) |

2. RCIC Turbine Operation (C002)

| | High-Pressure Condition | Low-Pressure Condition |
|---|---|---|
| Reactor pressure (saturated temperature) | 1184 psia | 165 psia |
| Steam inlet pressure | 1169 psia, minimum | 150 psia, minimum |
| Turbine exhaust pressure | 25 psia, maximum | 25 psia, maximum |
| Design inlet pressure | 1250 psig at saturated temperature | |
| Design exhaust pressure | 165 psig at saturation temperature | |

3. RCIC Orifice Sizing

| | |
|---|---|
| Coolant loop orifice (D009) | Sized with piping arrangement to ensure maximum pressure of 75 psia at the lube-oil cooler inlet and a minimum pressure of 45 psia at the spray nozzles at the barometric condenser. |
| Minimum flow orifice | Sized with piping arrangement to ensure minimum flow of 75 gpm with MO-F019 fully open. |
| Test return orifice (D006) | Sized with piping arrangement and drag valve E41-F011 to simulate pump discharge pressure required when the RCIC system is injecting design flow with the reactor vessel pressure at 165 psia. |

| | |
|---|---|
| Leak-off orifices (D008 and D010) | Sized for 1/8-in. diameter minimum; 3/16-in. diameter maximum. |
| Warm-up bypass orifice (D011) | Sized at 5/16-in. to insure sufficient steam supply to spin the turbine. |

4. Valve Operation Requirements

| | |
|---|---|
| Steam warm-up bypass valve (F095) | Open and/or close against 1169 psid pressure within 10 sec. |
| Steam supply valve (F045) | Open and/or close against maximum expected differential pressure within 45 sec. |
| Pump discharge valve (F013) | Open and/or close against maximum expected differential pressure within 30 sec. |
| Pump minimum flow bypass valve (F019) | Open and/or close against 1296 psid pressure within 25 sec. |
| Steam supply isolation valves (F007 and F008) | Close against maximum expected differential pressure within 15 sec. |
| Cooling water relief valve (F018) | Sized to prevent over-pressurizing piping, valves, and equipmentin the coolant loop in the event of failure of pressure control valve F015. |
| Pump discharge out-board isolation valve (F012) | Open against 1000 psid pressure within 15 sec (valve normally open and deenergized). |
| Pump test return valve (F022) | Capable of open and/or close against 1000 psi differential pressure. |
| Relief valve barometric condenser (F033) | Relief valve is capable of retiaing 10 in. of mercury vacuum at 140°F ambient, with a set pressure of 5 to 7 psig and flow of 20 gpm at 25 percent accumulation. |
| Turbine Exhaust isolation valve (F001) | Opens and/or closes against 50 psi differential pressure at a temperature of 267°F. Physically located as close to the containment as practical. |
| Vacuum pump discharge isolation (F002) | Opens and/or closes against 50 psi differential pressure at a temperature of 267°F. Physically located as close to the containment as practical. |
| Check valve turbine exhaust (F040) | Located at a high point in the line on a horizontal run, with adjacent piping arranged to provide a continuous downward slope, form the upstream side of the check valve to the turbine exhaust drain pot. |

| | |
|---|---|
| Vacuum breaker isolation valves (F062 and F084) | Open and/or close against a differential pressure of 50 psi. |
| Vacuum breaker check valves (F063 and F064) | Open with a minmum pressure drop (less than or equal to 0.5 psi) across the valve seat. |

5. Rupture Disk Assemblies (D001 and D002)

Used for turbine casing protection; includes a mated vacuum support to prevent rupture disk reversing under vacuum conditions.

| | |
|---|---|
| Rupture pressure | 150 psig ± 10 psig |
| Flow capacity | 60,000 lb/hr at 165 psig |

6. Condensate Storage Requirements

150,000 gap (Total reserve storage for both HPCI and RCIC systems, see Section 6.3.2.6.)

7. Piping RCIC Water Temperature

The maximum water temperature range for continuous system operation does not exceed 140°F. However, due to potential short-term operation at higher temperatures, piping expansion calculations were based on 170°F.

8. Turbine Exhaust Vertical Reaction Force

Unbalanced pressure due to discharge under the suppression pool water level is described in Reference 6.

9. Ambient Condition

| | Temperature | Relative Humidity |
|---|---|---|
| Normal plant operation | 60° to 100°F | 95 percent |
| Isolation conditions | 148°F | 100 percent |

### 5.5.6.3 Description

#### 5.5.6.3.1 General

The RCIC system consists of a steam-driven turbine-pump unit and associated valves and piping capable of delivering makeup water to the RPV. A schematic diagram is shown in Figure 5.5-7 and Figure 5.5-8. Logic diagrams are provided in Figure 7.4-1, Sheets 1 through 6.

The pump discharges either to the feedwater line or to a full flow test return line to the condensate storage tank. The discharge lines are full of water and remain flooded because they are connected to the feedwater line. The lines upstream of the normally closed HPCI and RCIC injection valves are kept full due to the static head provided by the condensate storage tank. The elevation of the injection valves is lower than the low level of the condensate storage tank, providing the static head. A minimum flow bypass line to the suppression pool is provided to protect the pump during startup and shutdown. The makeup water is delivered into the RPV through the feedwater line. Cooling water for the RCIC turbine lube-oil cooler and barometric condenser is supplied from the discharge of the pump, as shown in Figure 5.5-7.

Following any reactor shutdown, steam generation continues because of heat produced by the radioactive decay of fission products. Initially, the rate of steam generation can be as much as approximately 6 percent of rated flow and is augmented during the first few seconds by delayed neutrons and some of the residual energy stored in the fuel. Steam normally flows to the main condenser through the turbine bypass or, if the condenser is isolated, to the suppression pool. The fluid removed from the RPV is normally made up by the feedwater pumps supplemented by cooling water flow from the CRD system. If makeup water is required to supplement these primary sources of water, the RCIC turbine-pump unit starts automatically on receipt of a RPV low water level (L2) signal (Figure 7.4-1) or is started by the operator from the main control room. The RCIC delivers its design flow within 50 sec after actuation.

The RCIC makeup capacity is sufficient to avoid the need for the ECCS. Pump suction is usually lined up to the condensate storage tank but is automatically switched to the suppression pool on low condensate storage tank level. See Subsection 5.5.6.3.3.

Based upon normal condensate storage tank level of greater than 11’-0”, the volume of water stored for the RCIC (140,000 gal) is sufficient to allow operation for 8 hr after shutdown, assuming that none of the steam generated in the RPV is returned to the RPV as condensate. Other systems that use the condensate storage tank and could jeopardize the availability of this quantity of water can be isolated. However, manual actions are not required to protect the condensate storage tank inventory since, upon low level, RCIC suction is automatically transferred to the safety-related water source which is the suppression pool.

The RCIC system is sized to prevent actuation of the low level signal (L1) for RPV isolation incidents. Prevention of this signal ensures core cooling and prevents ADS actuation, thus preventing inadvertent blowdown of the RPV for this situation.

Quantitative information on steam and delivery water conditions is provided in Figure 5.5-6 for all operating modes of the RCIC system.

The backup supply of cooling water for the RCIC is the suppression pool. The turbine pump assembly is located below the level of the condensate storage tank and below the minimum water level in the suppression pool to ensure positive suction head to the pump.

All components required for initiating the RCIC are completely independent of auxiliary ac power, plant service air, and external cooling water systems. These components require only power derived from the station battery to operate the valves and logic. The power source for the turbine-pump unit is the steam generated in the RPV by the decay heat in the core. The steam is piped directly to the turbine, and the turbine exhaust is piped to the suppression pool.

Throughout the period of RCIC operation, the exhaust from the RCIC turbine is condensed in the suppression pool, which results in a slow temperature rise of approximately 3°F per hour in the pool. One RHR heat exchanger can be used to cool the suppression pool, if necessary. If for any reason the RCIC is unable to supply sufficient flow for core cooling, the emergency core cooling system (ECCS) provides the required boundary protection. A further discussion of this is found in Section 6.3.

The RCIC turbine-pump unit is located in a shielded area to ensure that personnel access areas are not restricted during RCIC operation. The turbine controls provide for automatic shutdown of the RCIC turbine on receipt of the following signals

a. RPV high water level - indicates that core cooling requirements are satisfied

b. Turbine overspeed - prevents damage to the turbine and turbine casing

c. Pump low suction pressure - prevents damage to the turbine pump unit that results from loss of cooling water

d. Turbine high exhaust pressure - indicates turbine or turbine control malfunction

e. System isolation signal - indicates need to shut down equipment.

Because the steam supply line to the RCIC turbine is a pressure containment boundary, certain signals automatically isolate this line and cause shutdown of the RCIC turbine.

The RCIC turbine has a speed governor that is positioned by the demand signal from the flow controller. Maximum output from the controller corresponds to maximum turbine speed.

The RCIC system may provide the ability to mitigate the consequences of small pipe breaks, but it is not provided primarily for such purpose. The ECCS provides redundant protection for the entire spectrum of pipe breaks. For small breaks this protection would be provided by HPCI and automatic depressurization.

Both the RCIC and HPCI systems provide decay heat removal capability when the main condenser is unavailable (i.e., isolated from the nuclear system) for heat sink purposes. The HPCI is a subsystem of the ECCS; however, the RCIC is not a subsystem of the ECCS.

Long-term heat removal capability may be provided by the RCIC or HPCI during the following operational events: scram, pressure relief, core cooling, RPV isolation, and restoration of ac power. The RHR system may be used for long-term heat removal during any long-term isolation. These events are all situations in which the RPV is isolated from the main condenser. None of these events are pipe break (loss of coolant) situations requiring immediate reactor water level restoration.

To ensure HPCI or RCIC system availability for the operational events noted previously, certain design considerations are used in the design of both systems.

#### 5.5.6.3.2 Reactor Core Isolation Cooling Following Main Condenser Isolation (See Figure 5.5-9)

A reactor shutdown is accompanied by the isolation of the main condenser from the reactor vessel; the fission product decay heat results in an increase in the reactor vessel pressure. The pressure increase is limited by or manual operation of the relief valves, which serve to dump steam to the suppression pool. In the event the feedwater pumps and control rod drive leakage cannot provide sufficient water to make up for that lost by the steam dumping, the RCIC begins to operate by either a low reactor water signal or a manual start. For normal operation, the RCIC turbine-driven pump takes water from the condensate storage tank and injects it into the feedwater line. The steam supply for the RCIC turbine is from a main steam line using decay-heat-generated steam; exhaust is to the suppression pool. During RCIC operation, the desired reactor vessel pressure is maintained by manual control of the relief valves.

When RCIC is initiated, automatic actions will take place as described in Subsections 5.5.6.3.6 and 5.5.6.7. Also for RCIC operation, the turbine control system must function properly and there can be no turbine trip signals present. The RCIC can deliver its design flow within 50 sec of the initiation signal. Based on normal condensate storage tank level of greater than 11'-0", the volume of water stored in the condensate storage tank for the RCIC (135,000 gal) is sufficient to allow operation of the RCIC for 8 hr after a shutdown. After this time, the system is sufficiently depressurized to allow the shutdown cooling mode of the RHR system to operate. The flow rate of water from the RCIC pump to the reactor vessel is 600 gpm, which is approximately equal to the reactor water boiloff rate 15 minutes after shutdown. This flow rate is sufficient to prevent the reactor vessel water level from dropping down to the top of the core.

