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4.1 Summary Description

The reactor coolant system (also called the reactor primary system) includes the reactor vessel; the 2-loop reactor coolant recirculation system with its pumps, pipes and valves; the main steam piping up to the main steam isolation valves; safety/relief valves; and the reactor auxiliary systems piping. The important parameters of the reactor coolant system are summarized in Table 4.1-1.

The piping susceptible to Intergranular Stress Corrosion Cracking (IGSCC) in the recirculation system, the residual heat removal system, and the core spray system has been replaced with material resistant to IGSCC or protected with a cladding of resistant weld metal. To further reduce susceptibility to IGSCC, a Hydrogen Water Chemistry System was placed in operation in 1988.

The NRC's evaluation of NSP's response to Generic Letter 88-01 concluded that the IGSCC inspection and mitigation program provides reasonable assurance of maintaining the long-term structural integrity of Austenitic stainless steel piping in the Monticello BWR Nuclear Power Plant (Reference 37). The materials of construction for the reactor coolant pressure boundary (RCPB) piping and safe-end materials at MNGP are identified in Appendices A and B of the NRC staff's evaluation of MNGP response to GL 88-01 that noted that all welds in the RCPB within the scope established by GL 88-01 are Category "A." All RCPB welds at MNGP are resistant to sensitization and intergranular stress corrosion cracking (References 37 and 134).

Table 4.1-1 Reactor Coolant System Data
(Page 1 of 3)

REACTOR VESSEL	
Internal height	63 ft 1-1/2 in.
Internal diameter	17 ft 1 in.
Design pressure and temperature	1250 psig @ 575°F
Maximum heatup rate	100°F/hr
Design lifetime	40 years
Base metal material	SA533 GR. B cc1339, CL. 1
Wall thickness	5-1/16 in. minimum
Base metal initial NDT temperature	40°F maximum
Cladding material	Weld deposited ER308ELC electrode
Cladding thickness	0.125 in.
Design Code	ASME Section III, Class A, 1965 Edition with Summer 1966 Addenda
RECIRCULATION LOOPS	
Number	2
Pipe Size	28-in. (nominal OD)
Material	Type 316 nuclear grade stainless steel
Design pressure and temperature	
Suction	1148 psig @ 562°F
Discharge	1248 psig @ 562°F
Design code[1]	ANSI B31.1, 1977 Edition through Summer 1978 Addenda
RECIRCULATION PUMPS	
Number	2
Type	Vertical, centrifugal, single stage, variable speed

1. The Nuclear Class I requirements have been verified in accordance with the rules of ASME Section III, NB-3600, 1980 Edition including Addenda through Summer 1982. Design conditions for Level A, B, C, and D service limits are considered in the analysis.

Table 4.1-1 Reactor Coolant System Data
(Page 2 of 3)

Power rating (MG Set Motor) (Pump Motor Nameplate)	4000 hp 3500 hp
Flow rate	32,500 gpm/pump
Design pressure and temperature	1380 psig @ 575°F
Total developed head	400 ft
Design code	ASME Section III, Class C
RECIRCULATION VALVES	
Number	Four 28-in.
Type	Motor operated gate
Design code	USAS B 31.1.0, 1967
JET PUMPS	
Number	20
Material	Type 304 stainless steel
Overall height (top of nozzle to diffuser discharge)	20 ft 10 in.
Diffuser diameter	14-3/4 in.
MAIN STEAM LINES	
Number	4
Diameter	18 in.
Material	Carbon steel
Design code	ASME Section I and III, 1977 Edition with Winter 1978 Addenda and USAS B31.1.0, 1967
SAFETY/RELIEF VALVES	
Number	8
Capacity (each)	800,000 lb/hr at 1100 psig (nominal)
Set Pressure Range Capability	1025-1155 psid
Set Pressure Setting	1109 psig ± 1%
Design code	ASME Section III, 1968 Edition with Winter 1968 Addenda; USAS B31.1.0, 1967 and B16.5, 1961

Table 4.1-1 Reactor Coolant System Data
(Page 3 of 3)

MAIN STEAM ISOLATION VALVES

Number	8 (2 ea. in 4 lines)
Size	18 in.
Material	ASTM A216GR WCB
Design Code	USAS B.31.1.0, 1967 and B.16.5, 1961 (Inboard) ANSI B.31.1, 1986 and B.16.34, 1981 (Outboard)

4.2 Reactor Vessel

4.2.1 Design Basis

The performance objectives of the reactor vessel are: (1) To contain the reactor core, the reactor internals, and the reactor core coolant-moderator; and (2) To serve as a high integrity barrier against leakage of radioactive materials to the drywell. To achieve these objectives the reactor vessel was designed using the criteria specified in Table 4.1-1.

The nominal operating pressure of 1000 psig was chosen on the basis of economic analyses for boiling water reactors. The reactor vessel design pressure of 1250 psig was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation, with additional allowances to accommodate transients above the operating pressure without causing operation of the safety/relief valves.

In addition to the above criteria, the reactor vessel was also originally designed for the transients which could occur during the design 40-year life. The reactor vessel was analyzed for the cycles listed in Table 4.2-1. Since these cycle numbers are based on fatigue usage factors less than one, they do not represent the absolute maximum number of cycles the vessel could survive, but rather the current design values. Subsequent analyses show fatigue usage factors remain less than one through at least 60 years of operation. Fatigue factors less than one indicate the vessel can support transients through 60 years of operation (Reference USAR Section K).

The reactor vessel was originally designed and fabricated for a 40-year life based upon the specified design and operating conditions. Design changes involving the feedwater nozzles and spargers assure that the feedwater nozzle design life is compatible with the reactor vessel design life. Subsequent analyses support operation for 60 years based upon specified design and operating conditions (Reference USAR Section K).

4.2.2 Description

The reactor vessel is a vertical cylindrical pressure vessel with hemispherical heads of welded construction as shown in Figure 4.2-1. The base plate material is high strength alloy carbon steel SA-533, Grade B. The vessel interior is clad with weld-deposited stainless steel using E-308 electrode. The vessel was fabricated in sections in the shop and shipped to the reactor site for field assembly. The head spray and head vent nozzle internal weld surfaces include the use of a corrosion resistant clad to mitigate potential for IGSCC (Reference 37).

The reactor vessel top head is secured to the reactor vessel by studs, nuts, and bushings. The vessel flanges are sealed with two concentric silver plated, stainless steel, self-energizing O-rings. The area between the O-rings is vented and monitored to provide an indication of leakage from the inner O-ring seal.

The control rod drive housings are inserted through the control rod drive penetrations in the reactor vessel bottom head and are welded to the stub tubes extending into the reactor vessel. The design of the stub tubes which consist of the control rod drive penetrations is described in GE-APED-5703, “Design and Analysis of Control Rod Drive Reactor Vessel Penetrations”, November 1968 (Reference 42). Each housing transmits

a number of loads to the bottom head of the reactor. These loads include the weight of a control rod and control rod drive, which are bolted to the housing from below, the weight of a control rod guide tube, one four-lobed fuel support piece, and the four fuel assemblies which rest on the top of the fuel support piece. The housings are fabricated of Type 304 Austenitic stainless steel.

The control rod drive housing support is designed to prevent a nuclear transient in the unlikely event of a control rod drive housing failure. This device consists of a grid structure located below the reactor vessel from which housing supports are suspended. The supports allow only slight movement of the control rod drive or housing in the event of failure.

The in-core neutron flux monitor housings are inserted up through the in-core penetrations in the bottom head of the reactor vessel and are welded to the inner surface of the bottom head. An in-core flux monitor guide tube is welded to the top of each housing and either a source range monitor/intermediate range monitor (SRM/IRM) drive unit or a local power range monitor (LPRM) is bolted to the seal-ring flange at the bottom of the housing.

The reactor vessel is supported by a steel skirt. The top of the skirt is welded to the bottom of the vessel. The skirt is then supported by a concrete and steel pedestal, which carries the load through the drywell to the reactor building foundation slab.

Stabilizer brackets, located below the vessel flange, are connected to tension bars with flexible couplings. The bars are then connected to stabilizer brackets located on top of the biological shield wall to limit horizontal vibration and to resist seismic and jet reaction forces. A truss structure between the biological shield wall and the concrete wall outside the drywell provides lateral support. The bars are designed to permit radial and axial expansion. Seismic analyses are included in Appendix A.

4.2.3 <u>Design Evaluation</u>

The reactor vessel is designed and built in accordance with ASME Boiler and Pressure Vessel Code, Section III, Class A (Reference 51). General Electric has specified additional requirements. Records of material properties were developed and are retained for future evaluation of the reactor vessel.

As required by the code, the reactor coolant system has been hydrostatically tested. Additional system pressure tests are performed in accordance with the Monticello ASME Section XI Inservice Inspection and Testing Program (see Section 13.4.6). An evaluation of the effects of cold working on the pressure vessel steel indicated negligible effect due to the degree of cold working used in this vessel. The effects of Intergranular Stress Corrosion Cracking (IGSCC) on reactor vessel internal components as well as the recirculation system piping are minimized by the installation of a Hydrogen Water Chemistry (HWC) system. The purpose of the HWC system is to minimize IGSCC by reducing the oxidizing environment by introducing excess hydrogen to the reactor coolant system which combines with the free oxygen produced by radiolysis.

The reactor vessel is stamped with a Code N-symbol verifying that a hydrostatic test has been satisfactorily made and all other required inspection and testing has been satisfactorily completed. Such application of the Code N-symbol together with final certification confirms that all applicable Code requirements have been complied with.

Overpressurization of the coolant system is prevented by the design of the reactor control systems and the reactor safety systems. These include:

a. High pressure reactor scram

b. High neutron flux scram

c. Turbine generator load rejection scram

d. Operation of the reactor core isolation cooling system

e. Operation of the turbine bypass system

f. Operation of the safety/relief valves

g. Operation of the high pressure coolant injection system.

Complete details of the reactor vessel design and fabrication are contained in the "Reactor Vessel Design Summary Report" (Reference 45) which was submitted as Amendment 13 to the FSAR. This report is supplemented by the "Reactor Vessel (Rapid Cycling) Stress Report" (GE 22A7227) (Reference 46) and the "Reactor Vessel (System Cycling) Stress Report" (GE 22A7454) (Reference 47) which were prepared in support of the feedwater sparger nozzle modifications (see Sections 3.6.2.9 and 4.2.3.3). The "Reactor Vessel Design Summary Report" is additionally supplemented by the "Stress Report ASME Section III Class 1 Analysis of Core Spray Nozzle Assembly for Monticello Generating Plant" (Bechtel 301-P-5) (Reference 48) (Section 4.2.3.7)

The reactor pressure vessel stress analysis was reevaluated in association with the program which resulted in NRC issuance of License Amendment 176 (Reference 137). This evaluation concluded that any changes associated with reactor power of 2004 MWt remain acceptable (Reference 137).

4.2.3.1 Design Temperature

The design temperature for various system components varies according to the specific operating condition of the system. The design temperature for the reactor vessel is based on the saturation temperature corresponding to the design pressure. Therefore, no specific system temperatures are designated as safety or operating limits.

4.2.3.2 Fracture Toughness of Reactor Pressure Vessel

4.2.3.2.1 Compliance with 10CFR50 Appendix G, (Reference 49) May 1983

A major condition necessary for full compliance to Appendix G is satisfaction of the requirements of the Summer 1972 Addenda to Section III of the ASME Code (Reference 50). This is not possible with components which were purchased to earlier Code requirements. The Monticello reactor pressure vessel (RPV) was manufactured to the 1965 Edition of the ASME Code, Section III, to and including the Summer 1966 Addenda (Reference 51).

Ferritic materials complying with 10CFR50 Appendix G must have both drop weight tests and Charpy V-Notch (CVN) tests with the CVN specimens oriented transverse to the principal material working direction to establish the reference temperature RT_{NDT}. The CVN tests must be evaluated against both an absorbed energy and a lateral expansion criteria. The maximum acceptable RT_{NDT} must be determined in accordance with the analytical procedures of the ASME Code Section III, Appendix G (Reference 50). Appendix G of 10CFR50 requires a minimum of 75 ft-lbs upper shelf CVN energy for unirradiated beltline materials, and at least 50 ft-lbs upper- shelf CVN energy at the end-of-life. It also requires at least 45 ft-lbs CVN energy and 25 mils lateral expansion for bolting material at the lower of the preload or lowest service temperature.

By comparison, materials for the Monticello RPV were qualified by drop weight tests and/or longitudinally oriented CVN tests (both not required), generally at only one temperature, confirming that the material nil-ductility transition temperature (NDTT) is at least 60°F below the lowest service temperature. There was no upper-shelf CVN energy requirement on the Monticello beltline materials. The bolting material was qualified to a 30 ft-lb CVN energy requirement at 60°F below the minimum preload temperature.

From the above comparison it can be seen that the fracture toughness testing performed on the Monticello RPV material cannot be shown to comply directly with 10CFR50 Appendix G. However, to determine operating limits in accordance with 10CFR50 Appendix G, estimates of the beltline materials RT_{NDT} values and the highest RT_{NDT} values of all other materials were made, as explained in Paragraph 4.2.3.2.1.1. The method for developing these operating limits is described in Paragraph 4.2.3.2.1.2.

On the basis of the last paragraph on page 19013 of July 17, 1973, Federal Register (Reference 52), the following is considered an appropriate method of compliance. Refer to USAR Appendix K for additional discussion of Upper Shelf Energy.

4.2.3.2.1.1 Method of Compliance

The intent of the proposed special method of compliance with Appendix G of the ASME Code is to provide operating limitations on pressure and temperature based on fracture toughness. These operating limits assure that a margin of safety against a non-ductile failure of this vessel is essentially the same as a vessel built to the Summer 1972 Addenda (Reference 50).

The specific temperature limits for operation are based on 10CFR50, Appendix G, May 1983 (Reference 49).

4.2.3.2.1.2 Methods of Obtaining Operating Limits Based on Fracture Toughness

Operating limits which define minimum metal temperatures versus reactor pressure during normal heatup and cooldown, and during inservice hydrostatic testing, were originally established using the methods of Appendix G of Section III of the ASME Boiler and Pressure Vessel Code, up to and including the Summer 1976 Addenda (Reference 53). Monticello later requested exemption from 10CFR50 Appendix G and received approval to use ASME Code Case N-640, ”Alternative Reference Fracture Toughness for Development of P-T

Limit Curves" as an alternative (Reference 114). Code Case N-640 changes the fracture toughness curve used for development of P-T limit curves from K_{Ia} to $K_{IC.}$ The other margins involved with the process remain unchanged (Reference 115).

The vessel and discontinuities such as the RPV flanges, nozzles and bottom head penetrations were evaluated and the operating limit curves are based on the limiting location. The original bolt-up limits for the flange and adjacent shell temperature were based on a minimum metal temperature of +60F. In 2013, Monticello was approved to use SIR-05-044-A "Pressure Temperature Limits Report Methodology for Boiling Water Reactors" (Reference 130) a NRC approved methodology for development of pressure temperature limits for the reactor vessel and transition of the P-T curves from the tech specs to a licensee controlled pressure temperature limits report (PTLR) (Reference 132). The new P-T limits were calculated for Monticello through end of current operating license in 2030. (Reference 129). The current bolt-up limits for the flange and adjacent shell temperature are based on the requirements in 10 CFR 50 Appendix G and SIR-05-044A. The bolt-up limits are the limiting material RTNDT of the regions affected by bolt-up stress or 60F, whichever is greater. The maximum through-wall temperature gradient from continuous heating and cooling at 100°F per hour was considered. The safety factors were as specified in the ASME Code Appendix G.

For the purpose of setting these operating limits the reference temperature, RT_{NDT}, was determined from the toughness test data taken in accordance with requirements of the Code and the General Electric RPV purchase specification to which the Monticello RPV was designed and manufactured. These toughness test data, Charpy V-Notch (CVN) and/or drop-weight nil ductility transition temperature (NDT) were analyzed to establish compliance with the intent of 10CFR50 Appendix G. Because all toughness testing needed for strict compliance with Appendix G was not required at the time of RPV procurement, some toughness results are not available. For example, longitudinal CVNs, instead of transverse, were tested, usually at a single test temperature of +10°F or +40°F, and only against an absorbed energy criteria. Also, at the time, either CVN or drop-weight testing was permitted; therefore, in some cases both tests were not performed as is currently required. To substitute for this absence of certain data, toughness property correlations were derived for the vessel materials in order to operate upon the available data to give a conservative estimate of RT_{NDT}, compliant with the intent of 10CFR50 Appendix G (Reference 49) criteria.

These toughness correlations vary, depending upon the specific material analyzed, and were derived from the results of WRC Bulletin 217, "Properties of Heavy Section Nuclear Reactor Steels" (Reference 69), and from toughness data for other BWR reactors. In the case of vessel plate material (SA-533 Grade B, Class 1), the predicted limiting toughness property is either NDT or transverse CVN 50 ft-lb temperature minus 60°F, whichever is greater.

As a matter of practice where NDT results are missing, NDT is estimated as the longitudinal CVN 35 ft-lb transition temperature. However, for the Monticello vessel plates, "no break" drop-weight information was available at GE specified temperatures, so the nil ductility transformation temperature was conservatively taken as 10 degrees below the "no break" test temperature. The transverse CVN 50 ft-lb transition temperature was estimated from longitudinal CVN data in

the following manner. The lowest longitudinal CVN ft-lb value was adjusted to derive a longitudinal CVN 50 ft-lb transition temperature by adding 2°F per ft-lb to the test temperature. If the actual data equaled or exceeded 50 ft-lb, the test temperature was used. Once the longitudinal 50 ft-lb temperature was derived, an additional 30°F was added to account for the orientation change form longitudinal 50 ft-lb to transverse 50 ft-lb. For forgings (SA-508 Class 2), the predicted limiting property is the same as for vessel plates, and the RT_{NDT} was estimated in the same way.

For the vessel weld metal the predicted limiting property is the CVN 50 ft-lb transition temperature minus 60°F, as GE experience indicates the drop-weight NDT values are typically -50°F or lower for these materials. The CVN 50 ft-lb temperature would be derived in the same way as for the vessel plate material, except the 30°F addition for orientation effects was omitted since there is no principal working direction in weld metal. If NDT values were available, they would also be considered and the RT_{NDT} would be taken as the higher of NDT or the 50 ft-lb transition temperature minus 60°F. However, no data, either Charpy or drop-weight, were available on the fracture toughness of the specific weld materials used in the Monticello RPV. The weld rod supplier, Alloy Rods Corporation, provided RT_{NDT} data for E8018NM weld metal, which was used to determine statistically conservative estimate of the beltline weld RT_{NDT}. The data showed a mean RT_{NDT} of -65.5°F, with a standard deviation of 12.7°F.

