APPENDIX I EVALUATION OF HIGH ENERGY LINE BREAKS OUTSIDE CONTAINMENT

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I.1 Evaluation Criteria

Regulatory Approved Analyses of the Consequences of High Energy Piping Failure Outside Containment, Licensing Basis

Standard Review Plant 3.6.1, Reference 3, Attachment Branch Technical Position ASB 3-1, "Protection Against Postulated Piping Failures In Fluid Systems Outside Containment", Section B.4.d, Implementation, "Designs of plants for which operating licenses are issued before July 1, 1975 are considered acceptable with regard to effect of piping failures outside containment on the basis of the analyses made and measures taken by applicants and licensees in response to the December 1972 letter from A. Giambusso, and the staff review and acceptance of these analyses and measures". Monticello operating license was issued on June 30, 1971.

Therefore, the licensing basis of the Monticello Nuclear Generating Plant HELB Program is:

- The Giambusso letter dated December 18, 1972 with Errata dated January 16, 1973, Reference 2.
- MNGP analyses and measure taken in response to the Giambusso letter dated September 7, 1973, Reference 8.
- AEC review and acceptance Safety Evaluation dated July 29, 1974, Reference 6.

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An additional MNGP HELB License Bases is the elimination of random breaks and cracks indicated in the above license documents. Revision 1 to Monticello License Amendment Request for Power Re-Rate, letter dated December 4, 1997, Reference 67, Exhibit E, paragraph 10.1 states, "Several previously postulated break locations were eliminated through the application of criteria established in Standard Review Plan 3.6.2 and Generic Letter 87-11". Exhibit E, Paragraph 10.6 gives the version of GL 87-11 used as June 19, 1987, Reference 21. This version of GL 87-11 contained the version of Branch Technical Position MEB 3-1 of SRP 3.6.2 used. This LAR provision was approved by the NRC in SER for power re-rate Amendment 102 dated September 16, 1998, Reference 66. The SER's approval was by reference, the SER's Reference 3.

Design Base Correspondence with NRC

In addition to the analysis that supports the Licensing Bases, the following subsequent correspondence concerning the Licensing Bases have occurred and supports the adequacy of the previous documented Licensing Bases adequacy which governs, but does not change the Licensing Bases. This correspondence is considered part of the HELB Design Bases.

The HELB issue was initiated by the Giambusso letter (Reference 2) with its enclosure "*General Information Required for Consideration of the Effects of a Piping System Break Outside Containment*." The letter is a "*request*" of Northern States Power (NSP) to provide "*analyses and other relevant information needed to determine the consequences of such an event, using the guidance provided in the enclosed general information request. … You should determine the applicability, for the Monticello facility, of the items listed in the enclosure*."

The Reference 2 enclosure's introductory paragraph states, "*applicants and licensees should determine on an individual basis the applicability of each of the following items for inclusion in their submittals*.

1. The systems (or portions of systems) for which protection against pipe whip is required should be identified. Protection from pipe whip need not be provided if any of the following conditions will exist:

(a) Both of the following piping system conditions are met:

(1) the service temperature is less than 200°F; and

(2) the design pressure is 275 psig or less."

The formal NSP response (Reference 8) to the Giambusso letter, submitted the Monticello HELB analysis report (dated August 1973), *Postulated Pipe Failures Outside Containment*, as part of the Monticello licensing basis. The Monticello August 1973 HELB report takes exception to the Giambusso letter's HELB selection criteria, as Section 3.0 specifically states:

"The analysis has been conducted using the following general procedures:

3.1 The high energy piping systems were identified using the criteria that the service temperature is greater than 200°F and the design pressure is greater than 275 psig."

After the AEC reviews were completed, the AEC transmitted their formal approval with related Safety Evaluation (SE) of the Monticello HELB analysis in Reference 6. Reference 6 specifically addresses the AEC acceptance of NSP's letter dated September 7, 1973 and its "*submitted report on Postulated Pipe Failures Outside Containment dated August, 1973".* The AEC SE states, "*Protection … should be provided … from ruptures in pipes carrying high energy fluid, where the temperature and pressure conditions of the fluid exceed 200°F and 275 psig, respectively*." The approved original licensing basis for Monticello's HELB selection criteria allows for an exception to the guidance in the Giambusso letter.

The Giambusso letter was written to have NSP address 10 CFR 50 Appendix A General Design Criterion 4 - *Environmental and dynamic effects design bases* with respect to HELB outside containment. Monticello was designed and built before the General Design Criteria (GDC) were issued. Therefore, Monticello was not designed as a GDC compliant plant. NRC NRR Office Instruction LIC-100, *Control of Licensing Bases for Operating Reactors* (Reference 9), provides guidance on how to interpret the licensing bases of older plants like Monticello with respect to meeting the GDC. LIC-100 Subsection 2.1.5.7, Appendix A, "General Design Criteria" states

> *"The General Design Criteria are not applicable to plants with construction permits issued prior to May 21, 1971. At the time of the promulgation of Appendix A, the Commission stressed that the GDC were not new requirements and were promulgated to more clearly articulate the licensing requirements and practice in effect at that time. While compliance with the intent of the GDC is important, each plant licensed before the GDC were formally adopted was evaluated on a plant specific bases, determined to be safe, and licensed by the Commission. Furthermore, current regulatory processes are sufficient to ensure that plants continue to be safe and comply with the intent of the GDC."*

The NRC SE acceptance letter (Reference 6) statement, *"Based on our review of the information you have submitted on the Monticello Nuclear Generating Plant, we have determined that after completion of the proposed modifications, the plant would withstand the consequences of postulated ruptures in high energy fluid piping outside the containment without loss of the capability to initiate and maintain the plant in a safe shutdown condition"*, demonstrates that Monticello complied with the intent of GDC 4.

The current licensing basis (CLB) requirement is for Monticello to comply with the intent of GDC 4 as described in the 1973 HELB analysis. 604000000250

Reference 10 reported the discovery of an applicable area of the turbine building that may have not been included in the 1973 HELB analysis. As part of resolution of that discovered condition, NSP thoroughly reviewed the original 1973 analysis. The results of that analysis are documented in the Reference 11 NUTECH Engineering report, and that report was submitted to the NRC in the Reference 25.

Section 3 of the NUTECH report defines the evaluation approach and how the Standard Review Plan (SRP) and ANSI/ANS-58.2 were used to clarify analytical assumptions. These helped to better define the assumptions for operating conditions, availability of off-site power, consideration of simultaneous events, single active failure, break locations, pipe whip effects, jet impingement effects and flooding effects.

Request for Additional Information (RAI) 1 from Reference 59 refers to the HELB selection criteria in Section 3.3 of the NUTECH report, and states, *"The selection criteria are less conservative than the Giambusso criteria (service water temperature exceeds 200°F or design pressure exceeds 275 psig) as well as the SRP 3.6.2 criteria (maximum operating temperature exceeds 200°F and maximum operating pressure exceeds 275 psig). Justify the deviation and provide a list of the piping systems that would be included in the Monticello HELB analysis if the Giambusso criteria were used and a list if the SRP 3.6.2 criteria were used."*

Reference 60 submitted the final NSP response to the HELB selection criteria issue in RAI 1. The response states:

> *'With one exception the criteria used for the selection of high energy lines was "operating temperature greater than 200°F and design pressure greater than 275 psig". This criteria is consistent with the criteria used in the original 1973 Monticello High Energy Line Break analysis. Section I.2-3 of Appendix I will be revised to reflect this selection criteria as part of the next annual update to the Monticello USAR.'*

USAR Appendix I has been updated to specifically include the HELB selection criteria (consistent with the criteria used in 1973 and 1986 analyses).

Although the NRC did not issue a formal SE, the NRC (in Reference 61) does acknowledge *"Additional information was provided June 18, 1986, December 4, 1986, September 25, 1987, December 14, 1987 (Reference 60 herein) January 25, 1988, and February 17, 1989." Reference 61 concludes that the "original non-conforming HELB conditions have been corrected"*.

Further review of the HELB criteria occurred with an extended power uprate (EPU) RAI response (Reference 62) with NRC review documented in Reference 63.

I.1.1 Definition of Terms

Circumferential Break - A pipe break perpendicular to the pipe axis with a break area equivalent to the internal cross-sectional area of the ruptured pipe.

Compartment - The collective term used to describe rooms, bays, hallways, etc., that makeup an enclosure in which a high energy pipe is routed or pipe break may take place.

Compartmental Pressurization - The condition of pressurization of a compartment above its normal pressure due to a high energy line break within the compartment.

Critical Crack - The single postulated crack on a high energy line that produces the most adverse effect on safe shutdown equipment within a compartment. Dimensionally the critical crack size is postulated to be one-half the pipe inside diameter in length by one-half the pipe thickness in width.

Equipment Qualification - The term used to describe the ability of equipment to remain functional in the abnormal temperature, pressure, humidity, and radiation environments generated by a postulated HELB within a compartment.

Flooding - The condition encountered within a compartment or adjoining compartments when liquid water or condensing steam escapes from a ruptured high energy line or other component at a faster rate than the fluid can be drained away.

High Energy Fluid - A fluid with a normal operating pressure greater than 275 psig and a normal operating temperature greater than 200°F. (Reference 2, 6, 8 and 64) 604000000250

High Energy System - Any mechanical system with piping containing high energy fluid.

High Energy Line - Any pipe which carries fluid with a normal operating pressure greater than 275 psig and a normal operating temperature greater than 200°F. (Reference 2, 6, 8 and 64) 604000000250

Jet Impingement - The forces exerted on targets by high pressure fluid exiting a break in the pipe.

Longitudinal Break - A pipe break parallel to the pipe axis and oriented at any point around the pipe circumference with a break area equal to the effective cross-sectional flow area immediately upstream of the break location.

Normal Operating Conditions - Plant operating conditions during reactor startup, operation at power, hot standby, or reactor cooldown to the cold shutdown condition.

Pipe Whip - The mechanical reaction caused by high energy fluid exiting a ruptured pipe.

Reactor/Plant Protective Actuation - A HELB that causes a Technical Specification specified trip. For HELB, this is a Containment Group Isolation and/or scram. (Reference 2, Errata) 604000000250

Safe Shutdown - Shutdown of the unit from power to subcritical condition with temperature in the reactor of less than or equal to 212°F, no resulting fuel damage, and no structural failure of primary or secondary containment.

Safe Shutdown System/Component - Systems and components required to safely shutdown the reactor and mitigate the consequences of a high energy line break without off-site power.

Single Active Failure - The malfunction or loss of function of an electrical or fluid system component, excluding passive failures, which rely on mechanical movement to complete its intended function upon demand.

Terminal End - Extremities of piping runs that connect to structures, components (e.g., vessels, pumps, valves), or pipe anchors that act as rigid constraints to piping motion and thermal expansion. A branch connection to a main piping run is a terminal end of the branch run, except where the branch run is classified as part of a main run in the stress analysis and is shown to have a significant effect on the main run behavior. In piping runs which are maintained pressurized during normal plant conditions for only a portion of the run (i.e., up to the first normally closed valve) a terminal end of such runs is the piping connection to this closed valve. (MEB 3-1, Footnote 3 – Reference 21) A normally closed valve is not considered a terminal end if the valve is part of the pipe stress analysis (which includes seismic) and the valve is not supported as an anchor. (Reference 64)

Volumes - The subdivision of the plant's buildings chosen to provide consistent identification of the compartments evaluated in this report. A similar but separate set of volume numbers is used for the Reactor Building and Turbine Building. See Figure I.1-1 through Figure I.1-10.

I.1.2 Assumptions

The following assumptions were used during the HELB evaluation in accordance with the evaluation criteria above:

I.1.2.1 A postulated break is assumed to occur during Normal Operating Conditions.

I.1.2.2 If Reactor/Plant Protective Actuation has occurred, then until initial vessel level has stabilized, which establishes adequate core cooling, loss of off-site power concurrent with the line break is assumed. Afterwards, off site power is assumed and shutdown is as directed by C.5 Emergency Operating Procedures. 604000000250

I.1.2.3 No fires or other simultaneous line breaks or accidents were considered.

I.1.2.4 If Reactor/Plant Protection Actuation has occurred, then until initial vessel level has stabilized, which establishes adequate core cooling, a single active component failure was assumed. Afterwards, single active failure is not assumed and shutdown is as directed by C.5 Emergency Operating Procedures. 604000000250

I.1.2.5 Circumferential breaks were assumed to result in complete pipe severance with full separation of the two severed pipe ends (i.e., guillotine). The break was assumed to be perpendicular to the longitudinal axis of the pipe. Circumferential breaks postulated at fittings were assumed to be at the fitting-to-pipe weld(s).

I.1.2.6 Longitudinal breaks were assumed to be oriented parallel to the longitudinal axis of the pipe oriented at any point around the pipe circumference with a break area equal to the effective cross-sectional flow area immediately upstream of the break location. Longitudinal breaks postulated at fittings were assumed to be at the center of the fitting.

I.1.2.7 Class I Systems, for which stress data was not available to determine break locations, were treated as Class II Systems.

I.1.2.8 A normally closed check valve in a piping line utilized as an inboard containment isolation valve was assumed to leak. The leakage was assumed to result in pressurization of that section of pipe out to the first normally closed valve outside the Primary Containment.

I.1.2.9 A normally closed check valve was assumed to remain closed in the event of a postulated line break between the check valve and the first normally closed valve outside the primary containment. For this condition, the check valve was assumed to be a passive component.

I.1.2.10 Breaks do not need to be postulated in sections of piping from containment up to the outboard isolation valve provided that the requirements of Branch Technical Position MEB 3-1 as identified in Standard Review Plan, NUREG-0800, 3.6.2 have been met. 604000000250

I.1.2.11 Break locations were postulated for all high energy piping with a nominal diameter greater than 1 inch, and a normal operation time exceeding 2% for the total unit operation time (Reference 25).

Table I.1-1 Volume Descriptions for HELBs in the Reactor Building

VOLUME	VOLUME DESCRIPTION
1	RHR and Core Spray Pump Room, Division I
2	RHR and Core Spray Pump Room, Division I Stairway
3	RHR and Core Spray Pump Room, Division II
4	RHR and Core Spray Pump Room, Division II Stairway
5	RCIC Room
6	Reactor Bldg Elevation 896′ Equipment and Floor Drain Tank
7	CRD Pump Room
8	HPCI Room
9	Suppression Pool Area - Northeast
10	Suppression Pool Area - Southeast
11	Suppression Pool Area - Southwest
12	Suppression Pool Area - Northwest
13	East Shutdown Cooling Room
14	CRD Hydraulic Control Unit Area - East 935′ Elevation
15	TIP Room
16	Steam Chase
17	TIP Drive Room
18	CRD Hydraulic Control Unit Area and HVAC Areas - NW 935′ El
19	CRD Hydraulic Control Unit Area and HVAC Areas - SW 935′ El
20	West Shutdown Cooling Room
21	PIPE Chase 974′
22	MCC and Standby Liquid Control System Area - East 962′ El
23	Contaminated Tool Storage - East 962′ El
24	Recirc M/G Set Airlock
25	962′ North of Reactor Shield Wall
26	Reactor Recirculation Pumps MG Set Room
27	Cooling Water Pump and Chiller Area - West 962′ El
28	RWCU Pump Room B and Hallway
29	RWCU Pump Room A
30	RWCU Heat Exchanger Area
31	RWCU Area Behind Hx Exchanger
32	RWCU Isolation Valve Room
33	MCC and Instrument Rack C-55 Area
34	CGCS-A Recombiner Area
35	Cooling Water Heat Exchanger and CGCS-B Recombiner Area
36	Standby Gas Treatment System B - Train Room
37	Standby Gas Treatment System Fan Room
38	Standby Gas Treatment System Airlock
39	Standby Gas Treatment System A - Train Area
40	Reactor Plenum Room
41	Reactor Recirculation MG Set Fan Room
42	Corridor Outside Main Exhaust Plenum
43	Skimmer Surge Tank and Fuel Pool Pumps Area
44	Snubber Rebuild and Decontamination Area
45	Northeast Stairway 1001′ El
46	Contaminated Equipment Storage Area
47	Northwest Stairway 1001′ El
48	Refueling Floor 1028′ El

Table I.1-2 Volume Descriptions for HELBs in the Turbine Building

VOLUME	VOLUME DESCRIPTION
1	Motor Control Center B-33A & B, and B-12
2	Turbine Building southeast Corner near MCC B-33
3	Lube Oil Reservoir and Reactor Feed Pump Area
4	Lube Oil Storage Tank Room
5	Turbine Building Corridor Northeast 911′ El.
6	Water Box Scavaging System Area
7	Turbine Building Sump and MCC B-31 Area
8	4KV and Load Center Division A
9	Hallway outside Air Ejector Room Entrance Door 15
10	Hydrogen Seal Oil Unit and Condensate Pump Area North
11	Hydrogen Seal Oil Unit and Condensate Pump Area South
12	Mechanical Vacuum Pump Area
13	Condensate Backwash - Receiving Tank Area
14	Air Ejector Room
15	Turbine Basement Condenser Area
16	Pipe Tunnel to Intake
17	Intake Entry Area
18	Intake Structure Pump Room
19	Circ Water Pump Area
20	Turbine Building 931′ El. East
22	Turbine Building 931′ El Vent Chase
27	Turbine Building Corridor Northwest and Hallway to No. 11 D.G. Entry 931′ El
36	Turbine Building Northwest Stairway from 941′ to 951′ El
33	Valve Operating Gallery and Condensate Demin Panels Area
21	FW Pipe and Cable Tray Penetration Room
23	Auxiliary Boiler Room
24	East Electrical Equipment Room and 13 EDG
25	13.8kV Switchgear Rooms
26	Turbine Building Corridor Southeast Corner 931′ El
35	Cable Chase 941′ El
29	No. 11 Diesel Generator Room
28	No. 11 Diesel Generator Room Entry Area
30	No. 12 Diesel Generator Room
32	Stator Water Cooling Area
31	4KV and Load Center Division B
39	Reactor Building Plenum Room
34	Turbine Building Railroad Car Shelter
38	Reactor Building Steam Chase
37	Turbine Deck 951′ to 1004′ El

I.2 High Energy Systems and Piping

High energy piping was assumed to be all piping with a normal operating temperature exceeding 200°F and a design pressure exceeding 275 psig (Reference 6 and 8). If the actual operating temperature was not known, the design temperature was conservatively substituted. All systems having piping containing high energy fluids are classified as high energy systems, and those portions of the piping containing the high energy fluid are identified as high energy lines. 60400000250

a. Main Steam

b. Feedwater

c. Condensate

d. HPCI (steam)

e. RCIC (steam)

f. RWCU (Reactor Water Clean-up)

g. Instrument and sample lines (identified with associated systems in Table I.2-1)

h. Core Spray

i. Residual Heat Removal

j. HPCI (injection)

k. RCIC (injection)

l. Standby Liquid Control

m. Off-gas

n. Control Rod Drive

o. GEZIP (Zinc Oxide Injection)

I.2.1 High Energy Line and System Descriptions

The primary high energy lines are described by system in the following paragraphs. These descriptions identify the piping containing the high energy fluids, the compartments containing the high energy lines, the boundaries of the high energy portions of the piping, and the justifications for excluding any of the piping from further HELB consideration.

