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May 10, 1983

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: Limerick Generating Station, Units 1 & 2

REFERENCE: Meeting between Containment Systems Branch (CSB) Reviewer, Mr. F. Eltawila, and Philadelphia Electric Company on April 7, 1983

Dear Mr. Schwencer:

The referenced meeting was held to discuss twenty issues of concern to the CSB. As a result of the discussions, we will make changes to the FSAR, the Design Assessment Report and to the responses previously provided to Questions raised by the CSB. Attached are drafts of the changes prepared in response to issues 1, 3a, 3b, 4, 7, 8, 10, 11, 12, 13, 15, 16 and 18. These draft changes will be formally incorporated into the FSAR revision scheduled for June 1983.

Also attached is a brief discussion of issue 17, indicating that a report on the functionality of the Limerick Purge & Vent Valves will be submitted to the NRC staff for review later in May 1983.

B001

Information responding to issues 5b, 5c, 5d and 5e will be provided later in May 1983, as will information on issue 20.

Very truly yours,

E. J. Bradley

JTR/cam c/1
Attachments
Copy to: See attached service list

cc: Judge Lawrence Brenner (w/o enclosure)
Judge Richard F. Cole (w/o enclosure)
Judge Peter A. Morris (w/o enclosure)
Troy B. Conner, Jr., Esq. (w/o enclosure)
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Atomic Safety and Licensing Appeal Board (w/o enclosure)
Atomic Safety and Licensing Board Panel (w/o enclosure)
Docket and Service Section (w/o enclosure)

Issue 1

The CSB review of the applicant's response to Question 480.7 has determined that the requirements of Appendix A to SRP Section 6.2.1.1.c concerning steam bypass capability have been met with one exception. The exception is that the applicant must commit to the leakage test and surveillance requirements stated in Positions B.2 and B.3 of Appendix A to SRP Section 6.2.1.1.c including the specified frequencies. (Open Item)

Response

The response to Question 480.7 and section 6.2.6.5.1 have been revised to resolve this concern and are attached.

flow rate of one spray system is 500 gpm. With two spray systems in operation, the required efficiency would be halved. The spray efficiency is typically on the order of 0.7 and, therefore, even with a single system is operation, the termination of the wetwell (and drywell) pressure increase is assured.

Spray Temperature	Required Efficiency of 1 Wetwell Spray System
70°F	0.22
90°F	0.24
120°F	0.28

c. The wetwell spray system is to be used to mitigate the consequences of suppression pool steam bypass high pressure. Limerick is in compliance with the guidelines of SRP 3.2.2 and Regulatory Guide 1.26 because the safety-related design basis for the containment spray system is that it provide a means of pressure reduction, not heat removal.

Therefore seismic Category I/Quality Group C standards are adequate for the wetwell spray headers. The containment spray system is also designed to be operable following a loss of offsite power plus a single failure.

As discussed in Section 6.2.1.1.5.2, use of the containment sprays is only one option available to the operator to respond to high pressure resulting from steam bypass of the suppression pool.

The Quality Group designations for the containment spray system have not changed since the PSAR was submitted.

d. Section 6.2.6.5.1 and Table 14.2-4 have been changed to provide the requested information.

e. A visual inspection will be conducted prior to each integrated leak rate test to detect possible drywell-to-suppression bypass leakage paths. A visual inspection of each primary containment vacuum relief valve assembly will be conducted during each refueling outage to verify that it is clear of foreign matter.

f. The vacuum relief valve position indicator system has adequate sensitivity to detect a total valve opening, for all valves, that is less than the bypass capability for a small break. Valve opening is detectable at a disk lift of 0.06 inches or greater above the valve

seat. Even assuming that all the vacuum breakers are open by 0.06 inches, the corresponding leakage area, $A/\sqrt{k}$, is well below 0.05 ft^2. Therefore, the valve leakage, which is based on the assumption that the valve opening is evenly divided among all the vacuum breakers, is well within the limits of acceptable bypass leakage.

g. Vacuum breakers will be ~~treated~~ tested for operability at an interval specified by the technical specifications. This surveillance testing will be in accordance with the BWR Standard Technical Specifications (4.6.4.2.b).

leakage rate tested with that liquid. The liquid leakage measured is neither converted to equivalent air leakage nor added to the Type B and C test totals. Isolation valves tested with liquid are identified in Table 6.2-25.

The acceptance criteria for all penetrations and isolation valves subject to Type B and C tests are given in Chaper 16.

6.2.6.4 Scheduling and Reporting of Periodic Tests

The periodic leakage rate test schedules for Types A, B and C tests are given in Chapter 16.

Type B and C tests can be conducted at any time during normal plant operations or during shutdown periods, so long as the time interval between tests for any individual Type B or C test does not exceed the maximum allowable interval specified in Chapter 16. Each time a Type B or C test is completed, the overall total leakage rate for all required Type B and C tests is corrected for any differences noted.

Provisions for reporting test results are given in Chapter 16.

6.2.6.5 Special Testing Requirements

6.2.6.5.1 Drywell Steam Bypass Test

Following the drywell structural integrity test, described in Section 3.8.1.7, a preoperational drywell leakage rate test is performed at drywell design pressure. Table 14.2-4 gives the test descriptions. Preoperational and periodic drywell leakage rate tests at a reduced pressure, defined in Chapter 16, are performed following the preoperational ~~and periodic~~ Type A test [and periodically thereafter.] ~~described above~~. These drywell leakage rate tests verify, over the design life of the plant, that no paths for gross leakage from the drywell to the suppression chamber air space bypassing the pressure suppression feature exist. The combination of the design pressure and reduced pressure leakage rate tests also verifies that the drywell performs adequately for the full range of postulated primary system break sizes. The drywell leakage rate limits specified in Chapter 16 are based on a value of 10% of the allowable bypass $A/\sqrt{K}$ for small breaks that are described in Section 6.2.1.1.5.4.

Drywell leakage rate tests are performed with the drywell isolated from the suppression chamber. Valves and system lineups are the same as for the Type A test except any paths for equalizing drywell and suppression chamber pressure open during the Type A test are isolated. The drywell atmosphere is allowed to stabilize for a period of one hour after attaining test pressure. Leakage rate test calculations, using the pressure decay method, commence after the stabilization period.

The pressure decay method is based on drywell atmosphere pressure and temperature observations and the known drywell free air volume specified in Table 6.2-22. Leakage rate is calculated from the pressure and temperature data, drywell free air volume, and elapsed time.

The periodic drywell leakage rate test pressures, test duration, and acceptance criteria are specified in Chapter 16. Periodic drywell leakage rate tests are performed at the intervals specified in Chapter 16. This surveillance testing will be in accordance with the BWR Standard Technical Specifications (4.6.2.1.d).

6.2.7 POST-ACCIDENT SYSTEM ISOLATION

Following an accident in which significant fuel damage is postulated to occur, a number of plant systems whose piping penetrates the primary containment may contain highly radioactive fluids. Adequate system isolation features exist to ensure that the integrity of these systems will be maintained.

6.2.7.1 System Isolation Provisions

The boundaries of potentially contaminated systems are adequately isolated by one of the following:

a) Two normally closed manual valves

b) One normally closed manual valve (low pressure piping)

c) One or two normally closed manual valves and a cap

d) One safety relief valve or one rupture disc

e) Two check valves

f) One remotely actuated valve and one check valve

g) Two remotely actuated valves

In cases where a remotely actuated valve is required to change position to provide system isolation, the valve receives an auto isolation signal. In some cases a system isolation valve does not receive a direct isolation signal but is interlocked to close when a containment isolation valve or other valve opens to permit fluid flow from the containment.

Table 6.2-26 lists remotely-actuated system isolation valves, their normal and required accident positions and their actuation signals. Containment isolation valves that also provide post-accident system isolation are not included in this table but are listed in Table 6.2-17.

Issue 3(a)

The applicant has not adequately demonstrated why inadvertent actuation of both drywell spray trains should not be considered in the evaluation of the drywell floor reverse pressure design basis and the drywell external pressure design basis. Also the drywell floor reverse pressure design basis must be stated because the FSAR and Design Assessment Report (DAR) give conflicting values. (Open Item)

Response

The response to question 480.4 has been changed to demonstrate why inadvertent actuation of both drywell spray trains should not be considered in the evaluation of the drywell floor reverse pressure design basis and the drywell external pressure design basis. The revised response is attached.

QUESTION 480.4 (Section 6.2.1.1)

Provide a detailed description of the administrative procedures that will preclude the actuation of both drywell spray networks whenever the suppression pool temperature is below 105°F (see FSAR Section 6.2.1.1.4).

RESPONSE

Operation of the drywell sprays at Limerick will be governed by appropriate emergency procedures, which will be written and revised to implement the BWR Owners Group Emergency Guidelines. There will be no other procedural requirements or administrative directives that will cause drywell sprays to be used. At present, the emergency procedures will be written to Revision 2 of the Owners Group Emergency Guidelines, which is currently under review by the Commission. Specifically, the following steps in the guidelines direct use of the drywell sprays: PC/P-3, PC/P-6, SP/L-3.3 and SP/L-3.4. At each of these steps, the operators will be directed first to determine if the present combination of suppression chamber temperature and drywell pressure fall below the drywell spray initiation pressure limit. If the combination of parameters is below the limit, the operator is directed to initiate drywell spray with a slow rate not to exceed the maximum drywell spray flow rate limit. Both of these limits will be calculated in accordance with Appendix C to the Owners Group Emergency Guidelines. The specific details of the calculations are given in Appendix C, Section 8.0 for the drywell spray initiation pressure limit and Section 9.0 for the maximum drywell spray flow rate limit. These limits prevent the generation of drywell negative pressures relative to secondary containments and suppression pools that could be damaging to the containment vessel.

The inadvertent actuation of both drywell spray trains is also prevented by the type and location of the control room switches. The outboard valves are normally closed and have keylocked switches that are approximately 8 feet apart.

Item 3(b)

The applicant has not adequately demonstrated why inadvertent actuation of both drywell spray trains should not be considered in the evaluation of the drywell floor reverse pressure design basis and the drywell external pressure design basis. Also the drywell floor reverse pressure design basis must be stated because the FSAR and Design Assessment Report (DAR) give conflicting values. (Open Item)

Response

FSAR Sections 3.8.3.3.1 and 6.2.1.1.3.1, FSAR Table 6.2-1, and DAR Tables 1.3-2 and 1.4-1 have been changed to specify the diaphragm slab design loadings as
ΔP(UP) = 20 PSID and
ΔP (DOWN) = 30 PSID

FSAR Table 6.2-5 has been changed to specify the diaphragm slab calculated loading as ΔP (DOWN) = 25.995 psid.

The revised sections and tables are attached.

Load or Phenomenon	NRC Acceptance Criteria	Criteria Source	LGS Position
b. Large Structures	None - Plant unique load where applicable.	NUREG-0487	Not Applicable No large structures
c. Grating	P drag vs. grating area correlation and pool velocity vs. elevation. Pool velocity from the PSAM. P drag multiplied by dynamic load factor.	NUREG-0487	Acceptable
4. Wetwell Air Compression			
a. Wall Loads	Direct application of the PSAM calculated pressure due to wetwell compression.	NUREG-0487	Acceptable
b. Diaphragm Upward Loads	5.5 psid for diaphragm loadings only.	NUREG-0808	Acceptable.
5. Asymmetric LOCA Pool	Use 20 percent of maximum bubble pressure statically applied to 1/2 of the submerged boundary.	NUREG-0487 Supplement 1	Acceptable
C. Steam Condensation and Chugging Loads			
1. Downcomer Lateral Loads			
a. Single-Vent Loads (24 in.)	Dynamic load to end of vent. Half sine wave with a duration of 3 to 6 ms and corresponding maximum amplitudes of 65 to 10 Klbf.	NUREG-0808	Acceptable
b. Multiple-Vent Loads (24 in.)	Prescribed variation of load per vent vs. number of vents. Determined from single vent dynamic load specification	NUREG-0808	Acceptable

CALCULATED ΔP UP = 10.6 PSID (FIG 4.2-3, 4.2-4).
DIAPHRAGM SLAB DESIGN ΔP UP = 20 PSID

TABLE 1.4-1

CONTAINMENT DESIGN PARAMETERS

	Drywell	Suppression Chamber
DRYWELL AND SUPPRESSION CHAMBER		
Internal design pressure, psig	55	55
External to internal design differential pressure, psid	5	5
Drywell deck design differential pressure, psid	30 DOWNWARD	20 UPWARD
Design temperature, °F	340	220
Drywell net free volume including downcomers, ft^3	248,950	
Suppression chamber free volume, ft^3		
Low level		161,350
High level		149,425
Suppression pool water volume, ft^3		
Low level		115,903
High level		127,756
Suppression pool net surface area, outside pedestal, ft^2		4974
Supression pool depth, ft		
Low level		22'
Normal level		23'
High level		24'-3"
VENT SYSTEM		
Number of downcomers	87	
Nominal downcomer diameter, ft	2	
Total vent area, ft^2	256.5	

Section 1.8 provides references to Regulatory Guides discussed in the FSAR. Regulatory Guides specific to this section are discussed in this section.

3.8.3.3 Loads and Loading Combinations

Tables 3.8-2 and 3.8-5 through 3.8-8 list the loading combinations used for the design and analysis of the containment internal structures.

The internal structures are also analyzed for hydrodynamic loads resulting from main steam relief valve discharge and LOCA phenomena. For a definition of loads and loading combinations (including hydrodynamic loads), see Refs 3.8-1 and 3.8-2.

3.8.3.3.1 Diaphragm Slab and Reactor Pedestal

Table 3.8-2 lists the loading combinations used for the design of the diaphragm slab and reactor pedestal. Descriptions of the loads are as follows:

a. Dead Load, Live Load, and Seismic Loads

For a description of dead load, live load, and seismic loads, see Section 3.8.1.3.

b. Design Basis Accident Pressure Load

The diaphragm slab and the reactor pedestal are designed for the following pressures:

1. Maximum pressure: 55 psig in the drywell and the suppression chamber

2. Maximum differential pressure: 30 psig DOWNWARD (55 psig in the drywell and 25 psig in the suppression chamber). 20 psig UPWARD (55 PSIG IN THE SUPPRESSION CHAMBER AND 35 psig IN THE DRYWELL)

c. Thermal Loads

The temperatures above and below the diaphragm slab for the operating and the postulated design accident conditions are shown in Table 3.8-3. The portions of the reactor pedestal above and below the diaphragm slab are designed for the drywell and suppression chamber maximum temperatures listed in Table 3.8-3.

Thermal effects anticipated at the time of the structural acceptance test are insignificant, since the difference in temperatures inside and outside the containment during the test is small.

Section 6.2.1.1.3.1 (cont'd)

pool. As the vapor formed in the drywell is condensed in the suppression pool, the temperature of the suppression pool water peaks and the suppression chamber pressure stabilizes. The drywell pressure stabilizes at a slightly higher pressure, the difference being equal to the downcomer submergence. During the RPV depressurization phase, most of the noncondensable gases initially in the drywell are forced into the suppression chamber. However, following depressurization the noncondensables redistribute between the drywell and suppression chamber via the vacuum relief valve system. This redistribution takes place as steam in the drywell is condensed by the relatively cool ECCS water which is beginning to cascade from the break causing the drywell pressure to decrease.

Two cases leading to potentially rapid drywell depressurization were considered for wetwell-to-drywell vacuum breaker sizing:

a. The inadvertent actuation of one drywell spray train (10,000 gpm @ 90°F, assumed)

b. Maximum ECCS spillage (7750 lbm/sec @ 140°F exit temperature, assumed) during the depressurization phase of the large recirculation outlet line break LOCA

Each case was considered to determine the number of vacuum breaker valve assemblies required to ensure that the maximum differential pressure across the diaphragm slab in the upward direction does not exceed allowables. For the analyses, a conservatively low 3 psid across the diaphragm slab was used, well below the present design allowable of ~~[illegible]~~ 20 psid upward.

In the analyses done for both cases, a. and b., it was conservatively assumed that all noncondensables have been removed to the wetwell vapor region prior to drywell depressurization. In addition to this, for the Case a. accident a 100% spray efficiency, together with a drywell temperature of 273°F, combine with the assumptions regarding spray rate and inlet temperature noted above, to render this analysis conservative. This results in a net drywell energy removal rate of approximately 321,000 Btu/sec.

The analysis for Case b. assumes a drywell saturation temperature of 262°F, an ECCS drop fall height of 42 feet, an average drop diameter of 1 inch (for calculating condensation heat transfer to the falling ECCS spillage), and an average heat transfer coefficient of 2300 Btu/hr-ft^2-°F (for calculating heat transfer from the drywell vapor region to the pool of ECCS spillage collected on the drywell floor). These considerations, combined with the assumptions regarding noncondensables and ECCS spillage rate and temperature, yield a net drywell energy removal rate of approximately 318,000 Btu/sec for an ECCS spillage spray effectiveness of 34%.

TABLE 6.2-1 (Page 1 of 2)

CONTAINMENT DESIGN PARAMETERS

	DRYWELL	SUPPRESSION CHAMBER
DRYWELL AND SUPPRESSION CHAMBER		
Internal design pressure, psig	55	55
External to internal design differential pressure, psid	5	5
Drywell deck design differential pressure, psid	30 DOWNWARD	20 UPWARD
Design temperature, °F	340	220
Drywell net free volume, ft³	248,950(1)	
Design leak rate, % by weight/day	0.5	0.5
Maximum allowable leak rate, % by weight/day	0.5	0.5
Suppression chamber free volume, ft³		
Low level		161,350
High level		149,425
Suppression pool water volume, ft³		
Low level		115,903(2)
High level		127,756(2)
Suppression pool surface area, ft²		4974(2) (outside pedestal)
Suppression pool depth, ft		
Low level		22'
High level		24' 3"
VENT SYSTEM		
Number of downcomers		87

TABLE 6.2-5

SUMMARY OF SHORT-TERM
CONTAINMENT RESPONSES TO RECIRCULATION LINE AND
MAIN STEAM LINE BREAKS

	RECIRCULATION LINE BREAK	MAIN STEAM LINE BREAK
Peak drywell pressure, psig	44.02	36.20
Peak drywell deck DOWNWARD differential pressure, psid	25.995	19.5
Time of peak pressures, sec	13.66	20.12
Peak drywell temperature, °F	290.9	330
Peak suppression chamber pressure, psig	30.57	30.55
Time of peak suppression chamber pressure, sec	34.75	50
Peak suppression pool temperature during blowdown, °F	135.7	136
Calculated drywell pressure margin, %	20	34
Calculated suppression chamber pressure margin, %	44	44
Calculated deck differential pressure margin, %	13	35
Energy released to containment at time of peak pressure, 10^6 Btu	262.23	-
Energy absorbed by passive heat sinks at time of peak pressure, 10^6 Btu	0	0

Issue 4

The applicant has not provided the analyses of the suppression chamber pressure and temperature transient resulting from postulated steam bypass of the suppression pool following a LOCA necessary to establish the environmental qualification conditions for the suppression chamber. (Open Item)

Response

Section 3.11 and Figures 3.11-3 and 3.11-4 have been changed to establish the conditions which are used for the environmental qualification of equipment in the suppression chamber and that bound the conditions resulting from postulated steam bypass of the suppression pool following a LOCA.

The revised section and figures are attached.

CHAPTER 3

DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

TABLES (Cont'd)

Rev 20 5/83

FIGURES (Cont'd)

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

Engineered safety feature (ESF) systems, including the reactor protection system (RPS), are safety-related equipment installed in accordance with mechanical and electrical separation requirements and designed and qualified, with appropriate margins, to function properly in the following service environments:

a. For all normal and upset environmental design conditions, including the maximum and minimum limits for temperature, pressure, relative humidity, and radiation (gamma and neutron), the equipment is required to perform its normal operational function and/or pass its periodic tests. Otherwise, the equipment must remain in a safe mode available for operation (excluding maintenance activities, if the equipment is part of a mutually redundant system or standby equipment). The normal and upset environmental requirements are specified in Tables 3.11-1 and 3.11-5.

b. In addition to the normal and upset operational environmental design bases stated above, the safety-related ESF equipment is designed to perform its safety function during exposure to the post-accident environment present in its operational area and/or remain in a safe mode after its safety function is performed.

Environmental design criteria for the design of mechanical and electrical components of the ESF system and RPS conform to 10 CFR Part 50, Appendix A, General Design Criteria 1, "Quality Standards and Records"; General Design Criteria 2, "Design Bases for Portection Against Natural Phenomena"; 4, "Environmental and Missile Design Bases"; 23, "Protection System Failure Modes"; 50, "Containment Design Basis"; and 10 CFR Part 50, Appendix B, Section XI. An evaluation of NUREG 0588, Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment, is currently in progress and will be discussed in a separate Environmental Qualification Report.

3.11.1 EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS

3.11.1.1 Nuclear Steam Supply System (NSSS) Engineered Safety Features and Reactor Protection System Equipment

An ESF is a safety-related system that provides a safety function to prevent, limit, or mitigate the consequences of a design basis accident (DBA) that may cause major fuel damage. An ESF includes the primary auxiliary systems of the safety system. The identification, location, and accident environmental design bases

Section 3.11.1.2 (cont'd)

2. Total integrated radiation doses (TID) for 40 years are calculated for a 100% load factor and rated power at various locations during normal operation, as shown in Table 3.11-5.

b. In addition to the normal and abnormal plant operation environmental requirements listed in a above, ESF components required to mitigate the consequences of a DBA and effect a safe shutdown are designed to remain functional during exposure to the applicable accident environmental conditions. Applicable accident environmental conditions are those anticipated to follow a DBA that the component is intended to mitigate and are listed below.

1. Components Inside Containment

Specific values for temperature, pressure, relative humidity, and TID inside containment following a DBA are listed in Table 3.11-5. The TID inside containment is calculated by assuming that 100% of the core noble gas inventory, 50% of the core halogen inventory, and 1% of the core solid fission product inventory are released. Insert A The duration of the DBA is assumed to be 180 days. ~~The service conditions were established in accordance with NUREG-0588~~.

2. Components Outside Containment

Specific values for temperature, pressure, relative humidity, and TID outside containment following a DBA are given in Table 3.11-5. The TID outside containment is calculated by assuming that 50% of the core halogen inventory and 1% of the core solid fission product inventory are in the emergency core cooling system (ECCS) water after a DBA. Insert B The duration of the DBA is assumed to be 180 days. ~~The service conditions were established in accordance with NUREG-0588~~.

3.11.2 QUALIFICATION TESTS AND ANALYSIS

3.11.2.1 NSSS Class IE Electrical and Mechanical Equipment Qualification

All components of the Class IE equipment are qualified, either by test or analysis, consistent with Institute of Electrical and Electronics Engineers (IEEE) 323-1971. Those components used in several systems, which can be located in different plant areas, are tested or analyzed for the worst environmental conditions in which they are required to function. Consideration of their

Insert A

. These source terms are consistent with those specified in NUREG-0588 and NUREG-0737.

The primary containment airborne dose calculations assumed that 50% of the 50% (i.e. 25%) halogen release from the core plates out instantaneously, as assumed implicitly in Regulatory Guide 1.3, Rev. 2. The airborne doses were calculated assuming source terms diluted by the primary containment (drywell and wetwell) free volume. These assumptions are consistent with those specified in NUREG-0737.

The beta doses and dose rates were calculated assuming an infinite cloud geometry. ~~In general, the beta doses in Table 1 can be reduced by performing a detailed study which accounts for the actual cloud geometry and thickness of the component~~. The LGS post-accident airborne radiation doses were calcualted in accordance with NUREG-0737 and not specifically based on NUREG-0588.

Insert B

Post-LOCA radiation doses for reactor enclosure areas containing radioactive pipes are conservatively based on the sum of the worst case piping contact dose and the secondary containment cloud dose. Liquid-carrying ECCS lines and RHR shutdown cooling lines were assumed to contain 50% of the core halogen inventory and 1% of the core particulate inventory, and the suppression pool water volume plus the reactor water volume was used as the dilution volume for these lines. The HPCI and RCIC steam lines were assumed to contain 100% of the core noble gas inventory and 25% of the core halogen inventory and the total reactor coolant system steam volume was used as the dilution volume for these lines. The recombiner system piping was assumed to contain 100% of the ~~noble~~ core noble gas inventory and 25% of the core halogen inventory and the primary containment (drywell plus wetwell) free volume was used as the dilution volume for this piping. The secondary containment atmosphere radiation doses assumed that 100% of the core noble gas inventory and 25% of the core

Insert B (cont'd)

halogen inventory are available for leakage into the secondary containment and the airborne activity is confined to spaces below the refueling floor. The liquid source terms used to calculate the radiation doses are in accordance with NUREG-0588 and NUREG-0737.

malfunction limits for critical parameters for different applications, where possible, is called for in the test procedures.

Systems containing Class IE components are analyzed to ensure compliance with IEEE 279-1971, paragraph 4.3, relating to the single failure criteria of IEEE 379-1972, and paragraph 4.4, relating to completion of protective actions.

3.11.2.1.1 NSSS Safety-Related Equipment Qualification

All components of safety-related equipment are tested and/or analyzed to meet the requirements of 10 CFR Part 50, Criteria 1, 4, 23, and 50. Satisfaction of Criterion 1 is achieved by reviews to assure that tests or analyses conform to the design, procurement, fabrication, and environmental qualification documentation. The environmental requirements of Criterion 4 are addressed in this section, while considerations relating to missiles are addressed in Section 3.5. The normal upset and abnormal postulated accident environments are shown in Tables 3.11-1 through 3.11-4 for Class IE electrical component qualification to meet the requirements of Criterion 23 for protection system failure modes, which are addressed in Chapter 7. The LOCA containment pressures and temperatures used for component tests and analyses to satisfy the requirements of criterion 50 and assure containment integrity are shown in Table 3.11-2.

Section 8.1.6.1 discusses Regulatory Guide 1.30, relating to the installation and related quality assurance of controls and instrumentation; Regulatory Guide 1.40, relating to continuous duty safety-related motors inside the primary containment; Regulatory Guide 1.63, relating to electrical penetrations; Regulatory Guide 1.131, relating to qualification of electrical cables, field splices, and connections.

The NSSS-supplied cable does not experience severe environmental conditions, since it experiences the control room environment. This cable is qualified as part of the power generation control center (PGCC) floor section module. The qualification information for the floor section module is contained in the PGCC NEDO-10466.

Compliance with IEEE-323(1974) and Regulatory Guide 1.89 were not design and qualification requirements for this plant. Class IE equipment supplied by General Electric has been tested in order to comply with IEEE-323(1971) criteria. A program to evaluate ~~conformance to the~~ ~~requirements~~ guidelines of NUREG 0588, Interim Staff Position on Environmental Qualification of Safety Related Electrical Equipment, is ~~under development~~ in progress. The results of that program will be ~~incorporated into the FSAR.~~ discussed in a separate Environmental Qualification Report.

Section 3.11.5.2.2 (cont'd)

Normal and maximum radiation exposures based on the above assumptions are presented in Table 3.11-5.