#### 5.5.6.3.3 Reactor Core Isolation Cooling Backup Mode (See Figure 5.5-10)

The RCIC can also take water from the suppression pool if the condensate storage tank level becomes too low. Transfer of the pump suction to the suppression pool is an automatic operation which follows the receipt of a low level signal from the condensate storage tank. The transfer requires the opening of normally closed valves (E51) F029 and (E51) F031 located in the pump suction line to the suppression pool. The opening of these valves causes the automatic closure of (E51) F010 located in the pump suction line leading to the condensate storage tank. Panel status information is provided for the operator in the form of valve position indication and an alarm if the operator closes either suppression pool suction valve while the condensate storage tank level is low.

#### 5.5.6.3.4 Reactor Core Isolation Cooling Test Flow Mode (See Figure 5.5-11)

The RCIC system is designed commensurate with the safety importance of the system and its equipment. Each component is individually tested to confirm compliance with system requirements. The system as a whole is tested during both the startup and preoperational phases of the plant to set a base mark for system reliability. To confirm that the system maintains this mark, functional and operability testing is performed at predetermined intervals throughout the life of the reactor plant.

A design flow functional test of the RCIC system may be performed during normal plant operation by drawing suction from the condensate storage tank and discharging through a full flow test return line to the condensate storage tank. The discharge valve to the feedwater line remains closed during the test, and reactor operation remains undisturbed. All components of the RCIC system are capable of individual functional testing during normal plant operation. System control provides automatic return from test to operating mode if system initiation is required. There are three exceptions:

a. Auto/manual initiation on the flow controller. This feature is required for operator flexibility during system operation

b. Steam inboard/outboard isolation valves. The closing of either or both of these valves requires operator action to properly sequence their opening. An alarm sounds when the controls for either of these valves is operated to direct the valves to close

c. Other parts of the system that have been bypassed or deliberately rendered inoperable. These shall be indicated in the control room at the system level.

5.5.6.3.5 Reactor Core Isolation Cooling Minimum Flow Mode (See Figure 5.5-12)

A minimum flow bypass line is provided for protection of the RCIC pump. A flowmeter in the pump discharge line provides a signal for initiating the minimum flow mode for low flow and stopping its operation for a sufficient flow. Pump discharge pressure also must be sensed by PS N020 to allow minimum flow bypass valve F019 to open.

5.5.6.3.6 Auxiliary Heat Removal Operation

If the main feedwater system is not operable, a reactor scram will automatically be initiated when reactor water level falls to Level 3. Reactor water level will continue to decrease from boil-off until the low-low-level setpoint, Level 2, is reached. At this point, the HPCI system and the RCIC system will be automatically initiated to supply makeup water to the reactor pressure vessel. These systems will continue automatic injection until the reactor water level reaches Level 8, at which time the HPCI and RCIC turbines are tripped. These systems (HPCI/RCIC) will restart automatically once the high-level trip signal clears and a low-low-level (Level 2) signal is received.

The RCIC system will start automatically upon receipt of the initiation signal from the reactor vessel low-water-level sensor. During startup from standby, the following events occur automatically. (See Figure 7.4-1.)

a. Turbine speed control given to RCIC system flow indicator controller

b. RCIC test bypass valve to condensate storage tank closes (if open)

c. Steam supply valves to turbine open

d. Barometric condenser condensate pump discharge isolation valve closes

e. Pump discharge valve to feedwater line opens

f. Barometric condenser vacuum pump starts

g. Cooling water supply valve to lube oil cooler opens.

The turbine starts as soon as the steam supply valve opens, since the turbine trip throttle valve and control valve are open. The minimum flow bypass valve to suppression pool opens when pump discharge pressure increases. System flow starts when pump discharge pressure exceeds feedwater line pressure. As pump discharge pressure and steam inlet pressure change, the control signal adjusts the turbine to maintain constant pump flow. When pump flow reaches a prescribed value, the minimum flow bypass valve closes.

On occurrence of a low water level in the condensate storage tank, the suction to the RCIC pump changes automatically from condensate storage tank to the suppression pool.

The operator can switch the flow controller to the manual position and decrease flow rate to stabilize the water level in the reactor vessel. This would be done before reaching the high-water-level isolation. Even if the operator does not manually take control and the RCIC trips on high level, the RCIC will restart automatically once the high-water-level isolation signal clears and a Level 2 low-low-water-level signal is received.

The following sequence of events occurs in the case of an automatic initiation of the HPCI system (see Figure 7.3-2).

a. Steam supply outboard isolation valve opens
b. HPCI suction valve from condensate storage opens (if closed)
c. HPCI pump discharge inboard and outboard isolation opens
d. deleted
e. HPCI steam inlet valve opens
f. HPCI lube-oil cooling water supply valve opens
g. HPCI auxiliary oil pump starts
h. HPCI condenser vacuum pump starts (if initiation is by Level 2 low-low-water-level signal only)
i. HPCI test return valves close (if open).

With the turbine stop valve and control valves open, steam is admitted to the turbine, accelerating it quickly to speed.

On the occurrence of either a low water level in the condensate storage tank or a high level in the suppression pool, the suction valve to the HPCI pump changes over from condensate storage tank to the suppression pool.

The operator can switch the flow controller to the manual position and decrease flow rate to stabilize the water level in the reactor vessel. This would be done before reaching the high-water-level isolation. Even if the operator does not manually take control and the HPCI trips on high level, the HPCI will restart automatically once the high-water-level isolation signal clears and a Level 2 low-low-water-level signal is received.

For the loss-of-feedwater transient, the HPCI/RCIC systems are used to automatically provide the required makeup flow. No manual operations are required.

With the MSIVs closed, reactor pressure may rise to the setpoint of the safety/relief valves that will operate to reduce reactor pressure.

The heat added to the suppression pool from the operation of the safety/relief valves and the RCIC and HPCI systems will cause the suppression pool to heat up. As the average temperature of the suppression pool rises, the operator will initiate the suppression pool cooling mode of the residual heat removal (RHR) system to reduce this temperature before reaching the Technical Specifications limit.

Reactor vessel heat removal may also be accomplished through the manual actuation of any of the 15 safety/relief valves. In the event that reactor vessel pressure reduction and heat removal is required through safety/relief valve operation, the remote actuation of the safety/relief valves is available and would be used in conjunction with the suppression pool cooling mode of the RHR system. The operator actions necessary to place the RHR system in the suppression cooling mode emergency operations are as follows:

a. Verify RHR and RHRSW systems are in Standby condition
b. Open the associated RHR Torus Isolation Valve
c. Start the associated RHR pump
d. Throttle open the associated RHR Torus Cooling Isolation Valve
e. Start the associated RHRSW Pumps
f. Throttle the associated RHR HX Bypass and RHR HX Outlet Valves to control cooldown rate.

With the RHR system in the suppression pool cooling mode, the operator may actuate the required safety/relief valves while maintaining the required suppression pool temperature and heat distribution limits.

During this mode of operation, the automatic depressurization system remains fully operational and will automatically initiate if the conditions necessary for automatic depressurization should occur.

##### 5.5.6.3.7 Physical Independence

The RCIC and HPCI systems are located in separate rooms in different corners of the reactor building. Piping runs are separated and the water delivered from each system enters the RPV via different nozzles.

##### 5.5.6.3.8 Control Independence

Control independence is secured by using different battery systems to provide control power to the RCIC and HPCI systems.

##### 5.5.6.3.9 Environmental Independence

The RCIC and HPCI systems are designed to meet Category I requirements. Environment in the equipment rooms is maintained by separate auxiliary systems.

#### 5.5.6.4 Safety Evaluation

To ensure that the RCIC operates when necessary and in time to prevent inadequate core cooling, the power supply for the system is taken from immediately available energy sources of high reliability. Added assurance is given in the capability for periodic testing during station operation. Evaluation of the instrumentation configuration for the RCIC shows that no failure of a single initiating sensor either prevents or falsely starts the system.

The RCIC piping within the drywell up to and including the outer isolation valve is designed in accordance with ASME B&PV Code Section III. The RCIC, including the RCIC turbine speed control system, is also designed as Category I equipment. (See Subsection 7.3.2 for isolation signals.)

### 5.5.6.5 Inspection and Testing

A design flow functional test of the RCIC is performed during plant operation by taking suction from the condensate storage tank and discharging through the full flow test return line back to the condensate storage tank. The discharge valve to the feedwater line remains closed during the test, and reactor operation is undisturbed. Control of the pump discharge valve is obtained by first closing the upstream discharge valve. Control system design provides automatic return from test to operating mode when system operation is required during testing. Routine maintenance and tests, based on the manufacturer's recommendations and/or operating/maintenance experience, will be scheduled in accordance with the plant preventive maintenance program, including periodic inspection of the RCIC suppression pool suction strainer. Valve position indication and instrumentation alarms are displayed in the main control room.

### 5.5.6.6 Isolation

Arrangements of isolation valves include the following.

a. Two RCIC lines penetrate the reactor coolant pressure boundary. The first RCIC line is the RCIC steam line that branches off one of the main steam lines between the reactor vessel and the main steam isolation valve. This line has two automatic motor-operated isolation valves. One is located inside and the other outside the primary containment. The isolation signals noted earlier close these valves

b. The RCIC pump discharge line is the other line; however, it indirectly penetrates the reactor pressure vessel. This line enters the main feedwater line, described elsewhere, which provides required isolation valves inside the primary containment. The RCIC system provides the automatic motor-operated valve outside the primary containment for isolation

c. The RCIC turbine exhaust line vacuum breaker system line has two automatic motor-operated valves and two check valves. This line runs between the suppression pool air space and the turbine exhaust line downstream of the exhaust line check valve. Positive isolation shall be automatic via a combination of low reactor pressure and high drywell pressure

   The vacuum breaker valve complex is placed outside the primary containment due to a more desirable environment. In addition, the valves are readily accessible for maintenance and testing

d. The RCIC pump suction line, minimum flow pump discharge line, and turbine exhaust line all penetrate the primary containment and are submerged in the suppression pool. The isolation valves for the lines are all outside the primary containment and require remote-manual operation, except for the minimum flow valves, which actuate automatically. Additionally, the turbine gland seal system vacuum pump discharges beneath the suppression pool after penetrating the primary containment. The isolation valve for the line is located outside the primary containment and requires remote-manual operation.