For convenience, the identification of specific plant areas, compartments, and rooms, was accomplished by subdividing the major buildings of the plant into discrete volumes for use in the HELB analysis. A list of volumes is provided in Tables I.1-1 and I.1-2.

I.2.1.1 Main Steam System

The high energy lines for the Main Steam System are located in 4 compartments; the main steam chase (Volume 16) in the Reactor Building, the condenser area (Volume 15), the steam jet air ejector (SJAE) room (Volume 14) and the turbine operation floor (Volume 37) in the Turbine Building.

The high energy lines on the Main Steam System include the four main steam lines, the equalization lines, the turbine bypass lines to the condenser, Main Steam to SJAE line, main steam to steam seal system, and the main steam drain lines.

The main steam line boundary points are containment penetrations X-7A, X-7B, X-7C and X-7D and the four nozzles to the high pressure turbine. Piping between the containment penetrations and the main steam outboard isolation valves has been evaluated per the criteria defined in I.3.1.1(3)(b). There are also terminal points at the inlet and outlet to the turbine stop and control valves for the four main steam lines. The equalization lines are considered branch lines from the four main steam lines. Therefore, their boundaries are at the tees to the main steam lines. The turbine bypass line boundaries are at the tees into the main steam lines, the penetrations at the condenser, and terminal points at the bypass control valves. The main steam to the SJAE’s line has boundaries at the tees to the main steam lines and at the SJAE pressure reducing station in the SJAE room. The steam seal regulator lines have boundaries on the steam bypass line and steam seal regulator. The main steam drain lines have boundaries at the outboard Primary Containment isolation valve, MO-2374, and at the 6-inch reducer just downstream of valve MO-2565.

I.2.1.2 Feedwater System

The high energy Feedwater System piping (FW2A-14″-ED and FW2B-14″-ED) begin in the feedwater pump area (Volume 3) at the discharge nozzles of feed pumps P-2A and P-2B. The two main feedwater lines, FW2A-14″ and FW2B-14″, pass through southeast corner of the Turbine Building (Volume 2), up into the Turbine Building pipe chase area (Volume 21) and then into the condenser area (Volume 15). Before entering the Reactor Building steam chase (Volume 16) and terminating at containment penetrations X-9A and X-9B, each feedwater line is connected to the respective high pressure feedwater heaters (E-14A & B and E-15A & B) on the turbine operating floor (Volume 37). Piping between the containment penetrations and the primary containment isolation valves FW-94-1 and FW-94-2 has been evaluated per the criteria defined in I.3.1.1(3)(b). The feedwater regulating station piping, FW2-6″-DE and FW2-14″-DE, is terminated at both of the main feedwater piping lines and is located in Volumes 2 and 3. The feedwater pumps recirculation lines to the condenser, FW3-8″-DE and FW4-8″-DE, are routed from the discharge of each feedwater pump to the condenser inlet nozzles. Their high energy boundaries are the feedwater lines and the condenser. These lines are routed from Volume 3 through the wall into the condenser area (Volume 15) to the condenser. The other feedwater piping, FW10-8″-DE and FW11-14″-ED are located in the condenser area (Volume 15). Their boundaries are the main feedwater lines and valve FW-88-3.

I.2.1.3 Condensate System

The high energy Condensate System lines are located in the condenser area (Volume 15), Turbine Building pipe chase area (Volume 21) and reactor feedwater pump area (Volumes 2 and 3). The high energy Condensate System piping includes the main condensate system piping (C4A-16″-GB and C4B-16″-GB) and the condensate cross-tie (C7-16″-GB). The boundaries of the condensate lines are the outlet nozzles of the second stage intermediate heaters E-13A & E-13B and the suction nozzles of the feedwater pumps P-2A and P-2B. The normal operating pressure of the condensate lines is <275 psig and is therefore not high energy. The piping down stream of the E-13 heaters is conservatively classified as high energy. 60400000250

I.2.1.4 HPCI (Steam) System

The high energy piping on the HPCI system is the steam supply line (PS18-8″-ED). The HPCI steam supply line is routed from containment penetration X-11 to the HPCI steam supply valve, MO-2036, located near the HPCI turbine. Piping between containment penetration X-11 and the outboard primary containment isolation valve MO-2035 has been evaluated per the criteria defined in I.3.1.1(3)(b). The remaining portions of this line, from the MO-2036 to the HPCI turbine require no further HELB evaluation because this portion of the line is pressurized less than 2% of the time.

The HPCI steam supply line begins in the main steam chase (Volume 16), is routed through the torus area (Volume 12) and terminates in the HPCI compartment (Volume 8).

I.2.1.5 RCIC (Steam) System

The only high energy line for the RCIC (Steam) System is the steam supply line PS17-3″-ED. This line is routed from the containment penetration X-10, to the turbine steam supply valve, MO-2078. Piping between containment penetration X-10 and the outboard primary containment isolation valve MO-2076 has been evaluated per the criteria defined in I.3.1.1(3)(b). The remaining portion of this line, from MO-2078 to the RCIC turbine require no further HELB evaluation because this portion of the line is pressurized less than 2% of the time.

The RCIC steam supply line begins in the main steam chase (Volume 16), is routed through the torus area (Volume 9), and terminates in the RCIC compartment (Volume 5).

I.2.1.6 Reactor Water Cleanup (RWCU) System

The high energy piping RWCU System consists of the RWCU supply line, from the primary containment penetration X-14 through the heat exchangers (REW3-4″-DBD/DCD/DB), the bypass piping around the regenerative heat exchanger (REW3-4″-DBD), the cross-tie from the supply line to valve RC-103, (REW3-4″-DBD), and the return line, from the regenerative heat exchanger to the HPCI and RCIC injection lines (REW6-3″ DBD/DC/ED). The piping is routed from the primary containment penetration in the RWCU area (Volume 32) through the heat exchangers in the RWCU area (Volume 28, 29, 30 and 31), through the Reactor Building at 962 foot elevation (Volume 27) and the MG set room (Volume 26), through the west side of the Reactor Building at the 935 foot elevation (Volume 18) and terminates back in the main steam chase (Volume 16). Piping between containment penetration X-14 and the outboard primary containment isolation valve MO-2398 has been evaluated per the criteria defined in I.3.1.1(3)(b).

I.2.1.7 Instrument and Sample Lines

The instrument sensing lines from the primary cooling system and reactor vessel to the instrumentation represent high energy lines for the portion of the routing outside of the Primary Containment. Since all of these lines are 1 inch or smaller in nominal size, neither circumferential breaks nor longitudinal breaks are required to be postulated. In addition, all lines are equipped with excess flow check valves which would mitigate any break within a few seconds of the break occurring.

All instrument and sample lines, which tee from other high energy lines, are identified and evaluated with the specific system.

I.2.1.8 Core Spray System

Within the Core Spray (CS) System, only a small portion of each injection line (TW7-8″-ED and TW11-8″-ED) was determined to be high energy piping. The portion of the injection lines identified as high energy, runs from the containment penetration out to the inboard containment isolation valves, MO-1753 and MO-1754 respectively. Line TW7-8″-ED from primary containment penetration X-16A to valve MO-1754 is located entirely within the reactor water cleanup compartment (Volume 31). The remaining Core Spray line TW11-8″-ED from primary containment penetration X-16B to valve MO-1753, is located within a block wall compartment (Volume 21) above the Reactor Building 962′ level. Both lines have been evaluated to meet the criteria defined in I.3.1.1(3)(b). The remaining portions of these lines from the inboard containment isolation valves to the Core Spray pump discharge nozzles were determined to require no further HELB evaluation because these lines are not pressurized more than 2% of the time. Portions of core spray piping have design temperatures that have increased from 196.7°F to 212°F as a result of EPU. Thus these lines are now classified as HELB lines, but require no further HELB evaluation because these lines are not pressurized more than 2% of the time.

I.2.1.9 Residual Heat Removal (RHR) System

The only portions of the RHR System considered high energy piping are on the LPCI injection lines (TW20-16″-DB and TW30-16″-DB) from the primary containment penetrations to the inboard containment isolation valves MO-2015 and MO-2014, respectively, and the shutdown cooling suction line REW10-18″-DB from the containment penetration to the outboard containment isolation valve MO-2030. Line TW20-16″-DB is located in the west shutdown cooling room (Volume 20) and terminates at penetration X-13A and valve MO-2015. Line TW30-16″-DB is located in the east shutdown cooling room (Volume 13) and terminates at penetration X-13B and valve MO-2014. Line REW10-18″-DB is located in Volume 13 and terminates at penetration X-12 and valve MO-2030. All of these lines have been evaluated per the criteria defined in I.3.1.1(3)(b). The remaining portions of these lines, from the inboard containment isolation valves to the RHR pump discharge nozzle, require no further HELB evaluation because these lines are pressurized less than 2% of the time.

The remaining RHR System lines were determined to not require further HELB evaluation. All of the piping comprising the shutdown cooling mode of RHR were identified as exempt from HELB evaluations because it is pressurized less than 2% of the time (Reference 12). Portions of core spray piping have design temperatures that have increased from 196.7°F to 212°F as a result of EPU. Thus these lines are now classified as HELB lines, but require no further HELB evaluation because these lines are not pressurized more than 2% of the time.

I.2.1.10 HPCI (Water) System

The high energy portion consists of the HPCI injection line, TW3-12″-ED, from the HPCI injection line valve (MO-2068) to the feedwater line. This portion of the line is located in the steam chase area (Volume 16). All other portions of the HPCI injection line are pressurized less than 2% of the time so they do not require further evaluation.

I.2.1.11 RCIC (Water) System

The high energy portion consists of the RCIC injection line, FW5-4″-ED, from the normally closed RCIC injection valve (MO-2107) to the Feedwater line. This portion of the line is located in the steam chase area (Volume 16). All other portions of the RCIC injection line and full flow test line are pressurized less than 2% of the time, so they do not require further evaluation.

I.2.1.12 Standby Liquid Control

Line CH2-1 1/2″-DC from containment penetration X-42 to check valve XP-6 is the only high energy portion of this system. It is located in the east shutdown cooling room (Volume 13). A terminal end break at the containment penetration X-42 has been evaluated per the criteria defined in I.3.1.1(3)(a).

I.2.1.13 Off-gas System

For the Off-gas System, the normal operating temperatures and pressures were used to determine the high energy lines on the system. However, no lines of the Off-gas System were excluded from the list of high energy piping based upon an operating pressure less than 275 psig.

With the exception of lines SHP101-4″ and GOLE-107-6″, all of the high energy lines in the Off-gas System are located in the Off-gas Storage and Compressor Building.

The steam supply line, SHP101-4″ is routed from the condenser area (Volume 15) through the SJAE Room (Volume 14). One boundary for this line is the tee to line PS9-3″-ED and the other is in the Off-gas Storage and Compressor Building. This line is the discharge line for safety valve PSV-7474B, and hence this line would only be pressurized if the PSV lifted. Line GOLE-107-6″ was excluded from consideration under the 2% criterion.

I.2.1.14 Control Rod Drive System

Within the CRD system, only a small portion of the CRD return line (CRD7-3″-DE) between check valve CRD-31 and the tee connection to the RWCU return line (REW6-3″-DC) is determined to be high energy piping. This line is located in the Reactor Building 935 foot floor (Volume 18). The remaining Scram Discharge Volume and associated piping are pressurized above 275 psig only after scram initiation. Therefore, these lines are excluded based on the 2% usage criteria (References 13 and 14).

I.2.1.15 Zinc Oxide Injection System (GEZIP)

The system consists of three 1 1/2 inch lines with one on the suction pipe of each feedwater pump and the third on feedwater pump P-2B discharge line. The entire GEZIP is considered high energy piping located in the feedwater pump area (Volume 3).

I.2.2 High Energy Line Designation List

Table I.2-1 provides a list of all high energy lines considered in this evaluation. In addition to identifying the high energy lines for each system considered, information is provided for each line's location, terminal points, normal operating or design conditions. Information for each HE line within a compartment was compiled by site inspection, and recorded in survey books (References 15 and 16).

The following descriptions identify the information provided in each column of Table I.2-1:

Column 1	System: Identifies the name of the plant system containing HE line.
Column 2	P&ID: Identifies the "M" number or drawing number for the P&ID containing the line.
Column 3	Volume: Identifies the volume in the Reactor Building (RB) or Turbine Building (TB) where the HE line is located. Volumes for piping excluded from HELB evaluation based the 2% criteria may not be shown.
Column 4	Line Designation: Identifies plant line number of HE line.
Columns 5 & 6	From/To: Identifies the segment of the HE line by official plant number. The From/To points may be a pump, heat exchanger, another pipe, tank, etc.
Columns 7-8	Design or Normal Conditions: Identifies the design pressure and normal operating temperature against which the line was identified as a HE line. If normal operating temperature is not known, the design temperature was conservatively used. The table is conservatively based on design pressure rather than operating pressure. 604000000250
Columns 9-10	HELB breaks and cracks postulated in Section I.3 are not postulated in HE lines that have the following exclusions: The exclusions included piping with operating pressure above 275 psig or its operating temperature above 200°F for less than 2% of normal operating time (Reference 21 - MEB 3-1, B.2.e). Moderate energy lines are not postulated per section I.3.1.3 (Reference 64). Exception to the 2% rule is noted in the table where used. 604000000250 "SP" (superpipe) was used to designate that breaks were not required to be postulated per section I.1.2.10. This portion of piping has been evaluated per the criteria defined in Section I.3.1.1(3)(b). Small diameter piping, 1 inch and less, is not included in this table.

Table I.2-1 High Energy Piping List

HIGH ENERGY LINE DESIGNATION LIST									
System	P&ID(s)	Volume(s)	Line Designation	From	To	Press psig	Temp F°	Exclusion(s) 2%	SP
Main Steam	M-115	RB-16	PS1-18-ED	X-7A	AO-2-86A	1110	582		X
Main Steam	M-115/102	RB-16 & TB-15	PS1-18-ED	AO-2-86A	SV1	1110	582		
Main Steam	M-115	RB-16	PS2-18-ED	X-7B	AO-2-86B	1110	582		X
Main Steam	M-115/102	RB-16 & TB-15	PS2-18-ED	AO-2-86B	SV2	1110	582		
Main Steam	M-115	RB-16	PS3-18-ED	X-7C	AO-2-86C	1110	582		X
Main Steam	M-115/102	RB-16 & TB-15	PS3-18-ED	AO-2-86C	SV3	1110	582		
Main Steam	M-115	RB-16	PS4-18-ED	X-7D	AO-2-86D	1110	582		X
Main Steam	M-115/102	RB-16 & TB-15	PS4-18-ED	AO-2-86D	SV4	1110	582		
Main Steam	M-102	TB-15 & TB-37	PS1-20	CV1	HP Turbine G-1	1110	582		
Main Steam	M-102	TB-15	PS2-20	CV2	HP Turbine G-1	1110	582		
Main Steam	M-102	TB-15	PS3-20	CV3	HP Turbine G-1	1110	582		
Main Steam	M-102	TB-15 & TB-37	PS4-20	CV4	HP Turbine G-1	1110	582		
Main Steam	M-102	RB-16	PS1-10-ED	PS1-18-ED	Cap	1110	582		
Main Steam	M-102	RB-16	PS2-10-ED	PS2-18-ED	Cap	1110	582		
Main Steam	M-102	RB-16	PS3-10-ED	PS3-18-ED	Cap	1110	582		
Main Steam	M-102	RB-16	PS4-10-ED	PS4-18-ED	Cap	1110	582		
Main Steam	M-102	TB-15	PS6-8-ED	BPCV12	E-1A	1110	582		
Main Steam	M-102	TB-15	PS8-8-ED	BPCV11	E-1A	1110	582		
Main Steam	M-102	TB-15	PS7-10-ED	PS11-6-ED	Cap	1110	582		
Main Steam	M-102	TB-15	PS7-10-ED	PS11-6-ED	PS7-8-ED	1110	582		
Main Steam	M-102	TB-15	PS7-10-ED	PS11-6-ED	PS10-5	1110	582		
Main Steam	M-102	TB-15	PS7-8-ED	PS7-10-ED	BPCV11	1110	582		
Main Steam	M-102	TB-15	PS7-8-ED	PS7-10-ED	BPCV12	1110	582		
Main Steam	M-102/104-1	TB-15/14	PS9-3-ED	PS1-18-ED	PS9-2-ED	1110	582		
Main Steam	M-102/104-1	TB-15/14	PS9-3-ED	PS2-18-ED	PS9-2-ED	1110	582		
Main Steam	M-104-1	TB-14	PS9-2-ED	PS9-3-ED	MS-23-1	1110	582		
Main Steam	M-104-1	TB-14	PS9-2-ED	PS9-3-ED	MS-24-1	1110	582		
Main Steam	M-104-1	TB-14	PS9-2-ED	PS9-3-ED	MS-23-2	1110	582		

Table I.2-1 High Energy Piping List (Cont'd)