For vessel weld heat affected zone (HAZ) material the RT_{NDT} was assumed the same as for the base material as ASME Code weld procedure qualification test requirements and post weld heat treatment indicates this assumption is valid.

Original closure bolting material (SA-540 Grade B24) toughness test requirements were for 30 ft-lb at 60°F below the bolt-up temperature. The purchase agreement for Monticello closure stud material was for 30 ft-lb at +10°F, and no deviations were reported.

4.2.3.2.1.3 Operating Limit Considerations Based on Surveillance Specimens

The Monticello reactor pressure vessel (RPV) originally contained several sets of reactor vessel specimens containing materials representative of the beltline region of the Monticello RPV (See Section 4.2.4). Two specimen sets were removed in 1981. These specimen sets had a low lead factor of 0.3 (ratio of specimen neutron fluence to highest neutron fluence experienced by the RPV wall). One of the two specimen sets was tested for RT_{NDT} at that time. The second set of specimens was later installed in the Unit 1 Prairie Island RPV for continued irradiation at an accelerated fluence (lead factor >10). These specimens were removed and tested in 1996.

The results from the 1996 surveillance material tests were evaluated utilizing guidance of Regulatory Guide 1.99, Rev 2, Position 2.1 (Reference 54). Based on this testing and evaluation, a more accurate chemistry factor, which is a function of copper and nickel content, was determined and new RPV pressure-temperature limit curves were developed. In addition, the curves for RPV core beltline (zero full power years) were revised to include the results of Oak Ridge National Laboratories (ORNL) testing of archived non-irradiated plate material (Reference 104).

The results of these surveillance specimen evaluations formed the basis for the revised Technical Specification operating limit changes reflected in License Amendment 106 (References 105, 106 and 107).

The BWRVIP Integrated Surveillance Program (ISP) which was initiated in 1999 was designed to replace the original plant specific surveillance capsule programs with representative capsules in host BWR plants. NRC has approved the ISP. MNGP committed to use the ISP in place of its existing surveillance programs, in the amendment issued by the NRC regarding the implementation of the Boiling Water Reactor Vessel and Internals Project Reactor Pressure Vessel Integrated Surveillance Program (Reference 111). The ISP meets the requirements for an integrated surveillance program in 10CFR50, Appendix H. Representative materials data that form the ISP capsules will be used for the Monticello vessel embrittlement predictions for lower intermediate shell plates 1-14 and 1-15 (plate heats C2220-1 and C2220-2) in accordance with ISP protocols. The Monticello surveillance program is currently covered by the ISP. Under the ISP program, a Monticello surveillance capsule was pulled in 2007 and tested in 2008 that contained representative vessel plate materials. The 2008 capsule test results were used to update the pressure-temperature limits for the reactor vessel for the period of extended operation through the end of the current license. The updated pressure-temperature limits use NRC approved methodology SIR-05-044-A (Reference 130). This methodology allows Monticello to transition the updated pressure-temperature limits from the technical specifications to a licensee controlled PTLR (Reference 132). The PTLR was approved for use under licence amendment 172 issued February 27, 2013 (Reference 131).

4.2.3.2.1.4 Calculated Values of Initial RT_{NDT}

The methods of Subsection 4.2.3.2.1.2 were used to calculate initial RT_{NDT} values for the core beltline plates and welds, closure flange region, nozzles and other discontinuities, and closure bolting material. The calculation methods conservatively estimate RT_{NDT}, in order to meet the intent of 10CFR50 Appendix G criteria. The core beltline plate and weld RT_{NDT} values are presented in Table 4.2-3. Regulatory Guide 1.99, Revision 2 (Reference 54) requires that the standard deviation δ be estimated from the data set evaluation performed. For the beltline welds, δ is estimated from the data set evaluation performed. For the beltline plates, actual measured values from each plate in question were obtained, and a conservative method of determining RT_{NDT} from that data was used. Therefore, δ for the beltline plates is estimated to be 0°F. Adjusted CVN data for the closure flanges and adjacent plates gave an RT_{NDT} of +10°F. Based on the NDT requirement in the purchase agreement, the RT_{NDT} of the welds at the closure flanges was +10°F also. Calculations for the nozzles and discontinuities gave an RT_{NDT} of +40°F based on purchase agreement NDT requirements. The closure bolting material RT_{NDT} used was +10°F, based on the purchase agreement NDT requirement.

4.2.3.2.1.5 Effect of Nozzles and Discontinuities on Operating Limits

The minimum temperature for bolt-up and pressurization was established using the requirements in 10 CFR 50 Appendix G (Reference 49) and SIR-05-044A (Reference 130). The bolt-up temperature limits are the limiting material RTNDT of the regions affected by bolt-up stresses or +60F, whichever is greater.

Above 20% test pressure, 10CFR50 Appendix G, Paragraph IV.A.2 requires the closure flange region to be 90°F above RT_{NDT} for hydrostatic pressure tests and leak tests, 120°F above RT_{NDT} for normal operation, and 160°F above RT_{NDT} when the core is critical. The 90°F requirement is met by adding 30°F at 20% test pressure to the 60°F bolt-up margin. The temperatures required at 20% test pressure for normal operation and core critical operation exceed the respective 120°F and 160°F margins required by 10CFR50 Appendix G.

Previously, the effect of the nozzle and bottom head discontinuities was considered by adjusting the results of BWR/6 reactor discontinuity analyses to the Monticello reactor. The adjustment was made by increasing the minimum temperatures required by the difference between the Monticello and BWR/6 RT_{NDT} values. The nozzle and bottom head adjustments were based on an RT_{NDT} of +40°F. During the most recent update to the operational limits approved under license amendment 172, postulated vessel discontinuities were evaluated using finite element analysis (FEA) (References 126 and 127) and the results were used in the development of the pressure-temperature curves in the PTLR.

4.2.3.2.1.6 Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity

NRC Generic Letter 92-01, Revision 1, “Reactor Vessel Structural Integrity” (Reference 55), was issued to obtain information needed to assess compliance with requirements and commitments regarding reactor vessel integrity in view of certain concerns raised in the staff’s review of reactor vessel integrity for the Yankee Nuclear Power Station. Monticello responded to the Generic Letter 92-01, Revision 1, and subsequent staff requests for information, by letters dated July 6, 1992, August 30, 1993, and May 26, 1994 (References 56, 57 and 58). Generic Letter 92-01, Revision 1, Supplement 1, (dated May 19, 1995) (Reference 59), was issued to require that all addressees identify, collect and report any new data pertinent to analysis of structural integrity of their reactor vessels and to assess the impact of the data on their reactor pressure vessel integrity analysis. Monticello provided a response to the Required Information of Generic Letter 92-01, Revision 1, Supplement 1, by letters dated August 17, 1995 and November 13, 1995 (References 60 and 61).

In response to Generic Letter 92-01, Revision 1, Supplement 1, an independent review was performed by NSP personnel of the data previously submitted and all Monticello reactor pressure vessel fabrication information made available to NSP pertaining to compliance with 10CFR50, Appendix G. This independent review has confirmed that all pertinent data has been considered in providing our response to Generic Letter 92-01, Revision 1 (References 105, 106 and 107).

Monticello has demonstrated regulatory compliance concerning reactor vessel structural integrity in our responses (Reference 97 for Generic Letter 88-11 and References 60 and 61 for Generic Letter 92-01 Rev. 1) to Generic Letter 88-11, “NRC Position On Radiation Embrittlement of Reactor Materials and Its Impact on Plant Operations” (Reference 62), and Generic Letter 92-01, Revision 1 (Reference 55). The evaluations performed in support of these previous responses were conservative, with considerable effort taken to ensure that all pertinent information was considered.

4.2.3.2.2 Pressure Temperature Limits

4.2.3.2.2.1 Limit Curves

The basis for setting operational limits on pressure and temperature for normal, upset and test conditions for the RPV is described in Section 4.2.3.2.1.2.

The fracture toughness analysis was done for the normal heatup or cooldown rate of 100°F/hour and it also included the effects of cold water injections into the nozzles and other operation transients. The temperature gradients and thermal stress effects are included. The results of the analyses are a set of operating limits for pressure testing, non-nuclear heatup or cooldown and core critical operation. These operating limits are documented in the licensee-controlled PTLR (Reference 132) and the Monticello Technical Specifications.

4.2.3.2.2.2 Temperature Limits for Bolt-up

A minimum temperature of +60F is required for the reactor head flange, reactor vessel flange, and the reactor vessel shell adjacent to the reactor vessel flange prior to applying tension to the reactor vessel head closure studs.

4.2.3.2.3 Operating Procedures

By comparison of the pressure vs. temperature limits in Paragraph 4.2.3.2.2 above, with intended normal operating procedures for the most severe upset transient, it is shown that the limits will not be exceeded during any foreseeable upset condition. Reactor operating procedures have been established such that actual transients will not be more severe than those for which the vessel design adequacy has been demonstrated. This includes guidance for cooling the reactor for disassembly (Reference 112). Of the design transients, the upset condition producing the most adverse temperature and pressure condition anywhere in the vessel head and/or shell areas yields a minimum fluid temperature of 250°F and a maximum pressure peak of 1180 psig. Scram automatically occurs with initiation of this event. For a temperature of 250°F, the maximum allowable pressure exceeds the intended margin against non-ductile failure. The maximum transient pressure of 1180 psig is therefore within the specified allowable limits.

4.2.3.2.4 Irradiation Effects

Estimated maximum changes in RT_{NDT} for 54 EFPY of the vessel beltline materials are shown in Table 4.2-3 along with other information and the references for that information. The methods in Regulatory Guide 1.99, Revision 2 (Reference 54) were used. Since predicted adjusted reference temperatures (ART) are less then 200°F, provisions to permit thermal annealing of the RPV in accordance with Paragraph IV.B of 10CFR50 Appendix G is not required.

Reactor vessel fracture toughness was reevaluated in association with the program which resulted in NRC issuance of License Amendment 172 (Reference 131). The 54 EFPY P-T limit curves satisfy the criteria of ASME Code, Section XI, Appendix G, as required by 10 CFR Part 50, Appendix G. The 54 EFPY P-T limit curves are bounding for all ferritic reactor coolant pressure boundary (RCPB) components and the curves were developed taking into consideration RV beltline and non-beltline components, including stress concentrators (e.g., RV nozzles) outside of the RV beltline shell region, as required by 10 CFR Part 50, Appendix G (Reference 137).

The Reactor vessel pressure-temperature operating limits evaluation based on RPV material specimen test results resulted in NRC issuance of License Amendment 172 (Reference 131 and 137). This amendment approved revised Monticello RPV pressure-temperature limit curves which are contained in the Pressure Temperature Limits Report. Additional surveillance specimens will be removed from the Monticello RPV in accordance with the Boiling Water Reactor Vessel Internals Project (BWRVIP) Integrated Surveillance Program (ISP) (Reference 111). The removal schedule developed by the BWRVIP ISP is included in BWRVIP-86-A (Reference 110). The technical basis for the ISP is discussed in the BWRVIP-78 (Reference 109). (References 109 and 110 are considered incorporated by reference in the MNGP USAR).

Fluence calculations will be performed in accordance with Regulatory Guide 1.190 “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence” (Reference 111). The calculated end of life (60 year) bounding estimate of reactor vessel inside wall fluence is 6.43×10^{18} n/cm^2 (Reference 125).

4.2.3.3 Thermal Shock Effects on Reactor Vessel Components

Several high stress points on the reactor vessel have been analyzed approximately and conservatively to determine the effect of LPCI cold water injection. The points examined were the mid-core inside of the reactor vessel wall, the control rod drive penetration and the recirculation inlet nozzle. The results at the mid-core inside of the vessel indicate a peak stress of 67,500 psi.

The results of the control rod penetration analysis are:

Amplitude of alternating stress	560,000 psi
Allowable ASME III cycles	14
Maximum strain range	3.7%

The recirculation inlet nozzle analysis resulted in a nozzle design with a thermal sleeve to reduce thermal shock effects. In 1984, reactor recirculation inlet safe ends were replaced in conjunction with recirculation piping replacement. The results reported in the GE Stress Report, 23A1627 Rev. 1, dated September 14, 1984 (Reference 65), show that replacement safe ends and thermal sleeves, including the attachment weld to the recirculation inlet nozzle, meet all the requirements of Article NB-3000.

The most recent calculations concerning thermal shock effects on reactor components continue to indicate that thermal stresses are well within established guideline values and, therefore, do not jeopardize the integrity or the function of either the reactor pressure vessel or the reactor internal components. A comprehensive discussion of the brittle fracture analysis performed for the BWR pressure vessel is contained in a General Electric topical report (Reference 1). This report shows that the reactor pressure vessel functions in a safe and reliable manner during and after all the postulated thermal shock conditions.

The feedwater spargers and nozzles were modified to minimize potential fatigue problems due to thermal cycling. The modifications featured improved thermal sleeve and sparger elbow designs and the selection of materials with improved fatigue resistance. (See Section 3.6.2.9 for further details.)

The reactor internal component that is most sensitive to thermal shock following a DBA is the shroud or baffle plate assembly. Analysis has shown that the leak-tightness of the shroud support plate and its joints to the vessel, shroud, and jet pump is not jeopardized by the DBA and that LPCI core flooding capability is maintained following the DBA. The results of shroud support plate analysis were reported in Section 3.6.3.3.

NRC Bulletin No. 88-08, Thermal Stresses in Piping Connected to Reactor Coolant Systems Action Item No. 1 (Reference 66) requested that licensees review systems connected to the RCS to determine whether isolable sections of piping connected to the RCS can be subjected to stresses from temperature stratification or temperature oscillations what could be induced by leaking valves and that were not evaluated in the design analysis of the piping. Monticello's response to NRC Bulletin No. 88-08 (Reference 28) concluded that for all identified reactor coolant system unisolable piping, the pipe thermal stress analysis showed that temperature differences are not large enough to cause excessive thermal stresses in the RCS unisolable piping per applicable codes.

4.2.3.4 Jet Reaction Forces

Jet reaction forces on the reactor vessel were analyzed, with the reactor vessel and support structures designed to withstand forces greater than those that would be created by full flow through any vessel nozzle at reactor design pressure plus the design earthquake force. The two largest jet reaction forces would come from shearing a recirculation nozzle - 658 kips, and shearing an outlet steam line – 330 kips. Thus even if a line shears, the vessel would not be moved by jet reaction forces sufficient to cause rupture of other connected pipes.

Piping connecting to the reactor vessel, except the recirculation piping (see Section 4.3.2), has sufficient inherent protection so as to preclude the need for restraints. Calculations have shown that pipe breaks which result in the maximum energy available for pipe whip are circumferential in nature and occur at the end of a

long run of pipe in the vicinity of an elbow. The worst break is one in which the jet force from the elbow acts in a lateral direction such that the long run of pipe acts as a cantilever. Long pipe runs such as this encounter obstructions enroute to the containment wall which minimize the probability of containment penetration. In addition, the smooth shape of a curved elbow and the relatively large area of impact imply that large deformation of the containment shell would be required before complete penetration would occur. Calculations made to determine if the piping could achieve the amount of energy necessary to penetrate the containment, using the restraint of other structural members (i.e., pipe, girders, beams, concrete, etc.) indicated that containment penetration was not possible.

Supplements to NRC Bulletin No. 88-08 were also reviewed for applicability to Monticello and it was again determined that no action is required (Reference 38). The staff informed NSP that based on our responses this issue was considered closed for Monticello (Reference 39).

4.2.3.5 Flow Induced Vibrations

Flow induced vibrations might occur in the primary recirculation system under abnormal operating conditions, and the system support structures have been designed to withstand these flow induced vibration forces.

4.2.3.6 Material Properties

Eight reactor pressure vessel nozzle safe ends were identified as being furnace sensitized prior to plant startup. Six of these safe ends were replaced prior to startup and two were weld clad. During the 1984 outage, the two treated by weld clad (the recirculation outlet nozzle safe ends) were replaced. There are no furnace sensitized stainless steel components in the associated reactor coolant systems.

Table 4.2-2 is a summary of the drop weight tests on the pressure vessel plates.

4.2.3.7 Reactor Pressure Vessel Design Report - Updates

4.2.3.7.1 Summary of Stress Analysis for Core Spray Safe Ends, Thermal Sleeves, and Nozzles

The Core Spray safe ends, thermal sleeves, and nozzles were originally analyzed by Chicago Bridge & Iron Company in their Stress Analysis Report - Section S7 (Reference 67). The core spray safe ends and thermal sleeves were reanalyzed by General Electric Company in the Reactor Vessel Core Spray Safe End Extension Stress Report No. 22A7490 (Reference 68). After replacement of the Core Spray Safe Ends in 1986, a stress report was prepared by Bechtel Power Corporation (Reference 48).

The Bechtel calculated maximum primary stress intensity for the safe end at 575°F design temperature is 11,175 psi compared to the allowable of 18,200 psi. For the nozzle, the calculated maximum general membrane primary stress intensity is 11,866 psi, and the allowable at design temperature is 26,700 psi.

The most critical point from the fatigue standpoint is Section 3, shown on Figure 4.2-2. The conservatively calculated value of the fatigue usage factor at this point is 0.0837.

4.2.3.7.2 Control Rod Drive Return Line Nozzle Capping

Originally exhaust water from the control rod drive return line entered the reactor vessel through the CRDRL nozzle (also referred to as nozzle N9 in Figure 4.2-1, or the CRDHSR nozzle in the Reactor Pressure Vessel Design Report). Thermal fatigue cracks were found immediately beneath this nozzle on several BWRs. As a result of this potential problem, the control rod drive return line nozzle was capped. Exhaust water from the control rod drives, which previously entered the reactor vessel through this nozzle, now flows through the directional control solenoid valves of control rods not being moved at the time (see Section 3.5.3.3.2.7) or may be directed to the reactor water cleanup system return line for emergency makeup to the reactor vessel.

4.2.4 Inspection and Testing

The reactor coolant system was given a system hydrostatic test in accordance with code requirements prior to initial reactor startup. Before pressurization the system was heated to NDT temperature +60°F.

A leakage test at operating pressure is made on the primary system following each removal and replacement of the reactor vessel head. The system is checked for leaks and abnormal conditions are corrected before reactor startup. The minimum vessel temperature during hydrostatic testing satisfies the pressure-temperature limitations for fracture toughness of 10CFR50, Appendix G. These limitations are presented as a series of curves in the PTLR.