### 5.5.6.7 Interlocks

The following define the various electrical interlocks (see Figure 7.4-1).

a. The steam line isolation valves, F007 and F008, are keylocked in the open position. The valves can still automatically close on a steam line isolation signal, but can be manually operated only when the keylock is placed in the "operate" position

b. The F029 and F031 limit switches activate when fully open, and close F010 and F022

c. The F001 limit switch activates when fully open, and clears the F045 and the F095 permissives so both F045 and F095 can open. The F045 and F095 valves are signaled to close if F001 moves to a position other than fully open

d. The F045 limit switch activates when fully closed and permits F004, F005, F025, and F026 to open, and closes F013 and F019

e. The turbine trip throttle valve (part of C002) limit switch activates when fully closed and closes F013 and F019

f. The combined pressure switches at reactor low pressure and high drywell pressure, when activated, close F062 and F084

g. A high turbine exhaust pressure, low pump suction pressure, or an isolation signal actuates and closes the turbine trip throttle valve

h. A 122.3 percent overspeed trips both the mechanical trip at the turbine and the trip throttle valve. The former is reset at the turbine and then the latter is reset by a combination of control room and local (near the RCIC skid) operator action

i. An isolation signal closes F007, F008, and other valves as noted above in Items e. and g

j. An initiation signal opens F010 and F012 if closed; opens F095, F045, F046, and F013 and starts the barometric condenser vacuum pump; and closes F022, if open. Drain isolation valves F004, F005, F025, and F026 will close automatically on receipt of F045 limit switch "not full closed" signal

k. High and low inlet RCIC steam line drain pot levels, respectively, open and close F054

l. The combined signal of low pump flow plus high pump discharge pressure opens F019. The F019 valve closes on a pump flow signal above the minimum flow setpoint

m. A reactor low water level (Level 1) or high drywell pressure signal trips the barometric condenser condensate and vacuum pumps

n. The F013 limit switch activates when not full closed and closes F022 and prevents F022 from opening

o. CST low level signal opens F029 and F031.

5.5.6.8 Limiting Single Failure

The most limiting single failure of the RCIC system and its HPCI backup system is the failure of HPCI. With an HPCI failure, if the capacity of the RCIC system is adequate to maintain reactor water level, the operator follows Subsection 5.5.6.3.2. If, however, the RCIC capacity is inadequate, the operator may also initiate the ADS system described in Subsection 6.3.2.2.2.

5.5.7 Residual Heat Removal System

5.5.7.1 Safety Design Bases

The RHR system meets the following safety design bases.

a. The system shall act automatically, in combination with other subsystems of the ECCS, to restore and maintain the coolant inventory in the RPV so that the core is adequately cooled to preclude fuel cladding temperatures from exceeding the acceptance criteria temperature of 2200°F following a design basis LOCA

b. The system, in conjunction with other subsystems of the ECCS, shall have such diversity and redundancy that only a highly improbable combination of events could result in the inability to adequately cool the core

c. The source of water for restoring RPV coolant inventory shall be so located within the primary containment as to establish a closed cooling water path

d. To ensure that the RHR system operates satisfactorily during a LOCA, each active component shall be testable during operation of the NSSS

e. A closed loop flow path between the suppression pool and the RHR heat exchangers shall be established so that the heat removal capability of these heat exchangers can be utilized for long-term containment heat removal.

See Subsection 3.1.2.4.5 for a discussion of conformance to General Design Criteria (GDC) 34. The RHR system design conforms to the single-failure requirement of GDC 34.

5.5.7.2 Power Generation Design Bases

The RHR system is designed to meet the following power generation design bases.

a. The system shall have enough heat removal capacity to cool down the reactor to 125°F within 20 hr after shutdown

b. Fuel pool connections shall be provided so that the RHR heat exchangers can be used to supplement the fuel pool cooling capacity

c. A closed loop flow path between the suppression pool and the RHR heat exchangers shall be established so that the heat removal capability of these heat exchangers can be used to cool the suppression pool.

5.5.7.3 Description

5.5.7.3.1 Summary

The RHR system is designed for three modes of operation to satisfy all the objectives and bases. To provide clarity to the information presented herein, each mode of operation is defined as a subsystem of the RHR system and is discussed separately. It is shown how each subsystem contributes toward satisfying all the objectives and bases of the RHR system.

The major equipment of the RHR system consists of two heat exchangers, four main system pumps, and four service water pumps. The equipment is connected by associated valves and piping, and the controls and instrumentation are provided for proper system operation. A schematic diagram of the RHR system is shown in Figure 5.5-13. A description of the controls and instrumentation is presented in Subsections 7.3.1.2.4 and 7.4.1.3. A description of how operation of the equipment in the RHR system in conjunction with other subsystems of the ECCS protects the core in case of a LOCA is presented in Section 6.3.

The main system pumps are sized for the flow required during LPCI operation, which is the subsystem that requires the maximum flowrate. Subsection 6.3.2 contains a discussion of the LPCI. The pumps are arranged and located so that adequate suction head is ensured for all operating conditions. The pump motor is air-cooled by the ventilation system.

The heat exchangers were originally sized on the basis of their required duty for the shutdown cooling function. The heat exchanger shell and tube sides are provided with drain connections. The shell side is provided with a vent to remove noncondensible gases. Thermal relief valves on the heat exchanger shell side and a relief valve on the RHR pump discharge protect the heat exchanger from overpressure.

The RHR heat exchanger duty for the shutdown cooling mode of operation is $41.6 \times 10^6$ Btu/hr.

The most limiting duty is that duty associated with torus cooling mode. See Section 6.2.2.3.

Detailed classification information for the RHR heat exchanger is presented in Table 3.2-1.

The RHR system can be connected to the fuel pool cooling and cleanup system, as shown in Figure 5.5-13, so that the RHR heat exchangers can assist fuel pool cooling during overload conditions. Subsection 9.1.3 contains a description of the fuel pool cooling and cleanup system.

One loop, consisting of a heat exchanger, two main system pumps in parallel, and associated piping, is located in one area of the reactor building. The other heat exchanger, pumps, and piping, all of which form a second loop, are located in another area of the reactor building to minimize the possibility of a single physical event causing the loss of the entire system. The two loops of the RHR system are cross-connected by a single header, making it possible to supply either loop from the pumps in the other loop. Water is supplied through a low pressure regulator and two check valves to ensure that the RHR discharge piping is continuously filled. This arrangement precludes water hammer effects. Figure 5.5-14 shows the RHR valve positions during normal reactor operation. Figures 5.5-15 through 5.5-17 show the RHR valve positions for the three RHR modes of operation as described in Subsections 5.5.7.3.2 and 5.5.7.3.3.

### 5.5.7.3.2 Shutdown Cooling

The shutdown cooling system is an integral part of the RHR system. It is operated during a normal shutdown and cooldown.

The RHR lines can be flushed prior to initiation of the shutdown cooling mode. Flushing is accomplished by establishing flow through the warm-up line to the suppression pool. Flow in this line is limited to approximately 500 gpm by a restricting orifice. The warm-up line isolation valve (F026B) is manually closed after flushing. If the operator fails to close the warm-up line valve, the potential loss of mass inventory could cause water level to drop. The low reactor water level isolation will automatically close the shutdown cooling valves and interrupt the outflow of water. Although it is preferred to flush the lines before RPV injection, no significant consequences will occur if flushing is omitted. The RHR piping will normally be filled with demineralized water or water from the suppression pool. The quality of the suppression pool water is maintained by the torus water management system.

The initial phase of nuclear system cooldown is accomplished by dumping steam from the RPV to the main condenser. When the nuclear system temperature has decreased to where the steam supply pressure is not sufficient to maintain the turbine shaft gland seals, the vacuum in the main condenser cannot be maintained and the RHR system is placed in the shutdown cooling mode of operation. The shutdown cooling system is able to complete cooldown to 125°F within 20 hours after the control rods have been inserted, and can maintain the nuclear system at 125°F for reactor refueling and servicing.

The allowable cooldown rate of the reactor coolant system should not exceed 100°F per hour. To achieve this condition, the heat exchanger's shell-side bypass valve (F048) is throttled to control the cooldown rate.

The RHR shutdown cooling mode is shown in Figure 5.5-15. Reactor coolant is pumped from one of the RRS loops by one or both of the RHR main system pumps and is discharged through the RHR heat exchangers, where cooling occurs by heat being transferred to the service water. Reactor coolant can be returned to the RPV through either RRS loop. When transitioning between the RRS and RHR shutdown cooling, a single RRS pump may be kept in operation while an RHR pump is started. During this time of simultaneous operation the operating RHR pump and the operating RRS pump may not inject into the same loop.

The high RPV water level provides conduction cooling to most of the mass of metal of the RPV and therefore limits thermal stress in the RPV during cooldown.

During a nuclear system shutdown following a scram, the decay heat level decreases rapidly enough that one RHR heat exchanger is capable of accommodating the entire shutdown cooling load.

FPCCS and natural circulation have been analyzed to be capable of serving as an alternate method of decay heat removal to enable RHR Shutdown Cooling to be taken out of service for maintenance during refueling (References 7 and 8). When operating in this alternate shutdown cooling mode, the fuel pool gates are removed and the RPV cavity is flooded. Entry into this mode requires satisfying the refuel technical specification associated with high RPV water level. FPCCS is normally operated with two pumps and two heat exchangers in service. In this capacity, FPCCS and natural circulation maintain FPCCS suction temperature less than 140°F, cooling both the old and freshly off-loaded assemblies in the fuel pool as well as those remaining in the RPV. RWCU may also be placed in operation with the regenerative heat exchanger bypassed to provide additional cooling and in-vessel mixing. This ability to enter this mode of FPCCS operation for RHR maintenance activities is evaluated on a per cycle basis using the expected vessel and spent fuel pool heat loads. The activity is managed such that normal shutdown cooling can be restored within 8 hrs. This is an arbitrary time frame that conservatively assures cooling can be restored prior to the onset of pool and core boiling. In addition, the operation of this mode restricts the operation of temporary auxilliary pool water filtration units such that the flow discharge does not interfere with the core exit flow and thereby impede natural circulation cooling.

#### 5.5.7.3.3 Containment Cooling Subsystem

The containment spray cooling subsystem provides containment cooling for postaccident conditions (see Figure 5.5-16). Water pumped through the RHR heat exchangers can be diverted to spray headers in the drywell and above the suppression pool. The spray removes energy from drywell atmosphere by condensing the water vapor. The spray collects in the bottom of the drywell until the water level rises to the level of the pressure suppression chamber vent lines. The water then overflows to the suppression pool. Approximately 5 percent of this flow can be directed to the suppression chamber spray ring to cool any noncondensible gases collected in the free volume above the suppression pool.

The RHR system is serviced by an automatic fill system that maintains the containment spray lines filled up to the outermost containment isolation valves.

NRC Generic Letter (GL) 2008-01, “Managing Gas Accumulation in Emergency Core Cooling , Decay Heat Removal, and Containment Spray Systems” requested each licensee evaluate the licensing basis, design, testing, and corrective action programs for the Emergency Core Cooling Systems (ECCS), Decay Heat Removal (DHR) systems, and Containment Spray systems, to ensure that the piping systems are maintained full of water, and that appropriate action is taken when gas accumulation is discovered. Fermi’s initial response to this GL is documented in Ref. 9. During the GL response effort, NRC clarified the meaning of the phrase “full of water” (Ref. 10). The NRC concluded that when some voids are discovered in piping, the system can be considered filled with water as long as reasonable expectation of the system’s ability to perform its specified function is established.