HIGH ENERGY LINE DESIGNATION LIST									
System	P&ID(s)	Volume(s)	Line Designation	From	To	Press psig	Temp F°	Exclusion(s)	
								2%	SP
Main Steam	M-104-1	TB-14	PS9-2-ED	PS9-3-ED	MS-24-2	1110	582		
Main Steam	M-102	TB-15	PS11-6-ED	PS1-18-ED	PS7-10-ED	1110	582		
Main Steam	M-102	TB-15	PS12-6-ED	PS2-18-ED	PS7-10-ED	1110	582		
Main Steam	M-102	TB-15	PS13-6-ED	PS3-18-ED	PS7-10-ED	1110	582		
Main Steam	M-102	TB-15	PS14-6-ED	PS4-18-ED	PS7-10-ED	1110	582		
Main Steam	M-115	RB-16	PS15-3-ED	X-8	MO-2374	1110	582		X
Main Steam	M-115/102	RB-16	PS15-3-EB	MO-2374	Support SR-721	1110	582		
Main Steam	M-115/102	RB-16	PS15-3-EB	Support SR-721	MO-2565	1110	582		
Main Steam	M-102	RB-16	PS15-3-EF	MO-2565	3″x 6″ Red. Elbow	600	490		
Main Steam	M-102	RB-16/TB-15	D4-6-EF/EB	3″x 6″ Red. Elbow	Condenser E-1A	600	490		
Main Steam	M-102	TB-15	PS21-4-ED	PS7-10-ED	RO-1618	1110	582		
Main Steam	M-102	TB-15	PS30-18-EBD	PS1-18-ED	PS2-18-ED	1110	582		
Main Steam	M-102	TB-15	PS30-18-EBD	PS1-18-ED	PS3-18-ED	1110	582		
Main Steam	M-102	TB-15	PS30-18-EBD	PS1-18-ED	PS4-18-ED	1110	582		
Main Steam	M-102	TB-15	PS30-10-EBD	PS30-18-EBD	Cap	1110	582		
Main Steam	M-102	TB-15	MS-8 Branch	PS30-10-EBD	MS-8	1110	582		
Main Steam	M-102	RB-16	D1-1.5-ED	PS1-10-ED	D4-2-ED	1110	582		
Main Steam	M-102	RB-16	D2-1.5-ED	PS2-10-ED	D4-2-ED	1110	582		
Main Steam	M-102	RB-16	D3-1.5-ED	PS3-10-ED	D4-2-ED	1110	582		
Main Steam	M-102	RB-16	D4-1.5-ED	PS4-10-ED	D4-2-ED	1110	582		
Main Steam	M-102	RB-16	D4-2-ED	Cap	MO-2564	1110	582		
Main Steam	M-102	RB-16	D4-2-EB	MO-2564	PS15-3-EB	1110	582		
Main Steam	M-102	TB-15	D23-2-EB	D23-1-EB	MO-1197	1110	582		
Main Steam	M-102	TB-15	D40-1.5-EBD	PS30-10-EBD	MO-4000	1110	582		
Main Steam	M-102	TB-15	D40-1.5-EF	MO-4000	D4-6-EF	600	490		
Main Steam	M-102	TB-15	D40-1.5-EBD	PS30-10-EBD	RO-4001	1110	582		
Main Steam	M-102	TB-15	D40-1.5-EF	RO-4001	D4-6-EF	600	490		
Main Steam	M-102/103	TB-15	PS10-5	PS7-10-ED	MO-1046	1110	582		
Main Steam	M-102/103	TB-15	PS10-3	PS10-5	MO-1045	1110	582		
Main Steam	M-103	TB-15	PS10-3	MO-1045	Steam Seal Regulator	1110	582		

APPENDIX I EVALUATION OF HIGH ENERGY LINE BREAKS OUTSIDE CONTAINMENT

Table I.2-1 High Energy Piping List (Cont'd)

HIGH ENERGY LINE DESIGNATION LIST									
System	P&ID(s)	Volume(s)	Line Designation	From	To	Press psig	Temp F°	Exclusion(s) 2%	SP
Condensate	M-105/106	TB-1514‘	C4A-16-GB	Heater E-12A	Heater E-13A	450	250	(1)	
Condensate	M-106	TB-15/21/23	C4A-16-GB	Heater E-13A	P-2A	450	321		
Condensate	M-106	TB-3	C8A-1.5-DE	C4A-16-GB	FW-179-1	450	321		
Condensate	M-105/106	TB-15	C4B-16-GB	Heater E-12B	Heater E-13B	450	250	(1)	
Condensate	M-106	TB-15/21/2/3	C4B-16-GB	Heater E-13B	P-2B	450	321		
Condensate	M-106	TB-3	C8B-1.5-DE	C4B-16-GB	FW-179-2	450	321		
Condensate	M-106	TB-2	C7-16-GB	C4A-16-GB	C4B-16-GB	434	321		
Feedwater	M-106	TB-2	FW2-14-DE	FW2A-14-DE	FW2B-14-DE	1550	323		
Feedwater	M-106	TB-2	FW2-6-DE	FW2-14-DE	FW2A-14-DE	1550	323		
Feedwater	M-106	TB-2	FW2-6-DE	FW2-14-DE	FW2B-14-DE	1550	323		
Feedwater	M-106	TB-3/2/21/15/37	FW2A-14-DE	P-2A	E-14A	1550	323		
Feedwater	M-106	TB-37/15	FW2A-14-DE	E-14A	E-15A	1550	357		
Feedwater	M-106	TB-37	FW2A-14-DE	E-15A	MO-1614	1550	410		
Feedwater	M-106/115	TB-37/25 & RB-16	FW2A-14-ED	MO-1614	FW-91-1	1250	410		
Feedwater	M-115	RB-16	FW2A-14-ED	FW-91-1	FW-94-1	1250	540		X
Feedwater	M-115	RB-16	FW2A-14-ED	FW-94-1	X-9B	1250	540		X
Feedwater	M-106	TB-3	V72B-1.5-DE	FW2B-14-DE	FW-199	1550	323		
Feedwater	M-106	TB-42/15	FW2B-14-DE	E-14B	E-15B	1550	357		
Feedwater	M-106	TB-37	FW2B-14-DE	E-15B	MO-1615	1550	410		
Feedwater	M-106/115	TB-37/15 & RB-16	FW2B-14-ED	MO-1615	FW-94-2	1250	410		
Feedwater	M-115	RB-16	FW2B-14-ED	FW-91-2	FW-94-2	1250	540		X
Feedwater	M-115	RB-16	FW2B-14-ED	FW-94-2	X-9A	1250	540		X
Feedwater	M-106/105	TB-3/15	FW3-8-DE	FW2A-14-DE	CV-3489	1550	323	X	
Feedwater	M-106/105	TB-3/15	FW4-8-DE	FW2B-14-DE	CV-3490	1550	323	X	
Feedwater	M-106	TB-37	FW10-8-DE	FW2A/B-14-DE	FW-88-1,2	1550	305		
Feedwater	M-106	TB-37/15	FW10-8-DE	FW-88-1,2	FW-88-3	1550	305		
Feedwater	M-106	TB-15	FW11-14-ED	FW2A-14-ED	FW2B-14-ED	1250	400		

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Table I.2-1 High Energy Piping List (Cont'd)

HIGH ENERGY LINE DESIGNATION LIST									
System	P&ID(s)	Volume(s)	Line Designation	From	To	Press psig	Temp F°	Exclusion(s) 2%	SP
Feedwater	M-106	TB-3/2	FW12A-1.5-DE	FW2A-14-DE	P-2A	1550	323		
Feedwater	M-106	TB-3	FW-66-1; 1.5″ Branch	FW12A-1.5-DE	FW-66-1	1550	323		
Feedwater	M-106	TB-3	FW12B-1.5-DE	FW2B-14-DE	P-2B	1550	323		
Feedwater	M-106	TB-3	FW-66-2; 1.5″ Branch	FW12B-1.5-DE	FW-66-2	1550	323		
CRD	M-118/128	RB-18	CRD7-3-DE	REW6-3-DC	1‘ west CRDH-44	1377	540		
CRD	M-118		CRD8-2-DC	CRD3-2-DC	CRD20A-1-DC	1785	570	X	
CRD	M-118		CRD8-2-DC	CRD3-2-DC	CRD20B-1-DC	1785	570	X	
CRD	M-118		CRD8-2-DC	CRD3-2-DC	CRD20C-1-DC	1785	570	X	
CRD	M-118		CRD8-2-DC	CRD3-2-DC	CRD20D-1-DC	1785	570	X	
CRD	M-118		CRD8-2-DC	CRD3-2-DC	Reducer	1785	570	X	
CRD	M-119		CRD13A-4-DB	CRD16A-12-CCD	CRD17A-1-CCD	1265	575	X	
CRD	M-119		CRD13B-4-DB	CRD16B-12-CCD	CRD17B-1-CCD	1265	575	X	
CRD	M-119		CRD14A-6-DB	CRD16A-12-CCD	CRD17A-1-CCD	1265	575	X	
CRD	M-119		CRD14B-6-DB	CRD16B-12-CCD	CRD17B-1-CCD	1265	575	X	
CRD	M-119		CRD15A-4-DB	CRD16A-12-CCD	CRD17A-1-CCD	1265	575	X	
CRD	M-119		CRD15B-4-DB	CRD16B-12-CCD	CRD17B-1-CCD	1265	575	X	
CRD	M-119		CRD16A-12-CCD	CRD16A-24-CCD	Blind Flange	1265	575	X	
CRD	M-119		CRD16A-24-CCD	CRD16A-12-CCD	CRD16A-2-CCD	1265	575	X	
CRD	M-119		CRD16A-2-CCD	CRD16A-24-CCD	CV3-33C	1265	575	X	
CRD	M-119		CRD16B-12-CCD	CRD16B-24-CCD	Blind Flange	1265	575	X	
CRD	M-119		CRD16B-24-CCD	CRD16B-12-CCD	CRD16B-2-CCD	1265	575	X	
CRD	M-119		CRD16B-2-CCD	CRD16B-24-CCD	CV3-33D	1265	575	X	
RHR	M-120		TW19-10-GE	TW19-14-GE	P-202B	510	330	X	
RHR	M-120		TW19-10-GE	TW19-14-GE	P-202D	510	330	X	
RHR	M-120		TW19-14-GE	TW19-10-GE	E-200B	510	330	X	
RHR	M-121		TW29-10-GE	TW29-14-GE	P-202A	510	330	X	
RHR	M-121		TW29-10-GE	TW29-14-GE	P-202C	510	330	X	
RHR	M-121		TW29-14-GE	TW29-10-GE	E-200A	510	330	X	
RHR	M-120		TW20-14-GE	E-200B	TW19-14-GE	510	330	X	
RHR	M-120		TW20-14-GE	E-200B	TW20-16-GE	510	330	X	
RHR	M-120		PAS-57-4 Branch	TW20-14-GE	PAS-57-4	510	330	X	

Table I.2-1 High Energy Piping List (Cont'd)

HIGH ENERGY LINE DESIGNATION LIST									
System	P&ID(s)	Volume(s)	Line Designation	From	To	Press psig	Temp F°	Exclusion(s) 2%	SP
RHR	M-120		TW20-16-GE	TW20-14-GE	MO-2013	510	330	X	
RHR	M-120		RHR-59-2 Branch	TW20-16-GE	RHR-59-2	510	330	X	
RHR	M-120		TW20-16-DB	MO-2013	MO-2015	1248	562	X	
RHR	M-120	RB-20	TW20-16-DB	MO-2015	X-13A	1248	562		X
RHR	M-120		TW21-2-GE	TW19-10-GE	RO-1999	510	330	X	
RHR	M-120		TW21-2-GE	TW19-10-GE	RO-2001	510	330	X	
RHR	M-120/121		TW22-16-GE	TW20-16-GE	TW22-14-GE	510	330	X	
RHR	M-120/121		TW22-14-GE	TW22-16-GE	TWH-168	510	330	X	
RHR	M-120/121		TW22-14-GE	TWH-168	TW22-16-GE	510	330	X	
RHR	M-120/121		TW22-16-GE	TW22-14-GE	TW30-16-GE	510	330	X	
RHR	M-121		TW30-14-GE	E-200A	TW29-14-GE	510	330	X	
RHR	M-121		RHR-79 Branch	TW30-14-GE	RHR-79	510	330	X	
RHR	M-121		TW30-14-GE	E-200A	TW30-16-GE	510	330	X	
RHR	M-121		PAS-57-5 Branch	TW30-14-GE	PAS-57-5	510	330	X	
RHR	M-121		TW30-16-GE	TW30-14-GE	MO-2012	510	330	X	
RHR	M-121		TW30-16-DB	MO-2012	MO-2014	1248	562	X	
RHR	M-121	RB-13	TW30-16-DB	MO-2014	X-13B	1248	562		X
RHR	M-121		TW31-2-GE	TW29-10-GE	RO-1998	510	330	X	
RHR	M-121		TW31-2-GE	TW29-10-GE	RO-2000	510	330	X	
RHR	M-121		TW36-4-GE	TW30-14-GE	Capped End	510	330	X	
RHR	M-121		TW37-4-GE	TW22-14-GE	MO-2407	510	281	X	
RHR	M-121	RB-13	REW10-18-ED	X-12	MO-2030	1248	570		X
RHR	M-121/135		FPW12-8-GE	TW22-14-GE	PC-18	510	263	X	
RHR	M-121/112		SW9-8-GE	TW33-12-GE	SW9-8-GF	510	263	X	
RHR	M-120		D80-2-GE	TW19-10-GE	RHR-30-2	275	281	X	
RHR	M-120		D81-2-GE	TW19-10-GE	RHR-30-4	275	281	X	
RHR	M-121		D84-3-GE	TW29-10-GE	RHR-47	510	330	X	
RHR	M-120		RV51-2-GE	TW20-16-GE	RV-2005	275	263	X	
RHR	M-121		RV52-2-GE	TW30-14-GE	RV-2004	275	263	X	
RHR	M-114-1		SC10-3-GC	CST-86	Branches	510	263	X	
RHR	M-114-1		SC10A-2-GC	SC10-3-GC	SC10A-2-GE	510	263	X	

Table I.2-1 High Energy Piping List (Cont'd)

HIGH ENERGY LINE DESIGNATION LIST									
System	P&ID(s)	Volume(s)	Line Designation	From	To	Press psig	Temp F°	Exclusion(s) 2%	SP
RHR	M-114-1/121		SC10A-2-GE	SC10A-2-GC	TW30-16-GE	510	263	X	
RHR	M-114-1		SC10B-2-GC	SC10-3-GC	SC10B-2-GE	510	263	X	
RHR	M-114-1/121		SC10B-2-GE	SC10B-2-GC	REW10-18-HE	510	263	X	
RHR	M-114-1		SC10C-2-GC	SC10-3-GC	SC10C-2-GE	510	263	X	
RHR	M-114-1/121		SC10C-2-GE	SC10C-2-GC	TW33-12-GE	510	263	X	
RHR	M-114-1		SC10D-2-GC	SC10-3-GC	SC10D-2-GE	510	263	X	
RHR	M-114-1/120		SC10D-2-GE	SC10D-2-GC	TW20-16-GE	510	263	X	
RHR	M-114-1		SC29-3-GC	CST-103-2	Branches	510	263	X	
RHR	M-114-1		SC29-2-GC	SC29-3-GC	SC29-2-GE	510	263	X	
RHR	M-114-1/122		SC29-2-GE	SC29-2-GC	TW7-10-GE	510	263	X	
RHR	M-114-1		SC29B-2-GC	SC29-3-GC	SC29B-2-GE	510	263	X	
RHR	M-114-1/120		SC29B-2-GE	SC29B-2-GC	TW23-12-GE	510	263	X	
RHR	M-120		TW23-12-GE	TW20	MO-2021	510	212	X	
RHR	M-120		TW23-12-GE	MO-2021	X39A	275	212	X	
RHR	M-120		TW24-12-GE	TW23	MO-2007	510	212	X	
RHR	M-120		TW24-12-GE	MO-2007	MO-2009	275	212	X	
RHR	M-120		TW25-4-GE	TW24-12	MO-2011	275	212	X	
RHR	M-121		TW34-12-GE	TW33-12	MO-2006	510	212	X	
RHR	M-121		TW34-12-GE	MO-2006	MO-2008	275	212	X	
RHR	M-121		TW35-4-GE	TW34-12	MO-2010	275	212	X	
Core Spray	M-122		TW11-10-GE	P-208A	MO-1751	500	212	X	
Core Spray	M-122		TW12-8-GE	TW11	RO-1747	500	212	X	
Core Spray	M-122		TW13-2 1/2-HE	TW11	RO-1743	500	212	X	
Core Spray	M-114-1/122		SC30-2-GE	TW11	CST-103-1	500	212	X	
Core Spray	M-122		TW7-10-GE	P-208B	MO-1752	500	212	X	
Core Spray	M-122		TW8-8-GE	TW7	RO-1748	500	212	X	
Core Spray	M-122		TW9-2 1/2-HE	TW7	RO-1744	500	212	X	
Core Spray	M-114-1/122		SC29-2-GE	SC29-3-GE	TW7-10-GE	500	212	X	
Core Spray	M-122	RB-31	TW7-8-ED	X-16A	MO-1754	1136	570		X
Core Spray	M-122		TW7-8-ED	MO-1754	MO-1752	1136	570	X	
Core Spray	M-122	RB-22	TW11-8-ED	X-16B	MO-1753	1136	570		X

Table I.2-1 High Energy Piping List (Cont'd)