Vessel material surveillance samples are located within the reactor vessel to enable periodic monitoring of material properties with exposure. The samples include specimens of the base metal, weld zone metal, heat-affected zone metal and standard specimens. These specimens receive neutron exposures more rapidly than the vessel wall material and, therefore, lead it in integrated neutron flux. The results of these tests and calculated nvt exposures are used in evaluating the adequacy of design values used in calculating the vessel pressure-temperature limitations.

Initially, 106 samples were inserted in the vessel and periodically removed for Charpy V-notch and tensile strength tests. Control specimens include 36 Charpy V and 9 tensile specimens.

The irradiation program for the specimens in the reactor consists of three baskets of specimens that are attached to brackets located on the I.D. of the reactor vessel. The baskets are at 30°, 120° and 300°, and are located at core mid-plane. Each basket contains 0.250 inch diameter tensile and standard Charpy V specimens. The specimens are sealed in helium filled capsules. Basket 1 contains 36 Charpy V and 8 tensile specimens. Each Charpy V specimen capsule (there are seven) contains copper, iron and nickel wires as flux dosimeters. Attached to basket 1 was a separate neutron dosimeter. This dosimeter was removed during the first refueling outage. Basket 2 contains 24 Charpy V and 8 tensile specimens. Basket 3 contains 24 Charpy V and 6 tensile specimens. The specimens in each basket, as well as controls and spares, are equally divided between base metal, weld metal and heat-affected zone specimens. Each individual specimen is given unique identification marks. The materials and the number and type of specimens to be installed in the reactor meet the recommendations of ASTM E185-66 (Reference 70).

The material for base metal specimens has been taken from a plate used in the vessel beltline region or from a plate of the same heat of material. The same plate used for base metal specimens is used for production of heat-affected zone specimens, and the weld specimens are produced by the identical weld practice and procedures used in the vessel fabrication. For vessels constructed from plate, the vessel longitudinal welds are represented; while for vessels fabricated from forged rings, the girth welds are represented. When widely varying weld practices such as submerged metal arc and electro-slag welding are used jointly in a vessel; both are represented in the surveillance program material. Thus, the surveillance specimens do represent the materials and processing of the vessel beltline region.

Additional information concerning the reactor vessel material surveillance program is provided in NEDO-24197, "Information on Reactor Vessel Material Surveillance Program," June 1979 (Reference 71) and EPRI Technical Report 1020231, BWRVIP-135, Revision 2: BWR Vessel and Internals Project Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluations (Reference 116).

Post hydro ultrasonic (UT) examination and mapping of the reactor pressure vessel was performed. Those portions of the reactor pressure vessel which are accessible for future examinations were examined prior to the facility being placed into commercial operation. The results of these examinations provide a baseline reference and record for future examinations.

A record was made of all the ultrasonic examinations performed on the baseline inspection program. A copy of these records is retained at the Monticello Plant Site. Ultrasonic examination and mapping similar to that performed at Dresden-2 has been performed on the CRD housing field welds. The details of the preoperational baseline ultrasonic inspection performed for Monticello is described in NEDC-10165, "Ultrasonic Baseline Inspection at the Monticello Power Plant," August 1970 (Reference 72). The details of the periodic inservice examination program are contained in the "Monticello Inservice Inspection and Testing Program" which has been submitted to the NRC under separate cover. The current revision of this document is specified in Section 13.4.6.

Following an Automatic Depressurization System (ADS) blowdown without HPCI, there is a concern for the integrity of welded connections in the core vicinity. These welded connections have a tendency to become embrittled due to their exposure to irradiation and thermal environment. Following an ADS actuation event without HPCI, the integrity of affected welded connections will be demonstrated by analysis or inspection before resuming operation (Reference 14).

General Electric has evaluated the testing/inspections that would be necessary following an ADS actuation event (Reference 15). Rapid cooldown of the reactor does not cause excessive stress in the reactor vessel, mainly because the range of rapid temperature change is only from 550°F to 300°F (and fatigue usage is less than 0.03 for this event). Also, cold water is not injected directly onto the hot reactor vessel surfaces. The reactor vessel nozzles have already been analyzed for this concern and this has been factored into the reactor operating limits. However, the bottom and top surfaces of the core top guide will need inspection if all of the following criteria are exceeded:

1. Core spray had injected water over the top of the fuel when the reactor water level was below the top of the active fuel.

2. The reactor was pressurized (i.e., not in cold shutdown) at the time of core spray injection with the reactor water level below the top of the fuel.

3. The fluence at the stainless steel core top guide was in excess of a threshold value of $5x10^{20}$ n/cm^2 (above 1 MEV). This fluence has currently been exceeded at the bottom surfaces of the Monticello core top guide (estimated value is $1.1x10^{21}$ n/cm^2).

A feedwater nozzle monitoring and inspection program was prepared and implemented to address concerns involving NRC Generic Technical activity A-10 (BWR Nozzle Cracking) (Reference 73) and NUREG-0619 (BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking) (Reference 74). The program, which involves monitoring of feedwater nozzle thermal sleeve leak detection system data and periodic UT and visual inspections of the nozzles (supplemented by PT inspection when warranted) was reviewed by the NRC and found acceptable in Reference 40.

Table 4.2-1 Reactor Vessel Analysis [1, 2]

Types of Cycles	Number of Cycles
Bolt up/Unbolt	120
Startup/Shutdown at 100°F/hr	289[3]
Scrams	270
Design Hydrostatic Test at 1250 psig	130
Reactor Overpressure at 1375 psig	1
Hydrostatic Test to 1560 psig[4]	3
Rapid Blowdown (2 or 3 SRVs) Terminated at or above 400°F/267 psig	1
Liquid Poison Flow at 80°F	10
Feedwater Heater Bypass	70
Loss of Feedwater Heater	10
Loss of Feedwater Pumps	30
Improper Start of Shutdown Recirculation Loop	10

1. Not all cycles apply to all locations in the RPV.
2. For further information, see Appendix H.
3. Includes eight cycles of the single SRV blowdown transient.
4. Per ASME Section III requirement of 1.25 x design.

Table 4.2-2 Monticello RPV Summary of Drop-Weight Tests Vessel Plates
(Page 1 of 2)

Part Name	Pc Mark	No Break Results, °F
Bottom Hd Dollar	1-28	+10* +50
	1-27	+10* +50
	1-26	+10* +50
Bottom Hd Side	1-18/21	+10* +50
	1-22/25	+10* +50
Lower Shell	1-17	+10 +50
	1-16	+10 +50
Lower Int Shell	1-14	+10 +10
	1-15	+10 + 0
Upper Int Shell	1-13	+40 +50
	1-12	+10 +50

* Specimens were taken from surface. Remaining Specimens 1/4 T from Tension surface where T is plate thickness

Table 4.2-2 Monticello RPV Summary of Drop-Weight Tests Vessel Plates
(Page 2 of 2)

Part Name	Pc Mark	No Break Results, °F
Upper Shell	1-11	+10 +20
	1-10	+10 +20
Top Hd Side	1-5/7	+10 +20
	1-2/4	-10 +20
Top Hd Dollar	1-1	+10 +20

Table 4.2-3 Determination of Limiting Beltline Material[1]

Material Identification	%CU	%Ni	Chemistry Factor	Initial RT_{NDT}	Sigma I σ_I	Sigma Delta σ_Δ	ΔRT_{NDT}	Margin[4]	ART
Plates:									
1-12 (C2089-1)	0.35	0.50	199.50°F	0°F	0°F	17°F	43.8°F	34°F[5]	77.8°F
1-13 (C2613-1)	0.35	0.49	198.25°F	27°F	0°F	17°F	43.5°F	34°F[5]	104.5°F
1-14 (C-2220-1)	0.16	0.64	180.0°F	27°F	0°F	8.5°F	142.6°F	17°F	186.6°F
1-15 (C-2220-2)	0.16	0.64	180.0°F	27°F	0°F	8.5°F	142.6°F	17°F	186.6°F
1-16 (A-0946-1)	0.14	0.56	98.20°F	27°F	0°F	17°F	68.2°F	34°F	129.2°F
1-17 (C-2193-1)	0.17	0.50	118.50°F	0°F	0°F	17°F	82.3°F	34°F	102.8°F
Welds:									
Limiting Case	0.10[2]	0.99[2]	134.9°F	-65.6°F	12.7°F	28°F	106.9°F	61.5°F	102.8°F
Nozzles:									
N2	0.18[3]	0.86[3]	141.9°F	40°F	0°F	17°F	51.2°F	34°F	

Note 1 - Table 4.2-3 values from CA 11-003,Table 3, Evaluation of Adjusted Reference Temperatures and Reference Temperature Shifts. (Reference 128)

Note 2 - Copper percentage for weld metal from Reference 98. Nickel percentage for weld metal from Reference 71, heat 659T174.

Note 3 - In the absence of Cu data for this nozzle, 0.18% is based upon heats of materials used for beltline nozzles at other plants. The mean from nine nozzles (0.119) plus one standard deviation (0.0617) was used to determine the value of 0.18%. CMTR data for the ten (10) MNGP N2 nozzles was averaged to determine the Ni content. CMTR data for the ten (10) MNGP N2 nozzles was used to determine the initial RT_{NDT}.

Note 4 - Margin calculated using Equation 4 from Reference 54.

Note 5 - Margin for Plates 1-12 and 1-13 is 34°F because there are not two sets fo credible surveillance data available for these plates.

4.3 Recirculation System

Cooling water is forced through the reactor core by the recirculation system. The system consists of two external loops together with associated pumps, valves and piping.

4.3.1 Design Basis

The recirculation system provides forced convection cooling of the reactor core. The design basis is:

Design Pressure, Suction	1148 psig at 562°F
Design Pressure, Discharge	1248 psig at 562°F
Recirculation System Design Code	ANSI B 31.1, 1977 Edition through Summer 1978 Addenda
Recirculation Pump Casing Design Code	ASME Section III, Class C

The suction pressure was selected relative to the reactor vessel design pressure. An additional margin of 20 percent above the allowable stress is permitted by Section I of the ASME Boiler and Pressure Vessel Code. The discharge pressure was established at a nominal 130 psi above the suction pressure to accommodate the pressure output of the recirculation pumps.

The recirculation pumps are exempt from the requirements of any sections of the ASME Boiler and Pressure Vessel Code or of the USAS Code for Pressure Piping because of their machinery classification. The Standards of the Hydraulic Institute are the only standards which are applicable; however, they are more pertinent to the testing and performance of the pump and consequently provide little or no guidance in the areas of casing quality and structural integrity. Therefore, to assure that the pump casing can contain pressure which is at least equivalent to reactor vessel pressure, the pump casing was designed in accordance with ASME Boiler and Pressure Vessel Code, Section III, Class C.

The jet pump assemblies are capable of withstanding without failure or loss of required functional integrity, the forces, loads and stresses calculated (including seismic loads) to be encountered during normal transient and accident conditions.

As the jet pump assemblies form a part of an inner vessel, they are sufficiently leak tight, allowing provisions for thermal expansion, to permit reflooding the reactor core to approximately 2/3 of core height following a design basis loss-of-coolant accident.

The key components which govern jet pump performance and which experience high velocities are designed to be removable for inspection and/or replacement.

Each component is able to withstand the combined loadings from differential pressure and temperature, dead weight, fluid movement, seismic acceleration and vibration. Allowable stresses as defined by the ASME Boiler and Pressure Vessel Code, Section III, are not exceeded for normal operation. Thermal expansion, corrosion and crud buildup allowances are provided in the design.

4.3.1.1 Performance Criteria

The jet pumps as a part of the reactor system provide stable, controlled coolant flow to the reactor core for forced convection cooling. The core flow supplied by multiple jet pumps operating in parallel is uniform and predictable from low to high flow conditions encountered during normal steady state and transient reactor operation with no flow discontinuities.

Motive fluid for ten jet pumps is supplied by each recirculation pump. There are two recirculation pumps. Each recirculation pump is a single-stage variable speed, vertical, direct motor drive, centrifugal pump equipped with mechanical shaft seal assemblies and is capable of stable and satisfactory performance while operating continuously at any speed corresponding to a power supply frequency of 11.5 to 57.5 cps. The motor-generator is provided with mechanical and electrical stops to prevent operation above 56 cps. System capability and performance for both the 57.5 cps rating and the 56 cps rating is provided in Reference 139. The recirculation pump motor design speed is 1800 rpm with a power supply frequency of 60 cps. A variable frequency a-c motor-generator (MG) set supplies power to each recirculation pump motor. The pump is electrically connected directly to the generator and is started by engaging the variable speed coupling between the generator and motor of the M-G set. Minimum speed corresponds to a frequency of 11.5 cps.

The reactor recirculation system is not a safety system since its purpose is operational. The system is safety related since it forms a part of the nuclear system process barrier. The water in this system is at a high temperature and pressure and contains a large amount of energy. To assure that the pump casing withstands conditions at least equal to the reactor vessel, the pump casing is designed in accordance with ASME Boiler and Pressure Vessel Code, Section III, Class C. This class is used because the pump casing does not experience pressure or temperature transients as severe as those that the reactor vessel experiences, so it is not necessary to make the cyclic analysis required by Class A of Section III.

The hydraulic characteristics of the jet pumps in combination with the remainder of the plant characteristics, nuclear instrumentation, and reactor protection system ensure safe conduct of operation of the plant during normal operation and ensure that no fuel damage results during operational transients caused by reasonably expected single operator error or equipment malfunction. Flow behavior characteristics are similar to those existing on forced circulation BWR plants relying on centrifugal recirculation pumps.

There is no recirculation line break which can prevent reflooding of the core to the level of the jet pump suction inlet. The core flooding capability is discussed in detail in the core standby cooling systems document filed with the NRC as a GE topical report (Reference 2) during initial plant licensing. A more recent evaluation is included in USAR section 14.7.2.3.6.

4.3.2 General Description

The recirculation system consists of two recirculation pump loops and twenty jet pumps. Each external loop consists of a motor-driven recirculation pump, two motor operated gate valves for pump isolation, piping, and required recirculation flow measurement devices. The loops are supported from radial beams and the sacrificial shield steel to minimize thermal stress and provide pipe whip restraints in the event of pipe rupture. The two external recirculation loops each discharge high pressure flow into an external manifold from which connections are ultimately made to the jet pump nozzles. The jet pumps are located inside the pressure vessel. Saturated water which is rejected from the dryers and steam separators is mixed with incoming subcooled feedwater above the core. The resulting subcooled mixture passes down the annulus area between the vessel and the core shroud. Approximately 42 percent of the flow goes through the recirculation loops and becomes the driving flow of the jet pumps. The remaining 58 percent of the flow is driven by the recirculated flow as it enters the jet pump at the suction inlet and is accelerated by momentum transfer from the driving water injected through the jet pump nozzles. Partial pressure recovery occurs in the mixing section. The balance of recovery is obtained in the jet pump diffuser section. Water flows out of the diffuser at sufficient pressure to recirculate through the core. A topical report APED 5460 “Design and Performance of GE-BWR Jet Pumps” (Reference 3) has been filed with the NRC. This report gives a detailed analysis and description of the jet pumps.

4.3.2.1 Physical Description

The main recirculating pumps are single-stage, centrifugal units with mechanical shaft seals. Each pump is rated to deliver 32,500 gallons per minute. These vertical pumps are arranged within the drywell to facilitate inspection, maintenance and/or removal during plant shutdown conditions. The pumps are driven by variable speed induction motors, which receive electrical power from variable frequency motor generator sets.

A seal injection system is provided to insure a cooling supply to the pump seals at all times. Seal isolation valves were also provided in the upper seal leak off lines to prevent leak off from the upper seal following a pump trip, thus minimizing the rate of hot water leakage from the primary system to the seals.

The reactor coolant recirculation system pumps and motors are located below the vessel in order to provide elevation head for the net positive suction head (NPSH) requirements.

In the event that one pump fails or is shut off, the discharge valve in the affected loop would be manually closed, then subsequently repositioned as desired for further plant operation.

The recirculation pumps are supported by three constant support hangers to accommodate the vertical loads. The constant support hangers are designed for the dead load of the pump, motor and water. The suspension system was designed to limit the weight effect as allowed in USAS B31.1.0 (Reference 44).

Six shock suppressors were provided to resist the seismic forces. These shock suppressors were designed to accommodate the design earthquake and the maximum earthquake (double the design earthquake).

The recirculation pump is constrained by a pump restraint which anchors the recirculation pump to the concrete pedestal. The pump restraint was designed to accommodate a jet force load due to a pipe rupture between the pump and the recirculation isolation valve. The restraint prevents the pump and motor from becoming a missile and thus protects the primary containment integrity. The pump restraint allows normal pump movement for thermal expansion.

The recirculation lines have been provided with a system of pipe restraints to limit pipe motion so that any reaction forces associated with a pipe split or circumferential break does not jeopardize containment integrity. These restraints allow for unrestricted movement of the piping due to thermal expansion and seismic response. Positioning of the restraints assures that strength of the pipe is maintained: a) on both sides of a circumferential break; and b) over the entire length of a split pipe.

The jet pumps, which have no moving parts, are located between the core shroud and the reactor vessel wall, as shown in Figure 4.3-1. Each pair of jet pumps is supplied driving flow from a single riser pipe. These risers have individual vessel penetrations and receive flow from one of two recirculation inlet manifolds.

The jet pump consists of a diffuser, a throat section, and a nozzle section as illustrated in Figure 4.3-2. The jet pump diffuser is a gradual conical section terminating in a straight cylindrical section at the lower end which is welded to the shroud support. The throat section, a straight section of tubing with a short diffuser entrance section at the lower end, and the nozzle section are clamped together. The throat and nozzle sections are attached to the riser and diffuser with brackets which provide structural rigidity, yet permit differential expansion between the carbon steel vessel and the stainless steel jet pump. The overall height from the top of the inlet nozzle assembly to the diffuser discharge is 20 ft 10 in., each diffuser has an outside diameter of 14-3/4 in. Replacement of the throat and nozzle section of the jet pump is possible.