Organic materials that exist within the containment are identified in Section 6.1.2. ~~The maximum radiation exposures identified in Table 3.11-5 are based on gamma radiation exposure only. Beta radiation is effectively attenuated by small amounts of shielding, such as conduits for cable and casings for equipment.~~

TABLE 3.11-5

CALCULATED NORMAL AND MAXIMUM PLANT ENVIRONMENTAL CONDITIONS

Unit 1 Room No.	Unit 2 Room No.	Area	NORMAL OPERATING CONDITIONS: Pressure	Min/Max Temp (°F)	Relative Humidity Avg/Max %	Dose[1] Rate (rads/hr)	Integrated[1] Dose (rads)	MAXIMUM CONDITIONS: Pressure[3] (psig)	Temp[4] (°F)	Relative[5] Humidity %	LOCA Dose Rate (See Notes)	LOCA[6] Dose (rads)	Total[1] Dose (rads)
		Reactor Enclosure											
102, 103 203, 204	173, 174, 200, 201	RHR pump compartments	-1/4-inch w.g.	65/115	50/90	1.45	5.09E+5	Atmos./1.5[i]	115/275[i]	100[i]	(13) (12)	gamma 4.78E+6 beta 1.38E+6	gamma 5.29E+6 beta 1.38E+6
108	179	RCIC pump compartment	-1/4-inch w.g.	65/115	50/90	1.30	4.91E+5	Atmos./2.7[a]	115/222[a]	100[a]	(20) (12)	gamma 1.95E+6 beta 1.38E+6	gamma 2.44E+6 beta 1.38E+6
109	180	HPCI pump compartment	-1/4-inch w.g.	65/115	50/90	2.41	8.46E+5	Atmos./2.7[b]	115/297[b]	100[b]	(18) (12)	gamma 3.95E+6 beta 1.38E+6	gamma 4.80E+6 beta 1.38E+6
110, 113, 114, 117	181, 184, 185, 188	CS pump compartments	-1/4-inch w.g.	65/115	50/90	2.5E-3	8.70E+2	Atmos.	115	90	(18) (12)	gamma 3.95E+6 beta 1.38E+6	gamma 3.95E+6 beta 1.38E+6
111	182, 189	Elevation 177 access area	-1/4-inch w.g.	65/104	50/90	2.5E-3	8.70E+2	Atmos.	120	90	(19) (12)	gamma 3.70E+6 beta 1.38E+6	gamma 3.70E+6 beta 1.38E+6
118		Elevation 177 access area	-1/4-inch w.g.	65/104	50/90	2.5E-3	8.70E+2	Atmos.	120	90	(12)	gamma 6.63E+4 beta 1.38E+6	gamma 6.71E+4 beta 1.38E+6
200, 207, 210	279, 284, 287	Elevation 201 access area	-1/4-inch w.g.	65/104	50/90	2.5E-3	8.70E+2	Atmos.	120	90	(19) (12)	gamma 3.70E+6 beta 1.38E+6	gamma 3.70E+6 beta 1.38E+6
309		Containment isolation valve compartment, el. 261 (Unit 1)	-1/4-inch w.g.	65/120	50/90	2.0	7.60E+5	Atmos.	120	90	(19) (12)	gamma 3.70E+6 beta 1.38E+6	gamma 4.46E+6 beta 1.38E+6
	286	Containment isolation valve compartment, el. 201 (Unit 2)	-1/4-inch w.g.	65/120	50/90	2.0	7.00E+5	Atmos.	120	90	(12)	gamma 6.63E+4 beta 1.38E+6	gamma 7.66E+5 beta 1.38E+6
209	285	RCIC piping area	-1/4-inch w.g.	65/115	50/90	1.30	4.84E+5	Atmos./5.3[a]	115/296[a]	100[a]	(22) (12)	gamma 1.87E+5 beta 1.38E+6	gamma 6.71E+5 beta 1.38E+6
208	283	HPCI piping area	-1/4-inch w.g.	65/115	50/90	2.41	8.46E+5	Atmos./6.1[b]	115/298[b]	100[b]	(21) (12)	gamma 8.42E+5 beta 1.38E+6	gamma 1.69E+6 beta 1.38E+6
304	370	Elevation 217 access area	-1/4-inch w.g.	65/104	50/90	2.5E-3	8.70E+2	Atmos.	120	90	(15) (12)	gamma 6.15E+6 beta 1.38E+6	gamma 6.15E+6 beta 1.38E+6
306, 307, 309	374, 375, 376	Containment isolation valve Compartment, el. 217	-1/4-inch w.g.	65/120	50/90	2.41	8.46E+5	Atmos./1.4[c]	120/264[c]	100[c]	(14) (12)	gamma 4.55E+6 beta 1.38E+6	gamma 5.40E+6 beta 1.38E+6

Rev. 20 5/83

T1001550-01V

TABLE 3.11-5 (Cont'd)

			NORMAL OPERATING CONDITIONS					MAXIMUM CONDITIONS					
Area			Pressure	Min/Max Temp (°F)	Relative Humidity Avg/Max %	Dose[1] Rate (rads/hr)	Integrated[1] Dose (rads)	Pressure[3] (psig)	Temp[4] (°F)	Relative[5] Humidity %	LOCA Dose Rate (See Notes)	LOCA[6] Dose (rads)	Total[1] Dose (rads)
Unit 1 Room No.	Unit 2 Room No.	Reactor Enclosure											
402	475	CRD hydraulic area	-1/4-inch w.g.	65/104	50/90	0.1	3.51E+4	Atmos.	120	90	(15) (12)	gamma 4.15E+6 beta 1.38E+6	gamma 4.19E+6 beta 1.38E+6
406		Neutron monitoring area (Unit 1)	-1/4-inch w.g.	65/104	50/90	3.72E+4	6.91E+7	Atmos.	120	90	(19) (12)	gamma 3.70E+6 beta 1.38E+6	gamma 7.28E+7 beta 1.38E+6
	479	Neutron monitoring area (Unit 2)	-1/4-inch w.g.	65/104	50/90	3.72E+4	6.91E+7	Atmos.	120	90	(24) (12)	gamma 4.08E+5 beta 1.38E+6	gamma 6.95E+7 beta 1.38E+6
407, 510	480, 587	Main steam tunnel	-1/4-inch w.g.	65/120	50/90	3.84	1.35E+6	Atmos./9.9[h]	120/311[h]	100[h]	(27) (12)	gamma 5.16E+6 beta 1.38E+6	gamma 6.51E+6 beta 1.38E+6
501	575	Drywell spray isolation valve compartment	-1/4-inch w.g.	65/115	50/90	0.225	7.89E+4	Atmos.	120	90	(16) (12)	gamma 3.95E+6 beta 1.38E+6	gamma 4.03E+6 beta 1.38E+6
502	576	Backwash receiving tank compartment	-1/4-inch w.g.	65/110	50/90	577	2.03E+8	Atmos.	120	90	(12)	gamma 6.63E+4 beta 1.38E+6	gamma 2.03E+8 beta 1.38E+6
503	577	RWCU system regenerative heat exchanger compartment	-1/4-inch w.g.	65/110	50/90	3.37	1.18E+6	Atmos./2.6[g]	120/216[f]	100[f]	(13)	gamma 6.63E+4 beta 1.38E+6	gamma 1.25E+6 beta 1.38E+6
504, 505	578, 579	RWCU system nonregenerative heat exchanger compartments	-1/4-inch w.g.	65/110	50/90	1.15	4.03E+5	Atmos./2.7[g]	120/217[d]	100[d]	(13)	gamma 6.63E+4 beta 1.38E+6	gamma 4.68E+5 beta 1.38E+6
506		Standby Liquid Control System Area	-1/4-inch w.g.	65/104	50/90	2.5E-3	8.78E+2	Atmos.	120	90	(16) (12)	gamma 3.95E+6 beta 1.38E+6	gamma 3.95E+6 beta 1.38E+6
	574	Standby Liquid Control System Area	-1/4-inch w.g.	65/104	50/90	2.5E-3	8.78E+2	Atmos.	120	90	(19) (12)	gamma 3.70E+6 beta 1.38E+6	gamma 3.70E+6 beta 1.38E+6
506A		Containment Hydrogen Recombiner Area (Area 11, Northwest side)	-1/4-inch w.g.	65/104	50/90	2.5E-3	8.78E+2	Atmos.	120	90	(19) (12)	gamma 3.70E+6 beta 1.38E+6	gamma 3.70E+6 beta 1.38E+6
	580A	Containment Hydrogen Recombiner Area (Area 14, Northeast side)	-1/4-inch w.g.	65/104	50/90	2.5E-3	8.78E+2	Atmos.	120	90	(16) (12)	gamma 3.95E+6 beta 1.38E+6	gamma 3.95E+6 beta 1.38E+6
506B		Corridor - Access Area (Area 15, Southwest side)	-1/4-inch w.g.	65/104	50/90	2.5E-3	8.78E+2	Atmos.	120	90	(12)	gamma 6.63E+4 beta 1.38E+6	gamma 6.71E+4 beta 1.38E+6
	580B	Corridor - Access Area (Area 18, Southeast side)	-1/4-inch w.g.	65/104	50/90	2.5E-3	8.78E+2	Atmos.	120	90	(16) (12)	gamma 3.95E+6 beta 1.38E+6	gamma 3.95E+6 beta 1.38E+6

T1001550-87Y

Rev. 20 5/83 ✗

Area			NORMAL OPERATING CONDITIONS					MAXIMUM CONDITIONS					
Unit 1 Room No.	Unit 2 Room No.	Reactor Enclosure	Pressure	Min/Max Temp (°F)	Relative Humidity Avg/Max %	Dose[1] Rate (rads/hr)	Integrated[1] Dose (rads)	Pressure[3] (psig)	Temp[4] (°F)	Relative[5] Humidity %	LOCA Dose Rate (See Notes)	LOCA[6] Dose (rads)	Total[1] Dose (rads)
506C		Containment Hydrogen Recombiner Area (Area 16, Southeast side)	-1/4-inch w.g.	65/104	50/90	2.5E-3	8.78E+2	Atmos.	120	90	(16) (12)	gamma 3.95E+6 beta 1.38E+6	gamma 3.95E+6 beta 1.38E+6
	508C	Containment Hydrogen Recombiner Area (Area 17, Southwest side)	-1/4-inch w.g.	65/104	50/90	2.5E-3	8.78E+2	Atmos.	120	90	(19) (12)	gamma 3.70E+6 beta 1.38E+6	gamma 3.70E+6 beta 1.38E+6
506D		Fuel Pool Service Water Booster Pump Area (Area 12, Northeast Side)	-1/4-inch w.g.	65/104	50/90	2.5E-3	8.78E+2	Atmos.	120	90	(16) (12)	gamma 3.95E+6 beta 1.50E+6	gamma 3.95E+6 beta 1.38E+6
	508D	Fuel Pool Service Water Booster Pump Area (Area 13, Northwest side)	-1/4-inch w. g.	65/104	50/90	2.5E-3	8.78E+2	Atmos.	120	90	(19) (12)	gamma 3.70E+6 beta 1.38E+6	gamma 3.70E+6 beta 1.38E+5
507, 508, 509	581, 582, 583	RWCU system pump compartments (Northeast side)	-1/4-inch w.g.	65/104	50/90	3.29	1.16E+6	Atmos./2.7[g]	120/217[e]	100[e]	(12)	gamma 6.63E+4 beta 1.38E+6	gamma 1.23E+6 beta 1.38E+6
510, 522	584, 597	RWCU system isolation valve compartment	-1/4-inch w.g.	65/115	50/90	55.5	1.26E+6	Atmos./2.7[g]	120/210[f]	100[g]	(15) (12)	gamma 4.15E+6 beta 1.38E+6	gamma 5.41E+6 beta 1.38E+6
511	585	Fuel Pool Cooling Water Pump and Heat Exchanger Area	-1/4-inch w.g.	65/104	50/90	2.5E-3	8.78E+2	Atmos.	120	90	(16) (12)	gamma 3.95E+6 beta 1.38E+6	gamma 3.95E+6 beta 1.38E+6
519		Access Area (El. 300')	-1/4-inch w.g.	65/104	50/90	2.5E-3	8.78E+2	Atmos.	120	90	(19) (12)	gamma 3.70E+6 beta 1.38E+6	gamma 3.70E+6 beta 1.38E+6
	594	Access Area (El. 300')	-1/4-inch w.g.	65/104	50/90	2.5E-3	8.78E+2	Atmos.	120	90	(16) (12)	gamma 3.95E+6 beta 1.38E+6	gamma 3.95E+6 beta 1.38E+6
523	593	Core spray isolation valve compartment	-1/4-inch w.g.	65/115	50/90	0.1	3.51E+4	Atmos.	120	90	(16) (12)	gamma 3.95E+6 beta 1.38E+6	gamma 3.99E+6 beta 1.38E+6
599	589	LPCI isolation valve compartment	-1/4-inch w.g.	65/115	50/90	0.1	3.51E+4	Atmos.	120	90	(15) (12)	gamma 4.15E+6 beta 1.38E+6	gamma 4.19E+6 beta 1.38E+6
602, 605	638, 641	North and South corridors, el. 313	-1/4-inch w.g.	65/104	50/90	2.5E-3	8.78E+2	Atmos.	120	90	(12)	gamma 6.63E+4 beta 1.38E+6	gamma 6.71E+4 beta 1.38E+6
612, 618	651, 653	RERS filter compartment	Atmos.	65/104	50/90	2.5E-3	8.78E+2	Atmos.	120	90	(23) (12)	gamma 1.04E+7 beta 1.38E+6	gamma 1.04E+7 beta 1.38E+6

Rev. 20 5/83 ✗

TABLE 3.11-5 (Cont'd)

Room No.	Area	NORMAL OPERATING CONDITIONS: Pressure	Min/Max Temp (°F)	Relative(5) Humidity Avg/Max %	Dose(1) Rate (rads/hr)	Integrated(1) Dose (rads)	MAXIMUM CONDITIONS: Pressure(3) (psig)	Temp(4) (°F)	Relative(5) Humidity Avg/Max (%)	LOCA Dose Rate (See Notes)	LOCA(6) Dose (rads)	Total(1) Dose (rads)
	Primary Containment											
	Zone I, above core	0.75 psig	65/150	20/100	gamma-24 neutrons-35	gamma-[illegible].4E+6 neutrons-1.2E+7	44.0	340	100	(7)	gamma-4.59E+7 neutrons-0 beta-1.10E+9	gamma-5.[illegible]3E+7 neutrons-1.2E+7 beta-1.10E+9
	Zone II, core region	0.75 psig	65/150	20/100	gamma-50 neutrons-30	gamma-1.8E+7 neutrons-1.1E+7	44.0	340	100	(7)	gamma-4.59E+7 neutrons-0 beta-1.10E+9	gamma-6.39E+7 neutrons-1.1E+7 beta-1.10E+9
	Zone III, under vessel	0.75 psig	65/150	20/100	gamma-12 neutrons-14	gamma-6.3E+6 neutrons-5.0E+6	44.0	340	100	(7)	gamma-4.59E+7 neutrons-0 beta-1.10E+9	gamma-5.02E+7 neutrons-5.0E+6 beta-1.10E+9
	Zone IV, near recirc. pumps	0.75 psig	65/150	20/100	gamma-50 neutrons-16	gamma-1.8E+7 neutrons-5.5E+6	44.0	340	100	(7)	gamma-4.59E+7 neutrons-0 beta-1.10E+9	gamma-6.4E+7 neutrons-5.5E+6 beta-1.10E+9
	Zone V, >15 ft from recirc. pumps	0.75 psig	65/150	20/100	gamma-25 neutrons-16	gamma-8.8E+6 neutrons-5.5E+6	44.0	340	100	(7)	gamma-4.59E+7 neutrons-0 beta-1.10E+9	gamma-5.5E+7 neutrons-5.5E+6 beta-1.10E+9
	Zone VI, suppression chamber	0.75 psig	65/150	20/100	gamma-11 neutrons-0.43	gamma-3.78E+6 neutrons-1.5E+5	44.0	340	100	(7)	gamma-4.59E+7 neutrons-0 beta-1.10E+9	gamma-4.96E+7 neutrons-1.5E+5 beta-1.10E+9
Room No.	Control Structure											
624	SGTS equip. compartment	Atmos.	65/104	50/90	2.35E-2	8.23E+3	Atmos.	120	50/90	(9)	3.22E+5	3.30E+5
533	Control room	+1/4-inch w.g.	76 ± 2	50/55	5.0E-4	1.76E+2	+1/4-inch w.g.	78	50/60	(10)	6.4	1.82E+2
502	Auxiliary equip. room	Atmos.	76 ± 2	50/55	5.0E-4	1.76E+2	Atmos.	78	50/60	(8)	1.3E+1	1.89E+2
449, 450	Cable spreading room	Atmos.	65/104	50/90	5.0E-4	1.76E+2	Atmos.	105	80/100	(8)	2.1E+1	1.97E+2

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Rev. 20 5/83

TABLE 3.11-5 (Cont'd)

		NORMAL OPERATING CONDITIONS					MAXIMUM CONDITIONS					
Room No.	Area	Pressure	Min/Max Temp (°F)	Relative Humidity Avg/Max %	Dose[1] Rate (rads/hr)	Integrated[1] Dose (rads)	Pressure[3] (psig)	Temp[4] (°F)	Relative[5] Humidity Avg/Max (%)	LOCA Dose Rate (See Notes)	LOCA[6] Dose (rads)	Total[1] Dose (rads)
	Control Structure											
619	HVAC equip. compartment, el. 304	Atmos.	65/104	50/90	2.5E-3	8.78E+2	Atmos.	104	50/90	(11)	1.26E+2	1.00E+3
625	HVAC equip. compartment, el. 332	Atmos.	65/104	50/90	2.5E-3	8.78E+2	Atmos.	104	50/90	(11)	1.26E+2	1.00E+3
323, 324 360, 361 425, 426 427, 436	Battery compartments	Atmos.	88/104[2]	50/90	5.0E-4	1.76E+2	Atmos.	104	50/90	(8)	2.1E+1	1.97E+2
428-435	Emergency switchgear room	Atmos.	65/104	50/96	5.0E-4	1.76E+2	Atmos.	104	50/90	(8)	2.1E+1	1.97E+2
452, 453	Inverter rooms	Atmos.	65/104	50/90	5.0E-4	1.76E+2	Atmos.	104	50/90	(8)	2.1E+1	1.37E+2
258, 263	Control structure chiller areas	Atmos.	65/104	50/90	2.5E-3	8.78E+2	Atmos.	111	80/90	(9)	2.1E+1	8.99E+2
713[25]	North Stack Radiation Monitoring Room	Atmos.	65/104	50/90	2.5E-3	8.78E+2	Atmos.	139[26]	50/90	-	4.1E+1	9.19E+2
	Diesel-Generator Enclosure	Atmos.	55/115	50/90	5.0E-4	1.76E+2	Atmos.	115	50/90	-	-	1.79E+2
	Spray Pond Pump Structure	Atmos.	60/115	50/90	5.0E-4	1.76E+2	Atmos.	115	50/90	-	-	1.79E+2

Notes:

(1) The normal total integrated doses are for 40 years. The present post-LOCA gamma doses are conservatively based on NUREG-0737 methodology. These conservative gamma doses would bound the total of the neutron doses and the post-LOCA gamma doses if the gamma doses were calculated by NUREG-0588 methodology. In view of this, it is not necessary to add the neutron doses to the present gamma doses for equipment qualification purposes.

(2) 88°F represents the battery compartment average temperature.

(3) First number corresponds to maximum LOCA/loss of ventilation pressure. Second number corresponds to maximum High Energy Line Break pressure, where applicable. Break identification is provided in note (4). The maximum LOCA pressure for each area should be assumed to last for 180 days for environmental qualification purposes, except for the primary containment. A bounding primary containment pressure profile is provided in Figure 3.11-2. The maximum High Energy Line Break (HELB) pressure for each area should be assumed to last until the isolation valve closes, which in all cases is significantly less than one minute. The maximum pressures listed for the control structure are maximum LOCA pressures. Maximum pressures resulting from other accidents will be evaluated later.

Rev. 20 5/83

Notes (Cont'd)

(4) First number corresponds to maximum LOCA/loss of ventilation temperature. Second number corresponds to maximum High Energy Line Break temperature, where applicable. Break identification is indicated as follows:

a. RCIC steam line break in RCIC pump room
b. HPCI steam line break in HPCI pump room
c. HPCI steam line break in isolation valve compartment
d. RWCU line break in RWCU nonregenerative heat exchanger compartment
e. RWCU line break in RWCU pump room
f. RWCU line break in RWCU regenerative heat exchanger compartment
g. RWCU line break in RWCU isolation valve compartment
h. Main steam line break in main steam tunnel
i. RHR steam line break in RHR compartment

LOCA temperature profiles as a function of time are provided in Figure 3.11-3 through Figure 3.11-6. For areas without LOCA temperature profiles, assume the maximum temperature lasts 180 days. The isolation valve compartment HELB temperature profile as a function of time is provided in Figure 3.11-7. Areas which have maximum HELB temperatures listed, but do not have HELB temperature profiles, only contain components which will have completed their safety function once the blowdown has ceased, which in all cases is significantly less than one minute. The maximum temperatures listed for the control structure are maximum LOCA temperatures. Maximum temperatures resulting from other accidents will be evaluated later.

(5) Those rooms which experience High Energy Line Break, identified by letters per note 4, will be subjected to 100% relative humidity for 72 hours after the start of the High Energy Line Break at which time the relative humidity will fall below 90%. For the normal primary containment relative humidity, the minimum relative humidity is provided instead of the average relative humidity.

(6) The LOCA total integrated doses are for 180 days. The beta doses and dose rates are conservatively based on infinite cloud geometry, and the doses for areas containing radioactive pipes are conservatively based on piping contact doses. Doses for specific components within these areas may be lower, depending on distance and spatial relationships.

(7) Primary Containment Cloud (See Table 3.11-5a).

(8) Control Structure Cloud, Adjacent Piping Shine (Dose rates not tabulated because of low Total Integrated Dose (TID)).

(9) SGTS Carbon Filter (See Table 3.11-5b).

(10) Adjacent Cloud Shine, Adjacent Piping Shine and SGTS Carbon Filter Shine (Dose rates not tabulated because of low TID).

(11) Control Structure Cloud and SGTS Carbon Filter Shine (Dose rates not tabulated because of low TID).

(12) Secondary Containment Cloud (see Table 3.11-5^{c})

(13) 30" Shutdown Cooling Piping (see Table 3.11-5^{j})

(14) 24" Shutdown Cooling Piping (see Table 3.11-5^{i})

(15) 18" Shutdown Cooling Piping (see Table 3.11-5^{h})

(16) 16" Shutdown Cooling Piping (see Table 3.11-5^{g})

(17) 6" Shutdown Cooling Piping (see Table 3.11-5^{e})

(18) 16" ECCS Piping (see Table 3.11-5^{g})

(19) 14" ECCS Piping (see Table 3.11-5^{f})

(20) 6" ECCS Piping (see Table 3.11-5^{e})

(21) 12" HPCI Steam Supply Piping (see Table 3.11-5^{m})

(22) 10" RCIC Steam Exhaust Piping (see Table 3.11-5^{l})

(23) RERS Carbon Filter (see Table 3.11-5^{k})

(24) 6" recombiner piping (see Table 3.11-5^{d})

(25) This room is located on the reactor enclosure roof.

(26) The North Stack Radiation Monitoring Room temperature drops to 130°F after 6 days

(27) 24" ECCS Piping (see Table 3.11-5i)

TABLE 3.11-5a

CALCULATED PRIMARY CONTAINMENT DOSE RATES

Gamma Dose Rates

TIME INT. HRS	DOSE RATE R/HR	INT. DOSE RADS	TOTAL RADS
.10E-01	1.07E+07	.00	.00
.50E+00	3.51E+06	3.16E+06	3.16E+06
.10E+01	2.60E+06	1.52E+06	4.68E+06
.20E+01	1.85E+06	2.20E+06	6.88E+06
.40E+01	1.23E+06	3.04E+06	9.92E+06
.80E+01	7.10E+05	3.79E+06	1.37E+07
.16E+02	3.66E+05	4.15E+06	1.79E+07
.24E+02	2.41E+05	2.39E+06	2.03E+07
.96E+02	7.17E+04	1.01E+07	3.03E+07
.24E+03	3.45E+04	7.32E+06	3.76E+07
.72E+03	4.37E+03	7.00E+06	4.46E+07
.22E+04	2.56E+01	1.22E+06	4.58E+07
.43E+04	6.56E+00	3.02E+04	4.59E+07

Beta Dose Rates

TIME INT. HRS	DOSE RATE R/HR	INT. DOSE RADS	TOTAL RADS
.10E-01	1.30E+08	.00	.00
.50E+00	4.48E+07	3.92E+07	3.92E+07
.10E+01	3.56E+07	2.00E+07	5.92E+07
.20E+01	2.68E+07	3.10E+07	9.02E+07
.40E+01	1.88E+07	4.51E+07	1.35E+08
.30E+01	1.26E+07	6.20E+07	1.97E+08
.16E+02	8.00E+06	8.10E+07	2.78E+08
.24E+02	5.92E+06	5.53E+07	3.34E+08
.96E+02	2.22E+06	2.72E+08	6.05E+08
.24E+03	1.01E+06	2.21E+08	8.26E+08
.72E+03	1.25E+05	2.03E+08	1.03E+09
.22E+04	3.00E+04	9.59E+07	1.13E+09
.43E+04	1.87E+04	5.16E+07	1.18E+09

TABLE 3.11-5b

SGTS CARBON FILTER
CALCULATED DOSE RATES

TIME INT. HRS	DOSE RATE R/HR	INT. DOSE RADS	TOTAL RADS
.10E-01	2.50E-03	.00	.00
.50E+00	1.00E+01	6.17E-01	6.17E-01
.10E+01	3.10E+01	9.47E+00	1.01E+01
.20E+01	7.47E+01	4.97E+01	5.98E+01
.40E+01	1.40E+02	2.08E+02	2.68E+02
.80E+01	2.10E+02	6.91E+02	9.58E+02
.16E+02	2.63E+02	1.88E+03	2.84E+03
.24E+02	2.81E+02	2.18E+03	5.02E+03
.96E+02	3.50E+02	2.26E+04	2.76E+04
.24E+03	4.51E+02	5.74E+04	8.50E+04
.72E+03	2.30E+02	1.58E+05	2.43E+05
.22E+04	3.41E+00	7.75E+04	3.21E+05
.43E+04	2.40E-03	1.01E+03	3.22E+05

TABLE 3.11-5c

CALCULATED SECONDARY CONTAINMENT DOSE RATES

Gamma Dose Rates

TIME INT. HRS	DOSE RATE R/HR	INT. DOSE RADS	TOTAL RADS
.10E-09	1.00E-10	.00	.00
.50E+00	2.49E+02	4.36E+00	4.36E+00
.10E+01	3.23E+02	1.42E+02	1.46E+02
.20E+01	3.93E+02	3.57E+02	5.03E+02
.40E+01	4.55E+02	8.47E+02	1.35E+03
.80E+01	4.46E+02	1.80E+03	3.15E+03
.24E+02	3.07E+02	5.96E+03	9.11E+03
.96E+02	1.87E+02	1.74E+04	2.65E+04
.24E+03	9.33E+01	1.94E+04	4.59E+04
.72E+03	6.40E+00	1.56E+04	6.15E+04
.43E+04	5.78E-02	4.85E+03	6.63E+04

Beta Dose Rates

TIME INT. HRS	DOSE RATE R/HR	INT. DOSE RADS	TOTAL RADS
.10E-09	1.00E-10	.00	.00
.50E+00	9.40E+02	1.57E+01	1.57E+01
.10E+01	1.39E+03	5.75E+02	5.91E+02
.20E+01	1.92E+03	1.64E+03	2.23E+03
.40E+01	2.50E+03	4.41E+03	6.64E+03
.80E+01	3.03E+03	~~4.46E+03~~ 1.10E+04	1.77E+04
.24E+02	3.50E+03	5.22E+04	6.99E+04
.96E+02	3.35E+03	2.47E+05	3.16E+05
.24E+03	1.75E+03	3.55E+05	6.71E+05
.72E+03	2.01E+02	3.44E+05	1.01E+06
.43E+04	4.23E+01	3.66E+05	1.38E+06

TABLE 3.11-5d

6 INCH RECOMBINER
CALCULATED PIPING DOSE RATES

TIME INT. HRS	DOSE RATE R/HR	INT. DOSE RADS	TOTAL RADS
.10E-09	7.53E+04	0	0
.50E+00	2.95E+04	2.44E+04	2.44E+04
.10E+01	2.25E+04	1.29E+04	3.73E+04
.20E+01	1.54E+04	1.87E+04	5.61E+04
.40E+01	9.75E+03	2.47E+04	8.08E+04
.80E+01	5.57E+03	2.99E+04	1.11E+05
.24E+02	1.82E+03	5.36E+04	1.64E+05
.96E+02	4.19E+02	6.86E+04	2.33E+05
.24E+03	2.08E+02	4.34E+04	2.76E+05
.72E+03	3.34E+01	4.59E+04	3.22E+05
.43E+04	8.80E-01	2.02E+04	3.42E+05

TABLE 3.11-5e

6 INCH ECCS
CALCULATED PIPE DOSE RATES(a)

TIME INT. HRS	DOSE RATE R/HR	INT. DOSE RADS	TOTAL RADS
.10E-09	5.79E+04	.00	.00
.50E+00	4.18E+04	2.47E+04	2.47E+04
.10E+01	3.57E+04	1.93E+04	4.40E+04
.20E+01	2.77E+04	3.15E+04	7.56E+04
.40E+01	1.97E+04	4.70E+04	1.23E+05
.80E+01	1.32E+04	6.49E+04	1.87E+05
.24E+02	5.91E+03	1.45E+05	3.32E+05
.96E+02	1.91E+03	2.55E+05	5.87E+05
.24E+03	1.12E+03	2.13E+05	8.00E+05
.72E+03	4.42E+02	3.50E+05	1.15E+06
.43E+04	7.05E+01	7.28E+05	1.88E+06

DELETE

(a) 6 inch shutdown cooling piping dose rates can be determined by multiplying the 6 inch ECCS piping dose rates by a factor of 8.425.