HIGH ENERGY LINE DESIGNATION LIST									
System	P&ID(s)	Volume(s)	Line Designation	From	To	Press psig	Temp F°	Exclusion(s) 2%	SP
Core Spray	M-122		TW11-8-ED	MO-1753	MO-1751	1136	570	X	
HPCI (Steam)	M-123	RB-16	PS18-8-ED	X-11	MO-2035	1110	582		X
HPCI (Steam)	M-123	RB-16/12/8	PS18-8-ED	MO-2035	MO-2036	1110	582		
HPCI (Steam)	M-123	RB-8	PS18-8-ED	MO-2036	HO-7	1110	582	X	
HPCI (Steam)	M-123	RB-16	6″ Heating Steam Conn.	PS18-8-ED	Cap	1110	582		
HPCI (Water)	M-124	RB-16	TW3-12-ED	FW2B-14-ED	MO-2068	1250	540		
HPCI (Water)	M-124	RB-16/12	TW3-12-ED	MO-2068	SR-597	1375	453		
RCIC (Water)	M-126	RB-16	FW5-4-ED	FW2A-14-ED	AO-13-22	1250	540		
RCIC (Water)	M-126	RB-16	FW5-4-ED	AO-13-22	MO-2107	1250	453		
RCIC (Water)	M-126	RB-16/9/5	FW5-4-ED	MO-2107	MO-2106	1375	453		
RCIC (Water)	M-126	RB-16/9	FW5-4-ED	MO-2107	MO-2110	1375	453		
RCIC (Steam)	M-125	RB-16	PS17-3-ED	X-10	MO-2076	1110	582		X
RCIC (Steam)	M-125	RB-16/9	PS17-3-ED	MO-2076	SR-567	1110	582		
RCIC (Steam)	M-125	RB-16	3″ Heating Steam Conn.	PS17-3-ED	Cap	1110	582		
RCIC (Steam)	M-125	RB-9/5	PS17-3-ED	SR-567	MO-2078	1110	582		
RCIC (Steam)	M-125	RB-5	PS17-3-ED	MO-2078	MO-2080	1110	582	X	
SLC	M-127	RB-13/14/22	CH2-1.5-DC	X-42	XP-6	1275	575		
SLC	M-127	RB-13/14/22	CH2-1.5-DC	XP-6	XP-11-14A	1275	575	X	
SLC	M-127	RB-13/14/22	CH2-1.5-DC	XP-6	XP-11-14B	1275	575	X	
SLC	M-127	RB-13/14/22	XP-30 Branch	CH2-1.5-DC	XP-30	1275	575	X	
RWCU	M-128	RB-32	REW3-4-ED	X-14	MO-2398	1136	570		X
RWCU	M-128	RB-32	REW3-4-DBD	MO-2398	RC-103	1136	570		
RWCU	M-128	RB-32	TE-12-92; 1.25″ Branch	REW3-4-DBD	TE-12-92	1136	570		
RWCU	M-128	RB-32/31/30	REW3-4-DBD	MO-2398	RC-102	1136	570		
RWCU	M-128	RB-32/31/30	REW3-4-DBD	MO-2398	E-201(11)	1136	570		
RWCU	M-128	RB-30	REW3-4-DBD	E-201(11)	E-201(12)	1136	570		
RWCU	M-128	RB-30	REW3-4-DBD	E-201(12)	E-201(13)	1136	570		
RWCU	M-128	RB-30	REW3-4-DBD	RC-102	E-201(13)	1136	570		
RWCU	M-128	RB-30	REW3-4-DBD	E-201(13)	E-202(11)	1136	570		
RWCU	M-128	RB-30	TE-12-94; 1.25″ Branch	REW3-4-DB	TE-12-94	1136	570		
RWCU	M-128	RB-30/28/29/27	REW6-3-DBD	E-201(11)	MO-2399	1377	540		

Table I.2-1 High Energy Piping List (Cont'd)

HIGH ENERGY LINE DESIGNATION LIST									
System	P&ID(s)	Volume(s)	Line Designation	From	To	Press psig	Temp F°	Exclusion(s) 2%	SP
RWCU	M-128	RB-30	TE-12-109;1.25″ Branch	REW6-3-DBD	TE-12-109	1377	540		
RWCU	M-128	RB-27/26/18	REW6-3-DC	MO-2399	CRD Tee	1377	540		
RWCU	M-128	RB-18/16	REW6-3-DC	CRD Tee	SR-727	1317	540		
RWCU	M-128	RB-16	REW6-3-DC	SR-727	FW5 Tee	1317	540		
RWCU	M-128	RB-16	REW6-3-DB	FW5 Tee	RC-7-1	1317	540		
RWCU	M-128/124	RB-16	REW6-3-ED	RC-7-1	TW3-12-ED	1250	540		
RWCU	M-128/126	RB-16	REW6-3-DB	FW5 Tee	FW5-4-ED	1317	540		
Off Gas	NH-54817-4	(Offgas Building)	CN103-2-EN1C	E-603A	V-808A	300	422		
Off Gas	NH-54818-1	(Offgas Building)	CN104-2-EN1C	E-603B	V-808B	300	422		
Off Gas	NH-54818-1	(Offgas Building)	GOLE107-6-HN2C	PSV-7574B	E1B	285	422	X	
Off Gas	NH-54817-4	(Offgas Building)	GOLE109-6-HN2C	PSV-7574A	GOLE107-6-HN2C	285	422		
Off Gas	NH-54817-4	TB-15/14	SHP101-4-HN1C	PS9-3-ED	TB Wall	1000	546		
Off Gas	NH-54817-4	(Offgas Building)	SHP101-4-HN1C	TB Wall	PCV-7497A	1000	546		
Off Gas	NH-54817-4	(Offgas Building)	SHP101-4-EN1C	PCV-7497A	J-1202A	300	422		
Off Gas	NH-54817-4	(Offgas Building)	SHP102-4-HN1C	SHP101-4-HN1C	PCV-7497B	1000	546		
Off Gas	NH-54817-4	(Offgas Building)	SMP101-3-EN1C	SMPE103-4-EN1C	E-603A	300	422		
Off Gas	NH-54818-1	(Offgas Building)	SMP102-3-EN1C	SMPE104-4-EN1C	E-603B	300	422		
Off Gas	NH-54817-4	(Offgas Building)	SMPE103-4-EN1C	J-1202A	MS-52-1	300	422		
Off Gas	NH-54817-4	(Offgas Building)	SMPE103-4-HN2C	MS-52-1	J-1201A	300	422		
Off Gas	NH-54818-1	(Offgas Building)	SMPE104-4-EN1C	J-1202B	MS-52-2	300	422		
Off Gas	NH-54818-1	(Offgas Building)	SMPE104-4-HN2C	MS-52-2	J-1201B	300	422		
Off Gas	NH-54817-4	(Offgas Building)	SMP109-4-EN1C	SHP101-4-EN1C	PSV-7574A	300	422		
Off Gas	NH-54818-1	(Offgas Building)	SMP110-4-EN1C	SHP102-4-EN1C	PSV-7574B	300	422		

Note 1: Normal operating pressure is <275 psig and therefore this segment is not high energy.

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I.3 Break Analysis

The postulated break locations for each high energy line are based on the criteria established in the Giambusso letter (Reference 2) SRP 3.6.2 (Reference 4) and Generic Letter 87-11 (Reference 21). The systems evaluated were the Main Steam, HPCI, RCIC, Feedwater, Condensate, RWCU, and miscellaneous sensing and sample lines. Calculations were performed in accordance with the “HELB” Evaluation Criteria (Reference 23) to determine where the break locations would be postulated for high energy piping (Reference 17). For each system, the calculation results are listed below. The break locations for high energy lines are shown on the drawings provided in the figures section of this appendix.

I.3.1 Selection of Break Locations

Break locations were postulated for all high energy lines specified in Table I.2-1 without an exclusion in accordance with the following criteria:

I.3.1.1 Class I (Lines for which piping analyses, that includes seismic, has been performed)

Breaks were postulated for each Class I high energy line as follows:

(1) At terminal ends (MEB 3-1, B.1.c(2)(3) - Reference 21);

(2) At any intermediate location where normal operating and seismic stresses exceeds 0.8(Sh + SA) or 0.8(SA) (MEB 3-1, B.1.c(2)(3) - Reference 21);

(3) At the terminal ends of the high energy piping that penetrates the Primary Containment boundary. The postulated HELB shall be either:

(a) At the primary containment penetration. This point is identified as the weld between the containment penetration and the piping. Evaluations were performed to demonstrate that the containment integrity is maintained with a HELB at the containment penetration.

or,

(b) At the weld between the outboard containment isolation valve and the piping not within the Primary Containment. Piping between the containment penetrations and the outboard containment isolation valves has been evaluated to meet the criteria found in References 2 and 4. This criteria does not require breaks to be postulated in these segments

(4) In accordance with Generic Letter 87-11 (Reference 21), seismically analyzed and supported high energy lines are not required to have postulated intermediate HELB or critical crack locations, provided that the stress level at each potential intermediate break location is less than the values identified in Section I.3.1.1 (2) above (Reference 21).

Critical cracks were postulated where normal operating and seismic stresses exceed 0.4 (1.2Sh + SA). Critical cracks are not required to be postulated in portions of piping from the containment wall to and including the inboard and outboard isolation valves (MEB 3-1, B.1.e(2) – Reference 21).

I.3.1.2 Class II (Lines for which no seismic analyses has been performed)

Break locations postulated for each Class II high energy line were selected as follows:

(1) At terminal ends.

(2) At each intermediate location of potential high stress or fatigue such as pipe fittings, valves, flanges and welded on attachments.

(3) Critical cracks were postulated at locations that result in the most severe environmental consequences (MEB 3-1, B.1.e(3) – Reference 21).

In those cases where a Class II high energy line is analyzed to Class I piping seismic requirements, break locations were selected using the criteria in Section I.3.1.1.

I.3.1.3 Breaks in Moderate-Energy and/or Non-High Energy Lines

MNGP does not specify Moderate-Energy lines described in Generic Letter 87-11 (Reference 21). High energy line break rules or high energy line crack rules do not specify to postulate high energy line breaks or high energy line cracks in non-high energy lines (References 2, 6, and 64).

I.3.2 Break Types

Circumferential breaks were postulated and evaluated at terminal ends. In addition, circumferential breaks were postulated at intermediate break locations on high energy piping with a nominal diameter exceeding 1 inch. Longitudinal breaks were postulated at all intermediate locations on high energy piping with a nominal diameter equal to or exceeding 4 inches. Longitudinal breaks were postulated with an orientation of any point around the pipe circumference and a break area equal to the effective cross sectional flow area immediately upstream of the break location. (Criterion 3 - Reference 2).

Critical cracks were postulated to occur in piping carrying high energy fluids. The critical crack size was taken to be 1/2 the pipe diameter in length and 1/2 the wall thickness in width (Criterion 2 - Reference 2)

I.3.3 HELB Interactions

The effects on Safe Shutdown Equipment from high energy lines can be divided into five (5) separate types of interactions. These interactions include:

- Pipe Whip
- Jet Impingement
- Flooding
- Environmental Effects
- Compartment Pressurization

The evaluation criteria for each of these interactions was performed by applying the following criteria:

I.3.3.1 Pipe Whip Effects

Pipe whip is defined as the mechanical reaction caused by high energy fluid exiting a ruptured pipe. For each postulated break, the effects of pipe whip were evaluated. Pipe whip movement was assumed to occur about a plastic hinge located at a point or points selected using the following guidelines (Reference 3):

(1) The pipe run terminal point;

(2) The nearest pipe fitting which will experience high bending moment. Generally this will occur at the second elbow from the break point.

(3) A pipe whip restraint;

(4) Any structure or equipment which can reasonably be expected to restrain the movement of the pipe (e.g., concrete wall or pressure vessel).

(5) Ordinary pipe supports were considered ineffective restraints during whip.

Circumferential breaks were assumed to cause pipe whip about the plastic hinge in a plane defined by the pipe geometry and in the direction of jet reaction (Criterion B.3.a.5 - Reference 4). Longitudinal breaks were assumed to cause pipe whip movement in the direction opposite of the fluid flow normal to the axis of the pipe at the point of the break (Footnote 8 to Criterion 3 - Reference 2). Additionally, it was assumed that the geometry of the pipe segment between the selected hinges remained unchanged throughout the pipe whip path.

The area of influence of pipe whip was considered to be the worst case area determined applying the criteria identified above. In general, circumferential breaks cause the worst case pipe whip conditions.

In evaluating the effect of pipe whip on safe shutdown components and structures, the following criteria were used:

(1) The energy level in a whipping pipe was considered to be insufficient to rupture an impacted pipe of:

(a) Equal or greater nominal pipe size and

(b) Equal or heavier wall thickness (Footnote 1 to Criterion 1.d - Reference 2).

(2) Impacted pipe of lesser nominal pipe size or thinner wall thickness was assumed to rupture or require further evaluation (Footnote 1 to Criterion 1.d - Reference 2).

(3) All electrical cables in cable trays, conduits, or other raceways impacted by a whipping pipe were assumed to be severed or require further evaluation. Other electrical and instrument and control (I & C) components were assumed to fail or require further evaluation (Reference 2).

(4) Structural components impacted by a whipping pipe were assumed to fail or require further evaluation (Reference 2).

I.3.3.2 Jet Impingement Effects

For each postulated break, the effects of jet impingement were evaluated. The criteria used to evaluate these effects were as follows:

(1) All jets were assumed to be influenced by gravity (Section 7.1 - Reference 7). However, because of the jet fluid velocities and the relative proximity of the targets, straight line jet travel was assumed.

(2) Jets from a circumferential break were assumed to sweep the arc traveled during the whip (Section 8.1 - Reference 7).

(3) A longitudinal break was assumed to occur at any azimuth location on the pipe circumference with a break area equal to the effective cross sectional flow area immediately upstream of the break location. (Footnote 8 to Criterion 3 - Reference 2).

(4) A jet discharging saturated steam, or a mixture of steam and water with a fluid temperature greater than the saturation temperature at the surrounding environmental pressure, was assumed to expand in a 10° half-angle cone.[1] (Section 7.2 - Reference 7 and page 3.6.2-8 Reference 4). (Note: This criterion is consistent with the previous Monticello HELB evaluation. References 1 and 6).

1. Because of the uncertainties associated with break orientation and jet geometry, an additional degree of conservatism was established to include potential targets beyond the 10° half angle cone. In some cases, targets up to 45° off the axis of the jet were considered hit and incapable of performing their safe shutdown function.

(5) A jet discharging saturated liquid water or subcooled water was characterized by a constant diameter jet for purposes of determining the jet force on targets. For the purpose of determining potential targets, saturated liquid water or subcooled water jets were assumed to expand, at least at a 10° half angle cone (Section 7.2, Reference 7 and page 3.6.2-8 - Reference 4). (Note: This criterion is more conservative than that used in the previous Monticello HELB evaluation - References 1 and 6).

(6) The magnitude of jet force was determined using the criteria of Item C.(4) on Page 3.6.2-7 of SRP 3.6.2 (Reference 4). The criteria for taking into account the geometric considerations of a fluid jet were provided in Appendix D of Reference 7.

(7) A postulated jet was considered effective until it struck a solid barrier upon which the jet momentum is reduced to zero. All safe shutdown components, pipes, and structures impacted by the jet were considered incapable of performing their safe shutdown function or required further evaluation (Section 7.2 - Reference 7).

(8) Where the jet must travel a significant distance to impact safe shutdown components or structures, simplified calculations were used to demonstrate that the jet impingement forces were negligible.

I.3.3.3 Flooding Effects

In each area or compartment where breaks were postulated, the potential adverse effects from flooding were identified using the following criteria (Section 10 - Reference 7):

a. Vulnerability of safe shutdown equipment due to flooding because of location and configuration.

b. Potential rupture sizes and the available quantity of water.

c. Absence of floor drains or doorways which could provide drainage.

d. Existence of flood protection in the form of dams or sumps with pumps.

Environmental effects are evaluated in the MNGP Environmental Qualification (EQ) program based on the break cases considered in the HELB analysis. Bounding profiles for environmental conditions are identified and flooding effects are evaluated based on these conditions.

I.3.3.4 Environmental Effects and Compartment Pressurization

A review was conducted to determine if the safe shutdown components in compartments or areas in which breaks have been postulated are qualified for worst case environmental conditions including the effects of compartment pressurization. Attention was focused on closed compartments which contained high energy lines. Environmental effects are evaluated in the MNGP Environmental Qualification (EQ) program based on the break cases considered for the HELB evaluations. Bounding profiles for environmental conditions are identified, including compartment pressurization, heatup and flooding effects.

I.3.4 System Break Locations

System break and critical crack locations are identified in References 26 through 58.

I.4 Safe Shutdown Requirements Reference 2, 6, 8 and 11

The systems required for safe shutdown of the reactor for the postulated pipe failure of each of the high energy systems were identified based on the analyses presented in USAR Section 6.0, Plant Engineered Safeguards, for LOCA and the original HELB Analyses of Reference 8. By verifying that these systems remain operational in the event of the postulated failure of the piping systems, it is assured that the plant can be safely shutdown and maintained in a safe shutdown condition. 604000000250

I.4.1 Safe Shutdown Systems

Safe shutdown systems are separated into two categories: one, those needed for Scramming and Initial Vessel Level Stabilization to demonstrate adequate core cooling; and two, those needed for Post Initial Vessel Level Stabilization for Containment Cooling, and Decay Heat Removal. 604000000250

I.4.1.1 Scram and Initial Vessel Level Stabilization Period

The following list identifies Emergency Core Cooling Systems, ECCS, credited for core cooling. Per Errata of Reference 2, where Reactor Protective Actuation has occurred (scram and/or containment group isolation) then concurrent with the postulated accident, single active failure and loss of off-site power is assumed. Otherwise, single failure and loss of off-site power is not assumed.

(1) Reactor Protection System (RPS), Control Rod Drive System (CRD) and Control Rods.

(2) High Pressure Coolant Injection (HPCI).

(3) Reactor Pressure Relief - Safety/Relief Valves (SRVs)

(4) Low Pressure Coolant Injection (LPCI) Mode

(5) Core Spray (CS). 604000000250

(6) Shutdown Instrumentation (See Item (8))

(7) Auxiliary Support

(a) On-site Power and Distribution System

(b) Emergency Service Water (ESW)

(c) Emergency Diesel Generator Auxiliary Systems

(d) DC Power Systems

(e) Heating, Ventilation and Air Conditioning (HVAC) Systems

(8) Shutdown Instrumentation

The minimum required process instrumentation to assure safe shutdown is as follows:

(a) Reactor Vessel Level Indication

(b) Reactor Pressure Indication

(c) Suppression Pool Temperature Indication

(d) Suppression Pool Level Indication

I.4.1.2 Post Initial Vessel Level Stabilization, Containment Cooling, and Decay Heat Removal Periods.

The following list identifies those systems for mitigation after vessel is stabilized. Single active failure is not assumed. Loss of off-site power is not assumed.

(1) RHR Shutdown Cooling (SDC) Mode

(2) RHR Suppression Pool Cooling (SPC) Mode

(3) RHR Service Water 604000000250

(4) ECCS

(5) Alternate Injection Systems, such as those suggested by Emergency Operating Procedure C.5-1100, RPV Control.

(6) All available torus cooling as suggested by Emergency Operating Procedure C.5-1200, Primary Containment Control.