During construction of the facility General Electric's investigation of castings used in the jet pump assemblies revealed cracks in the transition piece, 180° elbow, and nozzle castings. Subsequent sectioning and metallographic examination of the transition piece, nozzle, and coupling castings showed surface carburization to a maximum depth of 0.10 in. with the carbon content in these areas as high as 0.31 percent within approximately 0.020 in. of the surface. These castings were produced to General Electric Specification B50YP43A1 (Reference 75). This document specifies the requirements of ASTM A351 Grade CF8 (Reference 76), and additional supplemental requirements. The original castings as supplied conformed to all requirements of the General Electric Specification and the ASTM A351 Specification (Reference 76). Since the chemical analysis is specified to be taken from material removed not less than 1/4 in. below the surface all castings met the specified 0.08 percent maximum carbon content. New transition pieces were fabricated by a different process, not subject to the carburization problem. The collar castings were repaired by machining off the casting surfaces.

The remaining castings which contain the carburized surface are all a part of the removable inlet-mixer subassembly. The inlet consists of the 180° elbow and nozzle castings. The flange casting is welded to the mixer subassembly and contains the mating surface for the inlet-mixer connecting joint. The coupling casting is the member which joins the inlet to the mixer. The adapter casting is located near the bottom of the mixer and provides support for the locking wedge.

A review has been made to determine the suitability of these castings for reactor service. This review indicates that the castings are entirely satisfactory for reactor service. The principal factors leading to this conclusion are:

a. the low stress levels;

b. the inherent crack propagation resistance of high ferrite duplex Austenitic stainless steel;

c. the fact that all original jet pump design requirements are still met; and

d. the fact that the historical record indicates that there are no known failures of Austenitic stainless steel castings in the nuclear industry due to surface carburization.

4.3.2.2 Principles of Jet Pump Operation

The principle of operation of the jet pump is the conversion of momentum to pressure. The fluid emerging from the driving nozzle, called the driving or motive fluid, has a high velocity and a high momentum but low static pressure. The low-energy suction fluid is drawn into the mixing section or throat by the pressure difference between the downcomer fluid and the driving fluid.

In the throat, the two fluid streams combine by momentum transfer. During the process there is some static pressure recovery. However, the main function of the mixing chamber is to provide complete combination of the high and low energy stream so that a single high velocity stream is presented to the diffuser. For optimum operation, the velocity profile at the exit of the throat should be as flat as possible; i.e., the boundary layer entering the diffuser should be as thin as possible. This flat velocity profile insures maximum performance of the diffuser. In the diffuser, the relatively high velocity of the combined streams is converted to high static pressure. The resulting exit flow has the pressure required to provide the necessary recirculation flow through the core.

4.3.2.3 Recirculation System Modifications

4.3.2.3.1 Recirculation Piping Modification During the 1984 Outage

The recirculation piping was redesigned to utilize Intergranular Stress Corrosion Cracking (IGSCC) resistant material, eliminate as many fittings as possible, and to reduce the number of welds.

Design improvements consist of:

- Use of bent pipe for 12-in. risers. This eliminated 20 welds.

- Deletion of end caps on the supply manifolds by using a bent pipe and reducer on the ends of the 22-in. headers. This eliminated four welds.

- Use of seamless 12-in. pipe. This eliminated the need for periodic inspection of longitudinal welds.

- Solution annealing of all spools after shop welding.

A material was specified (316NG stainless steel) that has a low carbon, high nitrogen chemistry to minimize the possibility of IGSCC. To further minimize this problem, procedures required solution annealing of shop welds after fabrication and Induction Heating Stress Improvement (IHSI) treatment of all field welds.

The equalizing piping and valves that connected the two 22″ manifolds in the original design has been eliminated. Operational experience showed this connection to be without significant utility and Technical Specifications precluded operation of the reactor with the equalizing line open for extended periods of time. Four-in. intertie lines were subsequently installed on the RHR suction and discharge lines to facilitate cool down with an inoperable recirculation pump or an isolated loop. Use of these intertie lines will minimize the potential for RHR and Recirculation system steam condensation water hammer when the RHR system is placed into operation for shutdown cooling. The intertie lines are also provided for warmup of an idle recirculation loop (Reference 103).

The physical location of the field welds were improved to allow easier access to automated ISI equipment. This is expected to significantly reduce further inspection time and inspector radiation doses.

No significant difference in recirculation flow has occurred as a result of the replacement since piping size and effective pipe length are essentially unchanged. Hydraulic analyses have determined that the riser flows will change less than 1% and the system resistance change is negligible. Therefore core flow distribution remained unchanged.

No changes in operating parameters are anticipated as a result of these repairs. Replacement of existing 304 material with low carbon 316NG can be expected to enhance plant safety. This is because the possibility of reactor pressure boundary leakage will be significantly reduced as a result of the reduction or elimination of IGSCC that can be expected.

An Environmental Assessment of the Replacement of Recirculation System Piping at the Monticello Nuclear Generating Plant concluded the environmental impacts and costs of several reasonable alternatives were studied and found to be less desirable in terms of cost or exposure than the decontamination and replacement of piping (Reference 16).

4.3.2.3.2 Inlet Safe End Modification

The replacement safe ends and transition pieces for the recirculation inlet nozzles were furnished as ASME Section III, Class 1 material. For the safe end, this is consistent with the classification of the original safe end since it was supplied with the reactor vessel under the vessel ASME code stamp.

In the design of the safe end, mechanical pipe reaction loads and thermal sleeve reaction loads were specified. Due to the fact that the attached piping system was modified, the pipe reaction loads were established based on the structural capability of the attaching pipe material. This allowed the safe ends to be designed within the ASME code design requirements and provided flexibility in

completing the piping design. The thermal sleeve reaction loads that were used are the same as used in the original nozzle design.

The thermal cycles, which have been specified for the safe end thermal and fatigue analysis, represent only the operating cycles that involve a temperature transient in the nozzle. These transients have been categorized into service levels, which are required by the ASME Section III code, based on the frequency of the thermal transient and the function of the nozzle during the transient.

Piping-to-vessel inlet safe end nozzle stress were evaluated in accordance with ASME Section III, 1980 thru the Winter 1980 Addenda (Reference 79). Stress allowables were given in the piping design specification. Design and conditions for the inlet safe ends are those for the vessel (1391 psig and 575° per GE design specification for the safe ends).

To validate the overall design, the stress levels developed for B31.1 code (Reference 77) compliance were reviewed in accordance with NB-3600 of ASME Section III, 1980. During the Section III analysis, piping was evaluated against various operating pressures, temperatures and flow rates representative of time dependent thermal transients which the piping system may encounter during a 40-year lifetime. This analysis considered normal, upset and emergency conditions.

4.3.3 Performance Evaluation

4.3.3.1 System Design Test Data

Several series of tests have been conducted to study the performance of jet pumps under simulated reactor conditions. The tests were designed to verify analytical performance predictions and to supply additional design information regarding the effects of several jet pumps variables. The areas investigated included: mixing chamber to “throat” length, nozzle to throat spacing; nozzle eccentricity; nozzle size and configuration; simulated nozzle erosion; and diffuser configuration. Tests have been made using quarter scale models and actual full scale pumps. The quarter scale pumps were tested using groups of 4 pumps. The actual full scale pumps were tested individually. The tests made covered the pressure, temperature and subcooling ranges expected during reactor operation. This program yielded a better understanding of how the theory applies to reactor jet pump operation as well as the particular information required to verify the expected performance under specific operating conditions. A complete history of the jet pump testing is given in APED 5460 (Reference 3).

4.3.3.1.1 Performance Efficiency Tests

Early in the program it was verified that the maximum efficiency to be expected from any jet pump is a function of the design flow ratio. This ratio, designated M, is defined as the ratio of the driven mass flow (drawn from the downcomer annulus) to the driving mass flow through the nozzle; hence the efficiency is a function of the drive nozzle size. A jet pump was tested with nozzles designed for three flow ratios (M = 1, 1-1/4, 2-1/2). Figures 4.3-3 and 4.3-4 show the results for the nozzles designed to provide maximum efficiency at an M ratio of 2-1/2. On Figure 4.3-3 the upper curve shows the calculated performance of an idealized jet

pump with only mixing losses (no friction losses). This represents a maximum attainable efficiency for a simple jet pump. The second curve shows the calculated jet pump performance based on initial estimates of friction losses. The third curve shows the observed performance. Figure 4.3-4 shows the same data plotted as head ratio, N, versus flow ratio, M, where head ratio is defined as the ratio of the specific energy increase of the suction stream to the specific energy decrease of the driving stream.

From data generated with the three different nozzles, the following values of maximum efficiency were observed during the cold water phase of tests:

Design Flow Ratio	Efficiency, %
1	38-1/2
1-1/4	36-1/2
2-1/2	33-1/2

These data are plotted on Figures 4.3-5 and 4.3-6 which show calculated curves of the head ratio and flow ratio at peak efficiency as functions of the area ratio, R. The solid curves are calculated values. (These are the same curves as those shown on Figures 4.3-3 and 4.3-4 for "reasonable friction.") The dash curves through the data points show the experimental performance.

The performance tests have provided information regarding operation of the pumping system in the reactor both directly and indirectly. Direct study of the effect of fluid temperature variations and two-phase flow (carryunder), among other effects, was possible through Moss Landing Test simulation of these conditions. In addition, application of the performance data in various analytic models has made it possible to accurately evaluate the performance of the jet pump system under both normal and abnormal modes of reactor operation. The jet pump head-capacity characteristics curve for normal operating conditions is shown in Figure 4.3-7. The final jet pump system design characteristics, developed from analyses and application of these test data, are summarized in Table 4.3-1.

4.3.3.1.2 Cavitation Tests

Cavitation, caused by carryunder or insufficient subcooling has also been tested. Tests were performed at off-rated conditions in order to subject the jet pump to cavitation. This was done by reducing the subcooling and injecting superheated steam into the recirculation flow upstream of the jet pump suction inlet. These tests showed that when the jet pumps were caused to cavitate, the efficiency was increasingly degraded, but in spite of the presence of cavitation further increases in suction flow rate were still possible. Data from these tests clearly show the absence of an abrupt performance deterioration with cavitation; even when cavitation occurred, suction flow increases were still possible.

One test was designed to describe operation with superheated steam injected into the suction inlet. The objective was to determine jet pump performance with simulated carryunder introduced at the rate of 0.06% by weight (maximum available from the test facility). In addition, this test was performed with no subcooling. The resulting efficiency, 31.7% at M = 1.53, does not represent a serious degradation in jet pump performance even though the operating conditions

were particularly adverse; the jet pump was still capable of achieving a discharge head of 16.3 psi at full rated flow.

Data collected from this and other tests show the influence of subcooling on total flow. The data indicate that subcooling must be reduced to approximately 3-4 Btu/lb before cavitation begins to affect flow rate. This also reaffirmed the observation that the loss of subcooling does not bring about an abrupt loss in pumping capability. This cavitation threshold may be compared to normally available subcooling of about 20 Btu/lb.

4.3.3.1.3 Erosion-Corrosion Tests

Type 304 Austenitic stainless steel used in the jet pump assemblies has satisfactory corrosion resisting properties which are adequate for the 40-year design life. However, because of the high velocities existing in some regions of the jet pump, the potential problem of erosion-corrosion was investigated. The highest velocities occurring in the jet pump assembly are those at the nozzle where velocities may be as high as 180 fps. Erosion-corrosion tests have been performed on a small-scale nozzle configuration made from type 304 stainless steel. The test consisted of subjecting the nozzle to three consecutive 1000 hour periods of high temperature, high pressure flowing water simulating reactor conditions. Specifically the driving pressure was approximately 200 psi with the temperature ranging between 240°-415°F; the nozzle velocity was maintained at 460 - 470 fps. At the conclusion of each 1000 hour period, the internal diameter experiencing the high velocity flow was measured to determine the resulting enlargement from which the erosion rate was computed. The results of this test, and earlier tests conducted in substantially the same manner, indicate that the selected design value used in the jet pump design of 0.001 in./year is conservative. Using this design value, the nozzle I.D. would increase by no more than 0.08 in. during the 40-year design lifetime of the BWR. The effect of a slightly increased nozzle diameter (I.D.) is to force the system to run at a slightly lower M ratio. If the nozzle size originally were equal to or greater than the optimum nozzle size required for maximum jet pump efficiency, then the recirculation pump speed must be increased in order to maintain core flow. The design of the recirculation pumps includes sufficient margin to accommodate this lifetime effect. If the nozzle size originally were less than the optimum nozzle size, then the recirculation pump speed would be decreased in order to maintain design core flow.

The introduction of cavitation could accelerate erosion-corrosion in local areas. Cavitation is not expected to occur, however, because sufficient pressure or subcooling is available to suppress vaporization of the flowing liquid. Taking a typical example, the pressure in the 180° elbow (inlet subassembly) of the jet pump is more than 195 psi higher than the equivalent saturation pressure. At the nozzle where the highest velocity occurs, the pressure is still more than 120 psi above the saturation pressure. In the throat region (mixer) where the second highest velocities occur, approximately 20.4 Btu/lb subcooling is available for suppression of cavitation.

Other conditions which tend to aggravate the erosion-corrosion problem are not present in the jet-pump-reactor system:

a. the measurements taken from the full size prototype jet pump at reactor operating conditions indicate that high surface vibration does not exist on the jet pump components;

b. there are no points of direct impingement of the coolant flow on the jet pump subassemblies (the inlet subassembly also contains an internal vane, installed to reduce flow losses, which further reduces the possibility of impingement in the one subassembly where impingement could be considered most likely to occur);

c. the jet pump-reactor system contains no contaminants or particulate matter which can intensify erosion-corrosion; and

d. the seating surfaces at the inlet-riser joint and throat-diffuser joint contain Stellite-6 to minimize erosion-corrosion (wire drawing) due to leakage.

4.3.3.1.4 Stability Tests

Loop tests of a jet pump recirculation system were conducted at Moss Landing where simulation of reactor hydraulic conditions can be accomplished. The tests assessed the operational hydraulic stability of the system as constructed and provided data for evaluating the behavior of full scale jet pump recirculation systems installed in reactors. Because of an early jet pump system design which utilized an internally mounted manifold to distribute the drive flow to the various jet pumps, a developmental test program was structured to investigate the possibility of “manifold instability.” Analytical work had previously indicated that the probability of such a mode of flow oscillation was remote but verification was desirable. The term “manifold instability” describes a flow phenomena that sometimes exists when there are many exit flow ports from a small manifold. An example frequently cited is a manifold in a gas burner which has many gas ports to distribute the flows. A “dancing” of the flames is noted as gas flow increases through one or more ports while diminishing in others and then falls off as the flow through ports with previously low flow increases. Unsteady or unstable flow through the various gas ports is the result. The phenomena results because the pressure drops across the gas ports are very small and approximately the same order of magnitude as the internal manifold pressure drops between the gas ports. The increase in flow in one or more of the gas ports forces a redistribution of pressure drops internal to the manifold, such that new exit ports with low flow begin to be preferred.

The design of the jet pump system as used in BWRs is such that the “manifold instability” phenomena cannot exist. In order to obtain the high velocities through the drive nozzles necessary for the momentum exchange phenomena in the jet pump itself, the nozzle pressure drops are extremely high. In fact these pressure drops are orders of magnitude greater than the small pressure drops between the drive flow riser pipes within the now external manifold. As the necessary conditions are nowhere near present, “manifold instabilities” cannot exist in the jet pump system as designed. This fact was graphically demonstrated in the aforementioned developmental tests conducted in 1964 which attempted to force

a similar mode of flow instability. A bank of four quarter-scale jet pumps operating in parallel were tested under both cold and simulated reactor conditions. The drive flows for the four jet pumps were supplied from a common manifold. The discharge pressure of each jet pump could be individually controlled. No unstable performance was ever observed, even under conditions where efforts were made to drive the system into such a situation by imposing gross steady state and transient flow maldistributions upon the system.

Transient tests were conducted at the Moss Landing Test Facility which were structured to look for other possible modes of flow instability chargeable to the jet pump device. In May of 1967 tests were made on single unit quarter-scale jet pumps and in October of 1967, similar tests were conducted with quarter-scale dual unit jet pump assembly. The units tested were scaled down versions of the Dresden 2 production jet pumps.

The tests were designed to investigate the frequency response of the jet pump units for forced flow oscillations. Modifications were made to the equipment at the Moss Landing Test Facility such that valves in both the drive loop and total flow loop could be oscillated at frequencies from 0.1 Hz to about 7 Hz. This frequency range adequately covers the band of frequencies that are significant in BWR recirculation loop dynamics.

During both series of tests, flow instability modes were never observed. The total flow response to drive flow oscillations behaved in the manner that was predicted prior to running the tests. As the frequency of drive flow oscillation was increased, the ratio of the magnitude of the peak-to-peak total flow variation to the peak-to-peak drive flow variation became ever smaller. The phase shift between peaks in drive flow to peaks in total flow also increased as the oscillation frequency was increased. No resonant regions between 0.1 Hz and 7 Hz were ever observed for any of the tests conducted (Reference 3).

4.3.3.2 Normal Operation

4.3.3.2.1 Reactor Recirculation Flow Monitoring

The reactor core flow rate is monitored by direct measurement of differential pressure across the jet pump diffusers. For illustrative purposes, the twenty jet pumps can be divided into four groups of five each. A typical group is shown in Figure 4.3-8. In each group, one jet pump contains a diffuser with two static pressure taps. The remaining four units contain only one pressure tap. The "double tapped units" are calibrated by test prior to installation to determine the relationship between flow and differential pressure. This information is used to perform in-reactor calibration of the "top tap-to-lower plenum" pressure difference of all twenty jet pumps. The procedure can be summarized as follows:

a. Read the static pressure difference from each double tapped unit

b. Relate this pressure difference to flow rate by using the calibration information collected before installation of the double tapped units

c. Knowing the flow through each double tapped unit, determine the relationship between flow and the single tap to lower plenum pressure difference (see Figure 4.3-8).

d. Apply the single tap to lower plenum relationship found in the four calibrated double tap units to the remaining 16 single tap units.

After this calibration procedure has been completed, the total core flow is measured by analyzing the signals from the single tap-to-lower plenum pressure transducers on all twenty jet pumps. The resulting total core flow rate output signal is displayed on the reactor control board. The differential pressure signals are available in the Cable Spreading Room, and the double tap flows (1, 6, 11, and 16) are indicated in the Control Room.

4.3.3.2.2 <u>External Recirculation Loop Flow Monitoring</u>

The flow through the two external recirculation loops is continuously monitored by flow elements. The ΔP signals from these elements are each converted to flow signals, and transmitted to the reactor control room. In the control room, flow through each loop is both indicated and recorded.

In addition to the direct total external flow measurement and indication described above, an indirect measurement of external loop flow is available from the recirculation pump motor amperage which is indicated in the control room.