The containment spray cooling subsystem of the RHR system normally cannot be operated unless the core flooding requirements of the LPCI subsystem have been satisfied. The operator can bypass these requirements by using a keylock switch in the main control room (Subsection 7.3.1.2).

On initiation of the RHR containment spray mode, the inner isolation valve is fully opened. The outermost isolation valve is a throttling-type valve, and the extent of the valve opening is determined by the time the open pushbutton is kept depressed. The valve open, mid-open, or closed indications are provided on the control panels to inform the operator of the valve position. The operators for the outermost valves are designed to open slowly. After a steady-state condition has been reached, the outermost isolation valve is fully opened. In this manner, dynamic loadings imposed on the empty portions of the containment spray lines and on the system supports and restraints are limited to within design values during the initial spray period as well as during the steady-state operating condition.

The suppression pool cooling subsystem (see Figure 5.5-17) cools the suppression pool by using the RHR pumps and heat exchangers in a closed loop with the suppression pool. The suppression pool cooling subsystem is put into operation to limit the water temperature immediately after a blowdown to 170°F when reactor pressure is above 135 psig. During this mode of operation, water is pumped from the suppression pool through the RHR system heat exchanger and back to the suppression pool.

The equipment purchase specifications for the RHR heat exchangers that are used for the containment cooling and suppression pool cooling modes specify fouling factors.

The fouling factors are a function of the nature of the fluids, the temperatures involved, and the fluid velocities. The heat exchanger designer includes the fouling factor in calculating the overall thermal resistance and provides sufficient surface area to allow the required heat transfer rate while in the fouled condition.

The heat exchanger performance data sheets supplied by the heat exchanger designer/manufacturer show the expected (designed) performance of the heat exchanger under fouled conditions. Fouling beyond the extent specified in the purchase specification and used during the heat exchanger design will result in a decrease in the heat transfer rate.

5.5.7.3.4 Low Pressure Coolant Injection System

The LPCI system is an integral part of the RHR system. It operates to restore and, if necessary, maintain the coolant inventory in the RPV after a LOCA. A description of the salient features of the LPCI system is given in Sections 6.3 and 7.3.

The LPCI is a low-head, high-flow function that delivers its rated flow to the RPV through one of the RRS loops. It is designed to reflood the RPV to at least two-thirds core height and to maintain this level. After the core has been flooded to this height, the capacity of one RHR main system pump is sufficient to make up for shroud leakage and boiloff. The LPCI subsystem operates in con-junction with the HPCI system, ADS, and the core spray system to restore and maintain the coolant inventory in the RPV after a LOCA.

The HPCI is a high-head, low-flow system that can pump water into the RPV when the NSSS is at high pressure. If the HPCI fails to deliver the required flow of cooling water to the RPV, the automatic depressurization feature of the overpressurization protection system described in Subsection 5.2.2 functions to reduce nuclear system pressure so that LPCI and core spray may operate to inject water into the RPV. The HPCI turbine is manually shut down after both core spray and LPCI are in operation. When RHR is lined up in the shutdown cooling mode and RPV pressure is less than or equal to the cut in pressure, manual operation is required to permit LPCI to align and initiate. This includes manually lining up the suction path from the torus for the loop which is in shutdown cooling. Otherwise, these operations are carried out automatically.

During LPCI operation, the RHR system pumps take suction from the suppression pool and discharge to the RPV into the core region through one of the RRS loops. Instrumentation is provided to detect the undamaged path for injection of LPCI flow (Subsection 7.3.1.2). Any spillage through a break in the lines within the primary containment returns to the suppression pool through the pressure suppression vent lines. A minimum-flow bypass line to the suppression pool is provided so that the pumps are not damaged if operating with the discharge valves shut.

Service water flow to the RHR heat exchangers is not required immediately after a LOCA because heat rejection from the containment is not necessary during the time it takes to flood the reactor. Power for the main RHR and RHRSW pumps normally comes from an auxiliary ac power bus; but if offsite power is lost, power is made available from the standby ac power source to supply the RHR and RHRSW pumps.

To provide a source of water if any postaccident flooding of the primary containment is required, a cross tie exists from the piping on the discharge side of a pair of service water pumps to the discharge piping on the shell side of an RHR heat exchanger. This connection is provided with redundant valving appropriate to a primary containment penetration. The valves are remotely operable from the main control room. The pair of service water pumps that provide this function can add water to either RRS loop through the cross-connection between the piping of each RHR loop.

#### 5.5.7.3.5 Residual Heat Removal System Overpressure Protection

The design basis for overpressure protection in the RHR system is the conformance of the entire system to applicable portions of ANSI B31.7.

Failures due to overpressurization can result from the inadvertent opening of reactor coolant system (RCS) pressure boundary valves or RCS pressure boundary valve leakage. The RHR low-pressure piping is connected to the RCPB at the RHR shutdown suction and discharge connections to the recirculation system. Each of these lines is discussed in the following paragraphs.

a. The RHR suction from the recirculation system: This line has an inside containment isolation valve and an outside containment valve. Each valve is interlocked with a separate pressure switch that prohibits opening of the associated valve if the recirculation pressure exceeds the shutdown range. The design complies with GDC 55.

b. The RHR shutdown return line: This line has two valves outside containment. Each valve is interlocked to at least a control permissive of low reactor pressure. The line also has a testable check valve inside the containment that functions automatically to prevent outflow from the vessel. This design complies with GDC 55.

Reactor coolant system pressure boundary isolation valve leakage is accommodated by 1-in. or larger relief valves. This size of the valve is considered large enough to accommodate any postulated leakage. Valve F029 relieves shutdown cooling isolation valve leakage pressure; valves F025A and F025B relieve injection isolation valve leakage pressure. The heat exchangers contain their own relief valves, and the suction piping is relieved by valves F030A, F030B, F030C and F030D whenever the respective pool suction valves are closed.

### 5.5.7.4 Safety Evaluation

Because the LPCI and containment cooling subsystems act with other subsystems of the ECCS to satisfy the safety objective, they are evaluated in conjunction with the other subsystems of the ECCS in Section 6.3. The safety evaluation of the controls and instrumentation of the LPCI system is contained in Subsection 7.3.1.

There are two complete containment cooling systems. The RHR pumps in each of these systems receive power from ac power buses having standby power source backup supply. The two RHR pump motors and their associated motor-operated valves receive power from two separate buses. The pump's piping, controls, and instrumentation are separated and protected so that any single physical event or missile cannot make both loops inoperable.

The Fermi 2 design includes two parallel ac-powered inboard isolation valves (F009 and F608) fed from opposite electrical power divisions (F009 from Division I and F608 from Division II) and a dc-powered outboard isolation valve (F008) fed from Division II power. To prevent any inadvertent valve opening, the power fuses of the outboard isolation valve E1150F008 are removed during normal plant operation.

The following assumptions are used for the analyses of the procedures for attaining cold shutdown in the shutdown cooling mode.

a. The vessel is at about 70 psig and in a saturated condition

b. No offsite power is available

c. A worst single failure is assumed to occur (i.e., loss of a division of emergency power).

If a single failure (loss of Division II ac and dc power) were to cause an outboard suction valve (F008) to fail in the closed position, a handwheel is provided on the valve to allow manual operation. The shutdown would then continue in a normal manner using Division I of the RHR system.

Because manual operation cannot compensate for an electrical failure applied to inboard suction valve F009 (loss of Division I), the operator would open parallel valve F608, which is fed from the opposite division (Division II). Administrative controls would be used to enable the opening of valve F608 only when valve F009 could not be opened. These administrative controls require operation of a local key lock switch, the control room key lock switch and a push-button switch (in the control room) to open the valve. The local key lock switch prevents the valve opening from Multiple Spurious Operation (MSO). An auditory and visual feedback is provided by a control room alarm following the key lock switch operation. This is to prevent any inadvertent valve opening. Once valve F608 is open, the shutdown continues in the normal manner using Division II of the RHR system.

Thus, RHR system design conforms to the single-failure requirement of GDC 34.

### 5.5.7.5 Inspection and Testing

A design flow functional test of the RHR main system pumps is performed for each pump during normal plant operation by taking suction from the suppression pool and discharging through the test lines back to the suppression pool. The discharge valves to the RRS loops remain closed during this test, and reactor operation is undisturbed. An operational test of these discharge valves is performed by shutting the upstream valve after it has been satisfactorily tested, thereby establishing the RCPB at the downstream valve, and then operating the discharge valve. The discharge valves to the containment spray headers are checked in a similar manner by operating upstream and downstream valves individually. All these valves can be actuated from the main control room using remote manual switches. Control system design provides automatic return from test to operating mode if LPCI initiation is required during testing.

Testing of the sequencing of the LPCI mode of operation is performed after the reactor is shut down. Testing the operation of the valves required for the remaining modes of operation of the RHR system is performed as stated in the Technical Specifications and the pump and valve testing program (see Subsection 5.2.8.7).

Routine maintenance and tests, based on the manufacturers' recommendations and/or operating/maintenance experience, will be scheduled in accordance with the plant preventive maintenance program for the main system pumps, pump motors, and heat exchangers.

Preoperational tests are conducted during the final stages of plant construction prior to initial startup. These tests ensure correct functioning of all controls, instrumentation, pumps, piping, and valves. System reference characteristics such as pressure differentials and flow rates are documented during the preoperational testing and are used as base points for measurements obtained in subsequent operational tests.

For the containment spray cooling system, preoperational tests confirm that the containment spray headers and piping are clear of obstructions and the spray nozzles are capable of delivering rated flow. Air is injected into the drywell spray header via the blind flange connection on the outside of the primary containment. Unrestricted flow is verified through each spray nozzle. The spray nozzles in the suppression pool are checked with water during the suppression pool cooling tests.

For the suppression pool cooling system, the preoperational tests verify that the RHR heat exchanger shell-side design flow rate can be obtained while circulating water from the suppression pool. During the test, head versus flow curves are developed for reference in evaluating the future performance of the suppression pool cooling mode and the RHR pumps.

During plant operations, the pumps, valves, piping, instrumen-tation, wiring, and other components outside the primary containment can be inspected visually at any time. Components inside the primary containment can be inspected when the drywell is open for access. Testing frequencies are correlated with testing frequencies of the associated controls, and instrumen-tation is tested by the same action. When a system is tested, operation of the components is indicated by installed instrumentation.

The leak testing of all valves performing an isolation function between the high-pressure and the low-pressure boundary in the RHR system cannot be performed at the frequency prescribed in Section XI of the ASME Code. Because the testing removes one division of the RHR system from service, it is prudent to test only near the end of refueling outages or during maintenance on these systems. The Technical Specifications specify requirements for continued plant operation should the other division become inoperable.

Leakage tests are performed on these valves with high-pressure water. In every case, the low-pressure portion of the system is protected from overpressure with relief valves. The criterion for leakage tests is between 0.4 and 10 gpm which are values far below the capacity of the relief valves.

These valves cannot be exercised to any degree during plant operation. The exercising program for the gate and globe valves is part of the system functional tests described in the Technical Specifications. The check valves also are exercised at this time, using a mechanical exerciser as described in IWV-3522(b).