I.4.2 <u>Description of Safe Shutdown Paths</u>

Safe shutdown can be described as scramming and initial vessel level stabilization, followed by reducing RPV pressure and temperature from the normal operating conditions of approximately 1015 psia and 545°F to a RPV coolant temperature of less than 212°F and the RPV vented with the reactor in a subcritical condition and no fuel damage or breach of Primary Containment. The minimum equipment necessary for achieving and maintaining safe shutdown includes the following: 604000000250

(1) The Reactor Protection System and Control Rod Drive System (negative reactivity function)

(2) 3 SRVs (RPV overpressure protection and decay heat removal functions)

(3) 1 RHR pump (LPCI Mode) or 1 Core Spray pump and 1 RHR pump in SPC Mode (reactor vessel level maintenance and decay heat removal functions)

(4) 1 RHR Service Water pump (decay heat removal function)

(5) 1 Emergency Diesel Generator (EDG), EDG auxiliaries, 1 EDG Emergency Service Water pump, 1 Emergency Service Water pump, and 1 Division of safety related power distribution (for all functions except negative reactivity)

(6) Shutdown Instrumentation (from Reference 8) 604000000250

(a) Reactor Vessel Level

(b) Reactor Pressure

(c) Suppression Pool Temperature

(d) Suppression Pool Level

(7) Offsite power and non-ECCS equipment may be used as directed by the Emergency Operating Procedures, C.5 series. 604000000250

Evaluations performed to determine acceptable paths to safe shutdown utilized the above described minimum necessary equipment.

Additional safe shutdown paths can be described depending on the location of the postulated HELB and the single active failure taken. The HPCI system can be used to maintain RPV water level and depressurize the RPV, so that the low pressure high flow pumps (RHR or CS) can be used. Table I.4-1 identifies the systems available to support each shutdown performance goal.

For any postulated HELB with HPCI available, RCIC is not required, since HPCI could perform the same function. For any postulated HELB with HPCI not available due to a single active failure or HELB related damage, the unit is shut down using the SRVs (RPV depressurization), Core Spray (RPV water level maintenance), and RHR (decay heat removal and RPV water level maintenance). No single active failure could affect more than 1 SRV, 1 CS division, or 1 RHR division.

The RCIC System is not required for any postulated HELB in any of the identified compartments. However, for most postulated HELBs RCIC is available to support safe shutdown.

For any postulated HELB, at least one division of ESW is available to provide cooling to one division of RHR and CS systems. Hence, safe shutdown can be accomplished for any postulated HELB without requiring the RCIC System to support the safe shutdown.

I.4.3 Safe Shutdown System Component Locations

For each of the systems identified in Section I.4.1 of this report, the locations of the major components, main pipe routings, and power and control cable routings are listed in this section. The information is provided by system and HELB analysis volume for reference.

I.4.3.1 Reactor Protection System

The Reactor Protection System (RPS) consists of the motor generator set power supplies, the RPS power distribution panels, sensors and cabling. The two motor-generator sets are located in the Reactor Building on the 935-foot elevation in Volumes 14 and 17. The RPS power distribution panels are located in the Cable Spreading Room. There are sensors (i.e., high reactor pressure, etc.) included in the protection system that are located in a variety of sites throughout the plant.

I.4.3.2 Control Rod Drive System

The Control Rod Drive System consists of the control rods and the associated control rod drive hydraulic units. The electrical signals to the scram valves on the control rod drive hydraulic units are provided by the RPS. The control rods and Control Rod Drives are located inside of the Primary Containment. The control rod drive hydraulic units are located in two separate banks on either side of the Primary Containment on the 935 foot elevation of the Reactor Building in Volumes 14 and 18.

I.4.3.3 High Pressure Coolant Injection (HPCI) System

The HPCI System consists of a pump with a steam driven turbine and associated valving. It is located primarily in the HPCI room (Volume 8). The HPCI steam supply line and the HPCI injection line (PS18-8″-ED and TW3-12″-ED, respectively) are routed from main the steam chase (Volume 16) to the HPCI room (Volume 8). The valving for the HPCI System is located in the above identified compartments. Power cabling for the HPCI System is routed from the batteries in the EFT Building into the Turbine Building in the east corridor on the 931-foot elevation (Volume 20). The cables are then routed into the condenser bay (Volume 15), from there into the CRD pump room (Volume 7), down to equipment and floor drain tank area (Volume 6), and finally into the HPCI Room (Volume 8).

I.4.3.4 Reactor Pressure Relief System

The eight SRVs and associated piping are located inside of the Primary Containment on the main steam lines. There are four control panels and two instrument racks associated with the SRVs. The SRV control panel is located in the Cable Spreading Room. The SRV Low-Low Set control panel “A” is located in the cable spreading area. The “B” control panels are located in the main control room and on the third floor of the EFT Building. The SRV low-low set instrument rack “A” is located in the northeast corner of the torus area (Volume 9) and the “B” rack is located on the 935-foot elevation of the Reactor Building near the west side CRD hydraulic units (Volume 18).

I.4.3.5 Residual Heat Removal (RHR) System

The RHR System is a two divisional system with two motor driven pumps in each division. The valving for the system consists of a series of suction and discharge valves, which allow the RHR System to be operated in various modes. Two of the RHR pumps are located in the southeast corner room of the Reactor Building (Volume 1) and the other two in the southwest corner room of the Reactor Building (Volume 3). The suction valving is located in three separate compartments. Most of the RHR pump discharge valves are also located in the corner rooms with the following exceptions:

(1) Division I suppression pool cooling (SPC) valves (MO-2006 and MO-2008) and the Division II SPC valve (MO-2009) are located in the torus area (Volume 9-12).

(2) Division II SPC mode outboard valve (MO-2007) is located on the west side of Reactor Building, at the 935-foot elevation in Volume 19.

(3) The Division I LPCI injection and SDC containment isolation valves (MO-2014 and MO-2012) and the shutdown cooling line outboard isolation valve (MO-2030) are located in Volume 13 on the 935-foot elevation of the Reactor Building.

(4) The Division II LPCI Injection and SDC containment isolation valves (MO-2013 and MO-2015) are located in Volume 20 on the 935-foot elevation of the Reactor Building.

(5) The Division I containment spray isolation valves (MO-2020 and MO-2022) are located in Volume 14 on the 948-foot elevation of the Reactor Building and the Division II valves (MO-2021 and MO-2023) are located on the 962-foot 6-inch elevation in Volumes 31 and 33.

The power cabling for the RHR Pumps is routed from the respective safety-related switchgear room in the Turbine Building Volume 8 (Division I) and Volume 31 (Division II) underground to the Reactor Building. Division I cables are then routed to RCIC room (Volume 5) and continue underground to the southeast corner room (Volume 1). The Division II cables are routed underground to the Reactor Building drain tank room (Volume 6) and then continue underground to the southwest corner room (Volume 3). The power cables to the valves are routed from the respective MCCs in the Turbine Building Volume 1 (Division I) and Volume 20 (Division II), to the Cable Spreading Room, and into the Reactor Building via Volume 14 (Division I) and Volume 17 (Division II). From there the cables are routed to the respective valves. The control cables follow the routing of the valves' power cables. The power and control cables for Division II RHR valves MO-2007, MO-2009, and MO-2003 are routed from MCCs (Volume 24) to the third floor EFT Building. They then go underground around the south side of the Reactor Building to the 935-foot elevation (Volume 18) of the Reactor Building and then to the respective valves.

I.4.3.6 Core Spray (CS) System

The Core Spray System is a two divisional system capable of taking suppression pool water and spraying it over the top of the reactor core. Each division has a core spray pump, a valve on the suction line to the suppression chamber, and inboard and outboard injection valves.

The Division I pump (P-208A) is located in the southeast corner room of the Reactor Building (Volume 1) in the same room with the Division I RHR pumps. Its suction line, (TW6-12″-HE) is routed from the suppression chamber area (Volume 10) to the CS pump. The pool suction valve (MO-1741) is located in the southeast corner room (Volume 1) with the CS pump. The discharge piping is routed from the southeast corner room, up through the east side of the 935-foot elevation of the Reactor Building (Volume 14) and finally up to the 962-foot 6-inch elevation of the Reactor Building on the east side (Volume 22). The inboard and outboard injection valves (MO-1753 and MO-1751, respectively) are located in this area (Volume 22).

The Division II Core Spray System is located on the west side of the Reactor Building. The Division II core spray pump (P-208B) and the pool suction valve (MO-1742) are located in the southwest corner room (Volume 3). The CS pump suction piping is routed from the suppression chamber area (Volume 11) to the southwest corner room. The pump discharge piping is routed from the southwest corner room up through the 935-foot elevation of the Reactor Building (Volume 19), and up to the 962-foot 6-inch elevation of the Reactor Building (Volume 31). The inboard and outboard injection valves (MO-1754 and MO-1752, respectively) are located on the west side of the Reactor Building at the 962-foot 6-inch elevation in Volume 31.

The cabling for each division of the Core Spray System follows the same routing as the corresponding division of RHR.

I.4.3.7 RHR Service Water System

The RHR Service Water (RHRSW) System supplies water from the ultimate heat sink (river) to the tube side of the RHR heat exchangers and back to the river. The system has 4 pumps with 2 dedicated to each division. The pumps for both divisions are located in the Intake Structure pump room (Volume 18). The discharge piping is routed from the Intake Structure (Volume 18) through the access tunnel (Volume 16) into the Turbine Building north corridor (Volume 5). From the Turbine Building north corridor they are routed above the 931-foot elevation of the Turbine Building through Volumes 20 and 25, and into the condenser bay area (Volume 15). The lines exit the Turbine Building from the condenser bay and penetrate the Reactor Building in the TIP drive room, Volume 17. From the TIP drive room, the lines are routed to the torus area (Volumes 9-12) and then to the respective corner rooms.

Auxiliary air compressors are located on the 935-foot elevation of the Reactor Building, Division I in Volume 14 and Division II in Volume 19. Each divisions’ heat exchanger discharge pressure control valves (CV-1728 and CV-1729) are also located in the respective corner rooms.

Cabling for the system is routed entirely within the Turbine Building. The power cabling for the pumps is routed from the respective switchgear areas (Volumes 8, and 31) to the Intake Structure (Volume 18). Control cabling is routed to the Control Room via the respective divisional routing in the Turbine Building. For Division I this would be from the cable spreading room to Volumes 21, 2, 22 and 27. For Division II this would be from the cable spreading room to Volume 20 and then into the switchgear area.

I.4.3.8 Reactor Core Isolation Cooling (RCIC) System

While not required for safe shutdown, the RCIC System may be used if available. A description is included in this section for completeness.

The RCIC System, like the HPCI System, consists of a steam driven turbine-pump assembly and associated valving. The turbine-pump and most of the associated valves are located in RCIC room (Volume 5). The steam supply line (PS17-3.-ED) and the RCIC injection line (FW5-4.-ED) are routed from the RCIC room through the torus area (Volume 9) and up into main steam chase (Volume 16). The valving for the RCIC System is located in either the RCIC room or main steam chase. The RCIC power cables are routed from the batteries in the Plant Administration Building through the Cable Spreading Room into TIP drive room (Volume 17) in the Reactor Building and down to the RCIC room (Volume 5).

I.4.3.9 Shutdown Instrumentation

The Shutdown Instrumentation, like the other safety related equipment is divisionalized. The locations for the components for each division are given below:

Division	Parameter Measured	Equipment Designation	Compartment
I	Reactor Water Level	LT-2-3-112A	East side of Reactor Bldg. Elev. 935-foot
I	Suppression Pool Level	LT-7338A	Torus Area
I	Reactor Pressure	PT-6-53A	East side of Reactor Bldg. Elev. 962-foot 6-inch
I	Suppression Pool Temperature	TE-4073A to 4080A	Torus Area
II	Reactor Water Level	LT-2-3-112B	West side of Reactor Bldg. Elev. 935-foot
II	Suppression Pool Level	LT-7338B	Torus Area
II	Reactor Pressure	PT-6-53B	West side of Reactor Bldg. Elev. 962-foot 6-inch
II	Suppression Pool Temperature	TE-4073B to 4080B	Torus Area

The cabling from these instruments is routed through the same routing schemes back to the cable spreading area as are the other safety-related divisional separated cables. The only exceptions are the Division II suppression pool temperature cables and Alternate Shutdown System (ASDS) instruments. From the torus area they are routed into the condenser bay (Volume 15) and through Volume 20 on the 931-foot elevation of the Turbine Building and into the Emergency Filtration Building from there. ASDS instruments that have been rerouted are reactor water level LT-2-3-112B and suppression pool level LT-7338B. The routing is underground around the south side of the Reactor Building to the third floor, EFT.

I.4.3.10 Emergency Service Water Systems

The two emergency service water systems, consisting of the Emergency Diesel Generator Emergency Service Water (EDG-ESW) System and the Emergency Filtration Train Emergency Service Water (EFT-ESW) System, are both two divisional systems which collectively supply cooling water to the emergency diesel generators, the ECCS pump room coolers, the RHR & Core Spray pumps, and the EFT System. The systems, when combined, consist of 4 pumps, P-111A, B, C & D, located in the Intake Structure (Volume 18). Pumps P-111A and P-111C are both Division I components, and pumps P-111B and P-111D are both Division II components.

I.4.3.10.1 Emergency Diesel Generator Emergency Service Water System

Pump P-111A and P-111B are used to supply cooling water to the Division I and Division II emergency diesel generator, respectively. The discharge piping from each EDG-ESW pump tees in the intake structure with separate 4 inch lines routed underground to the respective EDG. The other branch of the tee on each line has a closed manual valve in the Intake Structure (ESW-61-1 and ESW-61-2). The cooling water lines to the EDGs are cross-tied in the Intake Structure. This allows cooling of either EDG from pump P-111A or P-111B.

The power cable for Division I ESW pump P-111A is routed from MCC-134 in the EFT Building, into the main feedpump area, (Volume 3), along the east corridor at the 911-foot elevation (Volume 3), along the north Turbine Building corridor (Volume 5), through the access tunnel (Volume 16), and into the intake structure. A control cable for P-111A is routed from the Control Room, through the Cable Spreading Room, through Volume 21, into the main feedpump area (Volume 3) and into the EFT Building. Another control cable for P-111A is routed from C-03 in the Control Room, through the Cable Spreading Room, through Volume 21, into the main feedpump area (Volume 3), along the east corridor at the 911-foot elevation, along the north Turbine Building corridor (Volume 5), and into the Division I essential switchgear area (Volume 8) before entering the Division I EDG compartment (Volume 29).

The power cable for Division II ESW pump P-111B is routed from MCC-143 located in the southeast corner of the Turbine Building on elevation 931-foot (Volume 20), along the east corridor of the Turbine Building (Volume 20), along the north Turbine Building corridor (Volume 26), and into the Division II essential switchgear area (Volume 31) before entering the Intake Structure through an underground duct bank. The control cables for P-111B originate in the EFT Building and are routed into the Turbine Building, southeast corner, at elevation 931-foot. From there, one of the control cables is routed directly into the Control Room, one is routed to MCC-143 (which is located in Volume 20), and one is routed along the east Turbine Building corridor at the 931-foot elevation (Volume 20), along the north Turbine Building corridor (Volume 26), and into the Division II essential switchgear area (Volume 31) before entering the Division II Emergency Diesel Generator compartment (Volume 30).

I.4.3.10.2 Emergency Filtration Train Emergency Service Water System

Pumps P-111C & D supply cooling water to the respective Division I and II ECCS pump room coolers, the RHR and Core Spray pumps, and the EFT System. The Division I ESW Line to ECCS room coolers and Division I RHR and Core Spray Pumps (ESW1-3″-HF) is routed from the Intake Structure, through the access tunnel (Volume 16), into the Turbine Building north corridor on the 911-foot elevation (Volume 5). From there, it is routed in the north corridor over into the reactor feed pump area (Volume 3). Within this compartment the lines tee again (Reference 18) with the piping line ESW1-3″-HBD routed into the Emergency Filtration Building. The line to the ECCS pump room coolers and RHR and Core Spray Pumps SW30A-3″-HF is routed from the main feed pump area into the condenser bay area (Volume 15) on the 911-foot elevation. The pipe is routed along the east wall over to the south wall of the room. The piping line exits the condenser bay through the south wall and into the Reactor Building. This line enters the Reactor Building in the TIP drive room (Volume 17) and is routed from there into the torus area (Volume 9). Once in the torus area the line tees again with part of the flow routed to the southeast corner room (Volume 1) and the remainder to the HPCI room (Volume 8) for the Division I HPCI room cooler.

The Division II ESW Line to the ECCS room coolers (ESW5-3″-HF) follows a parallel route to its Division I counterpart to the north turbine corridor at the 911-foot elevation (Volume 5). From there it is routed to Volume 20 on the 931-foot elevation of the Turbine Building. In Volume 20 the line tees (Reference 19) with line SW30B-3″-HF routed into Volume 21 and then into the condenser bay (Volume 15). From the condenser bay, the line enters the Reactor Building in the main steam chase (Volume 16) and from there is routed into the torus area (Volume 12). After entering the torus area, the line tees again with part of the flow going to the southwest corner room (Volume 3) and the remainder to the Division II HPCI room cooler in the HPCI room (Volume 8).

The cable for the other two ESW pumps (P-111C and P-111D) begins in the EFT Building. The power cable to P-111C is routed from MCC-134 in the EFT Building into the main feedpump area, Volume 3, along the east corridor at the 911-foot elevation, and then along the north Turbine Building corridor Volume 5 into the access tunnel. One control cable is located entirely in the EFT Building. The other enters the Turbine Building in the main feedpump area, then is routed into Volume 21 and into the Cable Spreading Room, before entering the Control Room. For pump P-111D, the power cable is routed from MCC-144 in the EFT Building into the Turbine Building southeast corner on the 931-foot elevation. It is then routed along the east corridor at that elevation into the Division II cable way (Volume 35), and ending in Division II essential switchgear area, Volume 31. From there the cable is routed outside underground to the Intake Structure. Three control cables for this pump are located entirely within the EFT Building and the other enters the Turbine Building at the 931-foot elevation in Volume 20. From there it is routed up to the turbine operating floor before entering the Control Room.

I.4.3.11 Emergency Diesel Generators and Auxiliaries

There is one Emergency Diesel Generator (EDG) for each essential division at Monticello. The EDGs are located in adjacent compartments with EDG-11 located in Volume 29 and EDG-12 located in Volume 30. The local control panels and air start systems are also located in the respective compartment with the EDG. The oil day tanks are located next to Volume 29 in separate day tank rooms. Cabling for power to the local control and electrical cabinets is routed in the same manner as the control cables for RHRSW pumps. The EDG power cabling is routed to the respective essential switchgear which are in adjacent compartments in the Turbine Building.