TABLE 3.11-5f

14 INCH ECCS
CALCULATED PIPING DOSE RATES

TIME INT. HRS	DOSE RATE R/HR	INT. DOSE RADS	TOTAL RADS
.10E-09	1.15E+05	.00	.00
.50E+00	8.30E+04	4.92E+04	4.92E+04
.10E+01	7.07E+04	3.84E+04	8.75E+04
.20E+01	5.49E+04	6.25E+04	1.50E+05
.40E+01	3.89E+04	9.29E+04	2.43E+05
.80E+01	2.58E+04	1.28E+05	3.71E+05
.24E+02	1.14E+04	2.83E+05	6.53E+05
.96E+02	3.62E+03	4.89E+05	1.14E+06
.24E+03	2.13E+03	4.05E+05	1.55E+06
.72E+03	8.54E+02	6.70E+05	2.22E+06
.43E+04	1.38E+02	1.41E+06	3.63E+06

TABLE 3.11-5g

16 INCH ECCS
CALCULATED PIPING DOSE RATES~~(a)~~

TIME INT. HRS	DOSE RATE R/HR	INT. DOSE RADS	TOTAL RADS
.10E-09	1.24E+05	.00	.00
.50E+00	8.93E+04	5.30E+04	5.30E+04
.10E+01	7.60E+04	4.13E+04	9.42E+04
.20E+01	5.90E+04	6.71E+04	1.61E+05
.40E+01	4.18E+04	9.98E+04	2.61E+05
.80E+01	2.77E+04	1.37E+05	3.98E+05
.24E+02	1.22E+04	3.03E+05	7.01E+05
.96E+02	3.85E+03	5.21E+05	1.22E+06
.24E+03	2.27E+03	4.30E+05	1.65E+06
.72E+03	9.12E+02	7.15E+05	2.37E+06
.43E+04	1.48E+02	1.51E+06	3.88E+06

DELETE

~~(a) 16 inch shutdown cooling piping dose rates can be determined by multiplying the 16 inch ECCS piping dose rates by a factor of 8.425.~~

TABLE 3.11-5h

18 INCH SHUTDOWN COOLING
CALCULATED PIPING DOSE RATES

TIME INT. HRS	DOSE RATE R/HR	INT. DOSE RADS	TOTAL RADS
.10E-09	1.32E+05	.00	.00
.50E+00	9.45E+04	5.60E+04	5.60E+04
.10E+01	8.04E+04	4.37E+04	9.97E+04
.20E+01	6.23E+04	7.10E+04	1.71E+05
.40E+01	4.42E+04	1.05E+05	2.77E+05
.80E+01	2.92E+04	1.45E+05	4.21E+05
.24E+02	1.28E+04	3.19E+05	7.39E+05
.96E+02	4.07E+03	5.47E+05	1.28E+06
.24E+03	2.37E+03	4.51E+05	1.73E+06
.72E+03	9.59E+02	7.50E+05	2.49E+06
.43E+04	1.55E+02	1.59E+06	4.08E+06

TABLE 3.11-51

24 INCH ~~SHUTDOWN COOLING~~ ECCS
CALCULATED PIPING DOSE RATES~~(1)~~

TIME INT. HRS	DOSE RATE R/HR	INT. DOSE RADS	TOTAL RADS
.10E-09	1.47E+05	.00	.00
.50E+00	1.05E+05	6.26E+04	6.26E+04
.10E+01	8.95E+04	4.85E+04	1.11E+05
.20E+01	6.93E+04	7.91E+04	1.90E+05
.40E+01	4.90E+04	1.17E+05	3.07E+05
.80E+01	3.24E+04	1.60E+05	4.68E+05
.24E+02	1.41E+04	3.53E+05	8.20E+05
.96E+02	4.40E+03	5.99E+05	1.42E+06
.24E+03	2.59E+03	4.93E+05	1.91E+06
.72E+03	1.05E+03	8.20E+05	2.73E+06
.43E+04	1.71E+02	1.74E+06	4.49E+06

~~(1) 24 inch ECCS piping dose rates can be determined by dividing the 24 inch shutdown cooling piping dose rates by a factor of 8.425.~~ DELETE

TABLE 3.11-5j

30 INCH SHUTDOWN COOLING CALCULATED PIPING DOSE RATES

TIME INT. HRS	DOSE RATE R/HR	INT. DOSE RADS	TOTAL RADS
.10E-09	1.57 ~~1.32~~E+05	.00	.00
.50E+00	1.12 ~~9.40~~E+05	6.65 ~~5.60~~E+04	6.65 ~~5.60~~E+04
.10E+01	9.48 ~~7.99~~E+04	5.15 ~~4.34~~E+04	1.18 ~~9.94~~E+05
.20E+01	7.34 ~~6.18~~E+04	8.37 ~~7.05~~E+04	2.02 ~~1.70~~E+05
.40E+01	5.19 ~~4.37~~E+04	1.23 ~~1.04~~E+05	3.25 ~~2.74~~E+05
.80E+01	3.42 ~~2.88~~E+04	1.70 ~~1.43~~E+05	4.95 ~~4.17~~E+05
.24E+02	1.48 ~~1.25~~E+04	3.70 ~~3.12~~E+05	8.66 ~~7.30~~E+05
.96E+02	4.59 ~~3.87~~E+03	6.29 ~~5.30~~E+05	1.50 ~~1.26~~E+06
.24E+03	2.72 ~~2.30~~E+03	5.15 ~~4.34~~E+05	2.01 ~~1.69~~E+06
.72E+03	1.11 ~~9.32~~E+03	8.59 ~~7.24~~E+05	2.87 ~~2.42~~E+06
.43E+04	1.80 ~~1.52~~E+02	1.84 ~~1.55~~E+06	4.71 ~~3.97~~E+06

TABLE 3.11-5k

CALCULATED RERS CARBON FILTER DOSE RATES

TIME INT. HRS	DOSE RATE R/HR	INT. DOSE RADS	TOTAL RADS
.10E-01	2.50E+03	.00	.00
.50E+00	7.99E+02	3.09E+01	3.09E+01
.10E+01	2.08E+03	6.70E+02	7.01E+02
.20E+01	4.20E+03	3.02E+03	3.72E+03
.40E+01	6.64E+03	1.07E+04	1.44E+04
.80E+01	8.90E+03	3.09E+04	4.52E+04
.24E+02	1.11E+04	1.60E+05	2.05E+05
.96E+02	1.18E+04	8.25E+05	1.03E+06
.24E+03	1.45E+04	1.89E+06	2.92E+06
.72E+03	7.40E+03	5.07E+06	7.99E+06
.43E+04	7.69E-02	2.32E+06	1.03E+07

TABLE 3.11-51

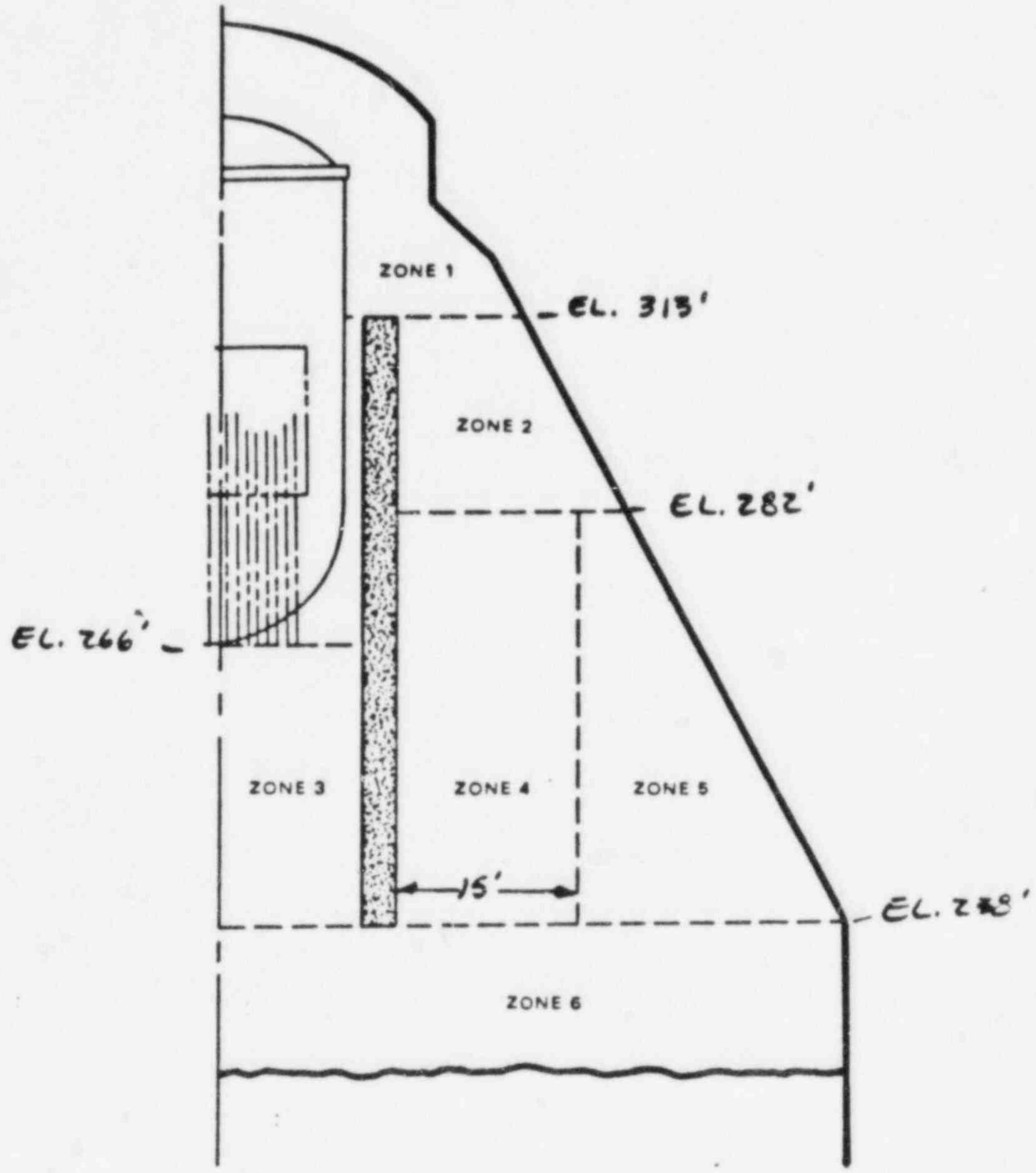
10 INCH RCIC STEAM EXHAUST CALCULATED PIPING DOSE RATES

TIME INT. HRS	DOSE RATE R/HR	INT. DOSE RADS	TOTAL RADS
.10E-09	2.06E+05	.00	.00
.50E+00	6.02E+04	5.93E+04	5.93E+04
.10E+01	3.41E+04	2.30E+04	8.23E+04
.20E+01	1.31E+04	2.19E+04	1.04E+05
.40E+01	2.56E+03	1.29E+04	1.17E+05
.80E+01	1.40E+02	3.33E+03	1.20E+05
.24E+02	3.92E-03	2.14E+02	1.21E+05

TABLE 3.11-5m

CALCULATED 12 INCH HPCI STEAM SUPPLY PIPING DOSE RATES

TIME INT. HRS	DOSE RATE R/HR	INT. DOSE RADS	TOTAL RADS
.10E-09	4.62E+06	.00	.00
.50E+00	2.25E+05	7.27E+05	7.27E+05
.10E+01	2.14E+04	4.33E+04	7.71E+05
.20E+01	2.28E+02	4.67E+03	7.76E+05
.40E+01	3.50E-02	5.20E+01	7.76E+05

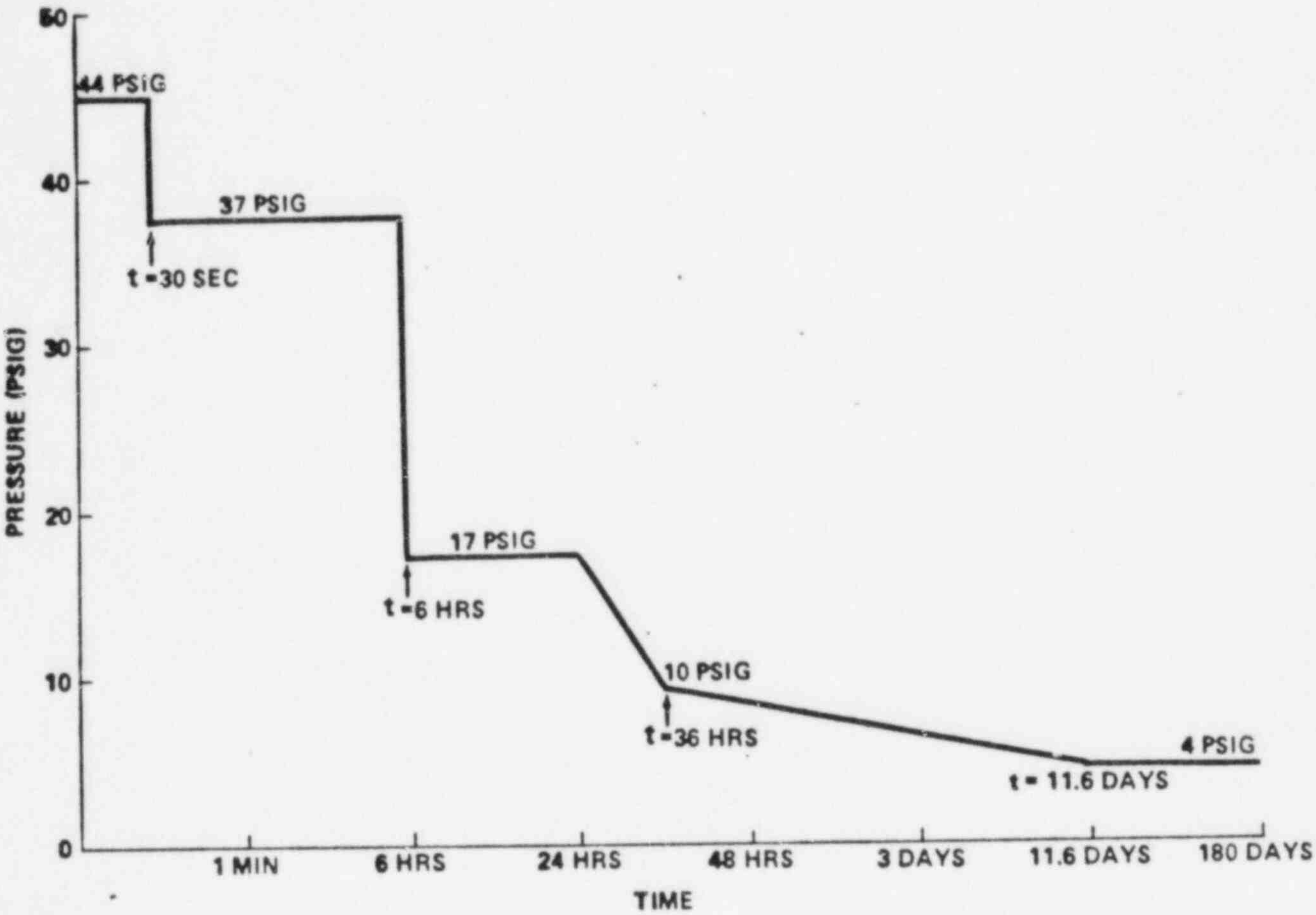


LIMERICK GENERATING STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

PRIMARY CONTAINMENT ZONES

FIGURE 3.11-1

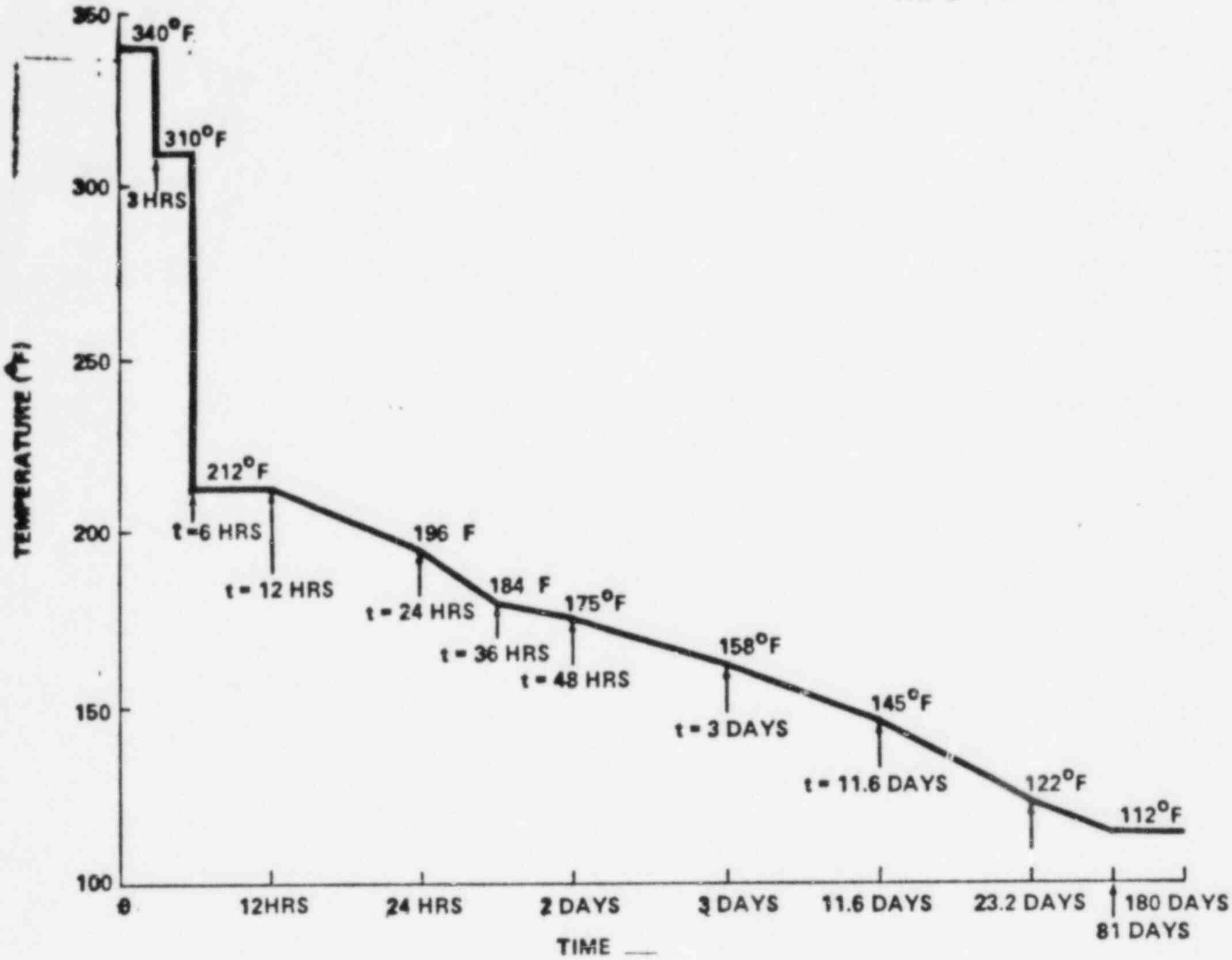
Rev 20 5/83



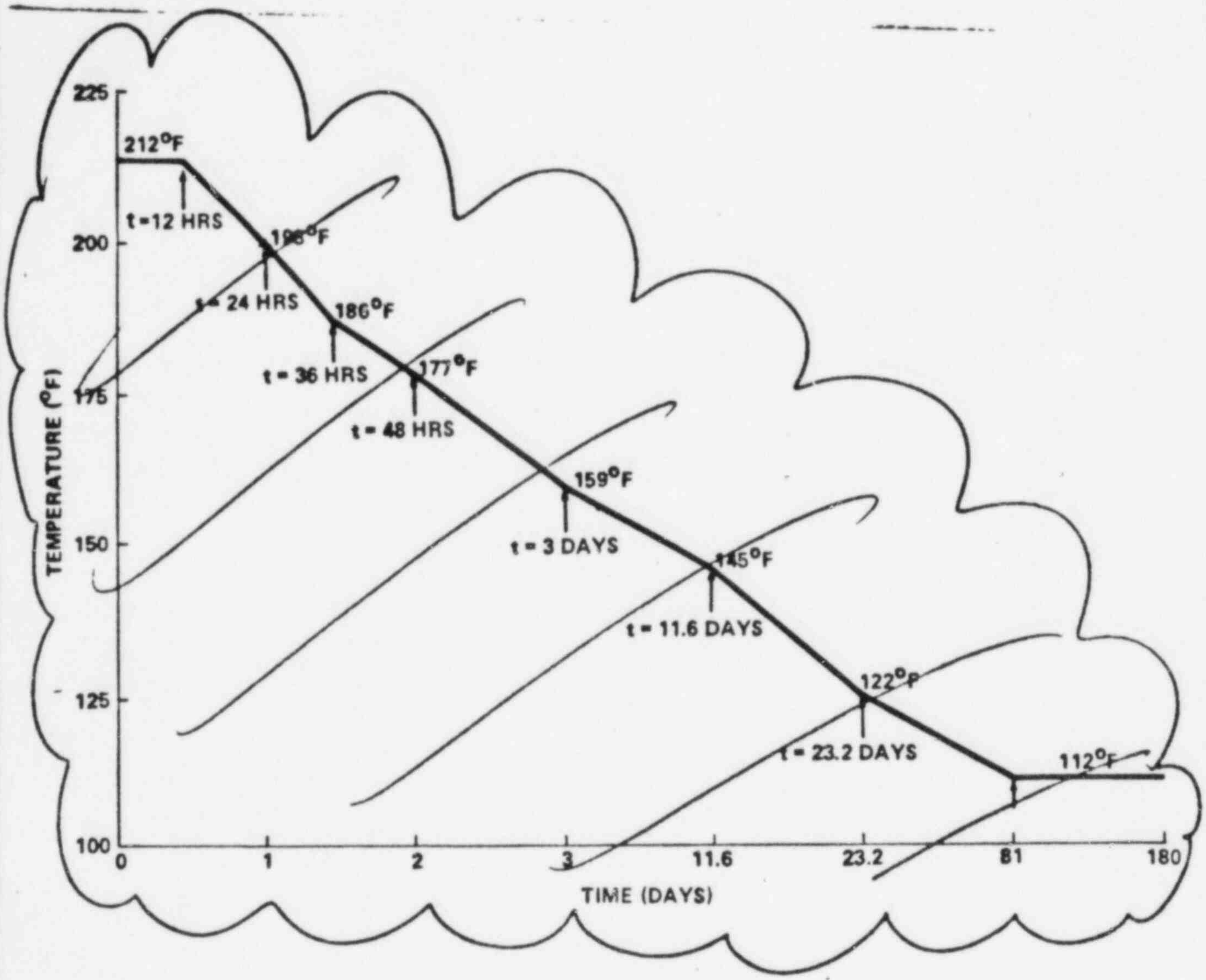
LIMERICK GENERATING STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

CALCULATED POST-LOCA
BOUNDING PRIMARY CONTAINMENT
PRESSURE PROFILE

FIGURE 3.11-2 Rev. 20 5/83



LIMERICK GENERATING STATION UNITS 1 AND 2 FINAL SAFETY ANALYSIS REPORT
CALCULATED POST-LOCA BOUNDING ~~DRYWELL~~ PRIMARY CONTAINMENT TEMPERATURE PROFILE
FIGURE 3.11-3 Rev 20 5/83



LIMERICK GENERATING STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

DELETED
~~CALCULATED POST-LOCA BOUNDING WETWELL TEMPERATURE PROFILE~~

FIGURE 3.11-4

Rev 20 5/83

Issue 7

The applicant must demonstrate why allowing a pressure greater than minus 0.25 inch water in the common refueling area following a LOCA is acceptable (i.e., no leakage or bypass leakage paths from the primary containment to the common refueling area, no LOCA impact on fuel storage facilities). (Note: This is a new issue resulting from the SGTS design change documented in FSAR Revision 15.)

Response

The response to question 480.22 has been changed to verify that no leakage or bypass leakage paths from the primary containment to the common refueling area exist. The revised response is attached.

QUESTION 480.22 (Section 6.2.3)

FSAR Section 6.2.3.2.1, page 6.2-40, states "An analysis of the post-LOCA pressure transient in the secondary containment will be performed to determine the length of time following isolation signal initiation of the SGTS that the pressure in the secondary containment would exceed minus 0.25 in. wg." Provide the results of this analysis of the pressure and temperature response of the secondary containment to a loss-of-coolant accident (LOCA) occurring inside the primary containment, and describe specifically how each of the guidelines of SRP Section 6.2.3 Item II.1 has been followed.

RESPONSE

Section 6.2.3.2.1 has been changed to provide the results of the post-LOCA secondary containment pressure transient analysis. The LOCA radiological analyses in Chapter 15 have been changed to account for the radiation released from the secondary containment during the time that the pressure exceeds minus 0.25 in. wg.

In addition to the pressure transient analysis, a detailed review has been performed to identify potential leakage paths from either the primary containment or the secondary containment to the common refueling area. This review resulted in the following changes which ensure that no leakage paths exist:

a. A vent path from the reactor well to the reactor enclosure was added.

b. Normally closed valves on the reactor well skimmer drain lines were added.

Periodic tests are performed in accordance with the plant Technical Specifications to verify that the secondary containment inleakage is less than 50 percent of its free volume per day at a negative interior pressure of 0.25 in. wg.

Issue 8

Regarding the identification of bypass leakage paths, CSB has three remaining concerns: (Open Item)

a. The applicant has not adequately demonstrated why it is realistic to assume that a water seal at a pressure greater than the containment accident pressure and lasting at least 30 days could be maintained in the feedwater line by water from the condensate storage tank following a feedwater line break inside containment.
b. The applicant has not demonstrated that the requirements of a closed system are met for the systems inside containment to which the recirculation pump cooling water supply and return lines and the drywell chilled water supply and return lines connect. The applicant's response to NRC Question 480.26 Part a has not resolved this open item.
c. CSB has been unable to confirm the presence of a vent line to the secondary containment located before two block valves and the secondary containment in the nitrogen line to the TIP indexing mechanism.

Response

The above bypass leakage concerns are addressed in order below.

a) Section 6.2.3.2.3.1 has been changed to demonstrate why it is realistic to assume that a water seal would be maintained by the condensate storage tank in the feedwater line after a feedwater line break inside containment.

b) Water seals would be maintained in the lines outside containment for the closed cooling water systems inside containment. This is discussed in the response to CSB item 15.

c) Section 6.2.3.2.3 has been changed to demonstrate why no significant amounts of radioactivity would be released to the environment from the TIP purge nitrogen supply line.

The revised sections are attached, as well as Table 6.2-15 which has also been revised to reflect the section changes.

The secondary containment design data are in Table 6.2-14.

6.2.3.2.2 Secondary Containment Isolation System

Isolation dampers and the plant protection signals that activate the secondary containment isolation system are described in Section 9.4.2.1.3.

6.2.3.2.3 Containment Bypass Leakage

Upon receipt of a reactor enclosure isolation signal, the reactor enclosure recirculation system (RERS) and the SGTS are automatically activated and begin to process all air flow streams from the reactor enclosure ventilation system. Therefore, if a LOCA occurs, radioactivity that exfiltrates the steel-lined primary containment or piping systems containing radioactive fluids is collected and passed through the RERS and SGTS as described in Section 6.5.