As might be expected, the core recirculation flow is essentially proportional to the total external loop flow rate during normal “flow control” manipulation of plant power output.

4.3.3.2.3 <u>Single Loop Operation</u>

The capability of operating at reduced power with a single recirculation loop is highly desirable from a plant availability/outage planning standpoint, in the event maintenance of a recirculation motor generator set or other component renders one loop inoperable. The resolution of Generic Issue B-19 regarding thermal-hydraulic stability has provided a basis to permit operation in the single loop mode with appropriate restrictions relating to stability concerns. Monticello submitted Technical Specification changes (Reference 22) to resolve the stability issue. Included in this resolution is single loop operation (Reference 23). Single loop operation is allowed within the Maximum Extended Load Line Limit Analysis operating domain as discussed here, however single loop operation is not permitted in the Maximum Extended Load Line Limit Analysis Plus (MELLLA+)/Extended Flow Window (EFW) operating domain. See USAR section 14.3. 01511192

The jet pump core flow measurement system is calibrated to measure core flow when both sets of jet pumps are in forward flow; total core flow is the sum of the indicated loop flows. For single loop operations, however, the inactive jet pumps will be back flowing. Therefore, the measured flow in the back flowing jet pumps must be subtracted from the measured flow in the active loop. In addition, the jet pump flow coefficient is different for reverse flow than for forward flow, and the measurement of reverse flow must be modified to account for this difference.

4.3.3.2.4 Pump Performance

The drive pumps are approximately 60 ft below normal reactor water level. This geometric head alone provides sufficient net positive suction head (NPSH) to prevent cavitation in the drive pumps at low speeds. At the higher speeds of normal operation, cavitation is prevented by the subcooling of the pump suction fluid; when returning feedwater (steam condensate) combines with the saturated fluid leaving the separators, the resulting mixture has more than enough subcooling to prevent cavitation. During initial operation with no feedwater flow, the drive pumps are started and brought up to minimum speed until operational pressure and steam production is established. As soon as steam is flowing, the downcomer flow is subcooled by the feedwater flow, and the drive pump speed may be increased to rated speed.

During normal operation, both the centrifugal pumps and the jet pumps require approximately 35 psi (110 ft) of net positive suction head (NPSH). At rated conditions the NPSH available (see Figure 4.3-9) is at least twice this amount.

Reduction of plant output by flow control maneuvering will cause a slight increase in this NPSH margin. The only time NPSH requirements approach the available NPSH is during high core flow and low thermal power operation. There is, however, no incentive to operate under high-flow, low-power conditions for sustained periods of time, and operation in extreme conditions is prevented procedurally as well as by interlocks.

Following refueling outages, performance of the pre-startup ASME Section XI reactor pressure test involves the operation of the recirculation pumps at high speed and no feedwater flow in order to heat up the coolant to the required test conditions. The feedwater low flow cavitation interlock on the recirc pump can be overridden when the reactor is shut down to allow pump operation in this manner. Provisions in the pressure test procedure provide controls to assure that available NPSH margins are maintained.

4.3.3.2.5 Recirculation Pump Seal Performance

In response to NUREG-0737, Item II.K.3.25 (Reference 80), the BWR Owners' Group submitted, in May 1981, an evaluation entitled "Effect of Loss of Alternating Current Power on Pump Seals" (Reference 81). A supplement to that evaluation providing test results concerning loss of pump seal cooling was submitted in September, 1981 (Reference 82). The endorsement of the BWR Owner's Group position in relation to the Bingham recirculation pumps installed at the Monticello plant was submitted in November, 1981 (Reference 83). The results of the evaluation and related tests which demonstrate the adequacy of the Monticello recirculation pump seal design are discussed below.

Under normal conditions, with the primary reactor system at or near rated temperature and pressure and the recirculation pumps either operating or secured, both reactor building closed cooling water system and seal purge system are operating. These two systems maintain the seal temperatures at approximately 120°F.

Recirculation pump vendor test data show that the pump seals may begin to deteriorate when seal temperatures exceed 250°F. If an event occurs where both closed cooling water to the pump seal heat exchanger and control rod drive seal purge flow are totally lost, the recirc pump seals will heat up. This will occur whether or not the pump is running. Vendor test data, taken while operating at approximately 530°F/1040 psia, indicate that the seals will heat up, reaching 250°F approximately 7 minutes after a total loss of cooling.

Similar test data indicate that if either one of the seal cooling systems is operating, the seal temperatures remain well below 250°F and no seal deterioration should occur.

If both closed cooling water and seal purge are totally lost, and if the seal temperatures exceed 250°F, seal deterioration may occur, resulting in primary coolant leakage to the drywell. An analysis of fluid loss through a degraded seal modeled the fluid leakage path as a series of fluid volumes with interconnecting junctions, each having appropriate initial conditions (Reference 4).

The model assumed gross degradation of the mechanical seals. Gross failure of these seals encompasses warpage, fractures and grooving of the seal faces due to excessive thermal gradients and dirt.

The results of this seal leakage analysis show that even with gross degradation of the seal, the leakage would be less than 70 gallons per minute per pump. This amount of leakage is within the capacity of normal or emergency vessel water level control systems which will easily compensate for it. A leakage of 70 gpm is equivalent to a liquid leak of 0.001 ft^2 flow cross-sectional area. Even if the seals on both of the recirculation pumps fail, the effects of such leakage do not influence the results of loss-of-coolant accident analyses. It is emphasized that the seal leakage analysis is extremely conservative and a leakage rate of 70 gpm is not expected upon seal failure. USAR Section 8.12 for Station Blackout assumes a nominal value of 18 gpm for seal leakage from each pump (Reference 140).

If an event occurred which led to seal degradation and subsequent leakage, the operator could isolate the pump by closing both the recirculation suction and discharge gate valves.

Based on fluid loss analysis of extremely degraded seals, the leakage is less than 70 gallons per minute. This amount of leakage will not lead to a safety concern, but may degrade the seals such that they would have to be repaired prior to resuming operation.

4.3.3.2.6 External Recirculation Loop Integrity Monitoring

Individual pressure differential sensors sense the differential pressure between the jet pump inlets of one recirculation loop and the jet pump inlets of the other recirculation loop. Excessive pressure differential, an indication of a failed recirculation line, is used to properly sequence the operation of the low pressure coolant injection subsystem (LPCI) described in Section 6.2.3.2.5.

4.3.3.3 Drywell Coolant Leak Detection System

Drywell leakage is broadly classified as being from identified or unidentified sources. Identified leakage is piped from the recirculation pump seals, valve stem leak-off, reactor vessel flange leak-off, and bulkhead and bellows drains to the drywell closed radwaste sump. All other leakage is collected in the drywell open radwaste sump and classified as unidentified leakage. Steam from a reactor vessel crack would condense in the drywell and collect in the open radwaste sump. Both the open and closed radwaste sumps are connected by piping to two smaller sumps containing a conductivity probe operated pump and level alarms. Both sump systems are of equal volume and nearly identical in construction. Drywell total leakage is the sum of drywell identified and unidentified leakage. If unidentified leakage has been identified and quantified, it may be classified as identified leakage. However, the total leakage limit would remain unchanged.

Each of the systems uses two timers to measure the interval between auto sump pump operation. One timer actuates an alarm if this interval decreases below an adjustable minimum. The other timer actuates an alarm if this interval corresponds to slightly less than 5 gpm for drywell unidentified leakage and 20 gpm for drywell identified leakage. Actual leak rate determinations are typically made twice a day using flow integrators in the pump discharge line of each system, or by using the level instrumentation, including computer points, associated with each sump. Other methods may be used to verify leakage. If both sump pumps or level and flow instruments associated with one sump are inoperable, such that one sump is overflowing to the other, all drywell leakage is assumed to be drywell unidentified leakage unless the leakage has been identified and quantified.

Three level switches are provided on the sump. One level switch starts the auto pump, the second level switch starts the standby pump. And the third actuates a high level alarm in the control room. Hand switches for manual operation of the pumps and indicating lights are provided in the control room.

The pumps are interlocked to the reactor protection system through two isolation valves on the sump discharge header. When these two valves are closed by a containment isolation signal, the two pumps will automatically trip. Indicating lights for each valve and one alarm for both valves are provided in the control room. These two valves will fail closed. A hand switch, located in the control room, can also be used to operate these valves.

A temperature element with a temperature indicating switch is provided to monitor the sump temperature. On a high temperature signal, the auto sump pump starts. The main discharge valve closes and the inlet valve opens allowing the sump effluent to be recirculated through a heat exchanger and back to the sump. This operation continues until the liquid is cooled enough to discharge to either the Waste Collector Tank (T24) or the Condensate Drip Tank (T22).

In addition to the sump leak measuring system, the following additional measurements are available to give indication of gross coolant leakage or changes in coolant leakage.

a. Drywell pressure

b. Drywell temperature

c. Drywell particulate airborne activity (CAM sampler)

4.3.3.4 Equipment Malfunctions and Systems Transients

Recirculation system malfunctions and transients are discussed in Section 14.7.

4.3.3.5 Carburized Jet Pump Castings

Subsequent to the discovering of carburized castings in the jet pump assemblies, a failure analysis was performed to determine the consequences of an extremely unlikely failure of the different castings involved. This analysis indicated that, during normal operation, reactor safety would not be jeopardized, and during the accident case, the core flooding capability would not be impaired.

The factors included in this review are summarized as follows:

a. Mechanical Integrity

1. The stress levels existing in these castings during normal operating conditions are quite low. The hydrostatic pressures during normal operation in the elbow and nozzle produce stresses less than 4000 psi which is one-quarter of the allowed design stress intensity value given in the ASME Code, Section III. The flange in normal operation experiences a maximum stress due to external pressure of 500 psi, and a tensile stress due to the dead weight of the Mixer of 32 psi. The adapter in normal operation experiences a maximum stress of 130 psi due to external pressure, and a tensile stress of less than 25 psi due to the dead weight of the remaining portion of the Mixer. The only exceptional stress occurs in the coupling where the bolt load produces a bending stress in the bolt lugs of approximately 16,000 psi. If a crack did occur across a bolt lug, the subsequent deflection would be in the direction of reducing the bending stress, and hence a total failure would be highly improbable.

2. During the accident case (use of the term “accident case” hereinafter refers to a recirculation line break), the hydrostatic pressures on the elbow and nozzle act externally rather than internally; these external pressures produce stresses of 7000 psi in the minimum wall regions of these castings. In addition, the irregular features of these parts, (struts of the nozzle, internal vane in the elbow, and heavy wall over the top of the elbow) provide additional margin against external pressures.

3. The stresses in the coupling and flange are approximately the same in the accident condition as they are in the normal operating condition.

4. The maximum stresses in the adapter in the accident condition are due to internal pressures and do not exceed 325 psi.

5. Many areas of these castings have been machined during subsequent fabrication operations. This includes all weld preps, the interior of the nozzle (excluding the suction inlet region), flange, and coupling, the top, outer, heavy wall portion of the 180° elbow where the 25,000 lb clamping load is applied, and all mating flange surfaces at the inlet-mixer joint.

6. The external as-cast surfaces of the coupling are loaded in compression which tends to negate the effects of the carburized surface. The internal tensile loaded surface is fully machined.

7. A crack propagation analysis was performed to estimate the extent of crack growth due to mechanical loading over the reactor lifetime in the jet pump elbow and nozzle castings. Using the assumptions of a 0.10 in. deep crack of indefinite length (5 in.) existing in 0.38 in. thick stainless steel at 120 startup shutdown cycles with the stress ranging between 20,000 to 30,000 psi, the predicted crack growth was 41 X 10^{-6} in. This crack growth rate was considered negligible. Therefore, this event is not limiting and is not included in the reactor cycle counting program.

b. Metallurgy

1. Duplex Austenitic stainless steel castings with ferrite exhibit a high degree of resistance to stress corrosion crack propagation. The term “duplex” in this case refers to the fact that the Austenitic stainless steel alloy contains delta ferrite in the Austenitic matrix. This high resistance is evidenced by observations of intergranular stress corrosion cracks, propagating in single phase wrought Austenitic stainless steel material, being terminated in weld metal composed of as-deposited duplex Austenitic stainless steel with ferrite similar to cast material. It is unlikely that the base metal of the stainless steel casting will permit deepening of any linear indications which may be existing in the carburized surface. Castings procured to General Electric specifications have controlled chemistry to produce Austenitic stainless steel with duplex structure.

2. Furnace sensitized and/or welded and unsolution heat treated Austenitic stainless steels have been shown by loop tests and service history to be acceptable for in-reactor and external loop service. The microstructure of much of the carburized zone is similar to sensitized stainless steel, and it is expected to show similar corrosion behavior.

3. Samples removed from actual carburized surfaces and from the interior of carburized castings were tested in the General Electric Vallecitos Nucleonics Laboratory’s accelerated corrosion test facility under constant tensile load conditions in excess of yield in 550°F, 1000 psi water with 100 ppm oxygen added. The carburized sample failed by intergranular cracking in times similar to those of furnace sensitized wrought Austenitic stainless steel. The interior specimen did not fail this test verifying that the subsurface uncarburized material is resistant to stress corrosion cracking.

c. Design Requirements

1. Despite the existence of the carburized surface condition on the previously described castings, the Jet Pump Assemblies meet all original design requirements relative to material specifications, fitting tolerances, leak tightness, removability, inspectability, and 40 year design life. The jet pumps are fully capable of meeting previously measured pumping efficiencies, and there will be no change in previously measured flow calibration constants.

2. All castings containing the carburized condition are part of the removable portion of the jet pump assembly. This condition does not alter the capability of removing the jet pump assemblies from the reactor pressure vessel. In addition, the fitting tolerances as established in the specification permit complete interchangeability of removable components.

d. Quality Control Program Summary

The following is a statement of the Quality Control Program for various castings.

1. Original Jet Pump Castings

   a) A review of vendor general capability for producing the types of castings, in-place quality control systems, heat treating facilities and procedures, non-destructive test techniques, etc., was conducted.

   b) Welding procedures for repair of defects were reviewed.

   c) The first piece was checked for complete conformance to the drawing (dimensional, radiographic, physical and chemical properties per ASTM A351, Liquid Penetrant Test) (Reference 76).

   d) Audits, during the production of the required quantities, were conducted by both General Electric and WISCO[1] personnel.

2. Replacement Jet Pump Castings

   In addition to the above, the following steps were taken to preclude the production of castings with a carburized skin:

   a) The casting process was evaluated after being revised by changing the additives to the sand.

   b) Procedures for this revised process were reviewed and approved.

   c) Periodic destructive examination of production castings is conducted to insure continued conformance to the procedures and to assure lack of a carburized skin.

   d) An additional liquid penetrant examination of the castings is being done at WISCO when the castings are received from the foundry.

   e) The use of integrally cast test lugs is also used on castings as an additional means of verifying the required process control.

[1] Williamette Iron and Steel Company

3. Repair of Original Castings

   Prior to the shipment of the original castings from WISCO to the site, all as-cast surfaces were liquid penetrant examined. The following controls were applied to the repair of all defects detected in this manner.

   a) All defects were identified by size, type and location.

   b) All defects were repaired either by complete defect removal with no material addition or the defect was repaired by welding.

   c) All repairs were examined by liquid penetrant to E5D-YP2.

   d) All repairs were conducted under the direction and witness of a General Electric Quality Control Engineer.

   e) It was unnecessary to do any repair of the transition castings because they were replaced with castings as described in b) above.

   f) All as-cast surfaces were removed from the collar casting by machining as previously described.

e. Failure Analysis

1. An unlikely jet pump failure during normal operation would be immediately detected by the core flow measuring equipment. The partial loss of core flow due to a jet pump failure would not jeopardize reactor safety because a loss of core flow for any reason always results in a flow control type power reduction due to the BWR void-power relationships; thermal margins always increase under these circumstances. APED 5460, “Design and Performance of GE-BWR Jet Pumps” (Reference 3) fully discusses loss of core flow due to jet pump failures.

2. An unlikely jet pump failure during an accident (recirculation line break) would not jeopardize reactor safety. Failure of the castings involved would present the following situation:

   a) A full circumferential break around the flange would permit the mixer subassembly to fall until bottoming occurred in the slip joint at the mixer-diffuser interface. The maximum slip joint gap is 0.56 in. This break could occur no more than 10-1/2 in. below the elevation of the suction inlet (2/3 core height). A break creating a gap no greater than 0.56 in. in axial length, no more than 10-1/2 in. below the 2/3 core height elevation would not bypass enough water to prevent flooding the core to the specified 2/3 core height. The leakage through this gap would be minor.

   b) A failure of the coupling would not result in consequences any more severe than described in (a) above. A complete failure of the coupling may permit separation at the inlet-mixer joint which is 6 in. below the suction inlet. This would permit the mixer to fall by the

same amount as in (a) above, resulting in a similar full circumferential gap in the mixer. The resulting leakage would be slightly less severe than a flange break because of a smaller static head.

c) A highly improbable full circumferential break around the lower end of the adapter casting would permit the remaining lower portion of the mixer to fall until bottoming occurred in the slip joint. This would result in a maximum gap of 0.56 in. at a maximum depth of 76 in. below the Jet Pump suction inlet. The leakage due to this static head is calculated to be no greater than 500 gpm.

d) A nozzle failure between the suction inlet and the coupling to the mixer would be no more severe than described in (a) and (b) above with an even smaller static head available to drive leakage flow. A coincident failure of all three symmetrical vanes would permit the suction inlet to drop a maximum of 0.56 in. which would have a negligible effect on the 2/3 core flooding capability.

e) A failure of the nozzle casting in the actual nozzle region or of the 180° elbow would cause consequences no more severe relative to core flooding than in (d) above under the three-vane failure. It could perhaps be postulated that a full circumferential break in a vertical plane across the neck of the 180° elbow would permit the remaining half-elbow, nozzle, and mixer sections to rise vertically with disengagement occurring at the slip joint thus preventing core flooding. A measure of the remoteness of this eventuality is found in the following facts.

1) The failure must occur across a narrow band in the neck of the 180° elbow approximately 2-1/2 in. wide with the break being a clean cut which permits relative motion between the two halves.

2) The bending stresses which could conceivably cause a failure in this segment of the 180° elbow are low in both the normal and accident conditions. In the normal operating condition, the bending stresses are calculated to be less than 600 psi. In the accident case, the bending stresses are found to be no greater than 2100 psi.