The RHR relief valves are removed as scheduled at refueling outages for bench tests and setting adjustments.

RHR heat exchanger tube leakage will be determined on a monthly basis by monitoring the service water return radiation levels. The effluent will be sampled such that significant leakage of reactor water into the RHR service water will be detected. Appropriate corrective actions will then be taken.

### 5.5.8 Reactor Water Cleanup System

#### 5.5.8.1 Power Generation Design Bases

The principal function of the RWCU system is to provide a means for reducing the concentration of radioactive and corrosive species in the reactor.

The RWCU system shall

a. Discharge excess reactor water during startup, shutdown, and hot standby conditions

b. Minimize reactor heat loss during system operation, except when used for Decay Heat Removal.

c. Remove solid and dissolved impurities from recirculated reactor coolant

d. Minimize temperature gradients in the RRS piping and vessel during periods of low flow rates.

e. Assist decay heat removal and coolant mixing during periods when the Reactor Pressure Vessel is under 250°F.

#### 5.5.8.2 Description

The RWCU system, shown in Figure 5.5-19, continuously purifies the reactor water. The system continuously removes water from the suction line of each RRS pump and from the reactor bottom head and returns it to the feedwater system. Water may also be sent to the main condenser (preferably) or to the radwaste system.

A regenerative heat exchanger is provided to maintain thermal efficiency during most operating modes of RWCU. However, a bypass line may be opened during times when the Reactor Pressure Vessel is under 250°F to allow cooled water to return to the reactor vessel. The RWCU system is operated at all times, when possible.

The major equipment of the RWCU system, located in the reactor building, includes pumps, regenerative and nonregenerative heat exchangers, and two filter-demineralizers with supporting equipment. The entire system is connected by associated valves and piping; controls and instrumentation provide proper system operation. Design data for the major pieces of equipment are presented in Table 5.5-2.

Reactor water is cooled in the regenerative and/or nonre-generative heat exchangers (or the nonregenerative heat exchangers alone when the shell side of the regenerative heat exchanger is bypassed), then filtered, demineralized, and returned to the reactor feedwater system through the shell side of the regenerative heat exchanger. A process diagram of the RWCU system is shown on Figure 5.5-20.

Because the maximum temperature of the filter-demineralizer units is limited by the ion exchange resin operating temperatures (Table 5.5-2), the reactor coolant must be cooled before being processed in the filter-demineralizer units. The regenerative heat exchanger transfers heat from the influent water to the effluent water. The nonregenerative heat exchanger cools the influent water further by transferring heat to the reactor building closed cooling water (RBCCW) system. The nonregenerative heat exchanger is designed to maintain the required filter- demineralizer operating temperature, even when the effectiveness of the regenerative heat exchanger is reduced by diversion of excess reactor water from the filter-demineralizer effluent to either the main condenser or the radwaste system or the regenerative heat exchanger is bypassed. A motor-operated valve in the suction line to the RWCU pumps automatically closes to prevent damage of the filter-demineralizer resins if the outlet temperature of the nonregenerative heat exchanger is high.

The filter-demineralizer units shown in Figure 5.5-21 are pressure-precoat type filters using mixed ion-exchange resins and fiber as a filter and ion-exchange medium. Spent resins are backwashed from a filter-demineralizer unit to a resin receiver tank from which they are transferred to the radwaste system for processing and disposal.

The suction line of the RCPB portion of the RWCU system contains two motor-operated isolation valves which automatically close in response to signals from the RCPB leak detection system. This action prevents the loss of reactor coolant and the release of radioactive material from the reactor. Subsections 7.6.1 and 5.2.7 and Table 5.2-11 describe the RCPB leak detection system.

The outermost isolation valve also automatically closes to prevent removal of liquid poison in the event of standby liquid control system actuation. These isolation valves may be remote manually operated to isolate the system equipment for maintenance or servicing.

A remote manually operated gate valve on the return line to the reactor provides long-term backup isolation of the system for the reactor. Instantaneous reverse-flow isolation is provided by two check valves in the RWCU return line, as shown in Figure 5.5-19. A motor operated isolation valve is provided in the RWCU line as shown in Figure 5.5-19. This valve automatically closes to isolate the RWCU system upon receipt of an isolation signal, or it may be remote manually operated.

#### 5.5.8.3 Safety Evaluation

To minimize the introduction of resins into the reactor in the event of septa failure in a filter-demineralizer, a strainer is installed on the outlet of each filter-demineralizer unit. Each strainer has a main control room alarm that is energized by high differential pressure. A bypass line is provided around the filter-demineralizer units for bypassing the units when necessary.

In the event of low flow or loss of flow in the system, flow is maintained through each filter-demineralizer by its own holding pump. Sample points are provided in the influent header and effluent line of each filter-demineralizer unit for continuous indication and recording of system conductivity. High conductivity is annunciated in the main control room. The alarm setpoints for the conductivity meters are 0.5 and 0.9 $\mu$ S/cm for the inlet and 0.09 $\mu$ S/cm for the outlet. The influent sample point is also used as the normal source of reactor coolant samples. Sample analysis also indicates the effectiveness of the filter-demineralizer units.

Operation of the RWCU system is controlled from the main control room except for the regenerative heat exchanger bypass. The manual bypass line isolation valve is administratively controlled and locked-closed during periods of nonuse. Figure 7.6-1 shows the RWCU system instrumentation and control logic.

Resin-changing operations, which include backwashing and precoating, are controlled from a local control panel in the reactor building.

#### 5.5.8.4 Inspection and Testing

Because the RWCU system is usually in service during plant operation, satisfactory performance is demonstrated without the need for any special inspection or testing beyond that specified in the manufacturers' instructions.

### 5.5.9 Main Steam Lines and Feedwater Piping

#### 5.5.9.1 Safety Design Bases

To satisfy the safety design bases, the main steam lines and feedwater piping have been designed

a. To accommodate operational stresses, such as internal pressures and earthquake loads, without a failure that could lead to the release of radioactivity in excess of the guideline values in published regulations

b. With suitable access to permit inservice testing and inspections.

#### 5.5.9.2 Power Generation Design Bases

The main steam lines and feedwater piping meet the following power generation design bases.

a. The main steam lines shall conduct steam from the RPV over the full range of reactor power operation

b. The feedwater lines shall conduct water to the RPV over the full range of reactor power operation.

5.5.9.3 Description

The main steam lines, consisting of four 24-in. diameter lines, are described in Section 10.3.

The feedwater piping is shown in Figure 10.4-10; at the drywell penetrations, it consists of two 20" lines. Each line includes two containment isolation valves. One simple check valve is inside the drywell. The isolation valve outside the drywell is an air actuated spring assist to close check valve. An additional check valve is located outside the drywell between the drywell wall and the spring assist to close check valve. In addition, a stop valve is provided between the isolation check valve and the reactor so that maintenance can be performed on the isolation valving and the HPCI system when the reactor is out of service. The design pressure and temperature of the feedwater piping between the reactor and the outermost isolation valve are 1275 psig and 450°F. The design pressure and temperature of the remaining reactor feedwater system are 1750 psig and 450°F. The Category I design requirements are placed on the feedwater piping from the reactor through the outboard isolation check valves and connected piping of 2-1/2 in. or larger nominal pipe size, up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation.

The materials used in the piping are in accordance with the applicable design code and supplementary requirements described in Section 3.2.

The reactor feedwater system is described in Subsection 10.4.7.

The penetration assemblies serve as a flexible joint, pressure boundary, and pipe jacket for process piping that penetrates the primary containment and its surrounding biological shield. The penetration assemblies, which are part of the primary pressure boundary, located between the inboard and outboard containment isolation valves, are rated as Class 1 in accordance with l0 CFR 50, Paragraph 50.55(a). Type I penetrations serve the primary pressure boundary process lines. Use of a flexible bellows and a penetration anchor is required because of fluctuations in operating temperature. Type I penetrations are provided with a hinged guard pipe around the process pipe to protect the bellows and the penetration sleeve from the effects of a postulated pipe rupture (Subsection 6.2.1.2.1.4).

The design, materials, and fabrication of the penetration assemblies are in accordance with the ASME B&PV Code Section III, 1971 edition, including the 1971 summer and winter addenda.

5.5.9.4 Safety Evaluation

Differential pressure on reactor internals under the assumed accident condition of a ruptured steam line is limited by the use of flow restrictors and by the use of four main steam lines. All main steam and feedwater piping is designed in accordance with the requirements defined in Section 3.2. Design of the piping in accordance with these requirements ensures meeting the safety design bases.

5.5.9.5 Inspection and Testing

Inspection and testing are carried out in accordance with Subsections 10.3.4 and 10.4.7. Inservice inspection is considered in the design of the main steam and feedwater piping. This consideration ensures adequate working space and access for the inspection of selected components.

The penetration assemblies are tested and inspected in accordance with the 1971 ASME Code Sections III and XI. They are designed for a 40-year service life.

5.5.10 Pressurizer

This subsection is not applicable to BWRs.

5.5.11 Pressurizer Relief Tank

This subsection is not applicable to BWRs.

5.5.12 Valves

Components beyond the RCPB that are part of systems or subsystems closely allied with the reactor coolant system consist of

a. Reactor feedwater system
b. RHR system
c. RCIC system
d. RWCU system
e. HPCI system
f. Standby liquid control (SLC) system
g. Core spray (CS) system.

5.5.12.1 Safety Design Bases

Line valves such as gate, globe, and check valves are located in the fluid systems to perform a mechanical function. Valves are components of the system pressure boundary and, having moving parts, are designed to operate efficiently to maintain the integrity of this boundary.

The valves shall operate under the internal pressure/temperature loading as well as the external loading experienced during the various system transient operating conditions. Table 3.9-27 lists the code class and design pressures and temperatures. The design criteria are described in Subsection 3.9.2.

5.5.12.2 Description

Class 2 and Class 3 line valves are designed in accordance with MSS-SP-66 or ANSI-B16.34. Original plant valves were procured in accordance with the then applicable ANSI-

B16.5 design. Materials used for Class 2 valves conform to the requirements of NC-3512 (a) and (b), and for Class 3 valves to ND-3512 (a) and (b). All materials, exclusive of seals and packings, are selected for 40-year plant operational life under full service conditions. Stress analyses show that Class 2 valves with motor, diaphragm, and piston operators only do not become inoperative under static seismic acceleration of 5g in the horizontal plane and 3g in the vertical plane.

Valve operators are sized to operate successfully under the maximum differential pressure determined in the design specification.

#### 5.5.12.3 Safety Evaluation

Line valves are shop tested by the manufacturer for performability. Pressure-retaining parts shall be subject to the testing and examination requirements of the appropriate ASME Code. To minimize internal and external leakage past seating surfaces, maximum allowable leakage rates are stated in the valve specifications for both the valve stem as well as the main seat for gate and globe valves.

Valve construction materials are compatible with the maximum anticipated radiation dosage for the service life of the valves.

#### 5.5.12.4 Inspection and Testing

Inspection and tests of line valves shall be in accordance with the applicable Code Class of the ASME B&PV Code Sections III and XI.