The potential paths by which leakage from the primary containment can bypass the areas serviced by the SGTS have been evaluated. Table 6.2-15 identifies all primary containment penetrations, the termination region of all lines penetrating primary containment, and the bypass leakage barriers in each line. It has been determined that no potential bypass leakage paths exist ~~for the entire spectrum of LOCAs except for a feedwater line break inside containment.~~ A water seal cannot be maintained in the broken feedwater line by the feedwater fill system (Section 6.2.3.2.3.2) for the case of a feedwater line break inside containment. For this case, containment leakage may travel past the broken feedwater line's containment isolation valves into the portion of the feedwater line located in the turbine enclosure. However, a water seal in this portion of the feedwater line would ~~realistically be expected to~~ be maintained by water from the condensate storage tank ~~. Therefore, no bypass leakage is postulated to reach the environment.~~ as discussed in Section 6.2.3.2.3.1.

Insert (A)

When designating the termination region, if either the system line that penetrates primary containment or any branch lines connecting to it penetrate the secondary containment, the termination region is listed in Table 6.2-15 as outside secondary containment (OSC). The types of bypass leakage barriers employed by these lines are:

1. Redundant primary containment isolation valves
2. Closed seismic Category I piping system inside containment
3. A water seal maintained for 30 days following a LOCA

Insert A

except for the feedwater line following a feedwater line break inside containment and the TIP purge nitrogen supply line. No significant amounts of radioactivity will be released to the environment in either case, as discussed below. ¶ Leakage through the containment isolation valves of the TIP purge line could be released from a break or equipment failure into the reactor enclosure, radwaste enclosure, or outdoors. Leakage into the reactor enclosure would not constitute a bypass leakage path since the reactor enclosure is serviced by SGTS. Any leakage into the radwaste enclosure would not result in a significant release of radioactivity because of plateout, deposition, and transport delay within the TIP purge piping. The long lines within the reactor enclosure (>240 feet of 1" pipe and >220 feet of 6" pipe) and the presence of numerous valves in these lines (2 containment isolation valves, a 1/4" check valve, a 1/4" solenoid valve, a 1/4" metering valve, a 1" check valve, and a 1" globe valve) provides a long and tortuous route between the primary containment and the radwaste enclosure. Any leakage through the reactor enclosure and radwaste enclosure piping would be into the liquid nitrogen facility (tanks, vaporizer, control valves, etc) located adjacent to the radwaste enclosure. The liquid nitrogen facility maintains system pressure at 50 psig.

4. The line beyond the outboard primary containment isolation valve is vented to secondary containment by use of a vent line located upstream of the two block valves.

5. A leakage collection system is provided.

6. The line contains a temporary spool piece that is removed during normal operation and replaced by blind flanges so that any leakage through the flange is into secondary containment.

Type 1 leakage barriers are considered to limit but not eliminate bypass leakage. Leakage barriers of types 2 through 6 are considered to effectively eliminate any bypass leakage.

Leakage from those lines terminating in the reactor enclosure is collected during the LOCA because the reactor enclosure is restored to and maintained at subatmospheric pressure and all exhaust is processed by the RERS and SGTS during these modes (Section 6.5). Therefore, lines terminating within the reactor enclosure are not considered potential bypass leakage paths.

Lines penetrating primary containment are isolated following a LOCA as described in Section 6.2.4. All containment isolation valves and penetrations are designed to seismic Category I requirements.

The primary containment and penetration leakage is monitored during periodic tests as discussed in Section 6.2.6. Those penetrations for which credit is taken for water seals as a means of eliminating bypass leakage (Table 6.2-15) are preoperationally leak-tested with water and Technical Specification leakage rates are given as water leak rates.

6.2.3.2.3.1 Water Seals

In each case where water seals are used to eliminate the potential of secondary containment bypass leakage, a 30-day water seal is assured because either a loop seal is present or the water for the seal is provided from a large reservoir. The water seals for all of these lines will be maintained at a pressure greater than the peak containment accident pressure. Each of the water seals listed in Table 6.2-15 is discussed below (some penetrations may be listed more than once due to the presence of multiple types of water seals).

a. Penetrations 9A & B and 44. The feedwater fill system (Section 6.2.3.2.3.2) is used to maintain a water seal in the lines downstream of these penetrations. Insert (B)

(Insert B)

The feedwater fill system cannot maintain a water seal in the feedwater lines (penetrations 9A & B) following a feedwater line break inside containment. However, for this case, a water seal will be maintained in the piping and equipment between the condensate storage tank and the feedwater penetrations since

- The elevation of the primary containment penetrations is 287',
- The elevation of the minimum CST water level is 245', and
- The low point of the makeup water line is at elevation 220'.

Therefore the water seal will be maintained in that portion of the piping system below elevation 245'. This water seal will be able to withstand the containment pressure resulting from a feedwater line break as discussed below.

Immediately after the line break, when containment pressure will be at its peak, the feedwater line will still have a pressure greater than the ~~containment~~ pressure in primary containment. The feedwater will flash into steam at the break until the line pressure matches the containment pressure, thus preventing any containment atmosphere from entering the line during this period. In addition there are several valves

Insert B (cont'd)

in the feedwater line between the primary containment and the water seal. A pressure decrease will occur across each valve, reducing the pressure the water seal will encounter. Figures 6.2-3 and 6.2-7 show the primary containment pressure transient following a recirculation line break. These figures show that the maximum long-term pressure post-LOCA is approximately 17 psig. The difference in elevation between the lowest point of the water seal and the minimum Condensate Storage Tank (CST) water level indicates a head of 20 psig, which is greater than the pressure the primary containment will exert on the water seal.

The piping and equipment which comprise this potential bypass leakage flow path is continuously inspected during normal operation when pressures are more than 10 times greater than post-LOCA pressures. Even considering liquid leakage from the components in the water seal, the volume of water contained in the CST is sufficiently large to assure that a water seal is maintained for at least 30 days.

 No Change

b. Penetrations 204A & B, 207A & B, 208B, 210, 212, 215, 216, 217, 226A & B, 235, 236, 238, 239 and 240. The lines associated with these penetrations all penetrate the wetwell above the suppression pool water level and terminate at least 4 feet below the minimum suppression pool water level. A 30-day water seal is therefore assured on the submerged portion of line.

c. Penetrations 13A & B, 16A & B, 17, 39A & B, 45A-D, 205A & B, and 225. Piping connected to these penetrations is normally full of water and will be kept full after a LOCA due to operation of the ECCS and/or safeguard piping fill system. The suppression pool is the water source for the ECCS and fill system, and therefore a 30-day water supply is assured.

d. Penetrations 203A-D, 206A-D, 209, 214 and 237. The lines associated with these penetrations all penetrate the wetwell at least 11 feet below the minimum water level of the suppression pool, and therefore a 30 day water seal is assured.

e. Penetrations 231A & B. The line to the containment isolation valves from the drywell floor drain sump is maintained full of water by an elevation difference between the sump and the valves. The line to the containment isolation valves from the drywell equipment drain tank is maintained full of water by an elevation difference between the tank and the valves.

f. Penetrations 10, 11, 12, 44, 228D and 241. Lines associated with these penetrations that pass through the secondary containment boundary and take credit for water seals are provided with loop seals inside secondary containment, which eliminates the possibility of bypass leakage.

g. Penetration 14. The minimum piping height inside primary containment of the RWCU supply line that branches off the recirculation loop is at El 267 ft. The primary containment penetration is at El. 297 ft and the RPV penetration is at El. 280 ft. This elevation difference ensures that a water seal is maintained in the line from the RPV to the containment isolation valves. The RWCU supply branch line that connects to the bottom of the vessel is normally full of water, and the water will be maintained in this line because it connects directly to, and below, the vessel.

h. Penetrations 37A-D and 28A-D. The CRD insert and withdraw lines are normally full of water. A water seal will be maintained in these lines after a LOCA due to

 No Change

the elevation difference between the containment penetrations (El. 265 ft) and the connections to the control rod drives (El. 215 ft).

6.2.3.2.3.2 Feedwater Fill System

The feedwater fill system prevents the release of fission products through the feedwater containment isolation valves after a LOCA by providing a water seal downstream of the valves.

6.2.3.2.3.2.1 Safety Design Bases

The feedwater fill system is designed with sufficient capacity and capability to prevent leakage through the feedwater lines under the conditions associated with the entire spectrum of LOCAs except for a feedwater line break inside containment.

The feedwater fill system conforms to seismic Category I requirements. Quality group classifications are shown in Table 3.2-1, Item XI.A. The system meets the intent of Regulatory Guide 1.96, where applicable.

The feedwater fill system is capable of performing its safety function considering the effects resulting from a LOCA, including missiles that may result from equipment failures, dynamic effects associated with pipe whip and jet forces, and normal operating and accident-caused local environmental conditions consistent with the design basis event. Furthermore, any portion of the feedwater fill system that is quality Group A and is located outside the primary containment structure is protected from missiles, pipe whip, and jet force effects originating outside the containment so that containment integrity is maintained.

The feedwater fill system is capable of performing its safety function following a LOCA and an assumed single active failure.

The feedwater fill system is designed so that effects resulting from a single active component failure do not affect the integrity or operability of the feedwater lines or the feedwater containment isolation valves.

The feedwater fill system is capable of performing its safety function following a loss of all offsite power coincident with a postulated design basis LOCA.

The feedwater fill system is designed to prevent leakage from the feedwater lines consistent with maintaining containment integrity for up to 30 days.

The feedwater fill system is manually actuated and is not required to be actuated sooner than 30 minutes after a LOCA.

TABLE 6.2-15

EVALUATION OF POTENTIAL SECONDARY CONTAINMENT BYPASS LEAKAGE PATHS

CONTAINMENT PENETRATION	SYSTEM	TERMINATION REGION[1]	BYPASS LEAKAGE BARRIERS[2]	POTENTIAL BYPASS PATH
1	Equipment access door	ISC	Double O-Ring	No
2	Equipment access door and personnel lock	ISC	Double O-Ring	No
3A	Main steam (MS) line D flow instrumentation	ISC	-	No
3B	Inst gas supply	OSC	1,4	No
3C	HPCI steam flow inst	ISC	-	No
3D	MS line A flow inst	ISC	-	No
3D	Instrument gas supply	OSC	1,4	No
4	Head access manhole	ISC	Double O-Ring	No
5	Spare	-	-	-
6	CRD removal hatch	ISC	Double O-Ring	No
7A-D	Primary steam	OSC	1,5	No
8	Primary steam line drain	OSC	1,4	No
9A&B	Feedwater	OSC	1,3	No[3],[4]
10	Steam to RCIC turbine	OSC	1,3,6	No
11	Steam to HPCI turbine	OSC	1,3,6	No
12	RHR shutdown cooling supply	OSC	1,3	No
13A&B	RHR shutdown return	OSC	1,3	No
14	RWCU supply	OSC	1,3	No
15	Spare	-	-	-
16A&B	Core spray pump discharge	OSC	1,3	No
17	RPV head spray	OSC	1,3	No
18	Spare	-	-	-
19	Spare	-	-	-
20A	RPV level inst	ISC	-	No
20A	LPCI ΔP inst	ISC	-	No
20B	LPCI ΔP inst	ISC	-	No
20B	RPV level inst	ISC	-	No
21	Spare	-	-	-
22	Drywell pressure inst	ISC	-	No
23	Closed cooling water supply	OSC	2	No
24	Closed cooling water return	OSC	2	No
25	Drywell purge supply	OSC	1,4	No
26	Drywell purge exhaust	ISC		No
27A	Instrument gas supply	OSC	1,4	No
27B	HPCI flow inst	ISC	-	No
28A	Recirc loop sample	ISC	-	No
28A	Drywell H_2/O_2	ISC	-	No
28B	LPCI ΔP inst	ISC	-	No
28B	Drywell air sample	ISC	-	No
29A	RPV flange leakage inst	ISC	-	No
29B	Core spray ΔP inst	ISC	-	No
30A	MS line D flow inst	ISC	-	No
30B	Drywell pressure inst	ISC	-	No
30B	MS line C flow inst	ISC	-	No

TABLE 6.2-15 (Cont'd)

CONTAINMENT PENETRATION	SYSTEM	TERMINATION REGION[1]	BYPASS LEAKAGE BARRIERS[2]	POTENTIAL BYPASS PATH
31A&B	Jet pump flow inst	ISC	-	No
32A&B	Jet pump flow inst	ISC	-	No
33A	Pressure above core plate inst	ISC	-	No
33A	Pressure below core plate inst	ISC	-	No
33B	RCIC steam flow inst	ISC	-	No
34A	MS line C flow inst	ISC	-	No
34B	Recirc flow inst	ISC	-	No
35A	Inst gas to TIP indexing mechanism	OSC	1,~~4~~ 8	NO[(4)]
35C-G	TIP drives	ISC	-	No
36	Spare	-	-	-
37A-D	CRD insert	OSC	1,3	No
38A-D	CRD withdraw	OSC	1,3	No
39A&B	Drywell spray	OSC	1,3	No
40A,B&C	Jet pump flow inst	ISC	-	No
40D	Pressure below core plate inst	ISC	-	No
40E	Drywell pressure inst	ISC	-	No
40F	RCIC steam flow inst	ISC	-	No
40F	Inst gas suction	OSC	1,4	No
40G	ILRT data acquis system	OSC	1,6	No
40H	Instrument gas supply	OSC	1,4	No
40H	Recirc pump cooler flow inst	ISC	-	No
41	LPCI ΔP inst	ISC	-	No
41	RWCU flow inst	ISC	-	No
42	Standby liquid control	ISC	-	No
43A	Recirc loop A ΔP inst	ISC	-	No
43A	Recirc pump seal pressure inst	ISC	-	No
43B	Main steam sample	ISC	-	No
44	CRD/RWCU return	OSC	1,3	No
45A-D	LPCI	OSC	1,3	No
46	Spare	-	-	-
47	RWCU flow inst	ISC	-	No
48A	RPV level inst	ISC	-	No
48A	Core spray ΔP inst	ISC	-	No
48B	RPV level inst	ISC	-	No
49A&B	MS line A&B flow inst	ISC	-	No
50A	Drywell pressure inst	ISC	-	No
50A	Recirc flow inst	ISC	-	No
50B	Recirc pump seal pressure inst	ISC	-	No
50B	Recirc pump cooler flow inst	ISC	-	No
51A	Recirc line flow inst	ISC	-	No
51B	Jet pump flow inst	ISC	-	No
52A	MS line B flow inst	ISC	-	No
52B	Recirc line flow inst	ISC	-	No
53	Drywell chilled water supply	OSC	2	NO
54	Drywell chilled water return	OSC	2	No
55	Drywell chilled water supply	OSC	2	No
56	Drywell chilled water return	OSC	2	No

CONTAINMENT PENETRATION	SYSTEM	TERMINATION REGION[1]	BYPASS LEAKAGE BARRIERS[2]	POTENTIAL BYPASS PATH

[1] The termination regions are: ISC - Inside Secondary Containment
OSC - Ountside Secondary Containment

[2] The bypass leakage barriers are defined as follows (see Section 6.2.3.3.3):
1. Redundant primary containment isolation valves
2. Closed piping system inside containment
3. A water seal maintained for 30 days following a LOCA
4. The line beyond the outboard primary containment isolation valve is vented to secondary containment by use of a vent line located between two block valves and the secondary containment.
5. A leakage collection system is provided
6. The line contains a temporary spool piece that is removed during normal operation and replaced by blind flanges so that any leakage through the flange is into secondary containment.

[3] The feedwater fill system will provide a water seal in the feedwater lines for all line breaks other than a feedwater line break inside containment. For a feedwater line break inside containment, a water seal is maintained by the condensate storage tank water supply as discussed in Section 6.2.3.2.3.1.a.

(4) No significant amounts of radioactivity will be released to the environment as discussed in Section 6.2.3.2.3.

Issue 10

Diverse containment isolation signals (i.e., reactor vessel level trip and high drywell pressure) are required to automatically isolate the nonessential main steam drain, main steam sample, recirculation loop sample, and RWCU system supply lines. (Open Item)

Response

Section 1.13 has been revised to provide justification for the isolation signals provided to the automatic isolation valves in each of the above lines. The revision is attached.

No change

adequate to account for instrument error. Any proposed values greater than 1 psi will require detailed justification. Applicants for an operating license and operating plant licensees that have operated less than one year should use pressure history data from similar plants that have operated more than one year, if possible, to arrive at a minimum containment setpoint pressure.

(7) Sealed-closed purge isolation valves should be under administrative control to assure that they cannot be inadvertently opened. Administrative control includes mechanical devices to seal or lock the valve closed, or to prevent power from being supplied to the valve operator. Checking the valve position light in the control room is an adequate method for verifying every 24 hours that the purge valves are closed.

Response

A description of compliance with each Position and Clarification is provided below.

Position (1), Clarification (1)

The containment isolation system design has been reviewed for compliance with SRP 6.2.4 regarding diversity in the parameters sensed for the initiation of containment isolation. Section 6.2.4 and Table 6.2-17 identify all containment isolation signals provided. There are eleven valves classified as nonessential that do not receive diverse containment isolation signals.

Two valves on the feedwater lines (HV-109A, HV-109B) are normally closed and will be opened only for startup of the feedwater system before the control rods are withdrawn.

The RCIC vacuum pump discharge line is provided with a stop-check valve (HV-F002) to prevent flow from the containment. A remote manually actuated motor operator ensures the long-term positive closure of the stop-check valve. This arrangement ensures that the essential RCIC pump-turbine will be ready to operate in the event of a reactor vessel isolation occurrence accompanied by loss of feedwater flow.

The recirculation pump cooling water supply and discharge isolation valves (HV-106, HV-107) and the drywell chilled water isolation valves (HV-122, HV-123, HV-128, HV-129) have provisions for remote manual isolation consistent with GDC 57. Closure of these isolation valves is undesirable unless the cooling water lines have failed.

The HPCI and RCIC steam supply line warmup valves (HV-F100, HV-F076, respectively) are provided with appropriate isolation signals to secure the line when system isolation is required. There is no adverse consequence associated with the valve opening or leaking while these systems are in operation.

Insert A

Position (2), Clarification (3)

All systems penetrating containment have been evaluated and identified as either essential or nonessential. Table 6.2-17 provides the results of this evaluation for each line, and Table 6.2-27 provides the basis for the selection of essential/nonessential systems.

Position (3), Clarification (2)

Systems determined to be nonessential are provided with diverse, automatic isolation signals, except as described in the response to Position (1). Manual valves are sealed closed as discussed in Section 6.2.4.3.

Position (4), Clarifications (4), (5)

The control systems for automatic isolation valves are such that resetting the isolation signal will not result in the automatic reopening of these valves. Ganged reopening of containment isolation valves is performed only where the operation of

Insert (A)

The main steam drain line isolation valves (HV-F016, F019) are normally closed during power operation. They provide a path from the steam lines to the main condenser for removal of condensation during shutdown and startup periods and during periods of low load. The six automatic isolation signals provided for these valves are the same as those provided for the MSIV's.

The main steam and recirculation loop sample line isolation valves (HV-F084, F085 and HV-F019, F020 respectively) are typically open less than one hour per year during normal plant operations. The two isolation signals which are provided for these valves assure their automatic closure before any fuel damage would occur for all anticipated periods of sample line use.

The RWCU supply line isolation valves (HV-F001, F004) are provided with the following signals to initiate automatic valve closure:

- Reactor low water level
- Line break in RWCU
  - high flow
  - heat exchanger high temp.
  - compartment high temp.
- Standby Liquid Control System operation

Insert (A) cont'd

These isolation signals are provided to protect the core in case of a possible break in the Reactor Water Cleanup System, to protect the ion exchange resin from damage due to high temperature, and to prevent the removal of boron by the ion exchange resin.

The RWCU system is described in FSAR Section 5.4.8. Closing times of the RWCU isolation valves have been chosen in order to prevent the reactor vessel water level from falling below the top of active fuel if a break were to occur in any of the RWCU lines. Diverse isolation signals are supplied to isolate the RWCU in the unlikely event of such a line break. The system is intentionally left in service whenever the above isolation signals are not activated in order to provide continuous purification of a portion of the recirculation flow.

Issue 11

The applicant must either demonstrate that design provisions to detect possible leakage from the main feedwater lines and RCIC and RWCU supply lines that connect to the main feedwater lines are provided or commit to administrative procedures to close the remote-manually actuated containment isolation valves on these lines shortly (e.g., within 20 minutes) following a LOCA or sooner if information indicates a degraded core condition exists. (Confirmatory Item)

Response

Section 6.2.4.3.1.2 has been changed to state that plant procedures will provide for remote manual closure of the above lines following a LOCA, when the lines are not in use for reactor coolant makeup. The revised section is attached.

No change

containment isolation barriers, are maintained. All power-operated isolation valves have position indicators in the control room. Discussion of instrumentation and controls for the isolation valves is included in Chapter 7.

6.2.4.3.1 Evaluation Against General Design Criteria

6.2.4.3.1.1 Evaluation Against General Design Criterion 54

All piping systems penetrating containment, other than instrument lines, are designed in accordance with Criterion 54.

6.2.4.3.1.2 Evaluation Against Criterion 55

Criterion 55 requires that lines which penetrate the primary containment and form a part of the RCPB must have two isolation valves; one inside the containment and one outside, unless it can be demonstrated that the containment isolation provisions for a specific class of lines are acceptable on some other basis.

The RCPB, as defined in 10 CFR Part 50, Section 50.2 (v), consists of the reactor pressure vessel, pressure retaining appurtenances attached to the vessel, and valves and pipes that extend from the reactor pressure vessel up to and including the outermost isolation valve.

6.2.4.3.1.2.1 Influent Lines

Influent lines that penetrate the primary containment and connect directly to the RCPB are equipped with at least two isolation valves, one inside the drywell, and the other as close to the external side of the containment as practicable.

6.2.4.3.1.2.1.1 Feedwater Line

The feedwater line is part of the RCPB as it penetrates the drywell to connect with the reactor pressure vessel. It has three isolation valves. The isolation valve inside the drywell is a check valve located as close as practicable to the containment wall. Outside the containment is an air-assisted check valve located as close as practicable to the containment wall, and farther away from the containment is a motor-assisted check valve on the feedwater line. Additional isolation valves are located on lines connecting to the feedwater line outside containment. Should a break occur in the feedwater line, the outboard check valves prevent significant loss of reactor coolant inventory and offer immediate isolation. (It is impractical to restrain the inboard check valve to withstand pipe whip resulting from a downstream feedwater line break; therefore it cannot be assumed to isolate for this case.) During a postulated LOCA, it is desirable to maintain reactor coolant makeup from all sources of supply. For this reason, the feedwater lines are not

automatically isolated upon signals from the protection system. Following a LOCA, plant procedures will require closure of the outermost valve, which is capable of being remotely closed from the control room to provide long-term leakage protection, unless the feedwater lines are providing reactor coolant makeup.
The air-assisted check valve is provided with a special actuator that performs the following functions:

a. The actuator is capable of partially moving the valve disc into the flow stream during normal plant operation in order to ensure that the valve is not bound in the open position. The actuator is not capable of fully closing the valve against flow, however, and there is no significant disruption of feedwater flow.

b. The actuator is capable of applying a seating force to the valve at low differential pressures and abnormal conditions. This improves the leaktightness of the valves. The actuator is not utilized during leak testing.

6.2.4.3.1.2.1.2 HPCI Line

The HPCI line connects to CS loop B that penetrates the drywell to inject directly into the RPV. Isolation is provided by two valves in the CS line, an air testable check valve inside the containment, and an air assisted check valve outside the containment, with positions of both indicated in the main control room. The core spray loop B line is also provided with a normally closed motor-operated globe valve which bypasses the inboard isolation valve for equalization during testing.

6.2.4.3.1.2.1.3 LPCI and CS Loop A

The LPCI lines and CS loop A line are provided with remote manually controlled gate valves outside and air testable check valves inside containment. Both types of valves are normally closed with the gate valves receiving an automatic signal to open at the appropriate time. The check valves are located as close as practicable to the RPV. The normally closed check valves protect against containment pressurization if there is a pipe rupture between the check valve and containment wall. The core spray loop A line and the LPCI lines are also each provided with a normally closed motor-operated globe valve which bypasses the inboard isolation valve for testing purposes.

6.2.4.3.1.2.1.4 RHR Head Spray Line

The RHR head spray line penetrates the drywell and discharges directly into the RPV. Isolation for this line is provided by a remote manually controlled gate valve inside containment and a remote manually controlled globe valve outside containment. Both valves are normally closed and receive an automatic isolation signal if there is an accident.

6.2.4.3.1.2.1.5 Recirculation Pump Seal Purge Line

The recirculation pump seal purge line extends from the CRD supply line outside primary containment, penetrates primary containment through an excess flow check valve outside and a check valve inside containment, and connects to the recirculation pump seal housing. The 1-inch recirculation pump seal purge line is Quality Group Classification A from the pump up to and including the excess flow check valve outside containment. Should this line be postulated to fail and either one of the check valves is assumed to fail open, the flow rate through a broken line outside containment is calculated to be substantially less than that permitted for a broken instrument line. Therefore, the two check valves in series provide sufficient isolation capability for the postulated failure of this line.

6.2.4.3.1.2.1.6 Standby Liquid Control System Lines

The SLC system line penetrates the drywell and connects to the RPV. In addition to a simple check valve inside the drywell, a motor-operated globe stop check valve is located outside the drywell. Since the SLC line is a normally closed, nonflowing line, rupture of this line is of extremely remote probability. An explosive-actuated valve provides an absolute seal for long-term leakage control, provided the SLC system has not been utilized.

6.2.4.3.1.2.1.7 RWCU System

The RWCU pumps, heat exchangers, and filter/demineralizers are located outside the drywell. The return line from the filter/demineralizers branches into three separate lines outside the drywell. One line connects to the RCIC line that connects to the B feedwater line penetrating the drywell and injecting directly into the RPV. The second branch connects directly to the A feedwater line that penetrates the drywell and injects directly into the RPV.

Isolation of both these lines is provided by the feedwater system check valves inside and outside the containment and an air-operated check valve in the connecting RWCU return line.

However, plant procedures will require remote manual closure of the isolation valves.

Following a LOCA, it is desirable to maintain reactor coolant makeup. For this reason, the above isolation valves are not provided with automatic isolation signals. A motor-operated globe valve is provided for long-term leakage control. Should a break occur in the RWCU return line, the check valves prevent significant loss of inventory and offer immediate isolation, while the motor-operated globe valve provides long-term leakage control.

The third line penetrates the drywell and then connects to the A feedwater line that injects directly into the reactor pressure vessel. This line is only used during outages. Isolation is provided by one locked-closed globe valve inside containment and one locked-closed globe valve outside containment.

6.2.4.3.1.2.1.8 RCIC Line

The RCIC line connects to the B feedwater line outside containment that penetrates the drywell to inject directly into the RPV. The feedwater line has a check valve both inside and outside the drywell. In addition to these two isolation valves, a motor-operated gate valve is located in the RCIC line that is normally closed and receives an automatic signal to open. ~~This valve can be remote manually isolated as deemed necessary by the operator.~~ Following a LOCA, plant procedures will require remote manual closure of this isolation valve unless RCIC is providing reactor coolant makeup.

6.2.4.3.1.2.1.9 RHR Shutdown Cooling Return

The RHR shutdown cooling return line penetrates primary containment and discharges into a recirculation pump discharge line that injects directly into the RPV. Isolation is provided by an automatically actuated motor-operated globe valve outside containment and an air testable check valve inside containment.

6.2.4.3.1.2.2 Effluent Lines

Effluent lines that form part of the RCPB and penetrate containment are equipped with at least two isolation valves; one inside the drywell and the other outside, located as close to the containment as practicable or justified on an alternate basis.