3) During normal operation, the internal hydrostatic pressures create stresses less than 3200 psi in the 180° elbow; during the accident case, the external hydrostatic pressures cause stresses of approximately 7000 psi. It is very unlikely that these stresses, acting coincidentally with the bending stresses given in (2), above, would produce a failure preferential to the zone described in (1) above.

4) The "clean break" type failure required must also include failure of the internal vane.

f. Radiography

1. All 180° elbows, nozzles, couplings and transition pieces have been radiographed in preselected areas imposed by drawing requirements. The areas of interest relative to radiography are generally those regions where thin sections join thick sections and, additionally, in the elbow, the region where the clamping force is applied. Because of the low stresses existing in these areas, the ASTM E71 Class 3 (Reference 84) acceptance standard is judged to be adequate.

g. Historical Record

1. Despite extensive use of Austenitic stainless steel casting in the nuclear industry, there remains no known failures of stainless steel castings in reactor service due to surface carburization.

2. A typical carburized Transition Piece Casting was used in jet pump tests at reactor conditions at Moss Landing. The total immersion time for this test was approximately 2000 hours. Inspection revealed no sign of crack generation or propagation during this test.

From the foregoing review, it is concluded that the Jet Pump castings under consideration are entirely satisfactory for reactor use. It is considered highly improbable that a failure would occur because of the low stresses which exist in these parts and the high crack propagation resistance of the material. In addition, all of the original jet pump design requirements are still met. A failure analysis indicates that in the extremely unlikely event of a casting failure during normal operation, reactor safety is not impaired, and, in the accident case, the core flooding capability is not significantly impaired. For these reasons, the castings are considered acceptable for use in the Boiling Water Reactor.

4.3.4 Inspection and Testing

Prior to installation, the head flow characteristics and the flow measurement calibration constants of the four double tapped jet pumps was determined by test. Extensive testing to verify and determine the jet pump performance was performed during the pre-operational test program.

During subsequent operation, the system is in continuous use and the instrumentation provided assures adequate monitoring of system performance.

An additional surveillance test program was also developed for the jet pumps for early identification of any jet pump hold-down beam failure. This test program was implemented at Monticello via Technical Specification changes reflected in License Amendment 42 (Reference 108) and maintained in Improved Technical Specifications Amendment 146 (Reference 113).

The recirculation piping can be inspected by removing the insulation. Periodic random visual inspection of areas of highest stress concentration or areas of more importance from a leak standpoint are made at regular scheduled refueling outages.

The criteria for inspection of recirculation piping are based on the probability of a defect occurring or enlarging at a given location. These include areas of known stress concentration and locations where cyclic strain or thermal stress might occur. A statistically significant portion of the system was inspected. The type of inspection at each location is dependent on the type and orientation of possible defects. Direct visual examination is conducted wherever possible since it is sensitive, fast and positive. Aids to direct visual examination such as liquid penetrant inspection are utilized wherever practical and where added sensitivity is required. Ultrasonic testing and radiography are performed where defects could occur on concealed surfaces.

A critical defect on the order of feet in length is required to cause a running crack in the materials used to construct the primary system. Considering the wall thickness of these systems, a crack cannot grow to such a length before it penetrates the wall and causes a leak. The inspection procedures are geared to the detection of leaks and small defects. These have a minor effect on plant safety but they may affect plant availability, thus there is a high incentive to detect them. Experience with operating plants to date shows the incidence of leaks to be very small; the probability of finding them at an early period is very high. Thus a small defect will not grow to a critical length prior to being detected. The specific details of the required periodic examinations are given in the current revision of the “Monticello Inservice Inspection and Testing Program,” which is specified in Section 13.4.6.

Table 4.3-1 Jet Pump System Characteristics

Number of Jet Pumps	20
Number of Jet Pumps per Loop	10
Description	6.07 in. throat I.D., 14.0 in. diffuser, I.D.
Nozzle internal Diameter	2.70 in.
Operating Parameters at Warranted Condition	
Diffuser Exit Velocity	15.9 fps
Driving Flow, per Loop	11.4×10^6 lb/hr
Diffuser Flow, per Loop	28.8×10^6 lb/hr
M Ratio	1.59
MN Efficiency (includes 180° Bend)	33.5
Jet Pump Head (Discharge - Suction Total Pressure)	96 ft

4.4 Reactor Pressure Relief System

4.4.1 Design Basis

The design basis for the Reactor Pressure Relief System is:

a. The Reactor Pressure Relief System prevents overpressurization of the nuclear system in order to prevent failure of the reactor system due to pressure.

b. The safety/relief valves which are part of the Automatic Depressurization System (RV-2-71A,C & D) provide automatic reactor depressurization for small breaks in the nuclear system occurring with a coincidental failure of the High Pressure Coolant Injection System (HPCI) so that Low Pressure Coolant Injection (LPCI) and the Core Spray System can operate to protect the fuel cladding.

c. The safety/relief valve discharge piping, quencher and supports are designed to accommodate the effects of normal earthquake, safety/relief valve discharge, and LOCA related loads. These loads have been defined by the topical report NEDO-21888, "Mark I Containment Program Load Definition Report" (Reference 86) and the Nuclear Regulatory Commission Safety Evaluation Report NUREG-0661 (Reference 87). The design of the safety/relief discharge piping in the wetwell also included the evaluation of the fatigue effects from the cyclic loads. These loads were verified to remain bounding for operation at 2004 MWt (Reference 137).

d. The Reactor Pressure Relief System is designed for testing and periodic verification of the operability of the system.

e. The safety/relief valve discharge lines are routed from the drywell area through the vent lines into the suppression chamber. Inside the suppression chamber, the safety/relief valve discharge lines are routed to standard Mark I T-quencher discharge devices (see Figure 5.2-30) located below the water level.

f. The safety/relief valves reclose following a plant isolation or any load rejections so that normal operation can be resumed as soon as possible.

g. The safety/relief valves which are also part of the SRV Low-Low Set System and have low-low set actuation (RV-2-71E,G & H) are prevented from subsequent manual or automatic actuation prior to the water leg receding in the discharge line to prevent excessive hydrodynamic loading of the discharge piping and suppression chamber components.

4.4.2 Description

The Reactor Pressure Relief System includes 8 safety/relief valves located on the main steam lines within the drywell between the reactor vessel and the first isolation valve. (See Table 4.4-1.) The safety/relief valves, which discharge to the suppression pool, provide two main protective functions: 01552261

a. Overpressure relief operation. The valves are opened (self-actuated) to limit the pressure rise.

b. Depressurization operation. The same valves are opened automatically or manually by applying air to a diaphragm-operated pilot valve to depressurize the reactor.

The main steam lines, in which the safety/relief valves are installed, are designed, installed and tested in accordance with the applicable codes. The safety/relief valves are distributed among the four main steam lines so that a single accident cannot completely disable a safety, relief, or automatic depressurization function. Section 15 Drawings NH-36241, NH-36243, and NH-36243-1 show the P&ID of the Reactor (Nuclear Boiler) System and shows the arrangements of the safety/relief valves.

4.4.2.1 Safety/Relief Valves

The safety/relief valves are designed, constructed, and marked with data in accordance with the ASME Boiler and Pressure Vessel Code, Section III, (Reference 43) Article 9, and in accordance with USAS B31.1.0 (Reference 44) and B16.5 (Reference 88). Popping-point tolerance (pressure at which valve “pops” wide open) is in accordance with ASME Section I, Paragraph PG-72(c) (Reference 89). Each valve is self-actuating at the set relieving pressure, but may also be actuated by admitting air pressure on a pilot valve which causes the main valve to open at lower pressures. For depressurization operation, the safety/relief valves are provided with this device. The safety/relief valve is designed for operation with saturated steam containing less than 1% moisture. The relieving pressures for overpressure relief and safety operating modes are adjustable between 1025 and 1155 psid. A simmer margin improvement program (References 18, 19, 20, and 21) resulted in increasing the maximum allowable safety/relief valve self-actuation setpoint of all eight valves. An increase in safety/relief valve simmer margin increases valve reliability by reducing the probability of valve leakage and spurious opening during operation. SRV simmer margins were evaluated for operation with a reactor pressure up to 1010 psig and verified to be adequate (Reference 135). The maximum elapsed time between reaching set pressure and start of main disc motion is 0.4 sec and the maximum main disc stroke time is 0.1 sec.

The dual safety/relief valve consists of three main sections. The pilot valve section is a relatively small, self-actuated relief valve, integral with the main valve, which provides pressure sensing and main valve control functions. The main element of this pilot valve is a precision-machined spring-bellows, the expansion of which accurately controls pilot valve opening pressure. The remote air actuator is another valve which controls the main element but is actuated by externally supplied pressure applied to a diaphragm. The main valve section is a hydraulically operated, reverse seating globe valve which, when actuated by the pilot valve, provides the pressure relief function by opening to discharge reactor system steam to the suppression pool.

A typical sequence of operation for overpressure relief self-actuation can be described as follows (refer to Figures 4.4-2 and 4.4-3):

a. In the closed position (Figure 4.4-2), the bellows is mechanically extended a slight amount by the preload spacer to provide a preload force on the pilot disc. This seats the first stage valve tightly and prevents reverse leakage at low system pressures or high back pressures. The main valve disc is tightly seated by the combined forces exerted by the main valve preload spring and the system internal pressure acting over the area of the main valve disc. In a closed position, the static pressures are equal in the valve body and in the chamber over the main valve piston. This pressure equalization is made possible by leakage through the piston orifice.

b. As system pressure increases, the preload force on the pilot disc is reduced to zero as the bellows is extended farther and the disc is held closed by the internal pressure acting over the pilot valve seat area. This hydraulic seating force, which is significantly greater than the initial preload, increases with increasing system pressure and prevents leakage or “simmering” at pressures near the valve set pressure.

c. As system pressure further increases, bellows expansion reduces the gap between the stem and the disc yoke. When the stem contacts the yoke, further pressure increase reduces the net pilot seating force to zero and lifts the first stage pilot valve from its seat.

d. Once the pilot valve starts to open, the hydraulic seating force is eliminated, resulting in a net increase in the force tending to open the pilot valve. This increase in net force produces the “popping” action during pilot valve opening (Figure 4.4-3).

e. Opening of the first stage pilot valve admits fluid to the operating piston of the second stage valve, causing it also to open.

f. Opening of the second stage pilot valve vents the chamber over the main valve piston and the downstream side of the valve. This venting action creates a differential pressure across the main valve piston almost equal to the system pressure and in a direction tending to open the valve. The main valve piston is sized so that the resulting opening force is greater than the combined preload and hydraulic seating force. Therefore, opening of the pilot opens the main valve.

g. As in the case of the pilot valve, once the main valve disc starts to open, the hydraulic seating force is reduced, causing a significant increase in opening force and the characteristic full opening or “popping” action.

h. When the pressure has been reduced sufficiently to permit the pilot valve to close, leakage of system fluid past the main valve piston repressurizes the chamber over the piston, eliminates the hydraulic opening force, and permits the preload spring to close the valve. Once closed, the additional hydraulic seating force due to system pressure acting on the main valve disc seats the main valve tightly and prevents leakage.

The diaphragm operated remote air actuator valve also displaces the second stage piston which in turn controls the main valve as shown in Figures 4.4-2 and 4.4-3. Using this feature, the safety/relief valve can be remotely opened by supplying pressure on the diaphragm of the remote air actuator.

The safety/relief valves are installed so that each valve discharge is piped through its own discharge line to a point below the minimum water level in the primary containment suppression pool to permit the steam to condense in the pool. The evaluation of the safety/relief valve discharge lines and supports for effects of LOCA-related loads and safety/relief valve discharge related loads was provided in the Monticello Mark I Containment Long Term Program Plant Unique Analysis Report (Reference 90). See USAR section 5.2.3 for further discussion on the Mark I analyses.

The ramshead devices at the end of the safety/relief valve discharge lines were replaced by (See Figure 5.2-30) T-quenchers. The T-quencher discharge device is used to ensure stable steam condensation at expected pool temperatures, and to mitigate the pressure, thrust, and hydrodynamic loads on the suppression chamber and safety relief valve discharge piping. The description and the evaluation of safety relief valve discharge and LOCA related loads for the discharge piping in the wetwell, the T-quencher, and the quencher supports was provided in the Mark I Containment Long Term Program Plant Unique Analysis Report (Reference 90). See USAR section 5.2.3 for further discussion on the Mark I analyses.

After a safety relief valve has closed, the steam in the discharge line is condensed causing a rapid pressure drop which draws water from the suppression pool back into the line. Each safety relief valve discharge line has an 8 in. relief valve to allow the drywell atmosphere to enter the discharge line when the pressure in the discharge line becomes negative with respect to the drywell pressure. The drywell atmosphere repressurizes the discharge line, minimizing excessive reflooding of the line, and subsequently restoring the water leg in the line to the suppression pool water level. The time between closure of a safety/relief valve and the normalization of the water leg in the discharge line has been determined to be 5.75 sec maximum. To prevent a valve from reopening within this time limit and causing significant hydrodynamic loads within the piping and suppression pool, a SRV Low-Low Set System has been implemented. A functional descriptive summary is provided in Section 4.4.2.3 and a detailed discussion of system operation is contained in Reference 17. The safety/relief valves are located on the main steam line piping, rather than on the reactor vessel top head, primarily to simplify the discharge piping to the pool and to avoid the necessity for removing sections of this piping when the reactor head is removed for refueling. In addition, the safety/relief valves are more accessible during a quick shutdown to correct possible valve malfunctions when located on the steam lines.

The set pressures and capacities of the 8 safety/relief valves are shown in Table 4.4-1. 01552261

During normal plant operation, pneumatic supply for the eight (8) safety/relief valve actuators is provided by the Instrument Nitrogen System. This pneumatic supply will automatically transfer to the Instrument Air System on high or low Instrument Nitrogen System pressure (Reference 13). Safety-related backup pneumatic supplies are also provided by the Alternate Nitrogen System, which automatically supplies pressure to the safety/relief valve actuators upon loss of the Instrument Nitrogen System. There are four (4) safety-related pneumatic supply sources which are identified as follows:

SAFETY RELIEF NUMBER	SAFETY-RELATED BACKUP PNEUMATIC SUPPLY SOURCE
RV-2-71A	Alternate Nitrogen System - Train A
RV-2-71B	Alternate Nitrogen System - Train A
RV-2-71C	Alternate Nitrogen System - Train B
RV-2-71D	SRV Accumulator Bank - East
RV-2-71E	Alternate Nitrogen System - Train A
RV-2-71F	Alternate Nitrogen System - Train B
RV-2-71G	SRV Accumulator Bank - West
RV-2-71H	Alternate Nitrogen System - Train B

The original design requirement for the safety-related backup pneumatic supplies (which originally consisted of an accumulator and check valve arrangement for each safety-relief valve) was that they be sized to hold the safety/relief valve open continuously for 30 minutes following loss of normal pneumatic supply. Subsequent requirements invoked for the pneumatic supplies are as follows:

a. The Automatic Depressurization System (ADS) safety/relief valves are capable of cycling 5 times over a 10 hr period (References 80 and 41).

b. The Low-Low Set System safety/relief valves are capable of 5 cycles or continuous safety/relief valve operation for 10 minutes following a LOCA or HELB.

c. The Automatic Depressurization System safety/relief valves are designed to withstand a hostile environment and still perform their function for 100 days following an accident (Reference 80).

d. A minimum of two SRVs are to be available for manual depressurization and alternate shutdown cooling for all fire events until cold shutdown conditions are achieved. Cold shutdown conditions are to be obtained within 72 hours while assuming loss of offsite power (Reference USAR Appendix J).

The four backup safety-related pneumatic supplies are designed, sized, and distributed among the eight safety/relief valves such that the above requirements are met. The minimum number of safety/relief valves will be available, when required, to fulfill Automatic Depressurization System, Low-Low Set System, manual depressurization, and alternate shutdown cooling functions under all postulated accident events and transients events coincident with the failures required to be assumed during these events. One clarification is that the 100 day NUREG-0737 requirement was satisfied by installation of an Alternate Nitrogen System to two (2) non-ADS safety/relief valves. This was accepted by the NRC and is documented in Reference 26. The safety-related pneumatic supplies are supported to Class I equipment criteria.

4.4.2.2 <u>Automatic Depressurization System</u>

The Automatic Depressurization System (ADS) feature of the Reactor Pressure Relief System serves as a backup to the High Pressure Coolant Injection (HPCI) System under LOCA conditions. If the HPCI System does not operate, the system is depressurized sufficiently to permit the Low Pressure Coolant Injection (LPCI) and Core Spray System to operate to protect the fuel cladding. Depressurization is accomplished through automatic opening of the safety/relief valves to vent steam to the suppression pool. Three safety/relief valves are included in the ADS. All three ADS valves are required to provide sufficient capacity for the ADS. Two ADS valves with bore size 5.03" or larger and one ADS valve of bore size 4.940" or larger are required for sufficient capacity. 01511192 Emergency depressurization within the Emergency Operating Procedures requires manual opening of three valves. Further description of the operation of the Automatic Depressurization System is found in Section 6.2.

4.4.2.3 <u>SRV Low-Low Set System</u>

The function of the SRV Low-Low Set System is to minimize the possibility of an SRV reopening with an elevated water leg in its discharge line. The elevated water leg occurs after an SRV closes. The condensing steam in the SRV discharge line creates a vacuum which draws torus water up into the discharge line. The water thrust load on the containment resulting from an SRV reopening while an elevated water leg is present in its discharge line would exceed the containment's design strength.

The system also reduces subsequent SRV actuation during plant transients and improves overall SRV performance.

The SRV Low-Low Set System controls the operation of three SRVs and is totally independent of the Automatic Depressurization System. The low-low set protective function is fully automatic and requires no operator action. The logic is also dual divisional.

As the name implies, the low-low set SRVs have lower opening setpoints than their mechanical setpoint. These lower setpoints are for electrical/pneumatic actuation of the low-low set SRVs. The low-low set SRVs will open if the following conditions occur; reactor SCRAM, reactor pressure greater than the low-low setpoint and the SRV low-low set hand switch in the auto position. The system operates on a 2 out of 2 once logic. For either division to operate, scram signals from both Reactor

Protection trip systems and two reactor high pressure signals must be received. See Figures 4.4-4 and 4.4-5.