Valves that serve as containment isolation valves and that must remain closed or open during normal plant operation may be partially exercised during this period to ensure their operability at the time of an emergency or faulted condition. Other valves, serving as system block or throttling valves, may be fully exercised without jeopardizing system integrity for the same reason.

### 5.5.13 Safety and Relief Valves

Overpressurization protection, in the form of relief valves, is provided to systems and subsystems closely related to the reactor coolant system, such as

a. CS system

b. HPCI system

c. RCIC system

d. RHR system and its subsystems

e. SLC system

f. Control rod drive (CRD) system

g. RWCU system

h. Reactor recirculation seal purge subsystem.

The safety/relief valves of the reactor primary coolant system are discussed in Subsection 5.2.2. Table 5.5-3 shows relief valve characteristics for the above systems.

5.5.13.1 Safety Design Bases

The piping systems that are normally isolated by at least two power-operated isolation valves or one check valve and one power-operated valve from the RCPB are provided with relief valves to protect the piping from overpressurization caused by one or more of the following mechanisms.

a. Isolation valve leakage

b. Pump operation with system isolation

c. External radiant heat

d. Hot fluid impingement from broken pipes.

The relief valves are conservatively sized and designed by taking into account all the possible causes of overpressurization and their effects.

These valves are designed in accordance with the requirements of ASME B&PV Code Section III, NC-7000. Relief valves in Group D piping are exempt from the ASME B&PV Code Section III requirements.

5.5.13.2 Description

5.5.13.2.1 Core Spray System Relief Valves

The core spray pump suction lines and discharge lines are equipped with relief valves. The setpoints and capacities for these valves are shown in Table 5.5-3. The core spray system is not subject to any kind of energy input, except pump motor energy when pumps are operating against closed valves. The piping system is designed to withstand the shutoff head of the pumps. All relief valves installed in the system provide thermal relief for isolable portions of the system, with sufficient capacities to relieve the volume change of the entrapped fluid due to thermal expansion.

5.5.13.2.2 High Pressure Coolant Injection System Relief Valves

The HPCI pump suction line and the line to the gland seal condenser are equipped with relief valves to prevent overpressurization of the lines.

The setpoints and capacities for these valves and rupture disks are listed in Table 5.5-3.

The HPCI system is not subject to any kind of energy input except the hydraulic oil pump motor and the motors for the gland seal condenser vacuum and drain pumps.

5.5.13.2.3 Reactor Core Isolation Cooling System Relief Valves

The RCIC pump suction line and the cooling water line to the gland seal condenser are provided with relief valves with the capacities and setpoints listed in Table 5.5-3.

There is a rupture disk on the steam turbine for the turbine casing protection with the setpoint at 150 ± 10 psig and the capacity of 60,000 lb/hr at 165 psig.

The RCIC system is not subject to any kind of energy input, except when the pumps operate with closed valves.

5.5.13.2.4 Residual Heat Removal System Relief Valves

The RHR pump suction and discharge lines are provided with a relief valve in each line. The setpoints and capacities are listed in Table 5.5-3.

The overpressure protection relief valves have sufficient capacity to relieve the volume change of the entrapped fluid that results from thermal expansion in isolable portions of the system. The piping is designed to withstand the shutoff head of the pumps.

The RHR heat exchangers are also provided with a relief valve in each heat exchanger as listed in Table 5.5-3.

5.5.13.2.5 Standby Liquid Control System Relief Valves

A relief valve is provided in the discharge line of each pump. The setpoint and capacity of each valve are listed in Table 5.5-3.

5.5.13.2.6 Control Rod Drive System Relief Valves

The CRD pump suction lines are equipped with relief valves. The setpoints and capacities are listed in Table 5.5-3.

5.5.13.2.7 Reactor Water Cleanup System Relief Valves

The relief valves are installed on the shell and tube sides of the heat exchangers and on the line to the condenser.

The setpoints and capacities of the relief valves are listed in Table 5.5-3.

5.5.13.2.8 Feedwater System Relief Valves

The feedwater system is designed to the maximum pressure of the reactor coolant system up to and including the outermost isolation valve. Beyond the outermost isolation valve, the system is designated as a nonsafety class. Details of the feedwater system are discussed in Section 10.4.

5.5.13.3 Safety Evaluation

The assumptions made in the evaluation of the adequacy of the relief valves provided are conservative, and the setpoints and capacities of the valves are sufficiently conservative to protect the system and subsystem pipings and components from the effects of overpressurization.

Some of the conservative assumptions made are

a. Conservative isolation valve leakage values are used in sizing the relief valves

b. The system is considered isolated with the pump(s) operating at shutoff conditions. A 100 percent energy conversion from the pump motor horsepower to heat is assumed, neglecting heat losses and mechanical work

c. Jet impingement of steam from a nearby broken pipe is taken into account in sizing the relief valves. To be conservative, heating of the piping is assumed to be from the condensation of steam by the piping

d. The piping subject to heating is assumed to be uninsulated

e. Reaction force on the piping from relief valve operation is assumed to be R = 2 x P x A, where R is the reaction force, P is the pressure setting of the valve, and A is the area of the valve inlet.

The radiation fields considered for the EQ Program relief valve designs are given in Table 3.11-5. Other valve characteristics can be found in Table 5.5-3.

5.5.13.4 Inspection and Testing

Inspection and testing were carried out in accordance with ASME (PTC) 25.2, ASME B&PV Code Section III. Inservice inspection of ASME Class 1, 2, and 3 valves will be performed in accordance with ASME B&PV Code Section XI.

5.5.14 Component Supports

Support elements are provided for those components beyond the RCPB that are in systems or subsystems closely allied with the reactor coolant system. These systems include

a. Reactor feedwater system

b RHR system

c. RCIC system

d. RWCU system

e. HPCI system

f. SLC system.

5.5.14.1 Safety Design Bases

The design procedures, design loading, and acceptability criteria are as described in Subsection 3.9.2. Flexibility calculations and seismic analysis for Class 2 and 3 components are made in accordance with NC/ND 3600 of the ASME B&PV Code Section III. Support types, materials used for fabricated support elements, and recommended pipe support spacing are in accordance with established industry practice and AISC Specifications.

5.5.14.2 Description

The use and location of rigid-type supports, variable or constant spring-type supports, and anchors or guides are determined by flexibility and stress analysis. Component support elements are manufacturers' standard items. Direct weldment to thin-wall pipe is avoided where possible.

#### 5.5.14.3 Safety Evaluation

Design loadings used for flexibility and seismic analysis toward the determination of adequate component support systems include all transient loading conditions expected by each component.

Provisions are made to restrain spring-type supports for the initial deadweight loading due to hydrostatic testing of steam systems to prevent damage to this type of support.

#### 5.5.14.4 Inspection and Testing

After completion of the installation of a support system, hanger elements will be visually examined to ensure that they are in correct adjustment to their cold setting position. Thermal expansion testing for selected piping systems will be conducted during the preoperational and startup phases. Spring-type hangers will be inspected to ensure that they will function properly between their hot and cold setting positions.

## 5.5 COMPONENT AND SUBSYSTEM DESIGN

## REFERENCES

1. P. W. Ianni, Effectiveness of Core Standby Cooling Systems for General Electric Boiling Water Reactors, APED-5458, March 1968.

2. Letter, "GE Recirculation Pump Potential Overspeed," Revision 2, March 30, 1979.

3. General Electric Company, Analysis of Recirculation Pump Under Accident Conditions, Revision 2, submitted to the NRC March 30, 1979.

4. J. Grimaldi, Analysis of Recirculation Pump Overspeed in a Typical General Electric Co. BWR, NEDO-l0677, October l972.

5. General Electric Company, Design and Performance of General Electric Boiling Water Reactor Main Steam Line Isolation Valves, APED-5750, March l969.

6. NUTECH, Fermi 2 Long Term Program Plant Unique Analysis Report for the Torus Attached Piping (PUAR-TAP), DET-19-076-6, transmitted to NRC per EF2-63925, June 10, 1983.

7. GE-NE-E11-00071-01, "Alternate SDC With Natural Circulation", June 1995.

8. JNL-95-8-01, “Alternate SDC Supplemental Analysis”, August, 16, 1995.

9. NRC-08-0064, dated 10-14-08, Fermi 2 Nine-Month Response to NRC Generic Letter 2008-01, “Managing Gas Accumulation in Emergency Core Cooling, Dacay Heat Removal, and Containment Spray Systems”

10. NRC Letter, dated 5-28-09, to James H. Riley from William H. Ruland, “Preliminary Assessment of Responses to GL 2008-01 and Future NRC Staff review Plans”

11. DTE Electric, NRC-14-0028, “Fermi 2 License Renewal Application”, dated April 24, 2014.

TABLE 5.5-1 REACTOR RECIRCULATION SYSTEM DESIGN CHARACTERISTICS

| | |
|---|---|
| External loops | |
| Number of loops | 2 |
| Pipe sizes (nominal O.D.) | |
| Pump suction, in. | 28 |
| Pump discharge, in. | 28 |
| Discharge manifold, in. | 22 |
| Recirculation inlet line, in. | 12 |
| Design pressure, psig/design temperature, °F | |
| Suction piping and valve up to and including pump suction nozzle | 1250/575 |
| Pump | 1525/562[a] |
| Discharge piping up to vessel | 1500/575 |
| Discharge valve | 1525/575 |
| Pump auxiliary piping and cooling water piping | 150/212 |
| Vessel bottom drain | 1250/575 |
| Operation at rated conditions | |
| Reactor recirculation system pump | |
| Flow, gpm | 45,200 |
| Flow, lb/hr | $17.1 \times 10^6$ |
| Total developed head, ft | 710 |
| Suction pressure (static), psia | 1033 |
| Required NPSH, ft | 135 |
| Water temperature, °F | 535.4 |
| Pump brake HP (min) | 7050 |
| Flow velocity at pump suction (approximate), ft/sec | 28.4 |

[a] The reactor recirculation system pump design pressure and temperature conditions envelop the system discharge piping design requirements.