6.2.4.3.1.2.2.1 Main Steam, RCIC and HPCI Steam Lines, and RHR Shutdown Cooling Supply Line

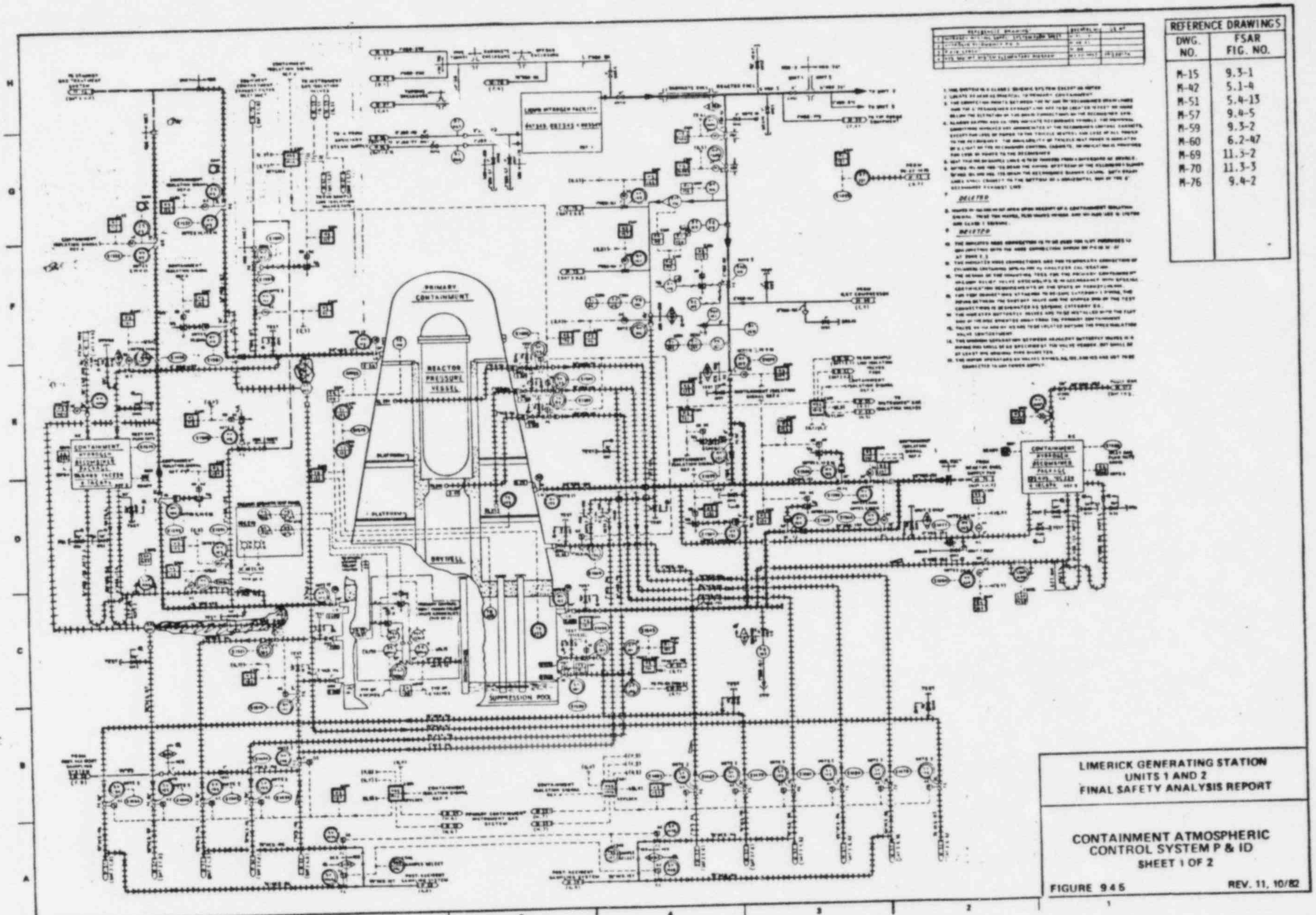
The main steam lines extend from the RPV to the main turbine and condenser system, and penetrate the primary containment. For these lines, isolation is provided by automatically actuated globe valves, one inside the containment and one just outside the containment. The main steam line isolation valves are spring-loaded, pneumatic, piston-operated globe valves designed to fail closed on loss of pneumatic pressure or loss of power to the solenoid-operated pilot valves. Each valve has two

Issue 12

The two H_2/O_2 sampling lines connected to the 24-inch and 18-inch drywell and suppression pool exhaust lines contain only a single containment isolation valve. Either a second containment isolation valve is required, or the applicant must demonstrate that the combustible gas analyzer package internal tubing meets the requirements of a closed system. (Open Item)

Response

The above points of connection of the H_2/O_2 sampling lines have been changed such that two isolation valves are provided as shown on revised Figure 9.4-5, which is attached.


REFERENCE DRAWINGS
DWG. NO. | FSAR FIG. NO.
M-15 9.3-1
M-42 5.1-4
M-51 5.4-13
M-57 9.4-5
M-59 9.3-2
M-60 6.2-47
M-69 11.3-2
M-70 11.3-3
M-76 9.4-2
PRIMARY CONTAINMENT
REACTOR PRESSURE VESSEL
DRYWELL
SUPPRESSION POOL
LIMERICK GENERATING STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT
CONTAINMENT ATMOSPHERIC CONTROL SYSTEM P & ID
SHEET 1 OF 2
FIGURE 9.4-5
REV. 11, 10/82

Issue 13

A second containment isolation barrier, such as a closed system inside or outside containment, must be provided for the drywell pressure, suppression pool level, suppression chamber pressure, and drywell sump level instrument lines in addition to the existing remote-manually actuated containment isolation valve located outside containment. Also, justification is required for not requiring the automatic isolation of the drywell sump level instrument lines since this instrumentation is not part of the reactor protection system or important to post-accident monitoring. (Open Item)

Response

Section 6.2.4.3.1.5 has been changed to describe the design provisions of the above instrument lines and is attached.

6.2.4.3.1.5 Evaluation Against Regulatory Guide 1.11

Instrument lines that penetrate the containment from the RCPB conform to Regulatory Guide 1.11 in that they are equipped with a restricting orifice located inside the drywell and an excess flow check valve located outside and as close as practicable to the containment. Should an instrument line that forms part of the reactor pressure boundary develop a leak outside the containment, a flow rate that results in a differential pressure across the valve of 3 to 10 psi causes the excess flow check valve to close automatically. Should an excess flow check valve fail to close when required, the main flow path through the valve has a resistance to flow at least equivalent to a sharp-edged orifice of 0.375 inch diameter. Valve position indication is provided in the reactor enclosure. Those instrument lines that do not connect to the RCPB conform to Regulatory Guide 1.11 in that they are either equipped with an excess flow check valve or an isolation valve capable of remote operation from the control room, and are sized or orificed to meet the criteria outlined in Regulatory Guide 1.11. The status of the isolation valves capable of remote operation from the control room is indicated in the control room.

Insert (A)

The TIP system lines as shown in Figure 9.3-2 and described below are considered instrument lines because (a) they function as instrument lines or support the operation of instrument lines, and (b) they are small diameter lines.

TIP system isolation valves are provided on each guide tube immediately outside the containment. Dual barrier protection is provided by a solenoid operated ball valve and an explosive actuated cable shearing valve. The ball valve is closed except when a TIP is inserted. These valves prevent loss of reactor coolant in the event that an incore guide tube ruptures inside the reactor vessel and prevents the escape of primary containment atmosphere.

The guide tube ball valve solenoid is normally de-energized and the valve is in the closed position. When the TIP starts forward, the valve solenoid is energized and the valve is held open against its spring. As the valve opens, it actuates a set of contacts which provide position indication at the TIP control panel and a permissive signal for TIP motion. Upon receipt of a containment isolation signal (reactor low water level or high drywell pressure), the TIP drive mechanism is signalled to retract the TIP. As the TIP is withdrawn into its shield chamber outside containment, a position switch signals the ball valve to close.

The shear valve is provided as a backup in the event that a TIP cannot be retracted or a ball valve sticks open when containment

Insert (A)

The drywell pressure, suppression pool level, suppression chamber pressure, and drywell sump level instrument lines are (1) provided with isolation valves capable of remote operation from the control room, (2) Q-listed, as discussed in Section 3.2, (3) designed to seismic Category I standards, (4) designed to withstand containment design pressure and temperature, and (5) terminate in the reactor enclosure, which is served by the SGTS.

Issue 15

Because the systems inside containment to which the recirculation pump cooling water supply and return lines and the drywell chilled water supply and return lines connect are not closed systems (i.e., not Safety Class 2 and/or not seismic Category I and not protected against missiles and pipe whip) the containment isolation design of these penetrations must be revised to meet GDC 56 requirements. The applicant's response to NRC Question 480.40 has not resolved this open item. (Open Item)

Response

The containment isolation system designs of the drywell chilled water system and the reactor enclosure cooling water system have been revised to meet the requirements of GDC 56. The following Sections, Tables, and Figures have been revised accordingly and are attached.

Sections 1.13, 3.2.2, 6.2.3, 6.2.4, 7.3.1.1.2

Tables 3.2-1, 6.2-15, 6.2-17, 6.2-25, 7.5-3

Figures 9.2-25, 9.2-27

Questions 480.26, 480.40

Reactor enclosure cooling water and

multiple valves is required for system operation. Sample inlet and return valve controls for the drywell radiation monitors and combustible gas analyzers are ganged as described in Sections 6.2.4.3.1.3.2.8 and 6.2.4.3.1.3.2.1. Drywell chilled water valve controls are ganged as described in Sections 6.2.4.3.1.~~2.3~~3.2.11. 6.2.4.3.1.3.2.10

Position (5), Clarification (6)

The setpoint for the drywell high pressure isolation signal is set at the minimum compatible with normal operation. Section 7.3.1.1.2.4.6 describes the selection of the drywell high pressure setpoint.

Position (6), Clarification (7)

Containment purge valves comply with Branch Technical Position CSB 6-4 as discussed below. Two purge isolation valves have closure times greater than 5 seconds: 2"-HV-105 and 2"-HV-111 have closure times of 30 seconds. An analysis of the radiological consequences of a LOCA that occurs during purging was performed to justify the line size and the valve closure time used in the purge system. Using the assumptions of BTP CSB 6-4, the resulting doses were a small fraction of the 10CFR100 limits. For local leak rate tests, the leakage rate of the purge isolation valves, combined with the leakage rate for all other penetrations and valves subject to Type B and C tests will be less than 0.60 La, in accordance with Appendix J to 10CFR50.

Position (7)

The containment purge isolation valves isolate on receipt of any one of the following signals:

a. high drywell pressure

b. reactor low water level

c. reactor enclosure high radiation

d. refueling floor high radiation

An analysis has been performed to demonstrate that the offsite doses that might result if a LOCA were to occur during purging operations would be less than both 10CFR100 and EPA Protection Action Guide limits. This analysis used the assumptions of NUREG 0800 Section 6.2.4 and Branch Technical Position CSB 6-4 and assumes a pre-existing spike that results in coolant activity levels in excess of Technical Specification limits. The analysis methodology was in accordance with the letter from T.J. Dente (BWR Owners Group) to D.G. Eisenhut (NRC) "Supplement to BWR Owners Group Evaluation of NUREG 0737 Item II.E.4.2(7)", dated June 14, 1982.

(2) Field audits are performed by representatives of the originating design group to ensure that the final installation of such items is in accordance with documents that formed the basis for the seismic analysis of the items.

(3) Such items are not included in the "Q" List.

3.2.2 SYSTEM QUALITY GROUP CLASSIFICATIONS

General Design Criterion 1 of 10 CFR Part 50, Appendix A, requires that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with their importance to safety. Components of the reactor coolant pressure boundary meet the requirements for Class 1 components of the American Society of Mechanical Engineers (ASME) B&PV Code, Section III, or equivalent quality standards, as required by 10 CFR Part 50.55.a. Regulatory Guide 1.26, Rev. 3, describes a quality classification system that may be used to determine applicable standards for other components in nuclear power plants. Quality group classifications are assigned to systems and components in accordance with the reliance placed on these systems to:

a. Prevent, or mitigate the consequences of, accidents and malfunctions originating within the RCPB

b. Permit shutdown of the reactor, and maintain it in the safe shutdown condition

c. Contain radioactive material

A tabulation of quality group classification for each component so defined is shown in Table 3.2-1 under the heading, "Quality Group Classification." The applicable codes and standards of each quality group are given in Table 3.2-2. The locations of these components, and the quality group classification of the piping, valves, and interfaces between components of different classifications, are indicated on the system piping and instrumentation diagrams in the pertinent section of the FSAR. A cross reference of system to FSAR figure number is provided in Section 1.7. A

System quality group classifications, and design and fabrication requirements as indicated in Table 3.2-1, meet the guidelines of Regulatory Guide 1.26, except as noted below.

The Limerick design is based on quality group commitments made before Regulatory Guide 1.26 was issued, and in some cases alternate approaches to the guide have been used, as follows:

as described by Regulatory Guide 1.26,

to full quality assurance requirements (Q-listed), and was designed to seismic Category I criteria.

e. Instrument tubing downstream of the containment isolation valve of instrument lines connected to the reactor coolant pressure boundary is Quality Group D for instruments that are "passive" (i.e., do not actuate safety systems), rather than Quality Group B or C as discussed in Paragraphs 1.e and 2.c of the guide. This is based on considerations given in Regulatory Guide 1.11 for instrument lines penetrating containment and having two restriction devices.

Insert (A) →

3.2.3 QUALITY ASSURANCE

Structures, systems, and components whose safety functions require conformance to the applicable quality assurance requirements of 10 CFR Part 50, Appendix B, are summarized in Table 3.2-1 under the heading, "Q-List." Quality assurance during construction is discussed in PSAR Appendix D. The quality assurance program during the operational phase is described in Chapter 17.

Insert (A)

f. The piping between the two containment isolation valves, and the outboard isolation valves, on the Drywell Chilled Water System are the equivalent of Quality Group B although they were originally designed and constructed as ANSI B31.1. Equivalency has been assured through the imposition of supplemental design, fabrication, and testing requirements:

- materials used are permitted by ASME Section III, Class 2.
- seismic and hydrodynamic accelerations were considered in design and analysis.
- design temperatures and pressures are greater than design values for the primary containment.
- documented quality control inspections were performed (by trained and qualified inspectors) on piping installation, welds, valves, and hangers
- the installation is of fully welded construction
- a hydrostatic test was performed at a pressure ~3 times greater than containment [redacted] design pressure

These provisions meet or exceed the committments made in PSAR Appendix A and Figure A.2.1.

TABLE 3.2-1 (Cont'd)

SYSTEM/COMPONENT [40]	FSAR SECTION	SOURCE OF SUPPLY [1]*	LOCATION [2]*	QUALITY GROUP CLASSIFICATION [3]*	PRINCIPAL CODES AND STANDARDS [4]*	SEISMIC CATEGORY [5]*	Q-LIST [6]*	COMMENTS
f. Ductwork and registers		P	R	-	AISI/AWS	I	Y	[23]
C. <u>Primary Containment</u>								
1. Drywell Cooling System	9.2.10, 9.4.5							
a. Piping and valves		P	T,R	D	B31.1	II,IIA	N	
b. Motors, fan		P	C	-	IEEE-334/ NEMA-MG-1	I	Y	
c. Fans		P	C	-	AMCA	I	Y	
d. Coils, cooling		P	C	-	ARI	IIA	N	
e. Ductwork		P	C	-	AISI/AWS	I	Y	[23]
f. Dampers		P	C	-	AMCA	I	Y	
g. Chilled water equipment		P	R	D	MF STD	II	N	
h. Chilled water isolation valves at primary containment		P	R	~~[illegible]~~ B	III-2	I	Y	
i. Piping associated with isolation valves at primary containment penetration		P	C	~~[illegible]~~ B	B31.1	I	Y	[48, 21
2. Purge System								
a. Piping and valves		P	R	B	III-2	I	Y	[48]
b. Piping and valves, beyond outermost containment isolation valves (smaller than 18-inch nominal diameter)		P	R	D	B31.1	IIA	N	
3. Hydrogen recombiner								
a. Piping and valves		P	R	B	III-2	I	Y	[48]
b. Reaction chamber		P	R	B	III-2	I	Y	
c. Blower		P	R	B	III-2	I	Y	
4. Vacuum relief system								
a. Valves		P	C	N	III-2	I	Y	
D. <u>Radwaste and Offgas Enclosure</u>	9.4.3							
1. Fans		P	RW	-	AMCA	II	N	
2. Coils, cooling		P	RW	-	ARI	II	N	
3. Heating coil, steam		P	RW	-	MF STD	II	N	
4. Ductwork		P	RW,T, CS	-	SMACNA	II	N	

TABLE 3.2-1 (Cont'd)

SYSTEM/COMPONENT [40]	FSAR SECTION	SOURCE OF SUPPLY [1]*	LOCATION [2]*	QUALITY GROUP CLASSIFICATION [3]*	PRINCIPAL CODES AND STANDARDS [4]*	SEISMIC CATEGORY [5]*	Q-LIST [6]*	COMMENTS
XI AUXILIARY SYSTEMS								
A. Safeguard Piping Fill System, Including Feedwater Fill System	6.3							
1. Piping and valves, from and including isolation valves, to feedwater lines		P	R	A	III-1	I	Y	[48]
2. Piping and valves, other		P	R	B	III-2	I	Y	[48]
3. Pumps		P	R	B	III-2	I	Y	
B. Suppression Pool Cleanup System Fig.	6.3-9							
1. Piping and valves, to second isolation valve		P	R	B	III-2	I	Y	[48]
2 Piping and valves, after second isolation valve		P	R	D	B31.1	IIA	N	
3. Pumps		P	R	D	MF STD	IIA	N	
C. Demineralized Water Makeup System	9.2.5							
1. Tanks		P	W	-	API-650	II	N	
2. Piping and valves		P	ALL	-	B31.1	II	N	
3. Pumps		P	W	-	B31.1/ HYD.I	II	N	
D. Drywell Chilled Water System	9.2.10							
1. Chillers		P	T	D	VIII-1	II	N	
2. Cooling coils		P	T	-	ARI	II,IIA	N	
3. Piping and valves, other		P	T,R	D	B31.1	II,IIA	N	
4. Valves, isolation to primary containment		P	R	B	III-2	I	Y	
5. Pumps		P	T	D	HYD.I/ B31.1	II	N	
6. Piping associated with isolation valves at primary containment penetration		P	C	~~D~~ B	B31.1	I	Y	[48, 21]
E. Control Structure Chilled Water System	9.2.10							
1. Piping		P	CS	D	B31.1	I	Y	[48]
2. Valves		P	CS	D	B31.1	I	Y	
3. Pumps		P	CS	C	III-3	I	Y	
4. Motors, pump		P	CS	-	IEEE-323, 344	I	Y	
5. Chillers (except condensers)		P	CS	D	VIII-1/ IEEE-323			
6. Chiller condensers		P	CS	C	III-3	I	Y	

[21] ~~Not used~~ The basis for classification of Non-ASME Section III equipment as Quality Group B is provided in Section 3.2.2.F.

[22] Diesel fuel oil storage tanks and transfer pumps were designed to ASME Section III, Class 3 but were not stamped.

[23] The structural design of seismic Category I and IIA HVAC ducts was verified by testing duct specimens as permitted by the AISI Code, to substantiate the duct width to duct sheet thickness ratio (w/t) and duct height to duct sheet thickness ratio (h/t) of up to 1500.

Seismic Category II ducts were designed and constructed in accordance with SMACNA.

[24] NRC Regulatory Guide 1.52, July 1976, suggests various industry standards and codes for this equipment. These references were used for system design, with exceptions as noted in Section 6.5.

[25] Dampers with electro-hydraulic operators were designed to IEEE-323. Fire dampers are labeled by Underwriters' Laboratories.

[26] Portions of ducts and dampers in the reactor enclosure and refueling floor HVAC system are seismic Category II, non Q-listed, and the remainder are seismic Category I, Q-listed.

[27] Deleted

[28] The main steam system (MSS) from its outer isolation valve up to, but not including, the turbine stop valve and bypass valve chest, and all branch lines 2-1/2 inches in diameter and larger up to, and including, the first valve (including their restraints), will be designed by the use of an appropriate dynamic seismic-system analysis to withstand the Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) design loads in combination with other appropriate loads, within the limits specified for Class 2 pipe in the ASME, Section III Code. The mathematical model for the dynamic seismic analyses of the MSS and branch line piping includes the turbine stop valves and the piping from the stop valves to the turbine casing. The dynamic input loads for design of the MSS are derived from a time history model analysis (or an equivalent method) of the reactor and

The secondary containment design data are in Table 6.2-14.

6.2.3.2.2 Secondary Containment Isolation System

Isolation dampers and the plant protection signals that activate the secondary containment isolation system are described in Section 9.4.2.1.3.

6.2.3.2.3 Containment Bypass Leakage

Upon receipt of a reactor enclosure isolation signal, the reactor enclosure recirculation system (RERS) and the SGTS are automatically activated and begin to process all air flow streams from the reactor enclosure ventilation system. Therefore, if a LOCA occurs, radioactivity that exfiltrates the steel-lined primary containment or piping systems containing radioactive fluids is collected and passed through the RERS and SGTS as described in Section 6.5.

The potential paths by which leakage from the primary containment can bypass the areas serviced by the SGTS have been evaluated. Table 6.2-15 identifies all primary containment penetrations, the termination region of all lines penetrating primary containment, and the bypass leakage barriers in each line. It has been determined that no potential bypass leakage paths exist for the entire spectrum of LOCAs except for a feedwater line break inside containment. A water seal cannot be maintained in the broken feedwater line by the feedwater fill system (Section 6.2.3.2.3.2) for the case of a feedwater line break inside containment. For this case, containment leakage may travel past the broken feedwater line's containment isolation valves into the portion of the feedwater line located in the turbine enclosure. However, a water seal in this portion of the feedwater line would realistically be expected to be maintained by water from the condensate storage tank. Therefore, no bypass leakage is postulated to reach the environment.

When designating the termination region, if either the system line that penetrates primary containment or any branch lines connecting to it penetrate the secondary containment, the termination region is listed in Table 6.2-15 as outside secondary containment (OSC). The types of bypass leakage barriers employed by these lines are:

1. Redundant primary containment isolation valves
2. Closed ~~seismic Category I~~ piping system inside containment
3. A water seal maintained for 30 days following a LOCA

4. The line beyond the outboard primary containment isolation valve is vented to secondary containment by use of a vent line located upstream of the two block valves.

5. A leakage collection system is provided.

6. The line contains a temporary spool piece that is removed during normal operation and replaced by blind flanges so that any leakage through the flange is into secondary containment.

Type 1 leakage barriers are considered to limit but not eliminate bypass leakage. Leakage barriers of types ~~2~~ 3 through 6 are considered to effectively eliminate any bypass leakage.

Leakage from those lines terminating in the reactor enclosure is collected during the LOCA because the reactor enclosure is restored to and maintained at subatmospheric pressure and all exhaust is processed by the RERS and SGTS during these modes (Section 6.5). Therefore, lines terminating within the reactor enclosure are not considered potential bypass leakage paths.

Lines penetrating primary containment are isolated following a LOCA as described in Section 6.2.4. All containment isolation valves and penetrations are designed to seismic Category I requirements.

The primary containment and penetration leakage is monitored during periodic tests as discussed in Section 6.2.6. Those penetrations for which credit is taken for water seals as a means of eliminating bypass leakage (Table 6.2-15) are preoperationally leak-tested with water and Technical Specification leakage rates are given as water leak rates.

6.2.3.2.3.1 Water Seals

In each case where water seals are used to eliminate the potential of secondary containment bypass leakage, a 30-day water seal is assured because either a loop seal is present or the water for the seal is provided from a large reservoir. The water seals for all of these lines will be maintained at a pressure greater than the peak containment accident pressure. Each of the water seals listed in Table 6.2-15 is discussed below (some penetrations may be listed more than once due to the presence of multiple types of water seals).

a. Penetrations 9A & B and 44. The feedwater fill system (Section 6.2.3.2.3.2) is used to maintain a water seal in the lines downstream of these penetrations.

Info only

b. Penetrations 204A & B, 207A & B, 208B, 210, 212, 215, 216, 217, 226A & B, 235, 236, 238, 239 and 240. The lines associated with these penetrations all penetrate the wetwell above the suppression pool water level and terminate at least 4 feet below the minimum suppression pool water level. A 30-day water seal is therefore assured on the submerged portion of line.

c. Penetrations 13A & B, 16A & B, 17, 39A & B, 45A-D, 205A & B, and 225. Piping connected to these penetrations is normally full of water and will be kept full after a LOCA due to operation of the ECCS and/or safeguard piping fill system. The suppression pool is the water source for the ECCS and fill system, and therefore a 30-day water supply is assured.

d. Penetrations 203A-D, 206A-D, 209, 214 and 237. The lines associated with these penetrations all penetrate the wetwell at least 11 feet below the minimum water level of the suppression pool, and therefore a 30 day water seal is assured.

e. Penetrations 231A & B. The line to the containment isolation valves from the drywell floor drain sump is maintained full of water by an elevation difference between the sump and the valves. The line to the containment isolation valves from the drywell equipment drain tank is maintained full of water by an elevation difference between the tank and the valves.

f. Penetrations 10, 11, 12, 44, 228D and 241. Lines associated with these penetrations that pass through the secondary containment boundary and take credit for water seals are provided with loop seals inside secondary containment, which eliminates the possibility of bypass leakage.

g. Penetration 14. The minimum piping height inside primary containment of the RWCU supply line that branches off the recirculation loop is at El 267 ft. The primary containment penetration is at El. 297 ft and the RPV penetration is at El. 280 ft. This elevation difference ensures that a water seal is maintained in the line from the RPV to the containment isolation valves. The RWCU supply branch line that connects to the bottom of the vessel is normally full of water, and the water will be maintained in this line because it connects directly to, and below, the vessel.

h. Penetrations 37A-D and 28A-D. The CRD insert and withdraw lines are normally full of water. A water seal will be maintained in these lines after a LOCA due to

Insert (B)

the elevation difference between the containment penetrations (El. 265 ft) and the connections to the control rod drives (El. 215 ft).

6.2.3.2.3.2 Feedwater Fill System

The feedwater fill system prevents the release of fission products through the feedwater containment isolation valves after a LOCA by providing a water seal downstream of the valves.

6.2.3.2.3.2.1 Safety Design Bases

The feedwater fill system is designed with sufficient capacity and capability to prevent leakage through the feedwater lines under the conditions associated with the entire spectrum of LOCAs except for a feedwater line break inside containment.

The feedwater fill system conforms to seismic Category I requirements. Quality group classifications are shown in Table 3.2-1, Item XI.A. The system meets the intent of Regulatory Guide 1.96, where applicable.

The feedwater fill system is capable of performing its safety function considering the effects resulting from a LOCA, including missiles that may result from equipment failures, dynamic effects associated with pipe whip and jet forces, and normal operating and accident-caused local environmental conditions consistent with the design basis event. Furthermore, any portion of the feedwater fill system that is quality Group A and is located outside the primary containment structure is protected from missiles, pipe whip, and jet force effects originating outside the containment so that containment integrity is maintained.

The feedwater fill system is capable of performing its safety function following a LOCA and an assumed single active failure.

The feedwater fill system is designed so that effects resulting from a single active component failure do not affect the integrity or operability of the feedwater lines or the feedwater containment isolation valves.

The feedwater fill system is capable of performing its safety function following a loss of all offsite power coincident with a postulated design basis LOCA.

The feedwater fill system is designed to prevent leakage from the feedwater lines consistent with maintaining containment integrity for up to 30 days.

The feedwater fill system is manually actuated and is not required to be actuated sooner than 30 minutes after a LOCA.

Insert (B)

i. Penetrations 23, 24, 53, 54, 55, and 56. Both the drywell chilled water system and the reactor enclosure cooling water system have a system vent in the refueling area. A 30-day water seal is provided between the system vent and the containment penetrations.

These systems are normally full of water and are inspected during normal operation, when system pressure is ~~much~~ greater than post-LOCA containment pressure, to assure that leakage is minimized. These systems have an interconnecting line between them which decreases the possibility of depleting their water seals. The systems each contain a head tank, with a seismically qualified water retaining boundary, which is located in the refueling area. These tanks increase the system water inventory and maintain the head of water in the system piping greater than the maximum containment pressure following a LOCA.

rupture of piping

d. Environmental design Section 3.11

Debris transported to the suppression pool by the emergency core cooling water is prevented from entering the ECCS suction lines by suction strainers. The suction strainers are described in Section 6.2.2.

Ensurance of the operability of valves and valve operators in the containment atmosphere under normal plant operating conditions and postulated accident conditions is discussed in Section 3.9.3.

connected to the RCPB which are

Provisions for detecting leakage from systems provided with remote manual isolation valves are discussed in Section 5.2.5.

The design provisions for testing the operability of the isolation valves and the leakage rate of the containment isolation barriers are discussed in Section 6.2.6.

A leakage control system is provided for the main steam isolation valves, and is discussed in Section 6.7. A seismic Category I fill system provides a water seal for the feedwater lines, as discussed in Section 6.2.3.2.3.