By opening the low-low set SRVs at a set-point lower than the mechanical setpoint, re-opening of a non-low-low set SRV will be prevented following a reactor isolation transient. Closing of a low-low set SRV occurs after an 80 psi blowdown of reactor pressure is detected. If an SRV has opened during a plant transient event, the 80 psi blowdown will ensure it mechanically recloses. The setpoints for the low-low set valves ensure that they will be the first SRVs to open and the last to close. After a low-low set SRV has opened and closed, a time delay relay prevents the plant operators or the low-low set logic from immediately re-opening the SRV to allow the water leg in the SRV discharge line to recede. Refer to the Technical Specifications for the current time delay setting (Reference 102).

4.4.2.4 Manual Depressurization

A manual depressurization of the nuclear system can be effected in the event the main condenser is not available as a heat sink after reactor shutdown. The steam generated by core decay heat is discharged to the suppression pool. The safety/relief valves are operated by remote-manual controls from the main control room or remote shutdown panel to control nuclear system pressure.

4.4.3 Performance Analysis

The reactor coolant system safety/relief valves assure that the reactor coolant system pressure safety limit is never reached. In compliance with Section III of the ASME Code, 1965 edition (Reference 51), the safety/relief valves are set to open at a pressure no higher than vessel design pressure (with at least one safety/relief valve set to open at a pressure no greater than design pressure) and they limit the reactor pressure to no more than 110% of the reactor pressure vessel design pressure. See Section 14.5.1 for further details.

A series of water discharge tests of safety/relief valves used in Boiling Water Reactors (BWR) were conducted to demonstrate the operational adequacy of the SRV and SRV discharge piping integrity, for expected operating conditions for transients and accidents. The tests were performed to satisfy requirement II.D.1 of NUREG-0737 (Reference 80) and are documented in NEDO-24988, October 1981 (Reference 91).

The tests were run at the Wyle Laboratories test facility in Huntsville, Alabama. The facility included a steam and water supply system, the test SRV mounted on a representative steam line, and a representative SRV discharge line routed to a pool of water. Opening and closing of the SRVs were monitored. Fluid conditions and flows were measured, as were strains, accelerations, temperatures, and pressures in SRV and associated piping. Monticello uses 3-stage Target Rock valves, which were included in these tests.

The water discharge test conditions simulated the alternate shutdown cooling condition, which is an operating condition which is considered in the design evaluation of many BWR plants. The results show that all of the tested SRVs opened and closed on command for all water tests. The measured SRV discharge line loads for water discharge were significantly less than those for the high pressure steam discharge condition for which the piping is designed.

The tests and analyses described herein verify the adequacy of SRV operation and the integrity of the SRV discharge piping under expected liquid discharge conditions, and satisfy all requirements of NUREG-0737, Item II.D.1 (Reference 80).

The SRV Low-Low Set system design basis is derived from two analyses as follows. A General Electric analysis (Reference 17) has shown that under the worst case reactor vessel transient only two SRVs are required to control reactor pressure following the initial opening of all valves on a reactor isolation transient. The SRV Low-Low Set system incorporates a third valve to allow for a single failure.

A Nutech Engineers analysis entitled “NSSS Transient Analysis Interaction With SRV Discharge Line Water Reflood Height” (Reference 92) determined that a maximum of 5.75 sec is required to clear an elevated water leg from a SRV discharge line. Based on the maximum vessel repressurization rate of approximately 4.3 psi per second given in the GE analysis, the 80 psi blowdown for the Low-Low Set SRVs will ensure that subsequent mechanical actuation of non-Low-Low Set SRVs will not occur and also ensures a subsequent actuation interval of greater than 5.75 sec for low-low set SRVs. The Low-Low Set system also incorporates time delay relays to block manual re-opening of Low-Low Set SRVs for subsequent lifts.

The number of SRV cycles increases with higher power levels due to a higher steaming rate at increased decay power levels. Operation at 2004 MWt will reduce the time between actuations to about 12 seconds following a scram with MSIV isolation. The time at which equilibrium height is re-established remains less than 6 seconds after the SRV closes, which is independent of reactor power level. The SRV low-low set logic includes a minimum Technical Specification required time delay after valve closure that prevents re-opening of the valve by remote actuation. The SRV low-low set logic therefore prevents subsequent SRV actuations until after the SRV discharge line reflood level stabilizes to the equilibrium height (Reference 136). The response of the Low-Low Set System to an MSIV closure event with direct scram has been evaluated for 2004 MWt rated power conditions (Reference 138). The results of the evaluation indicate that the system response was acceptable. The time between SRV closure and re-opening is sufficiently long to prevent undesirable thrust loads on SRV discharge piping. The evaluation included a sensitivity case to confirm that the inhibit timer set at a maximum value of 12 seconds would not result in opening of the non-low-low set valves. The evaluation assumed only two of the three low-low set SRVs were operable.

4.4.3.1 Environmental Qualification

Electrical SRV equipment required for core cooling is qualified for its required operating time per the 10CFR50.49 EQ Program.

4.4.4 Inspection and Testing

The safety/relief valves were tested in accordance with the manufacturer’s quality control procedures to detect defects and prove operability prior to installation. The following final tests were witnessed by a representative of the purchaser:

a. Hydrostatic test at USAS specified test pressure with cold water with additions of dye and a wetting agent.

b. Seat leakage test at design pressure with a maximum permitted leakage of 2 cc/in. of seat diameter per hour, and

c. Steam test: Valve pressurized with saturated steam with the pressure rising to the valve set pressure; the specified response time is verified when the valve opens (capacity and blowdown not tested with steam).

The set points were adjusted, verified, and indicated on the valves by the vendor. The valves were installed as received from the factory. Proper manual and automatic actuation of the safety/relief valves was verified.

It is recognized that it is not feasible to test the safety/relief valve setpoints while the valves are in place or during normal plant operation. The valves are mounted on 6-in.-diameter 1500-lb primary service rating flanges so that they may be removed for maintenance or bench checks and reinstalled during normal plant shutdowns. Pilot assemblies are bench checked or replaced with spare bench-checked pilot assemblies in accordance with the Monticello ASME Section XI, Inservice Inspection and Testing Program during each refueling outage (See Section 13.4.6). In addition, at least two safety/relief valve main bodies and associated main stage internals are normally replaced with refurbished spares during each refueling outage. The valve seats associated with replacement pilot assemblies and main stage assemblies are inspected during refurbishment.

Pressure switches are provided on the low pressure side of the machined pressure sensing bellows to detect possible leakage of the bellows on the safety/relief valves.

Steps taken to ensure that there is sufficient safety/relief valve safety-related pneumatic capacity include the following:

a. Leak testing of the safety-related backup pneumatic supplies from their source to respective safety/relief valves is included in the surveillance program and is performed once per refueling outage.

   Maximum allowable leakage rates for each of the four safety-related pneumatic supplies have been determined based on the most restrictive regulatory or design requirements. Acceptance criteria for leak rate tests have been established as an appropriate fraction of the maximum allowable leakage. Thus, margin is provided for test inaccuracies and for possible increases in leakage resulting from the effects of a harsh environment and/or seismic event.

b. Two of the backup supplies consist of an accumulator and check valve arrangement. These check valves have been changed to a soft-seated design. A preventative maintenance procedure has been implemented to periodically inspect these valve internals and replace software.

c. The bottled nitrogen supply racks used for the Alternate Nitrogen System are manually checked for adequate supply and pressure during plant operation at a frequency to assure minimum design capacity requirements of the system will be met, when required, assuming worst case leakage rates.

Table 4.4-1 Valve Set Point and Capacity

Safety/Relief Valves Self-Actuated Set Pressure (As-Left All 8 valves)	1109 psig +/- 11 psig
Safety/Relief Valves Self-Actuated Set Pressure (As-Found All 8 valves)	1109 psig +/- 33.2 psig
Safety/Relief Valve (SRV) Set Pressure Range	1025 psig to 1155 psig
Relief Capacity at Set Pressure Range for each 4.940" diameter bore SRVs	761 klbm/hr to 854 klbm/hr
Relief Capacity at Set Pressure Range for each 5.030" diameter bore SRVs	789 klbm/hr to 887 klbm/hr

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4.5 Reactor Coolant System Vents

The reactor vessel at Monticello can be vented near the top via the safety/relief valves, the HPCI steam supply line, the RCIC steam supply line. The safety/relief valves vent from all four of the main steam lines. HPCI vents from the "B" main steam line and RCIC vents from the "C" main steam line. The main steam lines are located 162.5 in. below the peak of the reactor vessel head. The Monticello plant fully satisfies the requirements of NUREG-0737, Item II.B.1 (Reference 80), regarding venting of non-condensible gases from the reactor coolant system. Venting the non-condensible gases in the reactor coolant system will ensure core cooling during natural circulation. At the same time the core cooling systems are used, the reactor coolant system will be vented. The procedures and technical specifications in effect provide for reactor coolant system venting.

There are eight safety/relief valves and each has the capability to discharge 821,000 lb of steam per hour at 1120 psig. This is considered a more than adequate venting capability. HPCI can vent between 53,000 and 112,000 lb of steam per hour (nominal design values based on GE/Terry Turbine data). RCIC can vent between 6,000 and 16,500 lb of steam per hour.

The safety/relief valves, the HPCI steam supply, and the RCIC steam supply are all larger than the definition of break size for a small LOCA. All three discharge to the suppression pool. Inadvertent actuation is a design-basis event and a demonstrated controllable transient.

There is an indication of valve position in the control room for each valve in the HPCI steam supply line and each valve in the RCIC steam supply line. For the safety/relief valves, there are thermocouples installed in each of the discharge pipes which give an indication of a safety/relief valve being open or leaking. There is a multiple channel recorder in the Control Room which receives output from the thermocouples and they are alarmed in the Control Room. There are also indicating lights, which indicate the state of the air actuator solenoid for each safety/relief valve, in the control room. Safety/relief valve position is also indicated by a differential pressure transmitter and analog trip unit monitoring the steam pressure in each discharge pipe. When pressure is sensed, the trip unit will indicate the valve is open by lighting an amber light and alarming in the Control Room.

The safety/relief valves, the HPCI steam supply isolation valves, the HPCI steam supply to turbine valve, the RCIC steam supply isolation valves and the RCIC steam supply to turbine valve each have a manual control switch in the Control Room. Starting the HPCI turbine auxiliary oil pump will open the HPCI turbine stop valve and allow the turbine speed governing valve to open when the steam admission valve is opened. The RCIC turbine governing valve and trip throttle valve are normally in the open position. The safety/relief valves and associated piping are seismically qualified, Class I. The HPCI steam supply and discharge lines and associated valves are seismically qualified, Class I. The RCIC steam supply and discharge lines and associated valves are seismically qualified; parts, Class I and the rest, Class II.

All of the reactor vent systems are safety grade per the requirements accepted when the plant was licensed.

The safety/relief valves, the HPCI system and the RCIC system all vent to the containment suppression pool, where discharged steam is condensed without causing a rapid containment pressure/temperature transient.

All the valves associated with the venting functions of the safety/relief valves, the HPCI steam supply line, and the RCIC steam supply line, with the exception of the skid mounted valves associated with HPCI and RCIC, are tested in accordance with the Monticello ASME Section XI Inservice Inspection and Testing Program (See Section 13.4.6).

Additional background information on the subject of reactor coolant system vents is given in References 5 through 11.

4.6 Hydrogen Water Chemistry

4.6.1 Design Basis

The presence of oxygen generated by radiolytic decomposition of water produces an environment favoring intergranular stress corrosion cracking (IGSCC) of the components exposed to the coolant. This mode of degradation can be controlled by suppressing the dissolved oxygen concentration with hydrogen injection and by maintaining high purity reactor coolant water. This process is called hydrogen water chemistry (HWC). The HWC system was installed in accordance with the recommendations of the BWR Owners Group, “Guidelines for Permanent BWR Hydrogen Water Chemistry Installation - 1987 Revision” (Reference 30). The NRC accepted these guidelines by letter dated July 13, 1987 and issued Safety Evaluation Reports on the Monticello design on January 7, 1988 and February 13, 1989 (References 31, 32 and 33).

The hydrogen water chemistry system is not safety related. Equipment and components are not class 1E or environmentally qualified. The hydrogen and oxygen piping are designed and installed in accordance with ANSI B 31.1 (1977 Edition with all addenda through Winter 1978 Addendum) (Reference 94). Where this piping is routed in the proximity of safety related equipment, the piping is supported in accordance with Class I seismic requirements, and hangers are classified as II over I.

The separation distance from the hydrogen storage tank to the Turbine and Reactor Buildings is more than 1,200 ft. A worst case hypothetical detonation of this tank will not endanger safety related structures and equipment.

The hydrogen and oxygen storage tanks are designed to remain at their original location during a design basis tornado or earthquake so that any liquid spill will originate from that location. The tanks are at an elevation higher than the 100 year flood; therefore, flood induced loads were not considered. The separation distances between bulk hydrogen and oxygen as described in NFPA 50 and NFPA 50B are maintained (References 95 and 96).

4.6.2 Description

The Hydrogen Water Chemistry System is divided into hydrogen injection, oxygen injection, instrumentation and control, hydrogen supply, and oxygen supply subsystems.

4.6.2.1 Hydrogen Injection Subsystem

The hydrogen injection system delivers hydrogen gas from the storage facility and injects it into the reactor coolant system at the feedwater pump piping. Hydrogen injection at this point does not degrade pump reliability or performance. Piping attached to the pumps is equipped with double isolation valves.

Design temperature and pressure is -40°F and 600 psig, respectively with nominal operating conditions of ambient temperature and 500 psig.

An excess flow device is installed between the hydrogen supply station and the turbine building to limit hydrogen flow in case of a pipe break. Individual pump injection lines contain double check valves to prevent feedwater from entering the hydrogen line. Automatic isolating flow control valves are provided in each injection line to prevent hydrogen injection into an inactive pump. Purge connections are provided near the feedwater pumps with a key locked handswitch to allow purging nitrogen through the flow control valves when maintenance is required and before hydrogen is introduced into the line.

4.6.2.2 Oxygen Injection Subsystem

The Oxygen injection subsystem delivers oxygen from the storage facility and injects it into the common off-gas line just before the recombiner to ensure that all excess hydrogen in the off-gas stream is recombined. It includes all necessary flow control and flow measurement equipment to maintain oxygen flow sufficient to recombine with all the hydrogen in the off-gas stream. The oxygen flow rate is about half the hydrogen flow rate.

Provision has also been made to inject oxygen into the condensate pumps to maintain acceptable dissolved oxygen levels in the condensate and feedwater systems for control of corrosion.

The design temperature and pressure is -40°F and 100 psig respectively with nominal operating conditions of -40° to 104°F and 35-75 psig.

The oxygen addition subsystem includes three standard oxygen bottles which act as an emergency backup supply in the event that the main supply is interrupted. This quantity of oxygen is sufficient to recombine the hydrogen remaining in the system following a hydrogen system isolation. This emergency supply system is equipped with a relief valve in the recombiner building which discharges to the atmosphere above the recombiner building roof.

4.6.2.3 Instrumentation and Control

Injection of hydrogen and oxygen is controlled from a control panel located in the northeast corner of the turbine building on the 931 ft elevation. The amount of hydrogen injected may be automatically controlled by plant load (main steam flow). It is also controllable by a manual control switch. The hydrogen injection system is automatically shutdown when any of the following conditions exist:

- high hydrogen flow
- low main steam flow
- area monitor high hydrogen concentration
- low oxygen pressure
- automatic termination of recombiner off-gas flow
- low oxygen concentration on off-gas
- manual trip from control room
- manual trip from remote control panel
- Reactor SCRAM

Hydrogen injection inhibits include feedwater pump not running. This will prevent injection into a nonrunning pump.

The oxygen injection system has redundant control systems. It is able to operate in two control modes: feedforward and cascade. In the feedforward mode oxygen flow is kept proportional to hydrogen flow. In the cascade mode, oxygen flow is controlled by the oxygen concentration at the recombiner outlet.

4.6.2.4 Hydrogen Supply

Liquid hydrogen is stored as a cryogenic liquid at approximately -420°F and 100 psig, in a vacuum insulated 9000 gal tank. Two redundant cryogenic pumps connected to the storage tank automatically withdraw liquid hydrogen and convert it to gas by passing it through a cryogenic heat exchanger. The gaseous hydrogen is stored in high-pressure receivers that provide surge volume and minimize pump start-stop cycles. Excess hydrogen gas is vented to the local vent stack. Safety considerations for the tank are satisfied by dual full flow safety valves and emergency backup rupture discs.

The liquid hydrogen storage tank is located east of the east cooling tower which meets the minimum separation distance between Class I structures and the liquid hydrogen tank as specified in Appendix B to EPRI Report NP-4500-SR-LD, "Guidelines for Permanent BWR Hydrogen Water Installation - 1987 Revision" (Reference 30).

[redacted]

Hydrogen delivery trucks momentarily pass at shorter distances from the reactor building using the site access road to the storage facility. The closest approach to the reactor building is 450 ft. It is estimated that hydrogen delivery truck will spend less than two hours per year on the plant access road, therefore, this is not a significant risk.

4.6.2.5 Oxygen Supply

Liquid oxygen is stored in a vacuum-jacketed vessel at pressures up to 250 psig and temperatures up to -250°F (saturated). Liquid oxygen is vaporized by ambient air vaporizers and routed through the pressure control station.

The liquid oxygen tank is in the same general area as the hydrogen tank, approximately 1100 ft from the nearest air intake to a safety related building. Figure 4.8 in Reference 31 allows a 9000 gal liquid oxygen tank (the largest that will be used) to be approximately 1060 ft from an air intake.

4.6.3 Performance Analysis

During normal plant operation with the hydrogen water chemistry system in operation, hydrogen injection rate is manually adjusted. Oxygen flow control is normally in the cascade mode to optimize consumption of oxygen gas.

If maintenance is required on in-line equipment after system shutdown, purging of the hydrogen or oxygen line by nitrogen may be required.

Hydrogen and oxygen injection is terminated automatically when one or more abnormal conditions occur. Manual reset of the automatic shutdown is required before the system can be reset.

Injecting hydrogen into the reactor coolant increases the fraction of volatile radioactive N-16, which is carried over in the steam. This results in increased radioactivity in certain areas of the plant. Surveys have been made and protective measures taken to ensure that exposure to plant workers remains ALARA.

4.6.4 Inspection and Testing

An Electrochemical Potential (ECP) system is used to provide evidence that hydrogen injection has reduced reactor materials ECP to levels consistent with IGSCC mitigation. The ECP System consists of a data acquisition system which processes input received from the following sources:

1. ECP probes installed in a flange on the reactor bottom head drain line.

2. A thermocouple sensing reactor bottom head drain line temperature.

3. Reactor Water Cleanup System inlet chemistry variables.

The ECP probes typically have a short life. Current operation does not require continuous ECP monitoring. However, if significant changes to reactor water chemistry are anticipated, installation of new ECP probes would be considered to determine the need for potential changes to the hydrogen injection rate.