TABLE 5.5-1 REACTOR RECIRCULATION SYSTEM DESIGN CHARACTERISTICS

| | |
|---|---|
| Pump motor | |
| Voltage rating | 4160 |
| Phase3 | |
| Frequency, Hz | 60 |
| Jet pumps | |
| Number | 20 |
| Total jet pump flow, lb/hr | $105 \times 10^6$ |
| Throat I.D., in. | 8.18 |
| Diffuser I.D., in. | 19.0 |
| Nozzle I.D. (representative), in. | 3.14 |
| Diffuser exit velocity, ft/sec | 15.8 |
| Jet pump head, ft | 87.8 |
| Reactor recirculating system loop valves | |
| Type | Gate valve |
| Actuator | Motor |
| Material | Austenitic stainless steel |
| Valve size diameter, in. | 28 |

## TABLE 5.5-2 REACTOR WATER CLEANUP SYSTEM EQUIPMENT DESIGN DATA

Reactor water cleanup system pumps

Number required - two

Capacity (each) - 50 percent of system flow

Discharge flow, gpm/pump - 180

Design temperature, °F - 575

Design pressure, psig – 1400

Heat exchangers

| | Regenerative | Nonregenerative |
|---|---|---|
| Reactor coolant flow rate, lb/hr | 133,000 | 133,000 |
| Shell-side pressure, psig | 1450 | 150 |
| Shell-side temperature, °F | 575 | 370 |
| Tube-side pressure, psig | 1450 | 1450 |
| Tube-side temperature, °F | 575 | 564 |

Filter-Demineralizers

Number required - two

Capacity (each) - 50 percent of system flow

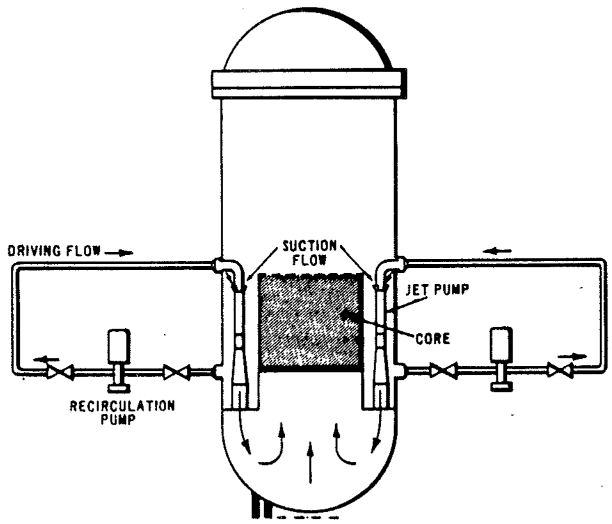
Flow rate/unit, lb/hr - 66,500 (Nominal)

Design temperature, °F - 150

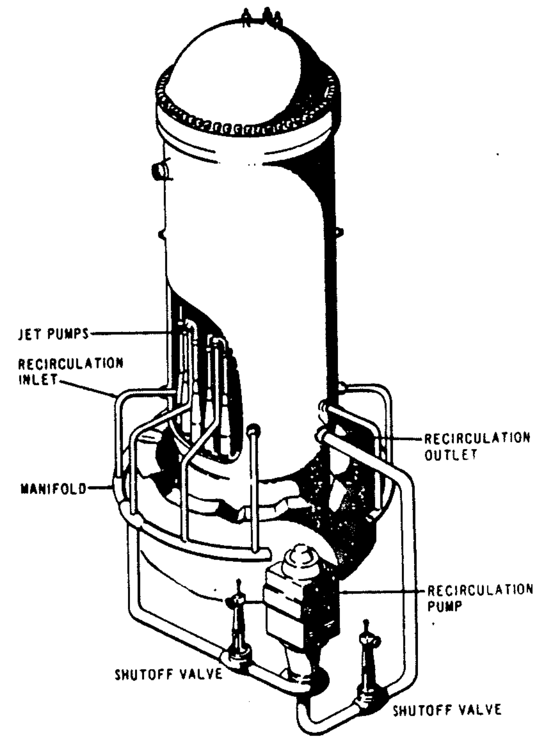
Design pressure, psig - 1400

## TABLE 5.5-3 RELIEF VALVES

| Valve | Location | Setpoint (psig) | Capacity |
|---|---|---|---|
| E1100F029 | RHR pump suction (Shutdown Cooling Header) | 140 | 20 gpm |
| E1100F030A | RHR pump suction | 150 | 20 gpm |
| E1100F030D | RHR pump suction | 150 | 20 gpm |
| E1100F030C | RHR pump suction | 150 | 20 gpm |
| E1100F030B | RHR pump suction | 150 | 20 gpm |
| E1100F025A | RHR pump discharge | 450 | 9,000 lb/hr |
| E1100F025B | RHR pump discharge | 450 | 9,000 lb/hr |
| E1100F001A | RHR heat exchanger | 450 | 46 gpm |
| E1100F001B | RHR heat exchanger | 450 | 38 gpm |
| C1100F001A | CRD pump suction | 250 | 90 gpm |
| C1100F001B | CRD pump suction | 250 | 90 gpm |
| G3300F036 | RWCU system to condenser | 150 | 270 gpm |
| G3300F023B | RWCU system nonregenerative heat exchanger – shell | 150 | 39 gpm |
| G3300F023A | RWCU system nonregenerative heat exchanger – tube | 1,450 | thermal relief |
| G3300F025C | RWCU system regenerative heat exchanger – tube | 1,450 | thermal relief |
| G3300F025A | RWCU system regenerative heat exchanger - shell | 1,450 | thermal relief |
| E5100F017 | RCIC pump suction | 100 | 10 gpm |
| E5100F018 | RCIC condenser cooling | 125 | 10 gpm |
| C4100F029B | SLCS pump discharge | 1,370 | 41 gpm |
| C4100F029A | SLCS pump discharge | 1,370 | 41 gpm |
| E2100F011A | CSS pump discharge | 500 | 100 gpm |
| E2100F011B | CSS pump discharge | 500 | 100 gpm |
| E2100F012A | CSS pump discharge | 500 | 100 gpm |
| E2100F012B | CSS pump discharge | 500 | 100 gpm |
| E2100F032B | CSS pump suction | 100 | 20 gpm |
| E2100F032A | CSS pump suction | 100 | 20 gpm |
| E4100F020 | HPCI pump suction | 100 | 10 gpm |
| E4100F050 | HPCI cooling water line | 125 | 10 gpm |
| B3100F015A | Reactor Recirculation Seal Purge Subsystem | 1,250 (Approximately) | thermal relief |
| B3100F015B | Reactor Recirculation Seal Purge Subsystem | 1,250 (Approximately) | thermal relief |
| E4150D003 | HPCI turbine exhaust rupture disk | 165 - 185 psig burst pressure | 43 lbm/sec |
| E4150D004 | HPCI turbine exhaust rupture disk | 165 - 185 psig burst pressure | 43 lbm/sec |




ELEVATION




ISOMETRIC

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**FIGURE 5.5-1**

**REACTOR RECIRCULATION SYSTEM ELEVATION AND ISOMETRIC**

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Refer to Plant Drawing M-2833

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FIGURE 5.5-2, SHEET 1

REACTOR RECIRCULATION SYSTEM
NUCLEAR BOILER SYSTEM

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Refer to Plant Drawing I-2106-01

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**FIGURE 5.5-2, SHEET 2**

**REACTOR RECIRCULATION SYSTEM P&ID**

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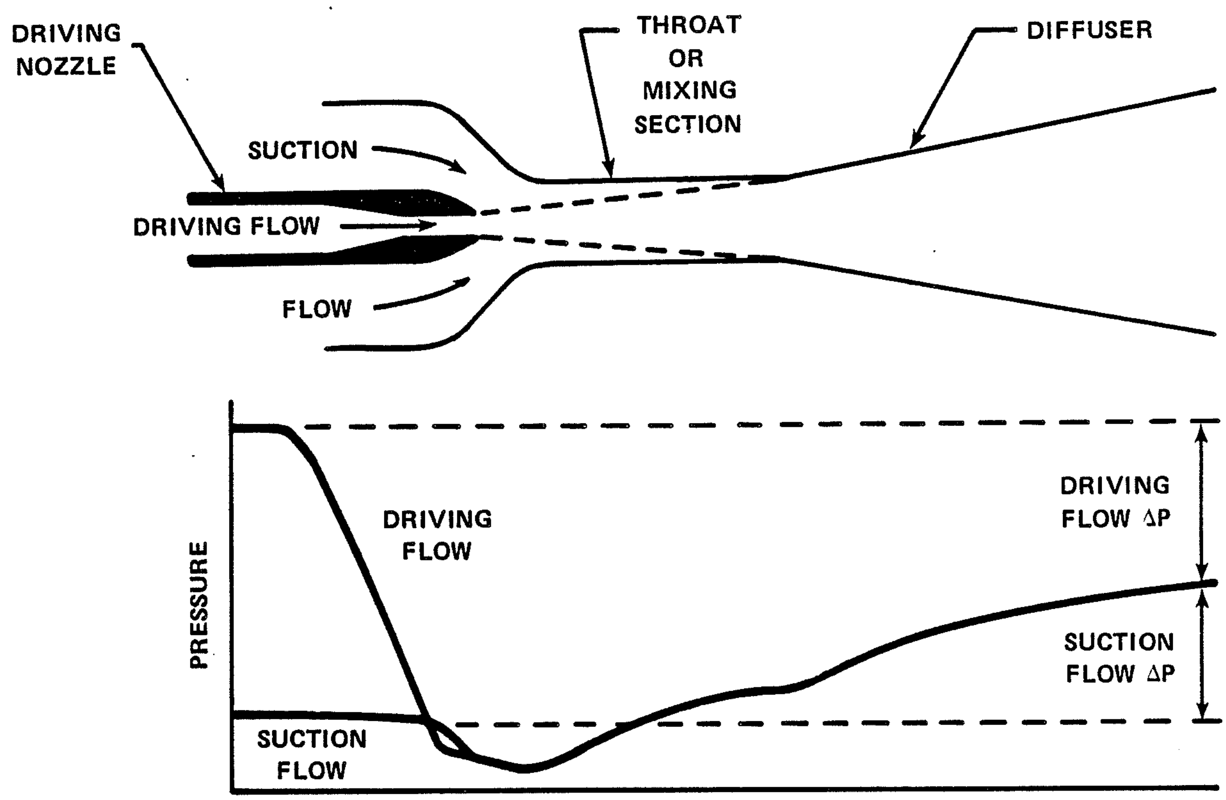