Containment isolation valve closure times are selected to ensure rapid isolation of the containment following postulated accidents. The isolation valves in lines that provide an open path from the containment to the environs have closure times that minimize the release of containment atmosphere to the environs to below 10CFR100 guideline values, mitigate the offsite radiological consequences, and ensure that ECCS effectiveness is not degraded. These valve closure times are identified with a double asterisk in Table 6.2-17. The isolation valves for lines in which high-energy line breaks can occur have closure times that minimize the resultant pressure and temperature transients as well as the radiological consequences. These valve closure times are identified with a single asterisk in Table 6.2-17.

All of the isolation valve closure times listed in Table 6.2-17 are the actual closure times that the isolation valves were purchased with, which in all cases are equal to or lower than the closure times necessary to meet the aforestated design requirements. Those closure times which are required to be met to satisfy isolation valve closure time design requirements are identified by a single or double asterisk in Table 6.2-17.

The essential/nonessential classification of containment isolation valves, as listed in Table 6.2-17, was based on the following: those systems identified as essential are regarded as indispensable or are backup systems in the event of an accident; nonessential systems have been judged to not be required after an

In addition, the piping is considered an extension of the containment boundary and, as such, is designed to the same quality standards as the primary containment. The drywell radiation sampling isolation valves have ganged controls for reopening. Inboard sample and return isolation valves SV-190A and SV-190C are ganged on HS-190A. Outboard sample and return isolation valves SV-190B and SV-190D are ganged on HS-190B.

6.2.4.3.1.3.2.9 Primary Containment Instrument Gas

The influent lines are provided with a normally-open power-operated globe valve outside the containment and a check valve inside the containment. Motor-operated valves are used on the influent lines that contain the ADS gas supply. These are essential lines that provide a long-term backup to the ADS accumulators inside containment. The valves on these essential lines are remote manually operated and automatically isolate only when flow out of containment through these lines would be possible (i.e., low differential pressure between the containment and the instrument gas line). The remaining influent lines are non-essential lines that use air-operated valves that are automatically closed on receipt of a containment isolation signal. The effluent lines are provided with normally-open air-operated globe valves inside and outside the containment that close automatically on receipt of a containment isolation signal.

Insert C

6.2.4.3.1.3.3 Conclusion on Criterion 56

In order to ensure protection against the consequences of accidents involving release of significant amounts of radioactive materials, pipes that penetrate the containment have been demonstrated to provide isolation capabilities on a case-by-case basis in accordance with Criterion 56.

In addition to meeting isolation requirements, the pressure retaining components of these systems are designed to the same quality standards ~~as the~~ equivalent to those used for the primary containment.

6.2.4.3.1.4 Evaluation Against Criterion 57

Criterion 57 describes criteria for closed system isolation valves.

Influent and effluent lines of this group are isolated by automatic or remote manual isolation valves located as close as possible to the containment boundary.

Insert (C)

6.2.4.3.1.3.2.10 Reactor Enclosure Cooling Water

Each [redacted] influent and effluent line is [redacted] provided with two gate valves located outside containment. The inboard valves are motor-operated. The outboard valves on the emergency service water intertie lines are ~~manually~~ locked closed. The outboard valves on the RECW lines (HV-108 and HV-111) are motor-operated and their controls are ganged [redacted] on HS-108. Each [redacted] of the motor-operated valves can be remote manually closed from the control room. Remote manual actuation is preferable because system operation is necessary to prevent recirculation pump seal failure, and subsequent system leakage.

Insert (E) - cont'd

6.2.4.3.1.3.2.11 Drywell Chilled Water

Each influent and effluent line is provided with two motor-operated gate valves located outside containment. The controls for these valves are ganged as indicated:

- Loop A inboard influent and effluent isolation valves HV-128 and HV-129 are ganged on HS-128.
- Loop B inboard influent and effluent isolation valves HV-122 and HV-123 are ganged on HS-122.
- Loop A outboard influent and effluent isolation valves HV-120A, 121A, 124A and 125A are ganged on HSS-121A.
- Loop B outboard influent and effluent isolation valves HV-120B, 121B, 124B and 125B are ganged on HSS-121B.

The inboard valves are provided with automatic isolation signals as indicated in Table 6.2-17. The outboard isolation valves are aligned to either the drywell chilled water system or the reactor enclosure cooling water system. By the end of the first refueling outage, the controls for the outboard valves will be modified to permit remote manual closure from the control room.

6.2.4.3.1.4.1 CRD Lines

The CRD system has multiple lines, the insert and withdraw lines, that penetrate primary containment.

The classification of these lines is Quality Group B, and they are designed in accordance with ASME Section III, Class 2. The basis on which the CRD insert and withdraw lines are designed is commensurate with the safety importance of maintaining the pressure integrity of these lines.

The CRD insert and withdrawal lines are not provided with automatic containment isolation valves in order to maximize the reliability of the scram function. A ball check valve located in the CRD flange housing automatically seals the insert line in the event of a line break. The insert and withdrawal lines terminate in hydraulic control units (HCUs) which contain multiple valves (manual, solenoid, air-operated, and check valves) to control CRD movement, minimize leakage, and provide isolation. All automatic valves in the HCUs are normally closed and are open only during rod movement. Because the scram valves in the HCU are normally open after a scram, the scram discharge volume is provided with redundant automatic vent and drain valves.

6.2.4.3.1.4.2 Reactor Enclosure Cooling Water and Drywell Chilled Water Supplies and Returns

The influent and effluent lines are provided with normally-open motor-operated gate valves that can be remote manually isolated from the control room. The drywell chilled water isolation valves have controls for ganged reopening. Loop A influent and effluent isolation valves HV-128 and HV-129 are ganged on HS-128. Loop B influent and effluent isolation valves HV-122 and HV-123 are ganged on HS-122.

The containment isolation system conforms to Regulatory Guide 1.141 except as discussed below:

American National Standards Institute (ANSI) N271-1976

Section 3.5

Criteria For Closed Systems Inside Containment. If a closed system inside containment is used as one of the two containment isolation barriers, it shall meet the criteria that follow...

(2) Be missile, pipe whip, and jet force protected from a LOCA or from a missile, pipe whip, or jet force that results in a requirement for containment isolation.

(3) Meet Safety Class 2 design requirements.

(7) Meet seismic Category I design requirements.

Limerick Design:

Closed systems such as reactor enclosure cooling water and drywell chilled water are not designed strictly in accordance with items (2), (3), and (7) of Section 3.5 of ANSI N271. The design criteria used for these systems are listed in Table 3.2-1.

Section 3.6.4 Single Valve and Closed System Both Outside Containment...

The single valve and piping between the containment and the valve shall be enclosed in a protective leaktight or controlled leakage housing to prevent leakage to the atmosphere.

Limerick Design:

For systems that fall into this category except for the ECCS pump suction lines, the single valve outside primary containment is not enclosed in a protective leaktight or controlled leakage housing. Moderate energy lines that fall into this category do not connect to the reactor coolant pressure boundary and are not postulated to break concurrent with a LOCA. Therefore, neither reactor coolant nor post-LOCA containment atmosphere are released. However, any leakage is contained within the secondary containment and is diluted and filtered prior to release. The ECCS pump suction isolation valves are enclosed in pump rooms adjacent to the containment that have provisions for the environmental control of any fluid leakage.

Section 3.6.5 Two Valves Outside Containment...

Section 5.3.2 - Leakage Rate Testing

Provisions and Methods. Provisions shall be made for leakage rate testing of containment isolation valves.

Limerick Design:

Individual leakage rate tests are performed for containment isolation valves as indicated in Table 6.2-25.

Note:

Regulatory Guide Paragraphs C.4 and C.6 refer to N271 Sections 4.4.8 (closed system design) and 4.14 (piping between isolation barriers) and adds the requirements of Section 3.5 to these sections. As discussed above, there is partial conformance with Section 3.5.

6.2.4.3.2 Failure Mode and Effects Analyses

A single failure can be defined as a failure of some component in any safety system that results in a loss or degradation of the system's capability to perform its safety function. Active components are defined as components that must perform a mechanical motion during the course of accomplishing a system safety function. Appendix A to 10 CFR Part 50 requires that electrical systems be designed against passive single failures as well as active single failures. Section 3.1 describes the implementation of these requirements as well as General Design Criteria 17, 21, 35, 41, 44, 54, 55, and 56.

In single failure analysis of electrical systems, no distinction is made between mechanically active or passive components; all fluid system components, such as valves, are considered electrically active whether or not mechanical action is required.

6.2.4.4 Tests and Inspections

The containment isolation system undergoes periodic testing during reactor operation. The functional capabilities of power operated isolation valves are remotely tested manually from the main control room. By observing position indicators and changes in the affected system operation, the closing ability of a particular isolation valve is demonstrated.

No change

A discussion of testing and inspection pertaining to isolation valves is provided in Section 6.2.1.6 and in Chapter 16. Table 6.2-17 lists all isolation valves.

Instruments are be periodically tested and inspected. Test and/or calibration points are supplied with each instrument.

Excess flow check valves (EFCVs) are periodically tested by opening a test drain valve downstream of the EFCV and verifying proper operation. As these valves are outside the containment and accessible, periodic visual inspection is performed in addition to the operational check.

Leak-rate testing for the containment isolation system is discussed in Section 6.2.6.

6.2.5 COMBUSTIBLE GAS CONTROL IN CONTAINMENT

Following a postulated LOCA, hydrogen gas may be generated within the primary containment as a result of the following processes:

a. Metal-water reaction involving the Zircaloy fuel cladding and the reactor coolant

b. Radiolytic decomposition of water in the reactor vessel and the suppression pool (oxygen also evolves in this process)

TABLE 6.2-15

EVALUATION OF POTENTIAL SECONDARY CONTAINMENT BYPASS LEAKAGE PATHS

CONTAINMENT PENETRATION	SYSTEM	TERMINATION REGION[1]	BYPASS LEAKAGE BARRIERS[2]	POTENTIAL BYPASS PATH
1	Equipment access door	ISC	Double O-Ring	No
2	Equipment access door and personnel lock	ISC	Double O-Ring	No
3A	Main steam (MS) line D flow instrumentation	ISC	-	No
3B	Inst gas supply	OSC	1,4	No
3C	HPCI steam flow inst	ISC	-	No
3D	MS line A flow inst	ISC	-	No
3D	Instrument gas supply	OSC	1,4	No
4	Head access manhole	ISC	Double O-Ring	No
5	Spare	-	-	-
6	CRD removal hatch	ISC	Double O-Ring	No
7A-D	Primary steam	OSC	1,5	No
8	Primary steam line drain	OSC	1,4	No
9A&B	Feedwater	OSC	1,3	No[3]
10	Steam to RCIC turbine	OSC	1,3,6	No
11	Steam to HPCI turbine	OSC	1,3,6	No
12	RHR shutdown cooling supply	OSC	1,3	No
13A&B	RHR shutdown return	OSC	1,3	No
14	RWCU supply	OSC	1,3	No
15	Spare	-	-	-
16A&B	Core spray pump discharge	OSC	1,3	No
17	RPV head spray	OSC	1,3	No
18	Spare	-	-	-
19	Spare	-	-	-
20A	RPV level inst	ISC	-	No
20A	LPCI ΔP inst	ISC	-	No
20B	LPCI ΔP inst	ISC	-	No
20B	RPV level inst	ISC	-	No
21	Spare	-	-	-
22	Drywell pressure inst	ISC	-	No
23	Closed cooling water supply	OSC	1,2,3	No
24	Closed cooling water return	OSC	1,2,3	No
25	Drywell purge supply	OSC	1,4	No
26	Drywell purge exhaust	ISC		No
27A	Instrument gas supply	OSC	1,4	No
27B	HPCI flow inst	ISC	-	No
28A	Recirc loop sample	ISC	-	No
28A	Drywell H_2/O_2	ISC	-	No
28B	LPCI ΔP inst	ISC	-	No
28B	Drywell air sample	ISC	-	No
29A	RPV flange leakage inst	ISC	-	No
29B	Core spray ΔP inst	ISC	-	No
30A	MS line D flow inst	ISC	-	No
30B	Drywell pressure inst	ISC	-	No
30B	MS line C flow inst	ISC	-	No

TABLE 6.2-15 (Cont'd)

CONTAINMENT PENETRATION	SYSTEM	TERMINATION REGION(1)	BYPASS LEAKAGE BARRIERS(2)	POTENTIAL BYPASS PATH
31A&B	Jet pump flow inst	ISC	-	No
32A&B	Jet pump flow inst	ISC	-	No
33A	Pressure above core plate inst	ISC	-	No
33A	Pressure below core plate inst	ISC	-	No
33B	RCIC steam flow inst	ISC	-	No
34A	MS line C flow inst	ISC	-	No
34B	Recirc flow inst	ISC	-	No
35A	Inst gas to TIP indexing mechanism	OSC	1,4	NO
35C-G	TIP drives	ISC	-	No
36	Spare	-	-	-
37A-D	CRD insert	OSC	1,3	No
38A-D	CRD withdraw	OSC	1,3	No
39A&B	Drywell spray	OSC	1,3	No
40A,B&C	Jet pump flow inst	ISC	-	No
40D	Pressure below core plate inst	ISC	-	No
40E	Drywell pressure inst	ISC	-	No
40F	RCIC steam flow inst	ISC	-	No
40F	Inst gas suction	OSC	1,4	No
40G	ILRT data acquis system	OSC	1,6	No
40H	Instrument gas supply	OSC	1,4	No
40H	Recirc pump cooler flow inst	ISC	-	No
41	LPCI ΔP inst	ISC	-	No
41	RWCU flow inst	ISC	-	No
42	Standby liquid control	ISC	-	No
43A	Recirc loop A ΔP inst	ISC	-	No
43A	Recirc pump seal pressure inst	ISC	-	No
43B	Main steam sample	ISC	-	No
44	CRD/RWCU return	OSC	1,3	No
45A-D	LPCI	OSC	1,3	No
46	Spare	-	-	-
47	RWCU flow inst	ISC	-	No
48A	RPV level inst	ISC	-	No
48A	Core spray ΔP inst	ISC	-	No
48B	RPV level inst	ISC	-	No
49A&B	MS line A&B flow inst	ISC	-	No
50A	Drywell pressure inst	ISC	-	No
50A	Recirc flow inst	ISC	-	No
50B	Recirc pump seal pressure inst	ISC	-	No
50B	Recirc pump cooler flow inst	ISC	-	No
51A	Recirc line flow inst	ISC	-	No
51B	Jet pump flow inst	ISC	-	No
52A	MS line B flow inst	ISC	-	No
52B	Recirc line flow inst	ISC	-	No
53	Drywell chilled water supply	OSC	1,2,3	NO
54	Drywell chilled water return	OSC	1,2,3	No
55	Drywell chilled water supply	OSC	1,2,3	No
56	Drywell chilled water return	OSC	1,2,3	No

No change

TABLE 6.2-15 (Cont'd)

CONTAINMENT PENETRATION	SYSTEM	TERMINATION REGION(1)	BYPASS LEAKAGE BARRIERS(2)	POTENTIAL BYPASS PATH

(1) The termination regions are: ISC - Inside Secondary Containment
OSC - Ountside Secondary Containment

(2) The bypass leakage barriers are defined as follows (see Section 6.2.3.3.3):

1. Redundant primary containment isolation valves
2. Closed piping system inside containment
3. A water seal maintained for 30 days following a LOCA
4. The line beyond the outboard primary containment isolation valve is vented to secondary containment by use of a vent line located between two block valves and the secondary containment.
5. A leakage collection system is provided
6. The line contains a temporary spool piece that is removed during normal operation and replaced by blind flanges so that any leakage through the flange is into secondary containment.

(3) The feedwater fill system will provide a water seal in the feedwater lines for all line breaks other than a feedwater line break inside containment.

TABLE 6.2-17 (Cont'd)

CONTAINMENT PENETRATION NUMBER	LINE ISOLATED	FLUID	LINE SIZE(in.)	NRC GENERAL DESIGN CRITERION	ESF SYSTEM	ESSENTIAL SYSTEM	VALVE NUMBER	VALVE TYPE(1)	VALVE LOCATION	VALVE ARRANGEMENT(2)	TYPE C TEST	LENGTH OF PIPE FROM CONT. TO OUTSIDE VALVES	PRIMARY METHOD OF ACTUATION(3)	SECONDARY METHOD OF ACTUATION	NORMAL VALVE POSITION(4)	SHUTDOWN VALVE POSITION	POST-ACCIDENT POSITION	POWER FAILURE VALVE POSITION	ISOLATION SIGNAL(5)	DIVERSE ISOLATION SIGNAL(12)	VALVE CLOSURE TIME(6)	POWER SOURCE(7)	REMARKS
X-20B	Instrumentation-RPV level	Water	1	55	-	-	F045C	XFC	Outside	(37)	No	2'-2"	Flow	-	O	O	O	-	-	-	-	-	
X-20B	Instrumentation-LPCI WP	Water	1	55	-	-	102C	XFC	Outside	(40)	No	13"	Flow	-	O	O	O	-	-	-	-	-	
X-21	Service air	Gas	3	56	No No	No No	1140 1139	GT GT	Inside Outside	(8)	Yes	- 0"	Manual Manual	- -	C C	C C	C C	- -	- -	- -	- -	- -	
X-22	Instrumentation - drywell pressure	Gas	1	56	-	-	147C	GB	Outside	(41)	No	8"	AC motor	Manual	O	O	O	AS IS	RM	-	30 sec	C	
X-23	Recirc pump cooling water supply	Water	4	56	No No No	No No No	106 108 109D	GT GT GT	Outside Outside Outside	(13)	Yes	0" 3'-0" 5'-2"	AC motor AC motor Manual	Manual Manual —	O O C	O O C	C C C	AS IS AS IS —	RM RM —	No No —	Standard Standard —	C	19 (30)
X-24	Recirc pump cooling water return	Water	4	56	No No No	No	107 111 109	GT GT GT	Outside Outside Outside	(13)	Yes	0" 3'-8" 5'-0"	AC motor AC motor Manual	Manual Manual —	O O C	O O C	C C C	AS IS AS IS —	RM RM —	No No —	Standard Standard —	C	19 (30)
X-25	Drywell purge supply	Gas	24	56	No Yes No Yes Yes No	No Yes No Yes Yes No	135 121 123 131 163 109	BF BF BF BF BF BF	Outside Outside Outside Outside Outside Outside	(5)	Yes	16'-7" 3'-11" 3'-4" 60'-7" 3'-8" 42'-2"	AC motor Comp air Comp air Comp air AC motor AC motor	Manual Spring Spring Spring Manual Manual	C C C C C C	O C O C C C	C C C C O C	AS IS C C C AS IS AS IS	B,H,R B,H,R B,H,R B,H,R B,H,R B,H,R	Yes NA Yes NA NA Yes	5** sec 5** sec 5** sec 5 sec 5 sec 5** sec	B A A A D B	
X-26	Drywell purge exhaust	Gas	24	56	No Yes No No Yes No	No Yes No No Yes No	115 145 111 114 161 117	BF GT GB BF BF GB	Outside Outside Outside Outside Outside Outside	(27)	Yes	53'-7" 66'0" 6'-6" 49'-7" 4'-5" 60'5"	AC motor AC coil AC motor Comp air AC motor Comp air	Manual - Manual Spring Manual Spring	C O C C C C	O O C O C C	C O C C O C	AS IS C AS IS C AS IS C	B,H,R B,H,R B,H,R B,H,R B,H,R B,H,R	Yes NA Yes Yes NA Yes	5** sec 2 sec 30** sec 5** sec 5 sec 5** sec	A D B B C A	
X-27A	Instrument gas supply	Gas	1	56	No No	Yes Yes	112B 151A	CK GB	Inside Outside	(48)	Yes	- 7"	Flow AC motor	- Manual	O O	O O	C C	- AS IS	- H	- NA	- 30 sec	- C	
X-27B	Instrumentation-HPCI	Steam	1	55	-	-	F024B	XFC	Outside	(40)	No	17"	Flow	-	O	O	O	-	-	-	-	-	
X-27B	Instrumentation-HPCI Flow	Steam	1	55	-	-	F024D	XFC	Outside	(40)	No	17"	Flow	-	O	O	O	-	-	-	-	-	

T1001550-01V

CONTAINMENT PENETRATION NUMBER	LINE ISOLATED	FLUID	LINE SIZE (in.)	NRC GENERAL DESIGN CRITERION	ESF SYSTEM	ESSENTIAL SYSTEM	VALVE NUMBER	VALVE TYPE(1)	VALVE LOCATION	VALVE ARRANGEMENT(2)	TYPE C TEST	LENGTH OF PIPE FROM CONT. TO OUTSIDE VALVES	PRIMARY METHOD OF ACTUATION(3)	SECONDARY METHOD OF ACTUATION	NORMAL VALVE POSITION(4)	SHUTDOWN VALVE POSITION	POST-ACCIDENT POSITION	POWER FAILURE VALVE POSITION	ISOLATION SIGNAL(5)	DIVERSE ISOLATION SIGNAL(12)	VALVE CLOSURE TIME(6)	POWER SOURCE(7)	REMARKS
X-50A	Instrumentation - drywell pressure	Gas	1	56	-	-	147B	GB	Outside	(41)	No	7"	AC motor	Manual	O	O	O	AS IS	RM	-	30 sec	B	
X-50A	Instrumentation - recirc flow	Water	1	55	- - - -	- - - -	F011A F011B F012A F012B	XFC XFC XFC XFC	Outside Outside Outside Outside	(40)	No	2'-5" 18" 2'-9" 18"	Flow Flow Flow Flow	- - - -	O O O O	O O O O	O O O O	- - - -	- - - -	- - - -	- - - -	- - - -	
X-50B	Instrumentation - recirc pump seal pressure	Water	1	55	-	-	F004A	XFC	Outside	(45)	No	4'-11"	Flow	-	O	O	O	-	-	-	-	-	
X-50B	Instrumentation - recirc pump cooler flow	Water	1	56	- -	- -	156A 157A	XFC XFC	Outside Outside	(44)	No	22" 20"	Flow Flow	- -	O O	O O	O O	- -	-	- -	- -	- -	
X-51A	Instrumentation - recirc line flow	Water	1	55	- -	- -	F009A,B F010A,B	XFC XFC	Outside Outside	(40)	No	18" 18"	Flow Flow	- -	O O	O O	O O	- -	- -	- -	-	- -	
X-51B	Instrumentation - jet pump flow	Water	1	55	- - -	- - -	F059T F051C F053C	XFC XFC XFC	Outside Outside Outside	(37)	No	12" 3'-4" 12"	Flow Flow Flow	- - -	O O O	O O O	O O O	- - -	- - -	- - -	- -	- - -	
X-52A	Instrumentation - main steam line B flow	Water	1	55	- -	- -	F070B F073B	XFC XFC	Outside Outside	(40)	No	12" 12"	Flow Flow	- -	O O	O O	O O		-	- -	- -	- -	
X-52B	Instrumentation - recirc line flow	Water	1	55	- - - -	- - - -	F011C F011D F012C F012D	XFC XFC XFC XFC	Outside Outside Outside Outside	(40)	No	2'-9" 18" 2'-3" 18"	Flow Flow Flow Flow	- - - -	O O O O	O O O O	O O O O	- - - -	- - - -	- - - -	- - - -	- - - -	
X-53	Chilled water supply "A"	Water	8	56 57b	No	No	17B 172A 175A	GT GT GT	Outside Outside Outside	(13)	Yes	0" 68'-10" 68'-10"	AC motor AC motor AC motor	Manual Manual Manual	O O O	O O O	C C C	AS IS AS IS AS IS	RM RM	~~No~~ Yes No No	Standard Standard Standard	[illegible]	|

C, H, R

Rev. 16, 01/83

T1001550-01Y

C,H,R

TABLE 6.2-17 (Cont'd)

CONTAINMENT PENETRATION NUMBER	LINE ISOLATED	FLUID	LINE SIZE(in.)	NRC GENERAL DESIGN CRITERION	ESF SYSTEM	ESSENTIAL SYSTEM	VALVE NUMBER	VALVE TYPE(1)	VALVE LOCATION	VALVE ARRANGEMENT(2)	TYPE C TEST	LENGTH OF PIPE FROM CONT. TO OUTSIDE VALVES	PRIMARY METHOD OF ACTUATION(3)	SECONDARY METHOD OF ACTUATION	NORMAL VALVE POSITION(4)	SHUTDOWN VALVE POSITION	POST-ACCIDENT POSITION	POWER FAILURE VALVE POSITION	ISOLATION SIGNAL(5)	DIVERSE ISOLATION SIGNAL(12)	VALVE CLOSURE TIME(6)	POWER SOURCE(7)	REMARKS
X-54	Chilled water return "A"	Water	8	56	No No No	No No No	129 131A 129A	GT GT GT	Outside Outside Outside	(13)	Yes	0" 67'-0" 67'-0"	AC motor AC motor AC motor	Manual Manual Manual	O C C	O C C	C C C	AS IS AS IS AS IS	[illegible] B,M B,M	Yes No No	Standard Standard Standard	B	[redacted]
X-55	Chilled water supply "B"	Water	8	56	No No No	No No No	122 120B 125B	GT GT GT	Outside Outside Outside	(13)	Yes	0" 46'-10" 46'-10"	AC motor AC motor AC motor	Manual Manual Manual	C O C	C O C	C C C	AS IS AS IS AS IS	[illegible] B,M B,M	Yes No No	Standard Standard Standard	B	[redacted]
X-56	Chilled water return "B"	Water	8	56	No No No	No No No	123 126B 124B	GT GT GT	Outside Outside Outside	(13)	Yes	0" 97'-2" 97'-2"	AC motor AC motor AC motor	Manual Manual Manual	C O C	C O C	C C C	AS IS AS IS AS IS	[illegible] B,M B,M	Yes No No	Standard Standard Standard	B	[redacted]
X-57	Instrumentation - RWCU flow	Water	1	55	-	-	102	CK	Outside	(40)	No	11"	Flow	-	O	O	O	-	-	-	-	-	
X-58A	Instrumentation - recirc loop B WP	Water	1	55	-	-	F040B	CK	Outside	(40)	No	14"	Flow	-	O	O	O	-	-	-	-	-	
X-61	Recirc pump seal purge	Water	1	56	No -	No -	1004A,B 103A,B	CK XFC	Inside Outside	(45)	Yes	- 20"	Flow Flow	- -	O O	C O	C O	- -	- -	- -	- -	- -	
X-62	H_2/O_2 sample return; drywell purge makeup	Gas	1	56	Yes No Yes	Yes No Yes	150 116 159	GT GB GT	Outside Outside Outside	(12)	Yes	3'-3" 6'-6" 15'-9"	AC coil AC motor AC coil	- Manual -	O C O	O C O	O C O	C AS IS C	B,H,R B,H,R B,H,R	NA Yes NA	2 sec 30** sec 2 sec	B B D	
X-63	Instrumentation-recirc loop WP	Water	1	55	-	-	F040D	XFC	Outside	(40)	No	3'-4"	Flow	-	O	O	O	-	-	-	-	-	
X-63	Instrumentation-recirc pump seal pressure	Water	1	55	- -	- -	F004B F003B	XFC XFC	Outside Outside	(45)	No	12" 17"	Flow Flow	- -	O O	O O	O O	- -	- -	- -	- -	- -	
X-65A,B	Instrumentation-RPV pressure	Water	1	55	- -	- -	F043B F049A	XFC XFC	Outside Outside	(37)	No	14" 14"	Flow Flow	- -	O O	O O	O O	- -	- -	- -	- -	- -	

TABLE 6.2-25 (Cont'd)