I.4.3.12 Auxiliary Power Distribution Systems

The Auxiliary Power Distribution System consists of the 4KV switchgear, the 480Vac load center, motor control centers and the transformers between them. The system is capable of distributing the electrical power generated by the EDGs under a loss of off-site power condition. For Division I, both the 4KV switchgear and 480Vac distribution equipment are located in Volume 8 on the 911-foot elevation of the Turbine Building in the northwest corner. The Division II load centers are located directly above the Division I load centers on the next floor (931-foot elevation) in Volume 31. Essential MCCs - 133A, 142A and 143A are located in the southeast corner of the Turbine Building with Division I MCC-133A located on the 911-foot elevation in Volume 1 and Division II MCCs - 142A and 143A located on the 931-foot elevation in Volume 20. Essential MCCs - 134 and 144 are located in the EFT. The EFT is not assigned a volume number because there are no high energy lines within the building.

Cabling from load centers to the MCCs is routed in separate compartments. Cables from Division I load centers are routed from Volume 8 through Volumes 35 and 26 terminating in Volume 20. Other Division I cables not terminating at the MCC are routed up into Volume 21 and into the cable spreading area. The Division II cables are routed from the load centers Volume 31 through Volumes 32, 35 and 26 and terminating in Volume 20. Cables going to the cable spreading area are routed through Volume 20.

I.4.3.13 DC Power Systems

With the exception of the 125 Vdc distribution panels located in the 4 KV switchgear rooms, the major DC power components required for safe shutdown are located in the battery rooms of the Administration or Emergency Filtration Buildings. The cabling is routed from Cable Spreading Room to the respective components along the divisional routings described previously. The only exception is the HPCI power cables, which are routed in a similar route to the Division II SPOTMOS cables. However, once routed into the Reactor Building, the HPCI cables are then routed to the HPCI area (Volume 8).

I.4.3.14 HVAC Systems

The only HVAC Equipment required for safe shutdown are the ECCS Room Coolers, V-AC-5 (Division I) and V-AC-4 (Division II), located in respective Reactor Building corner rooms on the 920-foot elevation and the EDG supply fan, V-SF-9 for EDG No. 12 and V-SF-10 for EDG No. 11. The cooling water for the ECCS room coolers is supplied from the respective Emergency Service Water System pump with V-AC-5 supplied by P-111C, and V-AC-4 by P-111D. Power cabling for the V-AC-4 and V-AC-5 is routed into the Reactor Building from the cable spreading area in the same manner as the other divisional cables.

The power and control cables for Division I EDG supply fan (V-SF-10) are routed from MCC-134 in the EFT Building, into the main feedpump area, (Volume 2), along the east corridor at the 911-foot elevation (Volume 3), along the north Turbine Building corridor (Volume 27) and into Division I switchgear area (Volume 7) before entering the Division I EDG compartment (Volume 29).

The power and control cables for Division II EDG supply fan (V-SF-9) are routed from MCC-143 located in the southeast corner of the Turbine Building at the 931-foot elevation, along the east corridor of the Turbine Building at the 931-foot elevation, along the north Turbine Building corridor, and into the Division II essential switchgear area (Volume 31) before entering the Division II EDG compartment (Volume 30).

Table I.4-1 Safe Shutdown Components Available

SAFE SHUTDOWN COMPONENTS AVAILABLE FOR EACH PERFORMANCE GOAL (1)				
Division	**Equipment Required to Achieve and Maintain Subcriticality**	**Equipment Required to Maintain RPV Inventory (4)**	**Equipment Required for RPV Depressurization**	**Equipment Required to Remove Decay Heat and containment spray/cooling**
I	Reactor Protection System and Control Rods and Control Rod Drives	1CS Pump and 2 RHR Pumps Available (5) • 1 CS Pump or 1 RHR Pump (LPCI Mode) (2)	4 SRVs Available (5) • 2 SRVs Needed for LPCI RPV Inventory Recovery or; • 2 SRVs Needed for CS Pump RPV Inventory Recovery	4 SRVs, 2 RHR Pumps, 1 CS Pump, 1 RHR Heat Exchanger, and 2 RHRSW Pumps Available • 1 RHR Pump (SDC Mode), 1 RHR Heat Exchanger, and 1 RHRSW Pump Needed or; • 1 CS Pump, 1 SRV, 1 RHR Pump (SPC Mode), 1 RHR Heat Exchanger, and 1 RHRSW Pump Needed or; • 1 RHR Pump (LPCI Mode), 1 RHR Heat Exchanger, 1 SRV, and 1 RHRSW Pump Needed
II	Reactor Protection System and Control Rods and Control Rod Drives	HPCI, 1 CS Pump, and 2 RHR Pumps Available • HPCI, 1 CS Pump or RHR Pump (LPCI Mode) Needed or; • 1 CS Pump or 1 RHR Pump (LPCI Mode) (3)	HPCI and 4 SRVs Available • HPCI and 1 SRV Needed or; • 2 SRVs Needed for LPCI RPV Inventory Recovery or; • 2 SRVs Needed for CS Pump RPV Inventory Recovery	4 SRVs, 2 RHR Pump, 1 CS Pump, 1 RHR Heat Exchanger, and 2 RHRSW Pumps Available • 1 RHR Pump (SDC Mode), RHR Heat Exchanger, and 1 RHRSW Pump Needed or; • 1 CS Pump, 1 SRV, 1 RHR Pump (SPC Mode), 1 RHR Heat Exchanger, and 1 RHRSW Pump Needed or; • 1 RHR Pump (LPCI Mode), 1 SRV, 1 RHR Heat Exchanger, and 1 RHRSW Pump Needed

1 Required Auxiliary Support Systems (i.e., EDG, EDG Auxiliaries, Essential Power Dist. Sys., ESW System, Shutdown Instrumentation) always required.
2 Following RPV depressurization by SRVs and RCIC not available.
3 Following RPV depressurization by SRVs and HPCI not available.
4 RHR pump used for RPV Inventory Control cannot be same RHR pump used for Decay Heat Removal, except for the final path described for either division under decay heat removal of this table.
5 Though not required safe shutdown equipment, RCIC if available, could be used.

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I.5 HELB and Safe Shutdown Evaluation

I.5.1 Compliance Evaluation (Reference 2, 6, 8, 11, 25, and 65)

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The objective of the compliance evaluation was to establish that, for any postulated HELB in any compartment, a path to safe shutdown existed using the mitigating systems identified in Section I.4. The methodology for showing this compliance was divided into four steps:

(1) Eliminate all compartments which did not contain both postulated HELBs and SSD equipment, or all compartments containing postulated HELBs which would not adversely affect SSD in adjacent compartments.

(2) Evaluate the loss of all SSD equipment from a postulated HELB in a specific compartment. If a path to safe shutdown exists, the evaluation of the compartment is complete and documented accordingly.

(3) Evaluate the loss of the SSD equipment affected by specific HELBs in a compartment to demonstrate compliance.

(4) Provide recommendations for proposed modifications to achieve compliance for specific postulated HELBs.

Step 1 was performed using the data obtained from the field surveys (References 15 and 16) and the HELB locations were identified. A review of all compartments was made to determine those compartments in which only SSD equipment or high energy lines were located. These compartments were identified as being in compliance. In addition, for the compartments containing only high energy lines, a review was performed to determine if any postulated HELBs could cause adverse effects on SSD equipment in adjoining compartments. If an interaction, due to the effects of a HELB, could be identified, an evaluation using the methods of Step 3 was performed.

Step 2 in the compliance evaluation was to evaluate the loss of all SSD equipment in a given compartment as a result of a HELB and evaluate whether a path to safe shutdown existed. This safe shutdown path was then applied to all postulated HELBs in the compartment.

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For any compartment in which compliance could not be demonstrated, an evaluation of specific HELBs was conducted (Step 3) to determine the SSD equipment adversely affected by a break and potential pathways to safe shutdown. Again compliance could be demonstrated by establishing a safe shutdown path. Any unacceptable interactions remaining were identified and changes to plant configuration and operating requirements were made to remove these interactions.

I.5.1.1 Identification of Potential Targets

The targets of the postulated HELBs were selected as those systems, components, and structures required to mitigate the consequences of postulated HELBs and accomplish and maintain a cold shutdown condition without fuel damage or break of primary containment as identified in the USAR, the Safe Shutdown Analysis, and the original HELB Report (Reference 8). 604000000250

Targets include floors, walls, compartment boundaries, and other systems where failures of the boundary or system in a postulated HELB could cause damage or create adverse conditions for primary safe shutdown equipment.

The primary and secondary containment boundaries and the Control Room are also considered to be targets of postulated HELBs.

I.5.1.2 Single Active Failure Evaluation

HELB Breaks in the Mainsteam line, Feedwater line, and RWCU line are automatically contained within the first 10 minutes by HPCI stabilizing vessel level after scramming. In this time frame, only ECCS components are assumed to mitigate the HELB with a single active failure in the ECCS components plus an assumed loss of off-site power. For environmental flooding purposes, off-site power for the FW break was assumed, but shut down is conservatively shown without off-site power. If HPCI is the single failure, then a division of low pressure injection systems, CS and/or RHR, are automatically available for vessel level stabilization after depressurization, at which time the HELB accident is contained. 604000000250

After initial vessel level stabilization, then off-site power is assumed and single active ECCS failure is not postulated. SDC and SPC, non-ECCS Modes of RHR, and ECCS equipment maybe used for containment cooling, vessel level control, and decay heat removal. Alternate Injection systems per C.5-1100 and All Available Torus Cooling per C.5-1200 are also available.

For each compartment containing high energy lines, where safe shutdown equipment was adversely affected, a single active failure review was conducted to determine the single active failures that could inhibit safe shutdown of the unit. A single active failure is defined as the malfunction or loss of function of an electrical or fluid system component, excluding passive failures, which rely on mechanical movement to complete its intended function upon demand. 604000000250

The determination of postulated targets and disabled systems revealed that for all HELBs within the plant, the SRVs and the RPS equipment would not be adversely affected. Also, there are no single active failures within these systems coincident with loss of off-site power which could prevent these systems from performing their intended safe shutdown function. Therefore, the reactivity control and reactor vessel depressurization function could always be achieved.

While RCIC is not an ECCS system its use is credited in Reference 8, Table 3. The reactor vessel level control function is achieved by using either the HPCI or RCIC systems before RPV depressurization and by using one division of the Core Spray or the LPCI mode of RHR after RPV depressurization. If neither HPCI nor RCIC is affected by a HELB, then either system can achieve the level control function, assuming a single active failure in either system. If both HPCI and RCIC are lost through a HELB or HELB and single active failure, the RPV can be depressurized with the SRVs and either the LPCI mode of RHR or CS can be used to restore and maintain RPV level. The LPCI and CS are redundant to each other, and each essential division has a CS and LPCI System. There are no single active failures in the CS, HPCI, RCIC and LPCI Mode of RHR which prevents the reactor vessel level function from being achieved. 604000000250

For the postulated HELBs, it can be concluded here that there are no coincident single active failures which would prevent safe shutdown of the unit. 604000000250

I.5.2 <u>HELB Evaluations by System</u>

Table I.5-1 identifies by compartment and system the locations of high energy piping subject to possible pipe rupture. By reading any row, a determination can be made of all compartments in which pipe ruptures of a specific system are postulated. Also, by examining any column, a determination can be made if any postulated high energy line breaks are located within a specific compartment. The last line of the table identifies those compartments which contain safe shutdown equipment.

The following section evaluates the effects of a postulated HELB on safe shutdown in each compartment identified in Table I.5-1 that has a postulated break location or has high energy piping traversing the compartment. There are no high energy lines in the EFT building. The evaluations are conducted on a system basis for each compartment.

Evaluations are provided for each high energy system with high energy piping that could not be excluded from further evaluation by Table I.2-1. For each system, an evaluation is provided for each compartment in which HELBs are postulated. The evaluations describe the effects of pipe whip, jet impingement, compartment pressurization, flooding, and environmental effects. Information on paths to safe shutdown is given, if the postulated HELBs damage SSD equipment. 604000000250

I.5.2.1 Main Steam System

The high energy lines for the Main Steam System are located in four compartments, the main steam chase (Volume 16) in the Reactor Building, the condenser area (Volume 15), the steam jet air ejector (SJAE) room (Volume 14), and the turbine operating floor (Volume 37) in the Turbine Building.

I.5.2.1.1 Main Steam Chase (Volume 16)

The main steam chase contains the four main steam lines and the associated drain piping. Pipe whip from the main steam lines is not considered a problem, since these lines are restrained at several locations in this compartment and pipe whip reactions are not toward any SSD equipment. Since the only postulated break locations are at the outboard containment isolation valves and the penetration acts as an anchor point, only circumferential breaks are required to be postulated. The resultant jets from the circumferential breaks do not impact any SSD equipment except for a small portion of the jet which would hit the compartment ceiling. Imbedded in the ceiling are Division II cables of SSD equipment, but these cables would be unaffected because of the concrete ceiling reinforcement below the conduits. HPCI and RCIC would both be available to support safe shutdown, since the steamline isolation valves and injection valves are qualified for the anticipated steam environment. In addition, all other SSD systems of both divisions would be available. Depressurization is accomplished by the HPCI or the SRVs, and LPCI and CS can be used for decay heat removal and RPV level maintenance. Therefore, safe shutdown can be accomplished.

The postulated main steam line break would rupture the blowout panels to the turbine operating floor (Volume 37), and fail open the railway door to the south side of the Reactor Building at elevation 935-foot. The effects of the pressurization would be from the same circumferential break, and the identical SSD equipment would be affected. No additional SSD equipment would be adversely affected as a result of the pressurization. Therefore, safe shutdown could be accomplished in the same manner as described above.

The consequences of compartment flooding was evaluated and no additional SSD equipment would be affected. The HPCI and RCIC injection valves would remain available due to the height of the operators above the floor (~ 5 ft.). Water would not leave the area since the bottom of the door opening is 4 feet above the compartment floor elevation.

The effects of the main steam line break in this compartment were used for environmental qualification purposes. Therefore, no additional SSD equipment will be adversely affected, and the previously identified path to safe shutdown can be utilized.

Breaks on the main steam drain line (PS15-3″-EB) were evaluated, and found to be bounded by the other main steam line break evaluations.

It is, therefore, concluded that a path to safe shutdown exists for any postulated main steam break in the main steam chase.

I.5.2.1.2 SJAE Room (Volume 14)

The main steam piping line (PS9-3″-ED) is routed to the SJAE Room from the Condenser area. No concern exists with respect to pipe whip and jet impingement, since there is no SSD equipment in the area. The consequences of compartment flooding was evaluated. Since there is no SSD equipment in the room, there was no negative effect. Also, the drains in the room can drain the entire volume to the sump area. In addition, the door to the SJAE room is barred and air tight. Any leakage as a result of the flooding would be negligible.

Compartment pressurization and environmental effects from this steamline break would not adversely affect any SSD equipment. The SJAE hatch plug has been modified and the SJAE has adequate vent areas to the Condenser bay (Volume 15) and to the 931-foot elevation of the Turbine Building. The door to the SJAE Room has also been modified to withstand the pressurization. This break does not cause Reactor Protective initiation since the break flow is less than the Group 1 flow trip point (Reference 2, Errata). Therefore, single failure and loss of offsite power considerations are not applicable.

SSD equipment adversely affected are Division I MCC-133A and MCC-133B due to high temperature. Division II equipment with HPCI are available for vessel inventory with subsequent safe shutdown. Division II CS is automatically available for Vessel Inventory makeup after SRV depressurization and RHR Suppression Pool cooling is available for containment and decay heat removal. Alternate Injection systems per C.5-1100 and All Available Torus Cooling per C.5-1200 are also available.

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I.5.2.1.3 Condenser Area (Volume 15)

(1) 18 Inch Steam Lines

This break causes Reactor Protective initiation because MSIV's close on a Group I due to flow. Therefore, single failure with loss of offsite power is applicable during the initial vessel level stabilization period.

SSD equipment adversely affected is: Division I MCC-133A and MCC-133B due to high temperature and being flooded, and DIV I RHRSW Line SW9-18"-GF due to pipe whip. It is assumed Division II LPCI injection is not available since the power for the LPCI injection valves in Div II MCC-143B have been hardwired to Div I MCC-133B for LPCI loop select logic during a LOCA.

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Initial Vessel Level Stabilization Period. Single failure with loss of offsite power.

- HPCI, if single failure, then
- SRV with Division II CS

Post Initial Vessel Level Stabilization Period. No single failure, no loss of offsite power.

- Vessel Level

  HPCI, or
  SRV with Division II CS, or
  Alternate Injection systems per C.5-1100

- Containment Cooling, and Decay Heat Removal

  Div II CS, SRV, Div II RHR (SPC), Div II RHR Heat Exchanger, Div II RHRSW Pump, and/or; Alternate Injection systems per C.5-1100, All Available Torus Cooling per C.5-1200.

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(2) 10 Inch Steam Line Bypass, PS7-10"-ED

This break by itself does not cause Reactor Protective initiation (Reference 2, Errata). Therefore, single failure and loss of offsite power considerations are not applicable.

SSD equipment adversely affected is: HPCI power cables and one division of SPOTMOS torus temperature instrument cable due to pipe whip and Division I MCC-133A and MCC-133B due to high temperature. Division II CS is automatically available for Vessel Inventory makeup after SRV depressurization and Division II RHR Suppression Pool cooling is available for containment and decay heat removal. Alternate Injection systems per C.5-1100 and All Available Torus Cooling per C.5-1200 are also available.

I.5.2.1.4 Turbine Operating Floor (Volume 37)

The only postulated HELBs in this compartment from the main steam system are breaks at the inlet to the High Pressure Turbine. A break at this location would not expose any SSD equipment to either pipe whip or jet impingement as there is no SSD equipment in this area. Environmental effects are evaluated in the MNGP Environmental Qualification (EQ) program based on the break cases identified in this Appendix. Bounding profiles for environmental conditions are identified. The environmental conditions include compartment pressurization, heatup and flooding effects. Any water condensing would drain either to the Condenser bay or lower elevations of the Turbine Building to areas where the water could be drained. The SSD equipment in these areas would not be adversely affected by the water produced. HELB barriers have been installed to prevent environmental effects from breaks on the turbine operating floor from affecting SSD equipment in the Div II 4KV and Div II Essential MCC areas (Volumes 31 and 20).

I.5.2.2 Feedwater System

The high energy lines for the Feedwater System are located in the feedwater pump area (Volumes 2 and 3), Turbine Building pipe chase (Volume 21), turbine operating floor (Volume 37), condenser area (Volume 15) in the Turbine Building and the main steam chase (Volume 16) in the Reactor Building.