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FIGURE 5.5-3

OPERATING PRINCIPLE OF JET PUMP

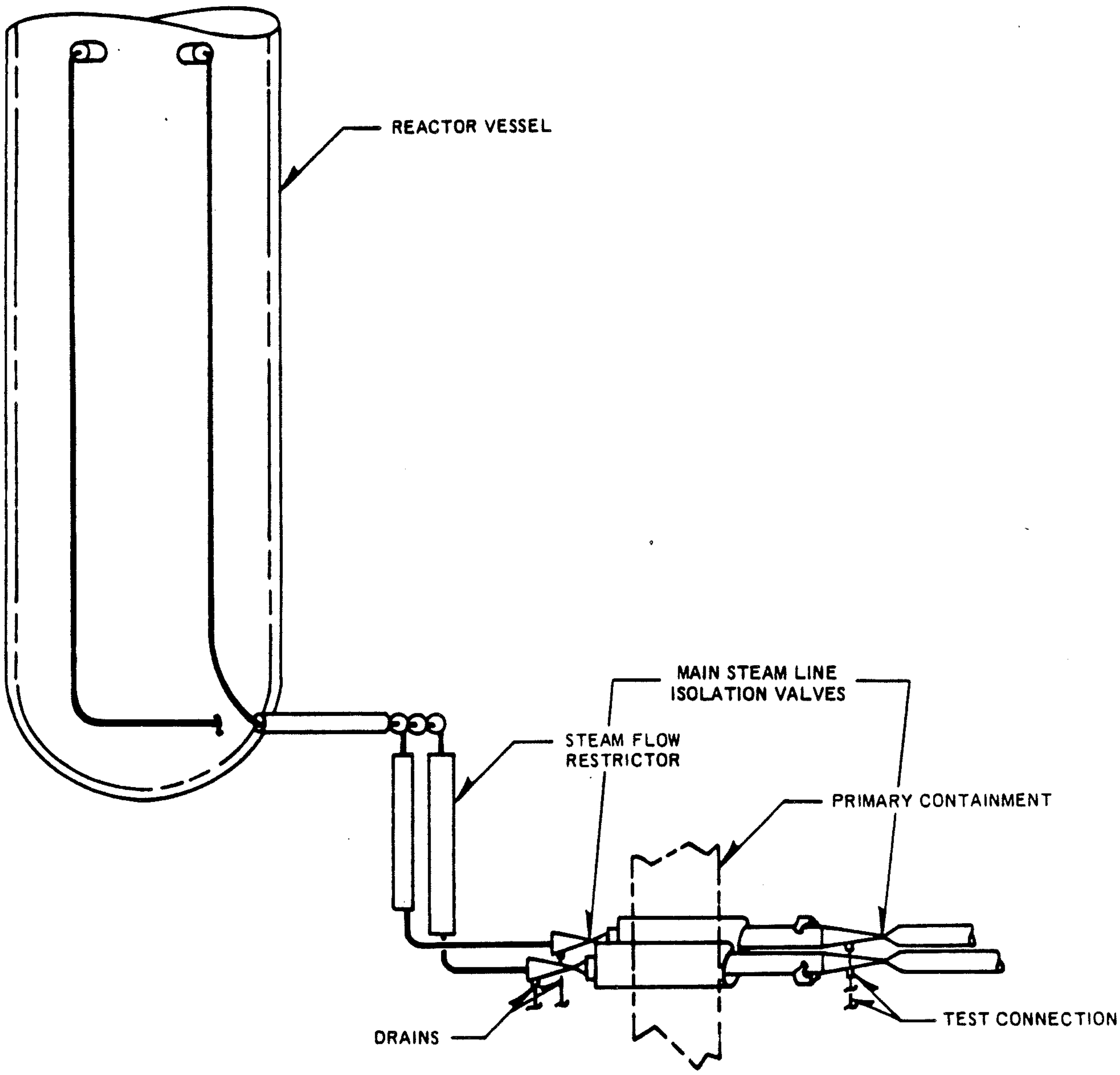


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FIGURE 5.5-4

MAIN STEAM LINE FLOW RESTRICTOR LOCATION

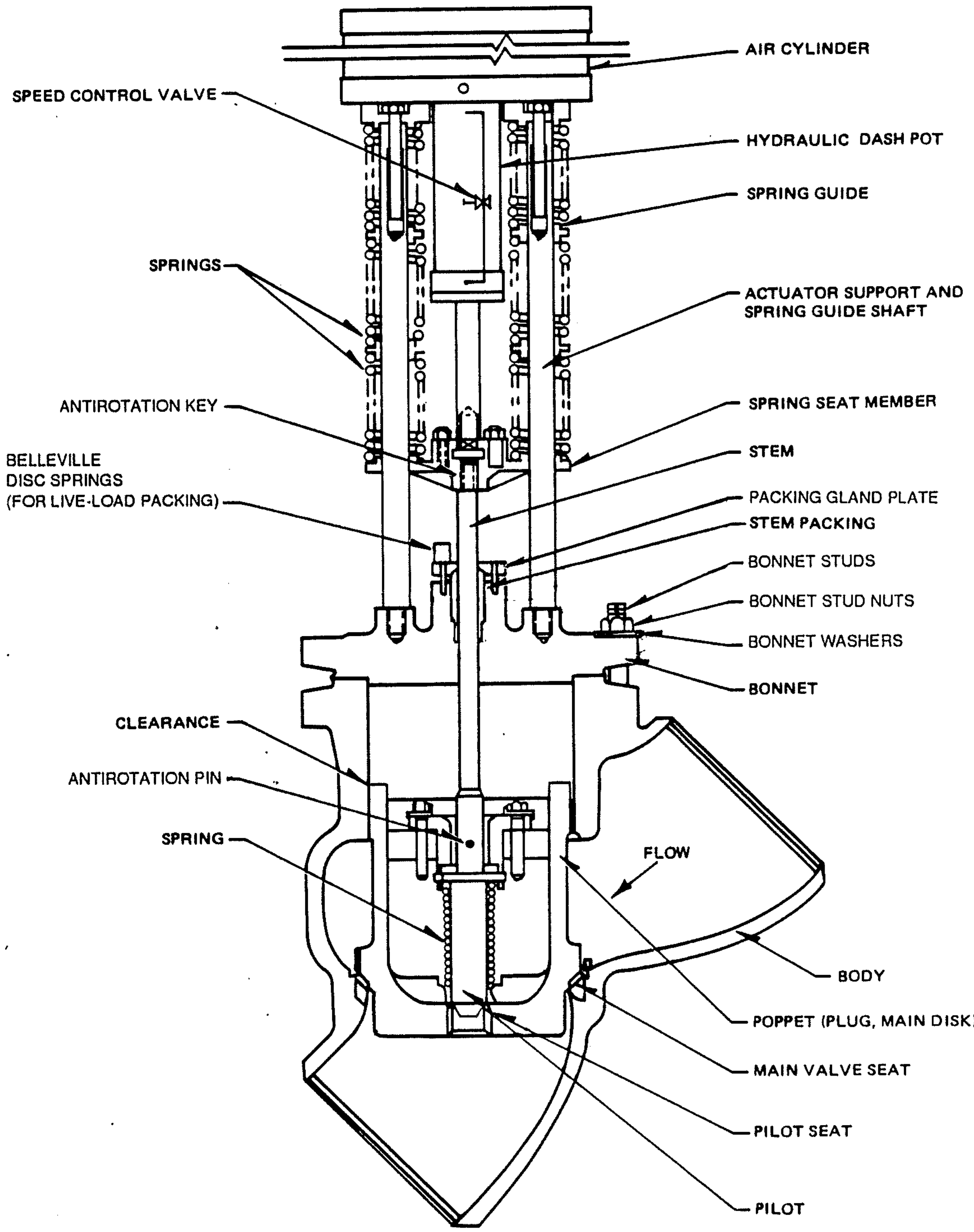


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FIGURE 5.5-5

TYPICAL MAIN STEAM LINE ISOLATION VALVE

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Refer to Plant Drawing M-5859

Figure Intentionally Removed
Refer to Plant Drawing M-2044

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FIGURE 5.5-7, SHEET 1

REACTOR CORE ISOLATION COOLING SYSTEM P&ID

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Refer to Plant Drawing M-2045

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FIGURE 5.5-7, SHEET 2
REACTOR CORE ISOLATION COOLING SYSTEM P&ID

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Refer to Plant Drawing M-5876

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FIGURE 5.5-8

REACTOR CORE ISOLATION COOLING VALVE POSITIONS DURING NORMAL OPERATION

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Refer to Plant Drawing M-5877

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FIGURE 5.5-9

REACTOR CORE ISOLATION COOLING INITIAL COOLING FOLLOWING MAIN CONDENSER ISOLATION

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Refer to Plant Drawing M-5878

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FIGURE 5.5-10

REACTOR CORE ISOLATION COOLING FOLLOWING MAIN CONDENSER ISOLATION USING SUPPRESSION POOL AS A BACKUP WATER SOURCE

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Refer to Plant Drawing M-5879

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FIGURE 5.5-11

REACTOR CORE ISOLATION COOLING TEST MODE

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Refer to Plant Drawing M-5880



FIGURE 5.5-12

REACTOR CORE ISOLATION COOLING MINIMUM FLOW MODE

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Refer to Plant Drawing M-2083

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FIGURE 5.5-13, SHEET 1

RESIDUAL HEAT REMOVAL SYSTEM P&ID

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FIGURE 5.5-13, SHEET 2

RESIDUAL HEAT REMOVAL SYSTEM P&ID

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Refer to Plant Drawing M-5862

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FIGURE 5.5-14

RESIDUAL HEAT REMOVAL VALVE POSITIONS DURING NORMAL REACTOR OPERATION

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Refer to Plant Drawing M-5863

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FIGURE 5.5-15

RESIDUAL HEAT REMOVAL SHUTDOWN COOLING

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Refer to Plant Drawing M-5864

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FIGURE 5.5-16

RESIDUAL HEAT REMOVAL CONTAINMENT COOLING MODE

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Refer to Plant Drawing M-5865

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FIGURE 5.5-17

RESIDUAL HEAT REMOVAL SUPPRESSION POOL COOLING MODE

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FIGURE 5.5-19

REACTOR WATER CLEANUP
REACTOR BUILDING

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Refer to Plant Drawing M-5858

Figure Intentionally Removed
Refer to Plant Drawing M-2047

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FIGURE 5.5-21

REACTOR WATER CLEANUP FILTER
DEMINERALIZER SYSTEM P&ID

## 5.6 INSTRUMENTATION REQUIREMENTS

The functional requirements for the reactor coolant system instrumentation are discussed in the following subsections. Details of the design and logic of the instrumentation are discussed in Chapter 7.

### 5.6.1 Neutron Monitoring System

This system is described in Subsection 7.1.2.1.4.

### 5.6.2 Reactor Pressure Vessel Instrumentation

Reactor pressure vessel (RPV) instrumentation is designed to provide the operator with sufficient indication of reactor core flow rate, RPV water level, RPV pressure, and nuclear system leakage to maintain proper operating conditions.

#### 5.6.2.1 Reactor Pressure Vessel Temperature

The RPV temperature is determined on the basis of reactor coolant temperature. Temperatures needed for operation and for compliance with the Technical Specifications operating limits are obtained from one of several sources, depending on the operating condition. During normal operation, either reactor pressure and/or the inlet temperature of the coolant in the RRS loops can be used to determine the RPV temperature. Below the operating span of the temperature detectors in the RRS loop, the pressure is used for determining the temperature. Below 212°F the coolant temperature in the RPV, and thus the RPV temperature, is reasonably determined by the reactor water cleanup (RWCU) system inlet temperature.

#### 5.6.2.2 Reactor Pressure Vessel Water Level

The number of RPV water level indications is sufficient to provide the operator with information to determine the adequacy of the coolant inventory to cool the fuel. In addition, by verifying that RPV water level is not rising to an abnormally high level, the operator is ensured that turbines are not endangered by the possibility of water carried into the steam lines. The common zero reference point for all vessel level instruments at Fermi 2 is the top of the active fuel.

#### 5.6.2.3 Reactor Pressure Vessel Coolant Flow Rates and Differential Pressures

Flow instruments, differential pressure instruments, and recorders are provided so that the core coolant flow rates and the hydraulic performance of RPV internals can be determined.

#### 5.6.2.4 Reactor Pressure Vessel Internal Pressure

Pressure switches, indicators, and transmitters detect RPV internal pressure from the same instrument lines used for measuring RPV water level.

### 5.6.2.5 Reactor Pressure Vessel Top Head Flange Leak

A connection is provided on the RPV flange into the annulus between the two metallic seal rings used to seal the RPV and top head flanges. This connection permits detection of leakage past the inner seal ring, and is described further in Subsection 5.2.7.