PENET. No.	SYSTEM	DRAW-ING(17)	TEST TYPE	INBOARD ISOLATION BARRIER DESCRIPTION/ VALVE NUMBER	OUTBOARD ISOLATION BARRIER INSTRUMENT/ VALVE NUMBER	NOTE
21	Service air	M-15	C	1140	1139	–
22	Instrumentation - drywell pressure	M-42	A	-	MO-147C	11
23	Closed cooling water supply	M-13	C	MO-106 ~~Closed system~~	MO-108 ~~[illegible]~~ 1090	12,22
24	Closed cooling water return	M-13	C	MO-107 ~~Closed system~~	MO-111 ~~[illegible]~~ 1089	12,22
25	Drywell purge supply	M-57	C	AO-121 MO-163 AO-123	MO-109 Closed system AO-131 MO-135	3, 12
26	Drywell purge exhaust	M-57	C	MO-161 MO-111 AO-114	SV-145 Closed system AO-117 MO-115	3, 5(MO-111 only), 12
27A	Instrument gas supply	-	C	CK-1128	MO-151A	-
27B	Instrumentation - HPCI flow	M-55	A	-	XFC-F024B	1
27B	Instrumentation - HPCI flow	M-55	A	-	XFC-F024D	1
28A	Recirc loop sample	M-43	C	AO-F019	AO-F020	-
28A	Drywell H_2/O_2 sample	M-57	C	SV-134	SV-144	12
28A	Drywell H_2/O_2 sample	M-57	C	SV-132	SV-142	12
28B	Drywell H_2/O_2 sample	M-57	C	SV-133	SV-143 SV-195	12
28B	Spare	-	A	-	-	-
29A	Instrumentation - RPV flange leakage	M-41	A	-	XFC-F009	1
29B	Instrumentation - CS ΔP	M-52	A	-	XFC-F018A	1
30A	Instrumentation - main steam line D flow	M-41	A	-	XFC-F071D XFC-F072D	1
30B	Instrumentation - drywell pressure	M-42	A	-	MO-147A	11

TABLE 6.2-25 (Cont'd)

PENET. No.	SYSTEM	DRAW-ING(17)	TEST TYPE	INBOARD ISOLATION BARRIER DESCRIPTION/ VALVE NUMBER	OUTBOARD ISOLATION BARRIER INSTRUMENT/ VALVE NUMBER	NOTE
46	Spare	-	A	-	-	-
47	Instrumentation - RWCU flow	M-44	A	-	XFC-102D	1
48A	Instrumentation - RPV level	M-42	A	-	XFC-F065B XFC-F047B	1
48A	Instrumentation - CS ΔP	M-52	A	-	XFC-F018B	1
48B	Instrumentation - RPV level	M-42	A	-	XFC-F065A XFC-F047A	1
49A,B	Instrumentation - main steam line A & B flow	M-41	A	-	XFC-F071A,B XFC-F072A,B	1
50A	Instrumentation - drywell pressure	M-42	A	-	MO-147B	11
50A	Instrumentation - recirc flow	M-43	A	-	XFC-F011A,B XFC-F012A,B	1
50B	Instrumentation - recirc pump seal pressure	M-43	A	-	XFC-F004A	1
50B	Instrumentation - recirc pump cooler flow	M-87	A	-	XFC-156A XFC-157A	[illegible]
51A	Instrumentation - recirc line flow	M-43	A	-	XFC-F009A,B XFC-F010A,B	1
51B	Instrumentation - jet pump flow	M-42	A	-	XFC-F059T XFC-F051C XFC-F053C	1
52A	Instrumentation - main steam line B flow	M-41	A	-	XFC-F070B XFC-F073B	1
52B	Instrumentation - recirc line flow	M-43	A	-	XFC-F011C,D XFC-F012C,D	1
53	Drywell chilled water supply	M-87	C	MO-128 ~~Closed system~~	MO-120A ~~[illegible]~~ MO-125A	12,22
54	Drywell chilled water return	M-87	C	MO-129 ~~Closed system~~	MO-121A ~~[illegible]~~ MO-124A	12,22
55	Drywell chilled water supply	M-87	C	MO-122 ~~Closed system~~	~~[illegible]~~ MO-120B MO-125B	12,22

TABLE 6.2-25 (Cont'd)

PENET. No.	SYSTEM	DRAW-INGS(17)	TEST TYPE	INBOARD ISOLATION BARRIER DESCRIPTION/ VALVE NUMBER	OUTBOARD ISOLATION BARRIER INSTRUMENT/ VALVE NUMBER	NOTE
56	Drywell chilled water return	M-87	C	MO-123 ~~Closed system~~	~~[illegible]~~ MO-121B MO-124B	12,22
57	Instrumentation - RWCU flow	M-44	A	-	XFC-102C	1
58A	Instrumentation - recirc loop B ΔP	M-43	A	-	XFC-F040B	1
58B	Spare	-	A	-	-	-
59A,B	Spare	-	A	-	-	-
60	Spare	-	A	-	-	-
61	Recirc pump seal purge	M-43	C	CK-1004A, B	XFC-103A,B	1, 19
62	H_2/O_2 sample return	M-57	C	SV-150	MO-116 SV-159	12
63	Instrumentation - recirc loop ΔP; recirc pump Seal pressure	M-43	A	-	XFC-F003B XFC-F004B XFC-F040D	1
64	Spare	-	A	-	-	-
65A,B	Instrumentation - RPV pressure	M-42	A	-	XFC-F043B XFC-F049A	1
66A	Instrumentation - RPV level	M-42	A	-	XFC-F045D	1
66A	Instrumentation - LPCI ΔP	M-51	A	-	XFC-102D XFC-103D	1
66B	Instrumentation - RPV level	M-42	A	-	XFC-F045A	1
66B	Instrumentation - LPCI ΔP	M-51	A	-	XFC-102A XFC-103C	1
67A,B	Instrumentation - RPV level; RPV pressure	M-42	A	-	XFC-F041 XFC-F043A XFC-F049B	1
100 A-D	Neutron monitoring system	M-60	B	Canister	-	8
101 A-D	Recirc pump power	M-60	B	Canister	-	8

TABLE 6.2-25 (Cont'd)

2. Penetration is sealed by a blind flange or door with double O-ring seals. These seals are leakage rate tested by pressurizing between the O-rings.

3. Inboard butterfly valve installed such that tested in the reverse direction is conservative.

4. Inboard gate valve tested in the reverse direction.

5. Inboard globe valve installed such that testing in the reverse direction is conservative.

6. The MSIVs are tested by pressurizing between the valves. Testing of the inboard valve in the reverse direction tends to unseat the valve and is therefore conservative. The valves are Type C tested at a test pressure of 25 psig.

7. Gate valve tested in the reverse direction.

8. Electrical penetrations are tested by pressurizing between the sea

9. The isolation provisions for this penetration consist of two isolation valves and a closed system outside containment. A single active failure can be accommodated. The closed system is missile protected, seismic Category I, quality group B, and is designed to the temperature and pressure conditions that the system will encounter post-LOCA. Valves will remain water covered during Type A testing. System leakage will be minimized in accordance with NUREG-0737, Item III.D.1.1. Any leakage out of the closed system will be into the reactor enclosure, thus faciliting collection and treatment.

10. The isolation provisions for this line consist of one isolation valve outside containment and a closed system outside containment. A single active failure can be accommodated. The closed system is missile protected, seismic Category I, quality group B and designed to the temperature and pressure conditions that the system will encounter post-LOCA.

System leakage will be minimized in accordance with NUREG-0737, Item III.D.1.1. Any leakage out of the closed system will be into the reactor enclosure, thus facilitating collection and treatment.

11. The valve does not receive an isolation signal but remains open to measure containment conditions post-LOCA. Leaktightness of the penetration is verified during the Type A test.

12. All isolation barriers are located outside containment.

In all cases the valve seating force is greater than the unseating force which post-LOCA containment pressures would exert.

TABLE 6.2-25 (Cont'd)

13. Isolation provisions for the CRD insert and withdrawal lines are described in Section 6.2.4.3.1.4.1. The scram discharge volume vent and drain valves are Type C tested.

14. The isolation provisions for this line consist of a suppression pool water seal, at least one isolation valve outside containment, and a closed system outside containment. The isolation valve is not exposed to the primary containment atmosphere because the line terminates below the minimum water level of the suppression pool. The closed system is missile protected, seismic Category I, quality group B, and designed to the temperature and pressure conditions that the system will encounter post-LOCA. Valves will remain water covered during Type A testing. System leakage is minimized in accordance with NUREG-0737, Item III.D.1.1. Any leakage out of the closed system will be into the reactor enclosure, thus facilitating collection and treatment.

15. The isolation barrier remains water filled post-LOCA and will be tested with water.

16. These lines penetrate the diaphragm slab and are not subject to Appendix J leakage rate testing.

17. Table 1.8-2 contains a cross-reference to figure numbers.

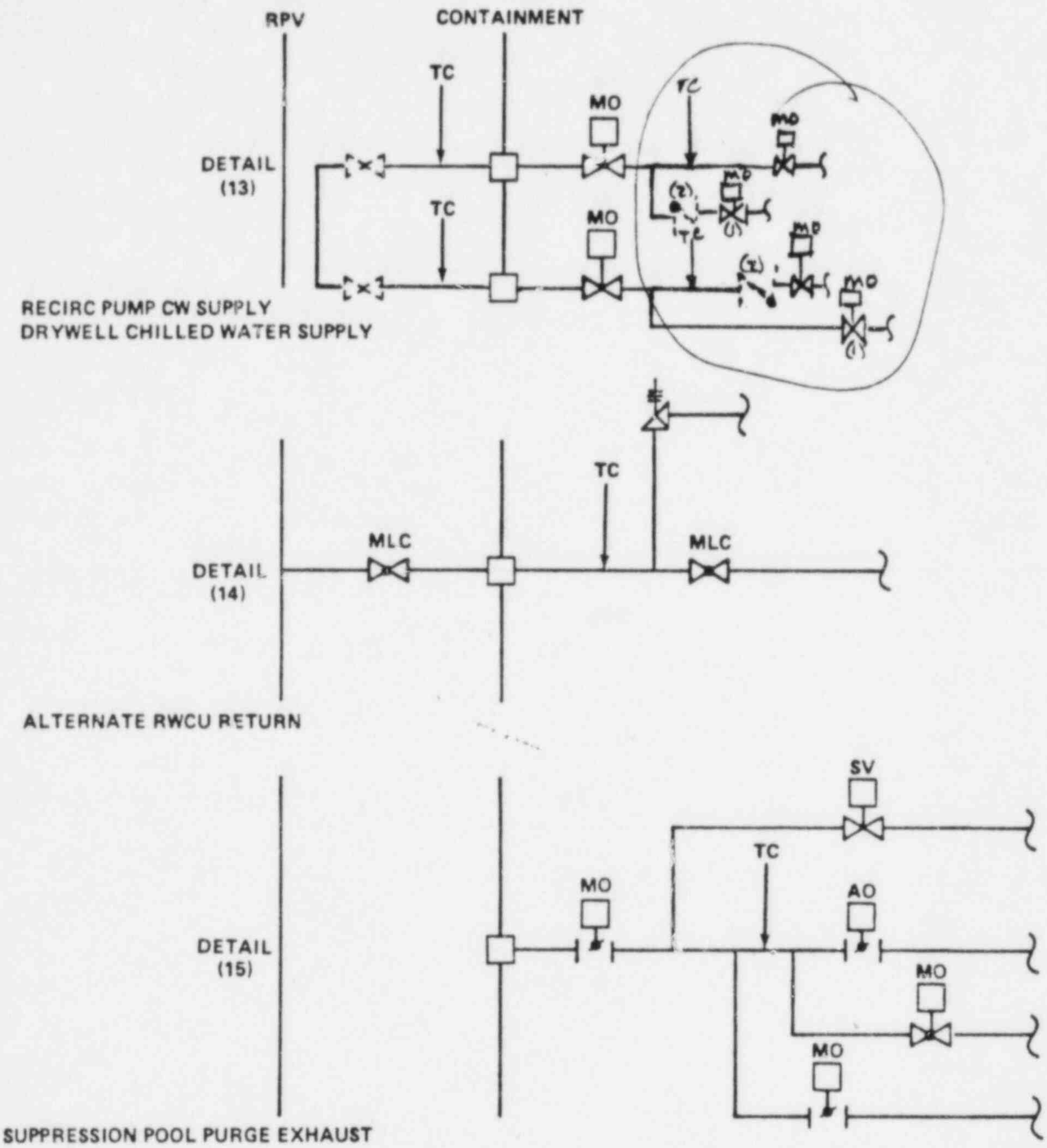
18. Feedwater penetrations will remain water filled post-LOCA as described in Section 6.2.3.2.3.

19. Check valve used instead of flow orifice.

20. Penetration is sealed by a flange with double o-ring seals. These seals are leakage rate tested by pressurizing between the oThe TIP drive tube is welded to the flange.

21. Seismic Category I, Quality Group B instrument line with an excess flow check valve. Because the instrument line is connected to a closed cooling water system inside containment, no flow orifice is provided. The line does not isolate during a LOCA and can leak only if the line or instrument should rupture. Leaktightness of the line is verified during the integrated leak rate test (Type A test).

22. The outboard isolation valves will be Type C tested with a head of water on the exterior side.



MO–MOTOR OPERATED
AO–AIR OPERATED
TC–TEST CONNECTION
MLC–MANUAL LOCKED CLOSED
SV–SOLENOID VALVE

(1) This valve is manual locked closed on RECIRC PUMP CW SYSTEM

(2) THIS VALVE IS ONLY ON RECIRC PUMP CW SYSTEM

LIMERICK GENERATING STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

CONTAINMENT PENETRATION DETAILS
SHEET 5 OF 16

FIGURE 6.2-36 REV. 11, 10/82

c. Closed system pipelines that are located inside the primary containment and penetrate primary containment have one isolation valve outside containment.

The MSIV controls include pneumatic piping and an accumulator for the air-operated valves as the isolation motive power source in addition to the springs. Pressure, temperature, and water level sensors are mounted on instrument racks or locally in either the reactor enclosure or the turbine enclosure. Valve position switches are mounted on motor- and air-operated valves. Switches are encased to protect them from environmental conditions. All signals transmitted to the control room are electrical (no pipe from the nuclear system penetrates the control room). The sensor cables and logic power supply cables are routed to cabinets in the auxiliary equipment room, where the system logic is located.

All instrument line penetrations of the containment are equipped with automatic isolation valves. These excess flow check valves automatically isolate if there is a break in the instrument line downstream of the valve.

7.3.1.1.2.4 PCRVICS Initiating Isolation Signals

The isolation trip settings of primary containment and reactor vessel isolation control system are listed in Chapter 16. The functional control diagrams (Figures 7.3-7, 7.3-8, 7.3-10, 7.4-1, 7.7-11) and the P&IDs (Figures 5.1-3, 5.1-4, 5.4-16, 5.4-13, 5.4-8, and 6.3-7) illustrate how these signals initiate closure of isolation valves. Additional logic is shown in Figures 7.3-14, 7.3-15, 7.3-16, 7.3-18, 7.3-19, and 7.3-20.

7.3.1.1.2.4.1 PCRVICS - Reactor Vessel Low Water Level

7.3.1.1.2.4.1.1 Subsystem Identification

A low water level in the reactor vessel could indicate that reactor coolant is being lost through a breach in the RCPB and that the core is in danger of becoming overheated as the reactor coolant inventory diminishes.

Three reactor vessel low water level isolation trip settings are used to complete the isolation of the primary containment and the reactor vessel.

The first (and highest) low water level setting level 3 (which is the RPS low water level scram setting) is selected to initiate isolation at the earliest indication of a possible breach in the RCPB, yet the setting is far enough below normal operational levels to prevent spurious isolation. The pipelines that are isolated when reactor vessel low water level falls to level 3 are RHR shutdown cooling and RPV head spray.

The second (middle) reactor vessel low water level isolation setting (level 2) is the same water level setting at which the RCIC and HPCI systems are placed in operation. The setting selected is low enough to allow the removal of heat from the reactor for a predetermined time following the scram and high enough to complete isolation in time for the operation of ECCS if there is a large break in the RCPB. The pipelines that are isolated when the reactor vessel water level falls to this second setting are listed below.

a. Reactor water cleanup

b. Containment atmospheric control including H_2O_2 sample lines

c. Traversing incore probe

d. Main steam sample

e. HPCI pump flush

f. RHR vacuum relief

g. Drywell sump drains

h. Suppression pool cleanup

i. Drywell radiation sample

j. Recirculation loop sample

k. Drywell chilled water

The third (and lowest) of the reactor vessel low water level isolation settings (level 1) is the water level setting used to initiate RHR, core spray, and automatic depressurization system, and to start the diesel generators. The pipelines that are isolated when the reactor vessel water level falls to this third setting are the main steam, main steam line drain, containment instrument gas, RHR heat exchanger vent valves, suppression pool spray and core spray pump test and flush.

Reactor vessel low water level signals are initiated from eight differential pressure sensors, four sensors for the level 1 and level 2 trip and four sensors for the level 3 trip, as shown in Figure 5.1-4. They sense the difference between the pressure caused by a constant reference leg of water and the pressure caused by the actual water level in the vessel.

Four pairs of instrument sensing lines, attached to taps above and below the water level on the reactor vessel, are required for the differential pressure measurement and terminate outside the drywell and inside the reactor enclosure. They are physically

7.3.1.1.2.4.6.3 Subsystem Initiating Circuits

Drywell pressure is monitored by four pressure sensors that are mounted on instrument racks outside the primary containment. Instrument sensing lines that terminate in the reactor enclosure connect the sensors with the drywell interior. Redundant sensors are physically separated and electrically connected to the isolation control systems so that no single event prevents isolation because of primary containment high pressure.

7.3.1.1.2.4.6.4 Subsystem Logic and Sequencing

When a predetermined increase in drywell pressure is detected, trip signals are transmitted to the PCRVICS. The PCRVICS isolates the following lines: drywell drains and drywell sump drains discharge to radwaste, primary containment purge and vent, traversing incore probe (TIP) system, containment atmosphere sampling, containment instrument gas, vacuum relief, HPCI pump flush and vacuum relief, RCIC vacuum relief, suppression pool cleanup, ~~and~~ drywell radiation sample, and drywell chilled water.

Four instrumentation channels are provided to ensure protective action when required and to prevent inadvertent isolation resulting from instrumentation malfunctions. The output trip signals of the instrumentation channels are combined in two-out-of-two logics. Instrumentation channels A and B or C and D are required to initiate isolation of either inboard or outboard valves, respectively. Thus, failure of any one channel does not result in inadvertent action.

7.3.1.1.2.4.6.5 Subsystem Redundancy and Diversity

Redundancy of trip initiation signals for drywell high pressure is provided by pressure switches installed at different locations around the drywell. Wiring from redundant instruments is separated. Each pressure switch is associated with one logic division. Two pressure sensors are supplied from RPS bus A, and the other two are supplied from RPS bus B.

Diversity of trip initiation signals for line breaks inside the primary containment is provided by drywell high pressure and reactor low water level. An increase in drywell pressure or a decrease in reactor water level initiates isolation.

7.3.1.1.2.4.6.6 Subsystem Bypasses and Interlocks

There are no bypasses or interlocks for drywell high-pressure trip signals.

7.3.1.1.2.4.6.7 Subsystem Testability

No change

isolation. Redundant channels for each monitored variable provide inputs to different isolation trip systems. The functions of the sensors in the isolation control system are shown in Figures 7.3-1 and 7.3-2. Table 7.3-5 lists instrument characteristics.

7.3.1.1.2.6 PCRVICS Initiating Circuits

The valves which are controlled by the PCRVICS for automatic isolation generally utilize actuator solenoids which are energized during normal service. The logic circuitry signalling automatic isolation is generally arranged such that when a monitored parameter reaches its trip setpoint, the associated trip logic contact opens. When the proper combination of logic trips occur, the actuator trip relay de-energizes, de-energizing the valve actuator solenoid. The system also has system level manual initiation switches that isolate all automatically controlled isolation valves. This general arrangement is applicable to the following valve discussions.

The MSIV actuators each have two actuator solenoids. For automatic valve closure, both solenoids must be de-energized. Each solenoid receives inputs from two separate division logics, either of which can de-energize the solenoid.

Four RPS instrument channels are provided for each monitored parameter used in the MSIV trip logic. The redundant instrument channels are independent and separate. Channels A and C actuator logic trip relays control one solenoid in each of the inboard and outboard MSIV's on each main steam line. Channels B and D actuator logic trip relays control the other solenoid in the inboard and outboard MSIV's.

Closure of the inboard and outboard main steam line drain isolation motor-operated valves is initiated by the MSIV actuator logic trip relays, utilizing two-out-of-two logic. Logic relays A and B initiate closure of the inboard isolation valve and relays C and D initiate closure of the outboard isolation valve.

Closure of the inboard and outboard RHR discharge to radwaste isolation valves and RHR process sampling isolation valves is initiated by low reactor vessel water level and high drywell pressure signals, utilizing two-out-of-two logic. Logic trip relay A closes the inboard valves and logic trip relay B closes the outboard valves.

Closure of the inboard RHR shutdown cooling suction and inboard RHR head spray isolation motor-operated valves, as well as the RHR shutdown cooling injection-testable check valve and bypass valve is initiated by logic trip relay A. Closure of the outboard RHR shutdown cooling suction, outboard RHR head spray isolation, and

Insert (D)

RHR shutdown cooling injection outboard throttling motor-operated valves is initiated by logic trip relay B. Tripping of the logic trip relay requires tripping of the two-out-of-two logic for low reactor vessel water level signal or the one-out-of-two logic for reactor low pressure signals.

Closure of the inboard and outboard reactor water sample motor-operated valves and reactor steam sample air-operated valves is initiated by low reactor vessel water level and high steam line radiation signals, utilizing two-out-of-two logic. Logic trip relay A closes the inboard valves and logic trip relay B closes the outboard valves.

Retraction of the TIP drives is initiated by low reactor vessel water level 2 and high drywell pressure in a two-out-of-two logic. Logic trip relay A initiates retraction of the TIP system drives.

Closure of the primary containment purge valves is initiated by any one of the following conditions in a two-out-of-two logic: reactor level below level 2 trip; high drywell pressure; high radiation in the reactor enclosure ventilation exhaust duct; high radiation in the refueling floor ventilation exhaust duct.

Closure of the inboard and outboard RWCU isolation motor-operated valves is initiated by low reactor vessel water level utilizing two-out-of-two logic or by high area temperature and high differential RWCU flow utilizing one-out-of-one logic. Logic trip relay A closes the inboard valve and logic trip relay B closes the outboard valves.

Closure of the inboard containment instrument gas system (CIGS) suction valve is initiated by any one of the following conditions in a two-out-of-two logic: Reactor level below level 1 trip; high drywell pressure; high radiation in the reactor enclosure ventilation exhaust duct; or high radiation in the refueling floor ventilation exhaust duct.

Closure of the primary containment atmosphere sample isolation valves and the post-LOCA hydrogen recombiner isolation valves is initiated by any one of the following conditions in a one-out-of-one logic: Reactor level below level 2 trip; high drywell pressure; high radiation in the reactor enclosure ventilation exhaust duct; high radiation in the refueling floor ventilation exhaust duct.

Closure of outboard CIGS isolation valves not on ADS gas supply lines is initiated by any one of the following conditions in a two-out-of-two logic: Reactor level below level 1 trip; high drywell pressure; high radiation in the reactor enclosure ventilation exhaust duct.

Insert (D)

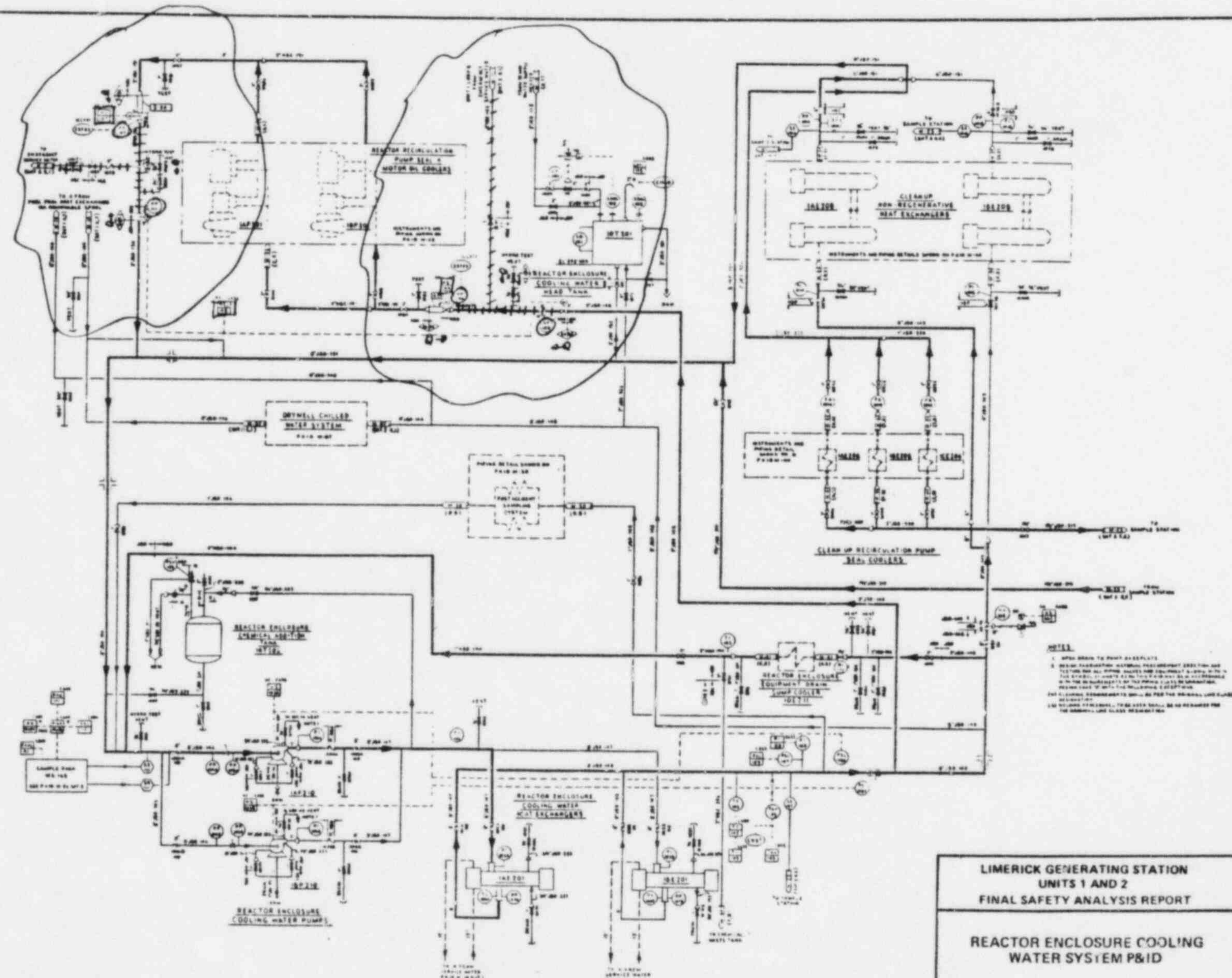
Closure of the drywell chilled water isolation valves is initiated by any one of the following conditions in a one-out-of-one logic: Reactor level below level 2 trip; high drywell pressure; high radiation in the reactor enclosure ventilation exhaust duct; high radiation in the refueling floor ventilation exhaust duct.