The two feedwater lines and the feedwater regulating station piping were seismically analyzed. Break locations were selected based upon the seismic analysis of the piping and the break location criteria established for seismic Class I piping. There were no break locations in the Turbine Building pipe chase (Volume 21).

I.5.2.2.1 Reactor Feedwater Pump Area - Volumes 2 and 3

This break causes Reactor Protective initiation because a loss of Feedwater causes vessel low water level that causes a scram. Therefore, single failure with loss of offsite power is applicable during the initial vessel level stabilization period.

SSD equipment adversely affected is: Division I MCC-133A and MCC-133B due to high temperature and being flooded. It is assumed Division II LPCI injection is not available since the power for the LPCI injection valves in Div II MCC-143B have been hardwired to Div I MCC-133B for LPCI loop select logic during a LOCA. Pipe whip restraints have been installed near the pipe terminal ends at the FW pumps to prevent damage to the ceiling above that provides floor support to the Div II MCC's.

Initial Vessel Level Stabilization Period. Single failure with loss of offsite power.

- HPCI, if single failure, then
- SRV with Division II CS

Post Initial Vessel Level Stabilization Period. No single failure, no loss of offsite power.

- Vessel Level

  HPCI, or
  SRV with Division II CS, or
  Alternate Injection systems per C.5-1100

- Containment Cooling, and Decay Heat Removal

  Div II CS, SRV, Div II RHR (SPC), Div II RHR Heat Exchanger, Div II RHRSW Pump, and/or; Alternate Injection systems per C.5-1100, All Available Torus Cooling per C.5-1200.

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I.5.2.2.2 Turbine Building Pipe Chase (Volume 21)

There are no postulated feedwater line breaks within this compartment. Thus, pipe whip, jet impingement, flooding and compartment pressurization effects result from breaks in other areas (Volumes 3 and 15). Since there are no breaks in this Volume, only temperature and compartment pressurization create environmental concerns in Volume 21. Environmental effects are evaluated in the MNGP Environmental Qualification (EQ) program based on the break cases identified in this Appendix. Bounding profiles for environmental conditions are identified. The environmental conditions include compartment pressurization, heatup, and flooding effects.

I.5.2.2.3 Turbine Operating Floor (Volume 37)

This break causes Reactor Protective initiation because a loss of Feedwater causes vessel low water level that causes a scram. Therefore, single failure with loss of offsite power is applicable during the initial vessel level stabilization period.

SSD equipment adversely affected is: Division I MCC-133A and MCC-133B due to high temperature and being flooded. It is assumed Division II LPCI injection is not available since the power for the LPCI injection valves in Div II MCC-143B have been hardwired to Div I MCC-133B for LPCI loop select logic during a LOCA.

Initial Vessel Level Stabilization Period. Single failure with loss of offsite power.

- HPCI, if single failure, then
- SRV with Division II CS

Post Initial Vessel Level Stabilization Period. No single failure, no loss of offsite power. 604000000250

- Vessel Level

  HPCI, or
  SRV with Division II CS, or
  Alternate Injection systems per C.5-1100

- Containment Cooling, and Decay Heat Removal

  Div II CS, SRV, Div II RHR (SPC), Div II RHR Heat Exchanger, Div II RHRSW Pump, and/or; Alternate Injection systems per C.5-1100, All Available Torus Cooling per C.5-1200.

The resultant flooding would drain back to the Condenser area (Volume 15). The cascading water would not adversely affect any SSD equipment.

No safe shutdown equipment is located on the turbine operating floor.

I.5.2.2.4 Main Steam Chase (Volume 16)

The only postulated break locations in this area consist of the terminal end at each outboard containment isolation valve (FW-94-1&2). For a break at the terminal point on line FW2A-14-ED at the FW-94-1 outboard weld, RCIC flow would be lost because the RCIC injection line (FW5-4″-ED) connects upstream of the postulated break. For feedwater line FW2B-14″-ED, a break at the terminal end at the FW-94-2 outboard weld would cause the loss of the HPCI System, because the HPCI injection point is upstream of the break. No safe shutdown equipment can be adversely affected by pipe whip or direct jet impingement from any pipe break of the Feedwater System in the Reactor Building steam chase. However, for conservatism, both HPCI and RCIC are assumed lost due to flooding (described below) and possible jet impingement due to compartment geometry.

Flooding in the Reactor Building steam chase, as a result of a feedwater line break, would cause the loss of both HPCI and RCIC, since the injection valves for these systems (MO-2068 and MO-2107) would be submerged above the motor operators at the peak HELB flood level. Water exiting the steam chase would flow along the floor of the Reactor Building 935-foot elevation. Some of the water would also flow down to the Control Rod Drive pump room (Volume 7) and into the Elev 896-foot tank and HPCI pump rooms. Additional safe shutdown equipment would not be affected by the water exiting the steam chase, because no SSD equipment required to mitigate this break is located in the affected areas. Therefore, the flooding from the postulated feedwater line break of either feedwater line would adversely affect only HPCI and RCIC. The path to safe shutdown would be the same as that described in Section I.5.2.1.1 with the exception that an RHR or CS pump would be required to maintain RPV level.

The main steam line break provides the bounding case in the steam chase for temperature and compartment pressurization. Flooding effects from the feedwater break remove RCIC/HPCI from the available safe shutdown equipment, however a safe shutdown path exists for both break cases.

In summary, a Feedwater Line break in the main steam chase (Volume 16) could adversely affect HPCI and RCIC. The emergency service water line, SW30B-3-HF, would not be adversely affected from a feedwater line break. Paths to safe shutdown would include using the SRVs to depressurize the RPV, using CS for RPV level maintenance and RHR for decay heat removal. No postulated single active failure prevents the other division from achieving safe shutdown.

I.5.2.2.5 Condenser Area (Volume 15)

This break causes Reactor Protective initiation because a loss of Feedwater causes vessel low water level that causes a scram. Therefore, single failure with loss of offsite power is applicable during the initial vessel level stabilization period.

SSD equipment adversely affected is: Division I MCC-133A and MCC-133B due to high temperature and being flooded. It is assumed Division II LPCI injection is not available since the power for the LPCI injection valves in Div II MCC-143B have been hardwired to Div I MCC-133B for LPCI loop select logic during a LOCA.

Initial Vessel Level Stabilization Period. Single failure with loss of offsite power.

- HPCI, if single failure, then
- SRV with Division II CS

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Post Initial Vessel Level Stabilization Period. No single failure, no loss of offsite power.

- Vessel Level

  HPCI, or
  SRV with Division II CS, or
  Alternate Injection systems per C.5-1100

- Containment Cooling, and Decay Heat Removal

  Div II CS, SRV, Div II RHR (SPC), Div II RHR Heat Exchanger, Div II RHRSW Pump, and/or; Alternate Injection systems per C.5-1100, All Available Torus Cooling per C.5-1200.

I.5.2.3 Condensate System

The high energy Condensate System lines are located in the condenser area (Volume 15), Turbine Building pipe chase area (Volume 21), and feedwater pump area (Volumes 2 and 3) in the Turbine Building.

Break locations for portions of the condensate system piping were selected based upon a seismic analysis of the piping and break location criteria established for Class I piping.

I.5.2.3.1 Condenser Area (Volume 15)

This break causes Reactor Protective initiation because a loss of Feedwater causes vessel low water level that causes a scram. Therefore, single failure with loss of offsite power is applicable during the initial vessel level stabilization period.

SSD equipment adversely affected is: Division I MCC-133A and MCC-133B due to high temperature and being flooded. It is assumed Division II LPCI injection is not available since the power for the LPCI injection valves in Div II MCC-143B have been hardwired to Div I MCC-133B for LPCI loop select logic during a LOCA.

Initial Vessel Level Stabilization Period. Single failure with loss of offsite power.

- HPCI, if single failure, then
- SRV with Division II CS

Post Initial Vessel Level Stabilization Period. No single failure, no loss of offsite power.

- Vessel Level

  HPCI, or
  SRV with Division II CS, or
  Alternate Injection systems per C.5-1100

- Containment Cooling, and Decay Heat Removal

  Div II CS, SRV, Div II RHR (SPC), Div II RHR Heat Exchanger, Div II RHRSW Pump, and/or; Alternate Injection systems per C.5-1100, All Available Torus Cooling per C.5-1200.

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I.5.2.3.2 Turbine Building Pipe Chase (Volume 21)

There are no postulated condensate line HELBs within this compartment.

I.5.2.3.3 Reactor Feedwater Pump Area (Volumes 2 and 3)

The postulated condensate line HELBs in these compartments would not affect any additional SSD equipment other than the SSD equipment described for a feedwater line break in the same compartment. The effects of pipe whip, jet impingement, flooding, compartment pressurization, and environmental effects are less adverse than for the feedwater line break. Therefore, the same evaluations, as provided in Section I.5.2.2.1 of this report, apply, and the same paths to safe shutdown can be used.

I.5.2.4 High Pressure Coolant Injection (Steam) System

The steam supply line (PS18-8″-ED) to the HPCI Turbine is located in the steam chase (Volume 16). The torus area (Volumes 9 and 12) and HPCI compartment area (Volume 8).

I.5.2.4.1 Main Steam Chase (Volume 16)

Possible pipe whip targets include the feedwater and main steam lines which are assumed to be unaffected, because they are larger and thicker walled than the HPCI steam line. No other SSD equipment is affected by a HPCI steam line pipe whip.

Jet impingement targets include the ceiling through which Division II embedded conduits are routed. The effects of jet impingement on the ceiling is discussed in Section I.5.2.1.1. However, the environmental effects from a HPCI steam line break are less severe than a break of one of the main steam lines.

For this postulated HELB, the path to safe shutdown would consist of using the SRVs for RPV depressurization and Division I CS to maintain reactor water level and Division I RHR to remove decay heat. No single active failure can prevent the use of either essential division to safely shutdown the unit.

Compartment flooding would not affect any SSD equipment. The HPCI steam line break is bounded by the main steam line break; therefore, any environmental effects will be less severe. Thus, a path to safe shutdown has been demonstrated.

Isolation of the HPCI steamline PS18-8″-ED is assured, since the terminal end break cannot target the outboard containment isolation valve MO-2035 or the cabling to the valve. Therefore, even with a single active failure to either containment isolation valve on line PS18-8″-ED, isolation of the break is accomplished with the remaining containment isolation valve.

I.5.2.4.2 Torus Area (Volumes 9 and 12)

No intermediate break locations were postulated on the HPCI Steam line in this compartment.

I.5.2.5 RCIC (Steam) System

The high energy piping for the RCIC System is located in the steam chase (Volume 16), the torus area (Volume 9) and RCIC compartment (Volume 5) in the Reactor Building.

I.5.2.5.1 Main Steam Chase (Volume 16)

For a postulated HELB on the RCIC steam line, the only SSD equipment affected by either pipe whip or jet impingement would be the ceiling to the compartment, in which Division II control and power cables are located, and the HPCI steam line (PS18-8″-ED). The effects on the ceiling are bounded by effects from a postulated main steam line break, and so have no impact on safe shutdown. Due to the potential jet orientation, the loss of the HPCI steamline, and hence, the HPCI System was assumed. For this postulated HELB, the path to safe shutdown would consist of using the SRVs for RPV depressurization and Division I CS to maintain reactor water level and Division I RHR to remove decay heat. No single active failure can prevent the use of either essential division to safely shutdown the unit.

Flooding from the RCIC steam line break in the main steam chase, assuming all steam condenses and remains in the compartment, would be bounded by the main steam line break. The environmental conditions generated both in the main steam chase and in adjoining compartments would be bounded by the effects of a postulated main steam line break. Therefore, no new adverse conditions have been produced, and paths to safe shutdown described for the postulated main steam line break are applicable here.

Isolation of the RCIC steamline PS17-3″-ED is assured, since the terminal end break cannot target the outboard containment isolation valve MO-2076, or the cabling to the valve. Therefore, even with a single active failure to either containment isolation valve on line PS17-3″-ED, isolation of the break is accomplished with the remaining containment isolation valve.

I.5.2.5.2 Torus Area (Volume 9)

No intermediate break locations were postulated on the RCIC Steam line in this compartment.

I.5.2.5.3 RCIC Compartment (Volume 5)

The only SSD components in this compartment are the RCIC equipment with the exception of the Division I RHR and Core Spray pumps' 4 KV power cables. There are no breaks or critical cracks in the RCIC compartment.

I.5.2.6 Reactor Water Clean Up (RWCU) System

The Reactor Water Cleanup high energy lines are located in the RWCU compartment (Volumes 28-32), the steam chase (Volume 16), the MG set room (Volume 26), the northwest side of elevation 962-foot (Volume 27) and the northwest side of elevation 935-foot of the Reactor Building (Volume 18).

The RWCU supply REW3-4″-ED and the return line REW6-3″-ED were seismically analyzed. Break locations were selected based upon the seismic analysis of the piping.

I.5.2.6.1 RWCU Area (Volume 28-32)

The possible RWCU supply line pipe whip or jet impingement targets consist of conduits which supply power to the Division II reactor sample outboard containment valve, both Division II Core Spray containment isolation valves, and both Primary Containment Atmospheric Control (PCAC) valve's air/nitrogen supply lines.

The Primary Containment Atmospheric Control (PCAC) valve and one Core Spray isolation valve are normally closed. Loss of power to these valves would cause no change of state and create no adverse concerns. In addition, the Division II Core Spray System has a check valve located inside primary containment unaffected by the HELB.

Jet impingement in the RWCU compartment can conservatively target any of the above valves. The RWCU outboard containment isolation valve is not a jet impingement target because of pipe geometry. Isolation of the RWCU supply line, REW 3-4″-ED, is assured, since the terminal end break cannot target the outboard containment isolation valve MO-2398, or the cabling to the valve. Therefore, even with a single active failure to either containment isolation valve on line REW 3-4″-ED, isolation of the break is accomplished with the remaining containment isolation valve.

Redundant valves inside the Primary Containment or located outside this compartment mitigate any concerns on loss of the above components. The air supply to the PCAC isolation valves AO-2386 and AO-2387 was upgraded and hardened against postulated critical cracks. For the affected valves, safe shutdown can be accomplished using the Division I equipment and the HPCI System. Sufficient redundancy exists to be able to shutdown the unit assuming a single active failure.

Compartment flooding would not be a problem, since no other SSD equipment would be adversely affected by flooding other than the SSD equipment directly affected by pipe whip or jet impingement. Environmental effects are evaluated in the MNGP Environmental Qualification (EQ) program based on the break cases identified in this Appendix. Bounding profiles for environmental conditions are identified. The environmental conditions include compartment pressurization, heatup, and flooding effects.

I.5.2.6.2 MG Set Room (Volume 26)

There are no postulated breaks or cracks in this area. The RWCU return piping outside the RWCU compartment is seismically analyzed in accordance with the criteria defined in Section I.3.1.1(4). There are no postulated intermediate RWCU HELBs or critical cracks outside the RWCU compartment.

I.5.2.6.3 Reactor Building 962-Foot Elevation, Northwest Area (Volume 27)

There are no postulated breaks or cracks in this area. There are no postulated intermediate RWCU HELBs or cracks outside the RWCU compartment as identified in I.5.2.6.2.

I.5.2.6.4 Reactor Building Open Area, 935-Foot Elevation, West Side (Volume 18)

There are no postulated pipe breaks or cracks in this area. There are no postulated intermediate RWCU HELBs or cracks outside the RWCU compartment as identified in I.5.2.6.2.

I.5.2.6.5 Main Steam Chase (Volume 16)

There are no pipe whip targets for SSD in this area. Any other concerns in this area are bounded by other larger pipes discussed in Sections I.5.2.1, I.5.2.2, and I.5.2.4.

Safe shutdown would be accomplished in the manner described in Section I.5.2.1.1, with potentially either HPCI or RCIC not available.

I.5.2.7 Core Spray System

As indicated in I.2.1.8, there are no postulated break locations for this system.

I.5.2.8 Residual Heat Removal System

As indicated in I.2.1.9, there are no postulated break locations for this system.

I.5.2.9 HPCI (Water) System

The high energy portion of this system is located entirely within the main steam chase (Volume 16). The environmental effects of this postulated HELB are bounded by the Feedwater line break in the steam chase.

I.5.2.10 RCIC (Water) System

The high energy portion of this system is located entirely within the main steam chase (Volume 16). The environmental effects of this postulated HELB are bounded by the Feedwater line break in the steam chase.

I.5.2.11 Standby Liquid Control

Line CH2-1 1/2″-DC from containment penetration X-42 to check valve XP-6 is not routed in the vicinity of any safe shutdown components. An inboard check valve will limit any energy release, and because of the small volume of water contained between the inboard containment check valve and valve XP-6, compartment (Volume 13) pressurization and flooding are not a concern because of an extensive drain system and a large free volume.

I.5.2.12 Off-Gas System

The steam supply line SHP101-4″ is located in the condenser bay area (Volume 15), and the SJAE room (Volume 14) in the Turbine Building. The effects of postulated breaks on this line in the condenser area are bounded by the postulated breaks on other lines such as Feedwater and Main Steam (see Sections I.5.2.1 and I.5.2.2). There is no SSD equipment in the SJAE Room to be affected by a pipe whip or jet impingement. Postulated breaks on line SHP101-4″ are bounded by other postulated HELBs, for which paths to safe shutdown have been demonstrated (see Sections I.5.2.1 and I.5.2.2). All other high energy lines in the Offgas System are located in the Offgas Storage and Compressor Building, in which no safe shutdown equipment is located. Since the occurrence of a HELB in this area can have no impact on systems required to mitigate accidents, no further evaluations need to be made.

I.5.2.13 Zinc Oxide Injection System (GEZIP)

The portions of the GEZIP system with postulated HELBs are the 1 1/2″ portions near the Feedwater and Condensate lines (Volume 3) in the Turbine Building. However, due to the proximity to the postulated Feedwater and Condensate Systems HELBs, any adverse effects from a GEZIP HELB would be enveloped by a postulated HELB on one of the Feedwater or Condensate lines. Since pathways to safe shutdown have been demonstrated for the Feedwater and Condensate postulated HELBs, these same pathways would exist for a much smaller GEZIP HELB.

I.5.3 <u>Table of System Effects</u>

Table I.5-2 shows the effect of specific high energy line breaks by volume and system. This table includes the required auxiliary systems which are considered potential HELB targets. Any system not affected by a postulated HELB is to be considered available to support safe shutdown. The meaning of the letter codes used in the table are as follows:

F - Primary failure as a direct result of a line break.

A - System is unaffected by a line break and is available to support safe shutdown.

U - The system is unavailable due to the failure of a required function or component associated with another system.

Table I.5-1 Location of High Energy Systems and Safe Shutdown Equipment by Volume

LOCATION OF HIGH ENERGY SYSTEMS AND SAFE SHUTDOWN EQUIPMENT BY VOLUME (REACTOR BUILDING)
PAGE 1 OF 2

HIGH ENERGY SYSTEM	VOLUME [1] (SEE PAGE 2 FOR ADDITIONAL VOLUMES)																	
	1	14	15	13	21	3	6	8	7	18	16	20	33	28-32	5	17	9-12	26-27
MAIN STEAM	—	—	—	—	—	—	—	—	—	—	HEBC	—	—	—	—	—	—	—
FEEDWATER	—	—	—	—	—	—	—	—	—	—	HEBC	—	—	—	—	—	—	—
CONDENSATE	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
HPCI (STEAM)	—	—	—	—	—	—	—	—	—	—	HEBC	—	—	—	—	—	HEC	—
RCIC (STEAM)	—	—	—	—	—	—	—	—	—	—	HEBC	—	—	—	—	—	HEC	—
REACTOR WATER CLEANUP	—	—	—	—	—	—	—	—	—	HE	HEBC	—	—	HEBC	—	—	—	HE
CORE SPRAY	—	—	—	—	HE	—	—	—	—	—	—	—	—	HE	—	—	—	—
RESIDUAL HEAT REMOVAL	—	—	—	HE	—	—	—	—	—	—	—	HE	—	—	—	—	—	—
HPCI (INJECTION)	—	—	—	—	—	—	—	—	—	—	HEBC	—	—	—	—	—	—	—
RCIC (INJECTION)	—	—	—	—	—	—	—	—	—	—	HEBC	—	—	—	—	—	—	—
STANDBY LIQUID CONTROL	—	—	—	HEB	—	—	—	—	—	—	—	—	—	—	—	—	—	—
OFF-GAS	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
CRD SYSTEM	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
GEZIP	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
INSTRUMENT AND SAMPLE LINES	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
DOES VOLUME CONTAIN SAFE SHUTDOWN EQUIPMENT?	YES	YES	YES	YES	YES	YES	YES	YES	YES	YES	YES	YES	YES	YES	YES	YES	YES	YES

[1] SEE SECTION I.2.1

Table I.5-1 Location of High Energy Systems and Safe Shutdown Equipment by Volume

LOCATION OF HIGH ENERGY SYSTEMS AND SAFE SHUTDOWN EQUIPMENT BY VOLUME (TURBINE BUILDING)

PAGE 2 OF 2

HIGH ENERGY SYSTEM	**VOLUME[1]**					
	3	2	21	15	14	37
MAIN STEAM	—	—	—	HEBC	HEB	HEBC
FEEDWATER	HEB	HEC	HE	HEBC	—	HEBC
CONDENSATE	HEB	HE	HE	HEBC	—	—
HPCI (STEAM)	—	—	—	—	—	—
RCIC (STEAM)	—	—	—	—	—	—
REACTOR WATER CLEANUP	—	—	—	—	—	—
CORE SPRAY	—	—	—	—	—	—
RESIDUAL HEAT REMOVAL	—	—	—	—	—	—
HPCI (INJECTION)	—	—	—	—	—	—
RCIC (INJECTION)	—	—	—	—	—	—
STANDBY LIQUID CONTROL	—	—	—	—	—	—
OFF-GAS	—	—	—	HEB	HEB	—
CRD SYSTEM	—	—	—	—	—	—
GEZIP	HEB	—	—	—	—	—
INSTRUMENT AND SAMPLE LINES	—	—	—	—	—	—
DOES VOLUME CONTAIN SAFE SHUTDOWN EQUIPMENT?	YES	YES	YES	YES	NO	NO

KEY

— - HIGH ENERGY PIPING OF THIS SYSTEM IS NOT LOCATED IN THIS COMPARTMENT, OR ALL HIGH ENERGY PIPING OF THIS SYSTEM HAS BEEN EXCLUDED BY 2% CRITERION, IS 1” NOMINAL PIPE SIZE OR SMALLER, OR BREAKS ARE NOT REQUIRED TO BE ASSUMED.

HE - CONTAINS HIGH ENERGY PIPING OF THIS SYSTEM (NOT EXCLUDED BY 2% CRITERION AND IS LARGER THAN 1, NOMINAL PIPE SIZE).

HEB - CONTAINS BREAK LOCATIONS FOR THE HIGH ENERGY PIPING OF THIS SYSTEM.

HEC - CONTAINS CRITICAL CRACK LOCATIONS FOR THE HIGH ENERGY PIPING OF THIS SYSTEM.

HEBC - CONTAINS BREAK AND CRITICAL CRACK LOCATIONS FOR THE HIGH ENERGY PIPING OF THIS SYSTEM.

APPENDIX I EVALUATION OF HIGH ENERGY LINE BREAKS OUTSIDE CONTAINMENT

Table I.5-2 Table of System Effects

SYSTEM	Safe shutdown system	STEAM CHASE (16)					CONDENSER AREA (15)					SJAE 14 ROOM	TURBINE BLDG. FEED PUMP AREA (2, 3)				RWCU COMP (28-32)	REACTOR BLDG. (13)	TURBINE (37) BLDG.	
		MAIN STEAM PS 1.2.3.4	HPCI PS18	RCIC PS17	FEEDWATER	RWCU REW6	CONDENSATE C4A	CONDENSATE C4B	FEEDWATER	STEAM BYPASS PS7	MAIN STEAM PS 4.3.2.1	MAIN STEAM PS 9	CONDENSATE C4B	CONDENSATE C4A	FEEDWATER FW2A	FEEDWATER FW2B	REW6 REW3	SBLC CH2	MAIN STEAM PS 4, 3, 2, 1	Feed Water
HPCI	□ ○	A	F	F	F	U	A	A	A	F	A	A	A	A	A	A	A	A	A	A
RCIC (12)	□	A	A	F	F	U	A	A	A	A	A	A	U	U	U	U	A	A	A	A
SRVS (7)	△ ○	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A
RHR-LPCI DIV1	□	A	A	A	A	A	U	U	U	U	U	U	U	U	U	U	A	A	A	U
RHR-LPCI DIV2	□	A	A	A	A	A	U (11)	U (11)	U (11)	U (11)	U (11)	U (11)	U (11)	U (11)	U (11)	U (11)	A	A	A	U (11)
RHR-SPC DIV1	△	A	A	A	A	A	U	U	U	U	U	U	U	U	U	U	A	A	A	U
RHR-SPC DIV2	△	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A
RHR-SDC DIV1	△	A	A	A	A	A	U	U	U	U	U	U	U	U	U	U	A	A	A	U
RHR-SDC DIV2	△	A	A	A	A	A	U (11)	U (11)	U (11)	U (11)	U (11)	U (11)	U (11)	U (11)	U (11)	U (11)	F (5)	A	A	U (11)
RHR-SERVICEWATER DIV1	△	A	A	A	A	A	U	U	U	U	F (3)	U	U	U	U	U	A	A	A	U
RHR- SERVICEWATER DIV2	△	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A
CORESPRAY DIV1	□ △	A	A	A	A	A	U	U	U	U	U	U	U	U	U	U	A	A	A	U
CORESPRAY DIV2	□ △	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	F	A	A	A
EMERGENCY SERVICE WATER DIV1	⬡	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A
EMERGENCY SERVICE WATER DIV2	⬡	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A
EMERGENCY POWER DIV1	⬡	A	A	A	A	A	F (4)	F (4)	F (4)	F (4)	F (4)	F (4)	F (4)	F (4)	F (4)	F (4)	A	A	A	F (4)
EMERGENCY POWER DIV2	⬡	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A
RPS CONTROL ROD VALVES CONTROL RODS	◇	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A
HVAC ECCS ROOM DIV 1	⬡	A	A	A	A	A	F (4)	F (4)	F (4)	F (4)	F (4)	F (4)	F (4)	F (4)	F (4)	F (4)	A	A	A	F (4)
HVAC ECCS ROOM DIV 2	⬡	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A

KEY

F= PRIMARY FAILURE AS A DIRECT RESULT OF A LINE BREAK.

A= SYSTEM IS UNAFFECTED BY A LINE BREAK AND IS AVAILABLE TO SUPPORT SAFE SHUTDOWN.

U= THE SYSTEM IS UNAVAILABLE DUE TO THE FAILURE OF A REQUIRED FUNCTION OR COMPONENT ASSOCIATED WITH ANOTHER SYSTEM.

SAFE SHUTDOWN SYSTEM

◇ — REACTIVITY CONTROL

□ — RESTORE AND MAINTAIN RPV WATER LEVEL

○ — RPV DEPRESSURIZATION

△ — DECAY HEAT REMOVAL / CONTAINMENT COOLING

⬡ — AUXILIARY SUPPORT

NOTES;

1. DELETED
2. DELETED
3. APPLIES TO PS4 ONLY.
4. MCC 133A,B.
5. DIV. 2 CONTAINMENT SPRAY ONLY.
6. THERE ARE NO SSD TARGETS IN THE AREA.
7. THE SRVS ARE RESPONSIBLE FOR RPV OVERPRESSURE PROTECTION.
8. DELETED.
9. DELETED.
10. DELETED
11. UNAVAILABLE UNTIL ISOLATED FROM MCC 133B.
12. Though not required safe shutdown equipment, RCIC if available could be used.

60400000250

I.6 References

1. Not used

2. Letter from A. Giambusso, Deputy Director for Reactor Projects, AEC, to Northern States Power Company, Subject: High Energy Breaks Outside of the Containment, December 18, 1972, plus Errata dated January 16, 1973. Roll 2943, Blip 0075. 604000000250

3. Standard Review Plan 3.6.1, Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment, Rev. 1, July 1981.

4. Standard Review Plan 3.6.2, Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping, Rev. 1, July 1981.

5. Not used.

6. United States Atomic Energy Commission - Safety Evaluation by the Directorate of Licensing, Docket No. 50-263, Monticello Nuclear Generating Plant - “Analysis of the Consequences of High Energy Piping Failures Outside Containment”, July 29, 1974. Roll GO 8009, Blip 1451. 604000000250

7. ANSI/ANS-58.2-1980, American National Standard, Design Basis for Protection of Light Water Nuclear Power Plants Against Effects of Postulated Pipe Rupture, December 31, 1980.

8. NSP (E C Ward) letter to AEC (A. Giambusso), “Monticello Nuclear Generating Plant E-5979 Postulated Pipe Failures Outside Containment,” Docket No. 50-263, dated September 7, 1973, with attachment, “Postulated Pipe Failures Outside Containment”, August 1973. Roll GO 8009, Blip 0127; Roll 2906, Blip 1489. 604000000250

9. NRC NRR Office Instruction, “Control of Licensing Bases for Operating Reactors”, LIC-100, Revision 1, January 7, 2004.

10. NSP (D M Musolf) letter to NRC, “Failure to Provide High Energy Line Break Protection for the Turbine Building Pipe Chase”, dated April 9, 1986.

11. NUTECH Engineering Report NSP-30-102, Revision 0, “Review of High Energy Line Analysis for the Monticello Nuclear Generating Station”, June 1986.

12. NSP letter to Terry Maxey; Subject: Evaluation of RHR Pressure and Temperature Increase for Environmental Qualification Impact, June 11, 1985.

13. NRC (R M Bernero) Generic Letter 86-01, “Safety Concerns Associated with Pipe Breaks in the BWR Scram System”, January 3, 1986.

14. Scram Discharge Volume Design Change Recommendations, SIL No. 331, October 13,1980.

15. Survey Book, High Energy Lines in Systems considered in 1972 HELB Analysis, May 1986, Calculation 86-600. Roll 2804, Blip 780, ARMS Number C920325220. 604000000250

16. Survey Sheets and Survey Notes for Identified Non-Bechtel High Energy Lines, June 1986, Calculation 86-595. Roll 2804, Blip 758, ARMS Number C920325219.

17. Nutech Calculation NSP730.0101-001 “Determination of High Energy Pipe Locations for the Systems, Bechtel Looked at in 1973”.

18. Drawing, ND-100482, ESW-EFT ESW “A” Division Crosstie Piping ISO, Revision 0, 10/14/87.

19. Drawing, ND-100483, ESW-EFT ESW “B” Division Crosstie Piping ISO, Revision 0, 10/14/87.

20. Deleted.

21. NRC (F J Miraglia) Generic Letter 87-11, “Relaxation in Arbitrary Intermediate Pipe Rupture Requirements”, June 19, 1987.

22. Deleted.

23. Nutech Calculation Package, NSP730.0020-001, Determination of “High Energy Pipe Break” Locations, June 1986.

24. Letter from David Musolf (NSP) to Director, Office of Nuclear Reactor Regulation, “Additional Information Related to High Energy Line Breaks Analysis”, December 4, 1986.

25. Letter from David Musolf (NSP) to Director, Office of Nuclear Reactor Regulation, “Resolution of Deficiencies in High Energy Line Break Protection for the Monticello Plant,” June 18, 1986.

26. ND-178628-1, Primary Steam System Break and Critical Crack Locations (PS1,2,3,4-18″-ED & PS30-18″-EDB).

27. ND-178628-2, Additional Primary Steam Break and Critical Crack Locations (PS6,7,8,11,12,13,14,21).

28. ND-178628-3, Additional Primary Steam Break and Critical Crack Locations (PS15-3″-EB).

29. ND-178628-4, HPCI System Steam Line Break and Critical Crack Locations (PS18-8″-ED).

30. ND-178628-5, Primary Steam Drain Break and Critical Crack Locations (D1,2,3-1.5″-ED & D4-2″-ED).

31. ND-178628-6, Additional Primary Steam Break Locations (PS9-3″-ED).

32. ND-178628-7, Additional Primary Steam Break Locations Air ejector E-2A (PS9-2″-ED).

33. ND-178628-8, Additional Primary Steam Break Locations Air Ejector E-2B (PS9-2″-ED).

34. ND-178628-9, Main Steam Drain Break Locations (D23-2″-EB).

35. ND-178628-10, Main Steam Equalizing Drain to Condenser Break Locations (D40-1.5″-EDB).

36. ND-178628-11, Condensate System Break Locations (C4A-16″-GB).

37. ND-178628-12, Condensate System (Continued) Break and Critical Crack Locations (C4B-16″-GB).

38. ND-178628-13, Condensate System (Continued) Break Locations (C4A-1.5″-EB).

39. ND-178628-14, Condensate System (Continued) Break Locations (C4B-1.5″-EB).

40. ND-178628-15, Feedwater System Break and Critical Crack Locations (FW2A-14″-DE, ED & FW2-6″-DE).

41. ND-178628-16, Feedwater System (Continued) Break Locations (FW2B-14″-DE, ED).

42. ND-178628-17, Standby Liquid Control System Break Locations (CH2-1 1/2″-DC).

43. ND-178628-18, Feedwater System (Continued) Break Locations Feedwater Heater E-15A (FW10-8″-DE & FW2A-14″-DE).

44. ND-178628-19, Feedwater System (Continued) Break Locations Feedwater Heater E-15B (FW10-8″-DE & FW2B-14″-DE).

45. ND-178628-20, Feedwater System (Continued) Break Locations Feedwater Recirculation lines (FW10-8″-DE).

46. ND-178628-21, Reactor Feed Pumps P-2A & P-2B (FW12A-1.5″-DE & FW12B-1.5″-DE).

47. ND-178628-22, Control Rod Drive Break Locations (CRD7-3″-DE).

48. ND-178628-24, HPCI Injection Line Break and Critical Crack Locations (TW3-12″-ED).

49. ND-178628-25, RCIC Injection Line Break and Critical Crack Locations (FW5-4″-ED).

50. ND-178628-26, RCIC System Steam Line Break and Critical Crack Locations (PS17-3″-ED).

51. ND-178628-27, Reactor Water Cleanup System Break and Critical Crack Locations (REW3-4″-DCD,DBD).

52. ND-178628-28, Reactor Water Cleanup Return Line Break Locations (REW6-3″-DBD).

53. ND-178628-29, Reactor Water Cleanup System (Continued) Break and Critical Crack Locations (REW6-3″-DC,DB,ED & REW6-1.5″-DB & REW6-4″-DB).

54. ND-178628-30, Off-Gas System Break Locations (SHP101-4″-HN1C).

55. ND-178628-31, Off-Gas System (Continued) Break Locations (SHP101-4″-HN1C).

56. ND-178628-32, Off-Gas System Break Locations (SHP101-4″-HN1C).

57. ND-178628-33, Steam Seal Regulator Piping Break Locations (PS10-5″ & PS10-3″).

58. ND-178628-34, GEZIP (OUT) System HELB Break & Critical Crack Locations (GZP 1 1/2"). 604000000250

59. NRC (D C DiIanni) letter to NSP (D M Musolf), “Request For Additional Information Related to High Energy Line Breaks”, dated September 09, 1987, with RAI enclosure.

60. NSP (D M Musolf) letter to NRC, “Additional Information Related to Updated High Energy Line Break Analysis”, dated December 14, 1987

61. NRC (W O Long) letter to NSP (T M Parker), “Monticello High Energy Line Break Analysis (TAC No. 61788)”, dated June 13, 1990

62. NSPM letter L-MT-09-044 (T J O'Connor) to NRC dated August 21, 2009, “Monticello Extended Power Uprate: Response to NRC Mechanical and Civil Engineering Review Branch (EMCB) Requests for Additional Informations (RAIs) dated March 28, 2009 (TAC No. MD9990)”, dated August 21, 2009.

63. NRC (T A Beltz) letter to NSPM (K D Fili), "Monticello Nuclear Generating Plant - Issuance of Amendment No. 176 to Renewed Facility Operating License Regarding Extended Power Uprate (TAC NO. MD9990) ", dated December 9, 2013.

64. EC 24933, Calculation 01-002 Rev 1, Temperature and As-built Analyses of CRD and RWCU Piping; Terminal End Definitions.

65. Change Request 604000000250, USAR Appendix I, HELB, Revision

66. NRC SER for Power Rerate, Amendment 102, dated September 16, 1998, Roll 4586, Blip 2110. 604000000250

67. MNGP License Amendment Request for Re-rate, Revision 1, dated December 4, 1997, Roll 4385, Blip 1109.

FIGURES

Withheld Security-Related Information

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