TABLE 7.5-3 (cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV.)	LOCATION
Containment Isolation Valve Position	B10	1					
HV13-106			Indicating lights	1 pair per valve	open/closed	HS13-106 (Div III)	Control room
HV13-107			Indicating lights	1 pair per valve	open/closed	HS13-107 (Div III)	Control room
SV26-190A, C			Indicating lights	1	open/closed	MSC lights (Div III)	Control room
SV26-190B, D			Indicating lights	1	open/closed	MSC lights (Div II)	Control room
HV40-1F001B, F, K, P			Indicating lights	1 pair per valve	open/closed	HS40-110B,F K,P (Div II)	Control room
HV41-1F022A, B, C, D			Indicating lights	1 pair per valve	open/closed	HS41-122A, B C,D (Div I)	Control room
HV41-1F028A, B, C, D			Indicating lights	1 pair per valve	open/closed	HS41-128A, B, C, D (Div II)	Control room
HV41-1F016			Indicating lights	1 pair per valve	open/closed	HS41-116 (Div I)	Control room
HV41-1F019			Indicating lights	1 pair per valve	open/closed	HS41-119 (Div II)	Control room
HV41-109A, B			Indicating lights	1 pair per valve	open/closed	HS41-109A, B (Div I)	Control room
HV41-133A			Indicating lights	1 pair per valve	open/closed	HS41-133A (Div I)	Control room
HV41-133B			Indicating lights	1 pair per valve	open/closed	HS41-133B (Div II)	Control room
HV41-1F084			Indicating lights	1 pair per valve	open/closed	HS51-186 (Div I)	Control room
HV13-108, 111			Indicating lights	1 pair per valve	open/closed	HS13-108 (Div IV)	Control room

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV.)	LOCATION
HV59-131			Indicating lights	1	open/closed	HS59-131 (Div II)	Control room
HV59-101			Indicating lights	1	open/closed	HS59-101 (Div I)	Control room
HV59-102			Indicating lights	1 pair per valve	open/closed	HS59-102 (Div II)	Control room
XV59-141A, B, C, D, E			Indicating lights	1 pair per valve	open/closed	Valve control monitor A, B, C, D, E	Control room
HV59-135			Indicating lights	1 pair per valve	open/closed	HS59-135 (Div II)	Control room
HV61-102, 112, 132			Indicating lights	1 per valve	open/closed	HS61-112 (Div I)	Control room
HV61-110			Indicating lights	1	open/closed	MSC lights (Div I)	Control room
HV61-130			Indicating lights	1	open/closed	MSC lights (Div I)	Control room
HV61-111			Indicating lights	1	open/closed	MSC lights (Div II)	Control room
HV61-131			Indicating lights	1	open/closed	HS61-131 (Div II)	Control room
HV87-128,129			Indicating lights	1 pair per valve	open/closed	HS87-128 (Div II)	Control room
HV87-122,123			Indicating lights	1 pair per valve	open/closed	HS87-122 (Div II)	Control room
HV87-120A,121A, 124A, 125A			Indicating lights	1 pair per valve	open/closed	HS87-121A (Div I)	Control room
HV87-120B,121B, 124B, 125B			Indicating lights	1 pair per valve	open/closed	HS87-121B (Div I)	Control room

REFERENCE DRAWINGS	
P & ID NO.	FSAR FIG. NO.
M-10	9.2-1
M-11	9.2-2
M-18	9.2-5
M-23	9.3-3
M-26	11.5-1
M-43	5.4-2
M-44	-
M-61	9.3-4
M-64	11.2-3
M-70	11.3-3
M-87	9.2-27

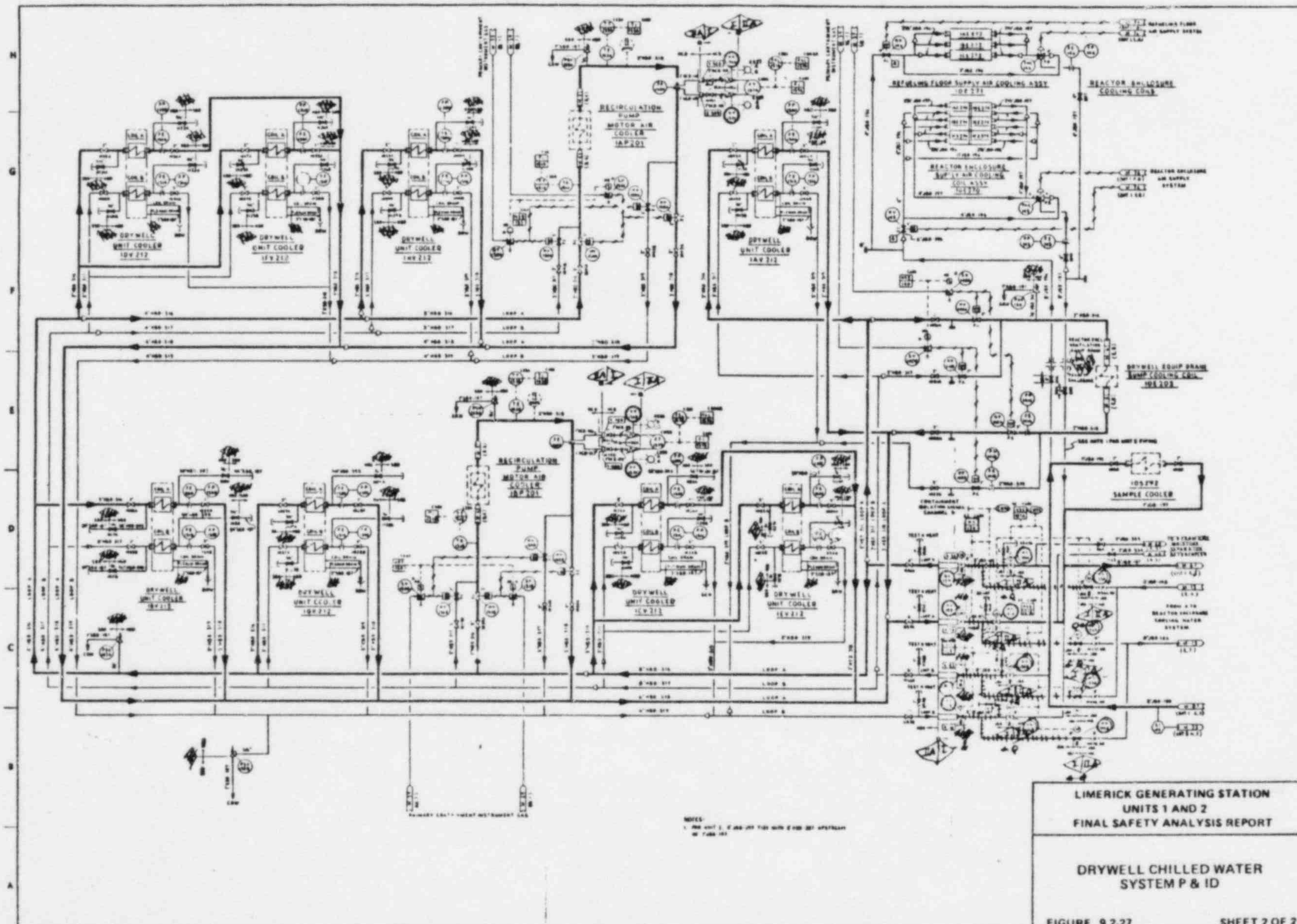


LIMERICK GENERATING STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

REACTOR ENCLOSURE COOLING WATER SYSTEM P&ID

FIGURE 9.2-25 REV. 10, 09/82

REFERENCE DRAWINGS	
P & I D NO.	FSAR FIG. NO.
M-13	9.2 - 25
M-23	9.3 - 3
M-43	5.4 - 2
M-59	9.3 - 2
M-60	6.2 - 43
M-61	9.3 - 4
M-76	9.4 - 2
M-87	9.2 - 27



LIMERICK GENERATING STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

DRYWELL CHILLED WATER
SYSTEM P & ID

FIGURE 9.2-27

SHEET 2 OF 2

No change

QUESTION 480.26 (Section 6.2.3)

Supplement your description of containment bypass leakage barriers listed in FSAR Table 6.2-15 in the following areas:

a. Penetrations X-23, 24, 53, 54, 55, and 56 rely on closed systems inside containment to preclude bypass leakage. Verify that each requirement listed in Branch Technical Position (BTP) CSB 6-3 Position B.9 is met for these systems.

b. Provide additional information describing the feedwater fill system and demonstrate that it conforms to the NRC guidelines commensurate to its safety function of eliminating bypass leakage (reference BTP CSB 6-3 Position B.8).

c. For all containment penetrations in FSAR Table 6.2-15 that rely on a water seal to prevent bypass leakage, provide additional information that demonstrates the water seal will be maintained for 30 days following a LOCA, i.e., that the water seal is maintained at a pressure greater than the peak containment accident pressure and that a sufficient water seal inventory to last at least 30 days following a design basis acccident is provided.

d. The recirculation pump seal purge lines (Penetration X-61) are vented to secondary containment by use of vent lines located before two block valves and the secondary containment (see FSAR Figures 5.4-2 and 4.6-5). The P&ID shows no automatic actuation signal to the valve operators for the normally closed vent valves (HV125 and HV126) or the normally open block valves (HV127 and HV128). Verify that these valve operators will receive containment isolation signals to open and close, respectively.

e. FSAR Table 6.2-15 indicates that the HPCI and RCIC vacuum relief lines (Penetrations X-228D and 241) contain temporary spool pieces that are removed during normal operation and replaced by blind flanges. These spool pieces are not shown on the P&IDs (FSAR Figures 6.3-7 and 5.4-8). Provide revised P&IDs that include the spool pieces, or describe alternative provisions to preclude bypass leakage.

NOTE: Any penetration through which bypass leakage cannot be precluded by an acceptably described bypass leakage barrier must be considered as a bypass leakage path and treated as described in BTP CSB 6-3 Positions B.6 and 7.

RESPONSE

Section 6.2.3.2.3 has been changed to supplement the description of containment bypass leakage barriers listed in Table 6.2-15. Each of the above areas has been addressed individually below.

a. Both the drywell chilled water (DCW) system and the reactor enclosure cooling water (RECW) system are classified as closed systems inside containment. A discussion on the basis for classifying the RECW and DCW as closed systems inside containment is included in the response to Question 480.40. Replace with Insert (E)

b. Section 6.2.3.2.3 and Table 3.2-1 have been changed to provide the requested information.

c. Section 6.2.3.2.3.1 has been added to provide the requested information.

d. The normally open block valves (HV127 and HV128) receive an automatic containment isolation signal to close, which is shown as Ref 18 on Figure 4.6-5. It is not necessary for the containment isolation signal to also automatically open the (normally closed) vent valves. The seal lines are normally filled with water, which will spill into secondary containment when the vent valves open. To preclude the possibility of this happening in the event of a false LOCA, the vent valves will be manually actuated to open when the operator has verified that isolation and venting of these lines is required. The water in the lines will provide a temporary seal to prevent bypass leakage until the vent valves are opened.

e. The temporary spool pieces are on branch lines to the HPCI and RCIC vacuum relief lines. The spool pieces are shown on Figure 6.3-7 (line 4" EBB-108) and on Figure 5.4-9 (line 3" EBB-109).

Insert (E)

The drywell chilled water system (penetrations 53, 54, 55, and 56) and the reactor enclosure cooling water system (penetrations 23 and 24) take credit for a water seal outside containment and redundant isolation valves as bypass leakage barriers. Sections 6.2.3.2.3.1 and 6.2.4.3.1.3 and Tables 6.2-15, 6.2-17, and 6.2-25 have been changed to provide information concerning these additional bypass leakage barriers.

QUESTION 480.40 (Section 6.2.4)

It is the NRC position that the requirements of GDC 56 and not GDC 57 must be met for the instrument gas supply lines (Penetration X-3B, X-3D, X-40H, and X-218), the recirculation pump cooling water supply and return lines (Penetrations X-23 and X-24) and the drywell chilled water supply and return lines (Penetrations X-53, X-54, X-55, and X-56) because the systems inside containment to which these lines connect do not meet the requirements of a closed system (see SRP Section 6.2.4 Item II.o and FSAR Section 6.2.4.3.1.6). Therefore demonstrate for these penetrations how the containment isolation requirements of GDC 56 will be met. (Note: For the instrument gas line penetrations, GDC 56 requirements will be met if both the outside valves (HV129B, HV151, HV129A, and HV135) and the inside check valves (1005B, 1112, 1005A, and 1001) are included in FSAR Table 6.2-17 as containment isolation valves for Penetrations X-3B, X-3D, X-40H, and X-218, respectively).

RESPONSE

Section 6.2.4 and Tables 6.2-17 and 2.1-3 have been changed to verify that the requirements of GDC 56 have been met for the instrument gas supply lines (Penetrations X-3B, X-3D, X-27A, X-40H, and X-218).

The drywell chilled water (DCW) system and the recirculation pump cooling water supply and return lines of the reactor enclosure cooling water (RECW) system form closed systems inside containment that meet the requirements of 10CFR part 50, Appendix A, General Design Criteria 54 and 57 and the intent of the guidelines in SRP Section 6.2.4, item II.O. These systems had ~~have~~ been designed with the following isolation provisions:

a. A least one isolation valve is provided on each line as close as practical to the containment

b. The isolation valves are capable of remote manual operation

c. Power supplies and controls for the isolation valves are safety grade

d. Piping systems from containment to, and including, the isolation valves are safety grade and quality group B.

e. The systems do not communicate with either the reactor coolant pressure boundary or the containment atmosphere

f. The closed systems are seismic Category IIA, as discussed in Section 3.2.1, and have been design to the same seismic loads as seismic Category I systems

g. The RECW and DCW systems have been designed to Quality Group C and D standards, respectively. These quality standards are supplemented by quality control inspections by trained and qualified inspectors that perform and document inspections on piping installations, welds, valves, and hangers. These systems are designed with fully welded joints between interconnecting piping and use the same materials that are used in Safety Class 2 piping systems

h. The systems have been designed to withstand the external pressure from the containment structural acceptance test and use materials capable of withstanding temperatures in excess of the containment design temperature

i. These systems do not connect to the environment except through a vent in the systems' head tanks.

The probability was of a release to the environment through these closed systems ~~is~~ low. For a release to the environment to occur, it ~~is~~ was necessary to assume all of the following:

a. Loss-of-coolant accident

b. Core damage

c. Failure of the closed system inside containment

d. Failure of the containment isolation valves

~~e. Failure of the system isolation valves (located outboard of the containment isolation valves)~~ stet

The design of these systems and their containment isolation provisions are in accordance with the design commitments made in the Limerick PSAR as well as industry and NRC guidance available at that time.

~~The closed systems are seismic Category IIA, as discussed in Section 3.2.1, and have been designed to the same seismic loads as seismic Category I systems.~~

~~The probability associated with a closed system failure resulting in a release to the environment is low. For a release to the environment, it is necessary to assume a LOCA resulting in both core damage and a closed system pipe break inside containment coincident with an isolation valve failure in the closed system.~~

In addition, system isolation valves are provided outside containment for each closed system. These valves are remote manually operated from the control room. The DCW system is also provided with low flow alarms and indicators in the control room. In addition, the DCW system will only use one loop at a time (either penetrations X-53 and X-54 or X-55 and X-56). The remaining DCW valves are normally closed.

~~Regardless of the abov~~ The DCW and RECW system designs have been changed to meet the requirements of GDC 56. Section 6.2.4 and Table 6.2-17 have been changed to indicate the basis for compliance.

Issue 16

The applicant has not provided a description and the results of an analysis which demonstrates the acceptability of the provisions made to protect structures and safety related equipment (e.g., fans, filters, and ductwork) located beyond the purge system containment isolation valves against loss of function from the environment created by escaping air and steam following a LOCA in accordance with BTP CSB 6-4 Position B.5.b., Reference NRC Question 480.42 Part h.

Response

The response to NRC Question 480.42 (h), which addresses this concern, is attached.

480.42(h)

Response

The Limerick containment purge and vent valves will be opened for a limited period of time during power operation for inerting and deinerting of the primary containment atmosphere. The inert atmosphere, (diluted with nitrogen to < 4% O_2) is purged from containment in anticipation of outages requiring containment access to allow inspections and limited repairs while the reactor is at some reduced power level. Inspections are also performed during power ascention prior to the inerting of containment. These inspections facilitate the early detection and location of coolant system leaks which could have an impact on unit reliability and safety if uncorrected.

The Containment Atmosphere Control System is described in Section 9.4.5 and illustrated in Figure 9.4-5. Purging and venting operations are normally performed through one 24" supply penetration and one 24" exhaust penetration. All gases purged from containment are processed through SGTS prior to release.

Because the purge and vent valves are opened during only a limited period of power operation (typically less than 90 hours per year), it is unlikely that a LOCA will occur while the valves are open. If a LOCA were to occur during this time, the containment isolation valves would close rapidly (less than 6 seconds after receipt of isolation signal) and terminate the release. Isolation will be complete long before any fuel damage or significant offsite exposure could occur. The containment isolation valves utilized have been specially designed and qualified for this service as described in Section 9.4-5. It is possible in such cases, however, for the downstream ductwork and/or SGTS filters ...

to be damaged by the pressure surge preceeding valve closure and/or the moisture content of the released gases.

Analyses have been performed to determine the potential for, and significance of, the above described sequence of events at Limerick. Calculations indicate that the design pressure of the ductwork and the design differential pressure of the SGTS filters ~~would~~ may be exceeded if the DBA LOCA occurred during purging. This could (1) result in the failure of the operating SGTS filter bank, and (2) possibly cause equipment failures in the Reactor Enclosure due to duct impact, impingement, and/or the resulting environmental conditions.

Failure of the operating SGTS filter bank is of little significance due to the limited benefit derived from SGTS for accident sequences related to plant risk and the possibility that the backup filter bank would be operable. The results of the Reactor Safety Study (see WASH-1400, Table 5-3) indicate that the failure of SGTS during a LOCA does not contribute to any significant releases in a BWR for the following reasons:

- LOCA sequences contribute little to radioactive releases relative to transient sequences.
- Consideration of SGTS failure is only relevant for small containment leaks (i.e. - compared with a containment overpressure rupture).

Because of the potential significance of equipment failures, calculations and detailed equipment location surveys have been performed

for Limerick to verify that (1) the environmental qualifications for the LGS equipment are sufficient to assure operability under the predicted environmental ~~conditions,~~ and (2) the potential does not exist for impact or impingement related damage to essential equipment.

The above described conclusions also apply but with significant conservatism to the situation when either a medium or small LOCA is postulated to occur during purging. For a small LOCA, the ductwork is not expected to rupture, and the SGTS heaters can be expected to reduce the relative humidity of the incoming gases. For a medium LOCA, if the ductwork should rupture, the amount of steam released into the reactor ~~building~~ enclosure and the energy available for impact or impingement will be considerably less than that associated with a DBA LOCA.

Issue 17

CSB's findings on the applicant's compliance with BTP CSB 6-4, "Containment Purging During Normal Plant Operations," is contingent on the acceptance of the valve operability assurance program for both high and low volume drywell and suppression chamber purge system line containment isolation valves by the Equipment Qualification Branch (EQB). (Contingency)

Response

Discussions have been held between the applicant, the Limerick valve manufacturer (Clow), and the NRC contractor reviewing the issue of purge and vent valve operability. The testing and analytical work performed by the manufacturer to demonstrate the functionality of the Limerick purge and vent valves has been described in detail. A report documenting this work is currently in preparation. Submittal to the NRC is currently scheduled for the week of May 23, 1983.

Issue 18

Containment isolation valves HV-109A and HV-109B (feedwater line penetrations X-9A and X-9B, respectively) must be sealed closed as defined in SRP Section 6.2.4 II.f. (Note: This is a new issue resulting from the FSAR Revision 16 change in footnote (13) of Table 6.2-17, page 19 of 19. The previous applicant's statement that these containment isolation valves will be "normally locked closed" has been revised to simply "normally closed" which is unacceptable.)

Response

Section 6.2.4.3.1.2.1.1 and Table 6.2-17 have been changed to state that the above isolation valves will be sealed closed. The revised section and table are attached.

containment isolation barriers, are maintained. All power-operated isolation valves have position indicators in the control room. Discussion of instrumentation and controls for the isolation valves is included in Chapter 7.

6.2.4.3.1 Evaluation Against General Design Criteria

6.2.4.3.1.1 Evaluation Against General Design Criterion 54

All piping systems penetrating containment, other than instrument lines, are designed in accordance with Criterion 54.

6.2.4.3.1.2 Evaluation Against Criterion 55

Criterion 55 requires that lines which penetrate the primary containment and form a part of the RCPB must have two isolation valves; one inside the containment and one outside, unless it can be demonstrated that the containment isolation provisions for a specific class of lines are acceptable on some other basis.

The RCPB, as defined in 10 CFR Part 50, Section 50.2 (v), consists of the reactor pressure vessel, pressure retaining appurtenances attached to the vessel, and valves and pipes that extend from the reactor pressure vessel up to and including the outermost isolation valve.

6.2.4.3.1.2.1 Influent Lines

Influent lines that penetrate the primary containment and connect directly to the RCPB are equipped with at least two isolation valves, one inside the drywell, and the other as close to the external side of the containment as practicable.

6.2.4.3.1.2.1.1 Feedwater Line

The feedwater line is part of the RCPB as it penetrates the drywell to connect with the reactor pressure vessel. It has three isolation valves. The isolation valve inside the drywell is a check valve located as close as practicable to the containment wall. Outside the containment is an air-assisted check valve located as close as practicable to the containment wall, and farther away from the containment is a motor-assisted check valve on the feedwater line. Additional isolation valves are located on lines connecting to the feedwater line outside containment. Insert A Should a break occur in the feedwater line, the outboard check valves prevent significant loss of reactor coolant inventory and offer immediate isolation. (It is impractical to restrain the inboard check valve to withstand pipe whip resulting from a downstream feedwater line break; therefore it cannot be assumed to isolate for this case.) During a postulated LOCA, it is desirable to maintain reactor coolant makeup from all sources of supply. For this reason, the feedwater lines are not

Insert (A)

The feedwater bypass line isolation valves are normally sealed closed using administrative control as discussed in NUREG-0800, Standard Review Plan section 6.2.4.II.f. These valves will only be opened for flushing of the condensate and feedwater systems during startup of the units (up to approximately 5 percent power).

No change

automatically isolated upon signals from the protection system. The outermost valve is capable of being remotely closed from the control room to provide long-term leakage protection.

The air-assisted check valve is provided with a special actuator that performs the following functions:

a. The actuator is capable of partially moving the valve disc into the flow stream during normal plant operation in order to ensure that the valve is not bound in the open position. The actuator is not capable of fully closing the valve against flow, however, and there is no significant disruption of feedwater flow.

b. The actuator is capable of applying a seating force to the valve at low differential pressures and abnormal conditions. This improves the leaktightness of the valves. The actuator is not utilized during leak testing.

6.2.4.3.1.2.1.2 HPCI Line

The HPCI line connects to CS loop B that penetrates the drywell to inject directly into the RPV. Isolation is provided by two valves in the CS line, an air testable check valve inside the containment, and an air assisted check valve outside the containment, with positions of both indicated in the main control room. The core spray loop B line is also provided with a normally closed motor-operated globe valve which bypasses the inboard isolation valve for equalization during testing.

6.2.4.3.1.2.1.3 LPCI and CS Loop A

The LPCI lines and CS loop A line are provided with remote manually controlled gate valves outside and air testable check valves inside containment. Both types of valves are normally closed with the gate valves receiving an automatic signal to open at the appropriate time. The check valves are located as close as practicable to the RPV. The normally closed check valves protect against containment pressurization if there is a pipe rupture between the check valve and containment wall. The core spray loop A line and the LPCI lines are also each provided with a normally closed motor-operated globe valve which bypasses the inboard isolation valve for testing purposes.

TABLE 6.2-17

CONTAINMENT PENETRATION DATA

CONTAINMENT PENETRATION NUMBER	LINE ISOLATED	FLUID	LINE SIZE (in.)	NRC GENERAL DESIGN CRITERION	ESF SYSTEM	ESSENTIAL SYSTEM	VALVE NUMBER	VALVE TYPE(1)	VALVE LOCATION	VALVE ARRANGEMENT(2)	TYPE C TEST	LENGTH OF PIPE FROM CONT. TO OUTSIDE VALVES	PRIMARY METHOD OF ACTUATION(3)	SECONDARY METHOD OF ACTUATION	NORMAL VALVE POSITION(4)	SHUTDOWN VALVE POSITION	POST-ACCIDENT POSITION	POWER FAILURE VALVE POSITION	ISOLATION SIGNAL(5)	DIVERSE ISOLATION SIGNAL(12)	VALVE CLOSURE TIME(6)	POWER SOURCE(7)	REMARKS
X-3A	Instrumentation-main steam line D flow	Water/steam	1	55	-	-	F070D	XFC	Outside	(40)	No	12"	Flow	-	O	O	O	-	-	-	-	-	
					-	-	F073D	XFC	Outside			12"	Flow	-	O	O	O	-	-	-	-	-	
X-3B	Instrument gas supply	Gas	1	56	No	No	1005B	CK	Inside	(22)	Yes	-	Flow	-	O	O	C	-	-	-	-	-	
					No	No	129B	GB	Outside			6"	Comp Air	Spring	O	O	C	C	C,H,S	Yes	4.4 sec	E	
X-3C	Instrumentation-HPCI steam flow	Steam	1	55	-	-	F024A	XFC	Outside	(40)	No	12"	Flow	-	O	O	O	-	-	-	-	-	
X-3C	Instrumentation-HPCI steam flow	Steam	1	55	-	-	F024C	XFC	Outside	(40)	No	12"	Flow	-	O	O	O	-	-	-	-	-	
X-3D	Instrumentation-steam line A flow	Steam	1	55	-	-	F070A	XFC	Outside	(40)	No	12"	Flow	-	O	O	O	-	-	-	-	-	
					-	-	F073A	XFC	Outside			12"	Flow	-	O	O	O	-	-	-	-	-	
X-3D	Instrument gas supply	Gas	1	56	No	Yes	1112	CK	Inside	(48)	Yes	-	Flow	-	O	O	C	-	-	-	-	-	
					No	Yes	151B	GB	Outside			7"	AC Motor	Manual	O	O	C	AS IS	N	NA	30 sec	D	
X-7A-D	Main steam	Steam	26	55	No	No	F022A-D	GB	Inside	(1)	Yes	-	Instr gas	Spring	O	C	C	C	C,D,E,F,P,Q	Yes	3 to 10* sec	A/W	
					No	No	F028A-D	GB	Outside			0"	Comp air	Spring	O	C	C	C	C,D,E,F,P,Q	Yes	3 to 10* sec	B/X	
					Yes	Yes	F001B,F,K,P	GB	Outside			4'-6"	AC motor	Manual	C	C	O	AS IS	EA	NA	30 sec	B	(8)
					-	-	101B,F,K,P	XFC	Outside		No	45'-8"	Flow	-	O	O	O	-	-	-	-	-	
X-8	Main steam drain	Steam/water mix	3	55	No	No	F016	GT	Inside	(33)	Yes	-	AC motor	Manual	C	O	C	AS IS	B,D,E,F,P,Q	Yes	Standard	A	
					No	No	F019	GT	Outside			0"	AC motor	Manual	C	O	C	AS IS	B,D,E,F,P,Q	Yes	Standard	B	
X-9A,B	Feedwater	Water	24	55	Yes	Yes	F010A	CK	Inside	(2)	Yes	-	Flow	-	O	C	O	-	-	-	-	[illegible]	
					No	Yes	F010B	CK	Inside			-	Flow	-	O	C	O	-	-	-	-	-	
					Yes	Yes	F074A	SLPACK	Outside			0"	Flow	Spring	O	C	C	-	-	-	-	B	(9)
					No	Yes	F074B	SLPACK	Outside			0"	Flow	Spring	O	C	O	-	-	-	-	B	
					No	No	F032A,B	CK	Outside			22'-2"	Flow	AC motor	O	C	O	AS IS	-	-	-	A	
					No	No	109A,B	GT	Outside			31'-6"	AC motor	Manual	C	C	C	AS IS	RM	No	Standard	[illegible]	(13)
					No	No	F039	SLPACK	Outside			29'-6"	Flow	Spring	O	O	C	-	-	-	-	B	(9)
					No	Yes	F013	GT	Outside			20'-1"	DC motor	Manual	C	C	O	AS IS	LFCC	NA	15 sec	A	
					No	Yes	1036A,B	CK	Outside			2'-6"	Flow	-	C	C	O	-	-	-	-	-	
					No	Yes	130A,B	GB	Outside			124'-6"	AC motor	Manual	C	C	C	AS IS	RM	NA	30 sec	A, B	
					No	Yes	133A,B	GB	Outside			70'-1"	AC motor	Manual	C	C	O	AS IS	RM	NA	30 sec	A, B	
					No	No	1016	GB	Outside			83'-2"	Manual	-	C	C	C	[illegible]	-	-	-	-	
					Yes	Yes	F105	GT	Outside			22'-8"	AC Motor	Manual	C	C	O	AS IS	RM	NA	20 sec	[illegible]	

T1001550-01V

Rev. [redacted] 18, 03/93

TABLE 6.2-17 (Cont'd)

(12) Only non-essential systems require diverse signals for automatic isolation. Therefore, this column is not applicable, (NA), for essential systems.

(13) These valves are normally sealed closed and will only be opened ~~for startup of the feedwater system, before the control rods are withdrawn. Closure times less than 60 sec and automatic isolation signals to these valves are therefore not required~~.

(14) Diverse isolation signals are not sensed as discussed in Section 6.2.4.3.1.

(15) These valves are normally closed, will be open only during reactor shutdown, are interlocked to open only on low reactor pressure, and connect to a closed system outside containment. Therefore, closure times less than 60 seconds are not required.

for flushing of the condensate and feedwater systems during startup (up to approxinately 5 percent power).