4.7 On-Line Noble Chemistry (OLNC)

4.7.1 Design Basis

The OLNC injection system function is to inject a platinum solution [$Na2Pt(OH)6$] into the feedwater system during normal operation. The platinum compound is applied to inside reactor surfaces by employing the feedwater as the medium of transport for depositing the platinum solution. Deposition of platinum on reactor interior surfaces, including inside existing cracks, is used to significantly reduce the Electrochemical Corrosion Potential (ECP) in the presence of excess hydrogen. Application of the platinum solution in conjunction with hydrogen injection (HWC) mitigates intergranular stress corrosion cracking of Monticello reactor pressure vessel internals and reactor piping by lowering ECP below that at which IGSCC crack initiation is mitigated and crack growth lowered.

On-line application of noble metal is preferential to application during shutdown. IGSCC crack growth rates have been observed in the industry that were suggested to be caused by flanking of a crack due to interrupted HWC injection during plant operation following application of noble metal chemistry during a shutdown. In addition, the Mitigation Monitoring System (MMS) samples reactor water to track the effect of injecting the platinum solution into the feedwater by monitoring noble metal durability and Electrochemical Corrosion Potential (ECP).

Noble metal addition allows for the use of less hydrogen in order to maintain a hydrogen to oxidant molar ratio >2 to establish low ECP to mitigate IGSCC initiation and growth in vessel internals and recirculation piping. The use of less hydrogen decreases main steam line radiation, which reduces dose rates in areas impacted by steam line radiation, which supports reduction in personnel cumulative radiation exposure.

ONLC is proprietary knowledge owned by General Electric Hitachi Nuclear Energy (GEH). GEH has provided a proprietary Technical Safety Evaluation (TSE) to support implementation of ONLC at Monticello (Reference 133). Certain effects of ONLC are evaluated within the TSE. These include; noble metal effect on components and equipment, effect of noble metal migration, effect of deposits on fuel clad surfaces, and radiological effects, among others. The TSE primarily concentrates on the potential effects of NobleChem to ensure: (i) the integrity of the reactor coolant pressure boundary; (ii) the capability to shut down the reactor and maintain it in a safe shutdown condition; and (iii) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable regulatory guide exposures.

4.7.2 Description

A prefabricated injection skid for OLNC is located on the 935'-0" elevation of the reactor building. The OLNC injection skid is portable. Tie down links are used to connect the Injection Skid to expansion bolts in the floor slab and the installation has been analyzed as Seismic 11/1. The OLNC injection system function is to inject a platinum solution [Na2Pt(OH)6] into the feedwater. The platinum compound is applied to in-reactor surfaces by employing into the feedwater as the medium of transport for depositing the platinum solution. A portion of the OLNC injection system is classified as safety related because it will be part of the reactor coolant system pressure boundary. The portion of the OLNC injection system that is classified as safety related is from the feedwater tie-in points to the double isolation valves. The OLNC injection skid is classified as a non-safety related equipment.

The OLNC Mitigation Monitoring System (MMS) consists of three panels, the MMS Durabiliy Monitoring (DM) panel, the MMS Data Acquisition System (DAS) panel, and the MMS Automatic Flow Control Module (AFCM) panel. MMS panels are permanent installed on the 962'-6" elevation of the reactor building. The OLNC MMS function is to sample reactor water to track the effect of injecting the platinum solution into the feedwater by monitoring noble metal durability and Electrochemical Corrosion Potential (ECP). The system consists of a durability monitor section, an ECP manifold, isolation valves, flow meters, temperature and pressure instrumentation, and a junction box from which signals are routed to the Data Acquisition System (DAS). The MMS DM panel contains a once-through flow path with the outlet routed back to the Reactor Water Clean-up system piping. All the MMS panels are designed as non-seismic and non-safety related equipment. The piping/tubing is insulated and the distance from plant to the MMS DM minimized to keep the temperature difference between the reactor and the MMS water to less than 5°F.

4.8 Zinc Water Chemistry (GEZIP)

4.8.1 Design Basis

Based on experience at “natural” zinc plants, i.e., plants that contain 5-15 ppb zinc as a result of admiralty brass condensers, a strong correlation exists between the concentration of zinc in reactor water and recirc pipe dose rate buildup. Studies performed by General Electric show that small concentrations of zinc in the reactor water will result in a reduction in the amount of cobalt-incorporated into the oxide film established on stainless steel piping. This reduction in cobalt-60 incorporation will provide substantial reductions in dose rates, particularly in primary containment.

4.8.2 Description

A prefabricated skid is located at the 911 ft level of the turbine building. The passive GEZIP System is designed to continuously inject a dilute water solution of ionic zinc into the reactor feedwater. A stream of water taken downstream from the feedwater pump discharge is routed through a dissolution vessel containing depleted zinc oxide (DZO) pellets. The sintered DZO pellets dissolve in the diverted feedwater stream providing the ionic zinc. The stream containing the dissolved DZO is returned upstream of the feedwater pump suction and is mixed with the main feedwater flow.

The reactor water zinc concentration is periodically measured using normal plant chemistry sampling procedures. The amount of zinc leaving the dissolution vessel and entering the feedwater stream (the injection rate) is adjusted to maintain the zinc concentration in the reactor water at the desired level.

The injection rate of the zinc is adjusted by controlling the rate of water flow through the dissolution vessel by manually positioning the main flow control valve. The dissolution vessel is normally filled with sufficient DZO to last through one complete fuel cycle. Plants vary in the amount of DZO required, therefore additional DZO additions during a cycle may be necessary.

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113. NRC (Terry A. Beltz) letter to NMC (John T. Conway), “Monticello Nuclear Generating Plant (MNGP) - Issuance of Amendment for the Conversion to the Improved Technical Specifications with Beyond-Scope Issues (TAC Nos: MC7505, MC7597 through MC7611, and MC8887)”, dated June 05, 2007.

114. NRC (Darryl S. Hood) letter to NMC (Jeffrey S, Forbes), ”Monticello Nuclear Generating Plant- Exemption from the Requirements of 10 CFR Part 50, Section 50.60(a) and Appendix G (TAC No. MB 5078)” Dated February 21, 2003.

115. ASME B&PV Code Case N-640, ”Alternative Reference Fracture Toughness for Development of P-T Limit Curves” Dated February 26, 1999.

116. EPRI Technical Report 1020231, BWRVIP-135, Revision 2: “BWR Vessel and Internals Project Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluations”.

117. Deleted.

118. Deleted.

119. Deleted.

120. Deleted.

121. Deleted.

122. Deleted.

123. NSP (Timothy J. O'Connor) letter to the NRC, L-MT-12-002, License Amendment Request: Revise and Relocate Pressure Temperature Curves to a Pressure Temperature Limits Report, dated January 20, 2012

124. Deleted.

125. Calculation 11-039, Monticello Neutron Flux and Fluence Evaluation for Extended Power Uprate, Rev. 0

126. Calculation 11-004, Finite Element Stress Analysis of Monticello RPV Feedwater Nozzle, Rev. 0

127. Calculation 11-020, Finite Element Stress Analysis of Monticello RPV Recirculation Inlet Nozzle, Rev. 0

128. Calculation 11-003, Evaluation of Adjusted Reference Temperatures and Reference Temperature Shifts, Rev. 1

129. Calculation 11-005, Revised P-T Curves Calculation, Rev. 1

130. SIR-05-044-A, Pressure Temperature Limits Report Methodology for Boiling Water Reactors, Rev. 0, dated April 2007

131. NRC letter to NSP(Mark A. Schimmel), Monticello Nuclear Generating Plant - Issuance of Amendment to Revise and Relocate Pressure Temperature Curves to a Pressure Temperature Limits Report (TAC NO. ME7930), dated February 27, 2013

132. Monticello Nuclear Generating Plant Pressure and Temperature Limits Report (PTLR) up to 54 Effective Full-Power Years (EFPY)

133. On-Line NobleChem (OLNC) Application Technical Safety Evaluation Monticello, Rev 0 (GE Proprietary).

134. NSPM letter L-MT-08-043 (T J O'Connor) to NRC, "Monticello - Extended Power Uprate, Acceptance Review Supplemental Information Package 6", dated June 12, 2008 (ADAMS Accession No. ML081640435).

135. Monticello Engineer Change (EC) 13638, "EPU - Overall EC for EPU Implementation".

136. GE Hitachi EPU Project Task Report GE-NE-0000-0060-9229-TR-R3, Revision 3, "Task T0400: Containment System Response", February 2011 (Monticello calculation number 11-173).

137. 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.

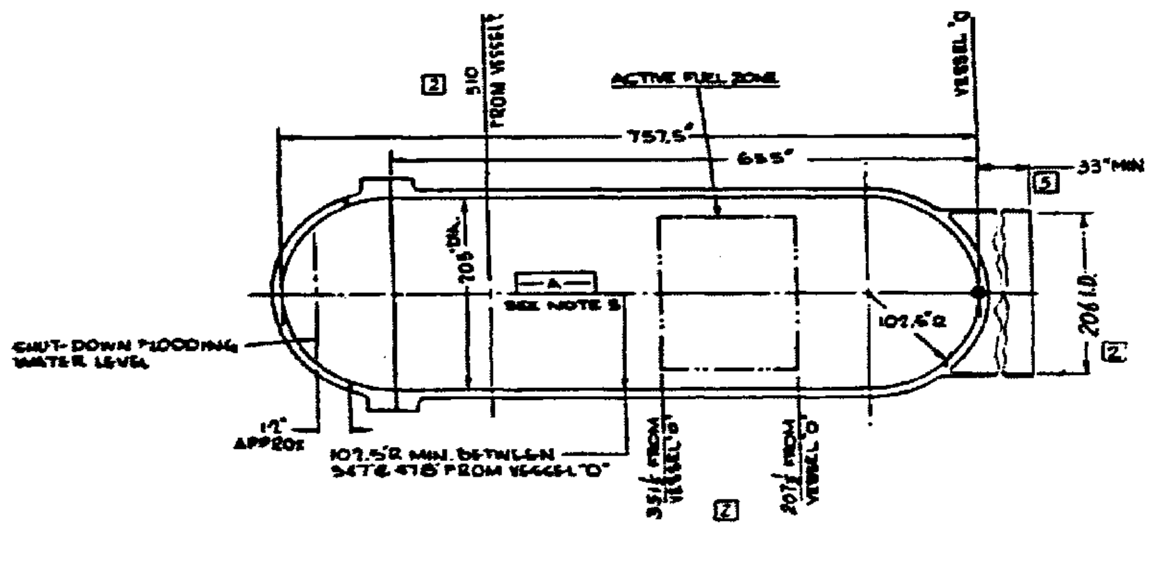
138. GE Hitachi EPU Project Task Report GE-NE-0000-0062-2954-TR-R0, Revision 0, "Task Report T0900: Transient Analysis", December 2007 (Monticello calculation 11-250).

139. GE Hitachi EPU Project Task Report GE-NE-0000-0060-4280-TR-R0, Revision 0, "Task T0307: Recirculation System BILBO Code Output to Task T0307", May 2007 (Monticello Calculation 11-182).

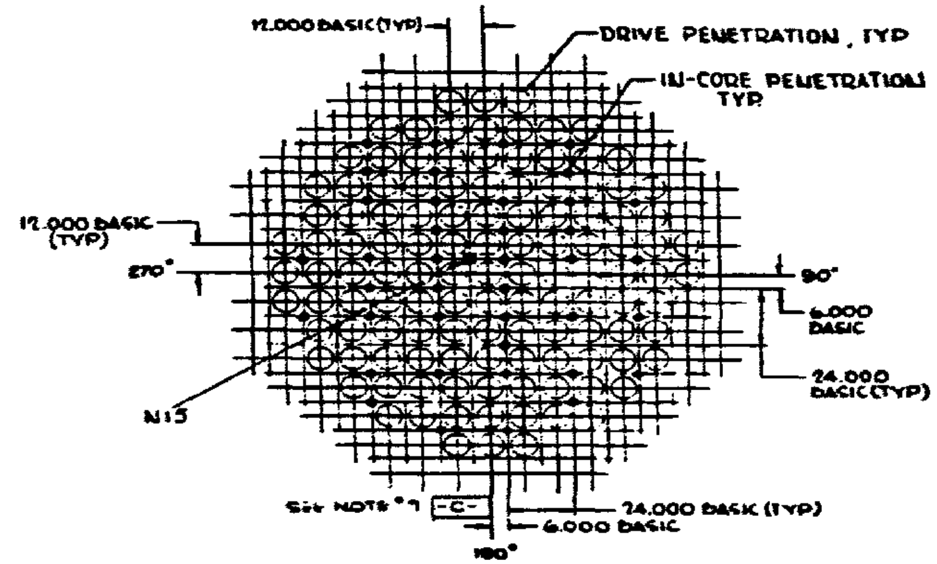
140. GE Hitachi EPU Project Task Report GE-NE-0000-0073-2176-TR-R0, Revision 0, "Task Report T0903: Station Blackout", March 2008 (Monticello calculation number 11-184).

FIGURES

Figure 4.2-1 Reactor Vessel Outline and Nozzles



LONGITUDINAL SECTION



BOTTOM HEAD PLAN VIEW [2]

REFERENCE INDEX	
CONTROL ROD DRIVE PENETRATION	
IN-CORE PENETRATION	
SHROUD SUPPORT	
VESSEL SHELL FLANGE	
STABILIZER BRACKET	
NOZZLES	
BRACKETS	
THERMOCOUPLE PADS	
VESSEL INSULATION SUPPORTS	
VESSEL LOADINGS	
SEISMIC LOADINGS	

NOZZLE SCHEDULE [2]

NOZZLE NO.	QUAN.	SIZE	FUNCTION	AZIMUTH ± 0°-5'	DISTANCE FROM VESSEL 0 ± 1/4	END PREP.	MATING PIPE O.D.	MATING PIPE SCH OR WALL	MATING PIPE MATERIAL	REMARKS	NOZZLE SAFE END O.D.	NOZZLE SAFE END I.D.
N1A, N1B	2	28"×26"	RECIRCULATION OUTLET	A-0°, B-180°	150 [1]	107C5505	28.000	1.084	SA 240-TP304	[2]	28.400	25.930
N2A THRU N2K	10	12"	RECIRCULATION INLET	A-30°, B-60°, C-90°, D-120°, E-150°, F-210°, G-240°, H-270°, J-300°, K-330°	186"	↓	12.750	SCH.80	ASTM A358, CL-1, GR 316	[2] NUCLEAR GRADE	12.840	11.505
N3A, N3B, N3C, N3D	4	18"	STEAM OUTLET	A-72°, B-108°, C-252°, D-288°	595"		18.000	SCH.80	SA 333-GR6	———	18.100	16.318
N4A, N4B, N4C, N4D	4	10"	FEEDWATER	A-45°, B-135°, C-225°, D-315°	460"	↓	10.750	SCH.120	SA 106-GRB	[2]	10.840	9.669
N5A, N5B	2	8"	CORE SPRAY	A-90°, B-270°	481"	107C5505	8.625	SCH 80	SA 312-TP304		5.690	7.709
N6A, N6B	2	6 [2]	HEAD SPRAY & INSTRUMENTATION	A-0°, B-180° ON TOP HEAD AT 30°	—		—	—	—	6"-1500 LB FLG.		
N7	1	4"	VENT	TOP DEAD CENTER.	—		—	—	—	4"-1500 LB FLG.		
N8A, N8B	2	4"	JET PUMP INSTRUMENTATION	A-60°, B-240°	129 [1]	107C5505	4.500	SCH.80	SA 312-TP304	———	4.560	4.045
N9	1	3"	SPARE	65°	459"	↓	3.500	SCH.80	SA 312-TP304	NOZZLE CAPPED	3.530	2.934
N10	1	2"	CORE DIFF. PRESS & LIQUID CONTROL	350°	79 1/2		2.375	SCH.80	SA 312-TP304			
N11A, N11B	2	2"	INSTRUMENTATION	A-40°, B-220°	547"	↓	2.375	SCH.80	SA 312-TP304	——— 2" FEMALE S.W. FITTING		
N12A, N12B	2	2"	INSTRUMENTATION	A-40°, B-220°	422	107C5505	2.375	SCH 80	SA 312-TP304	——— 2" FEMALE S.W. FITTING		
N13	1	1"	HIGH PRESSURE SEAL LEAK DET.	BETWEEN HEAD BOLTS 0° APPROX.	—	—	—	—	—	1"-3000 LB FEM. SW FITTING		
N14	1	1"	SPARE	BETWEEN HEAD BOLTS 5° APPROX.	—	—	—	—	—	SIDE CAPPED		
N15	1	2"	DRAIN	SEE BOTTOM HEAD PLAN VIEW	—	107C5505	2.375	SCH.160	SA106-GRB	[X] 2" FEMALE S.W. FITTING		
	121	6"	CONTROL ROD DRIVE	SEE BOTTOM HEAD PLAN VIEW	—	—	—	—	—			
	40	2"	FLUX MONITOR	SEE BOTTOM HEAD PLAN VIEW	—	—	—	—	—			

NOTES:

1. DESIGN PRESS.-1250 PSIG AT VESSEL "0"
2. DESIGN TEMP.-575°F
3. NORMAL OPERATING PRESS.-1000 PSIG AT 655 FROM VESSEL "0" ELEVATION.
4. NORMAL OPERATING TEMP.-SATURATION TEMP. AT 1000 PSIG.
5. VESSEL CENTERLINE [-A-] SHALL BE DEFINED AS A VERTICAL (PLUMB) LINE THRU THE CENTER OF THE CENTRAL CONTROL ROD DRIVE PENETRATION. THE GENERAL LOCATION TOLERANCE OF THE CENTERLINE OF ANY STRAKE OR FEATURE IS .016 IN/FT OF VESSEL ELEVATION EXCEPT WHERE SPECIFICALLY NOTED [4] [2]
6. FOR SUPPLEMENTARY INFORMATION SEE SPEC. 21A1112.
7. VESSEL AZIMUTH AXIAL PLANE [-C-] DETERMINED BY CENTER LINE DATUM [-A-] & 0° OUTER CONTROL ROD DRIVE BORE AT TOP OF PENETRATION. [2]

Figure 4.2-2 Core Spray Safe End

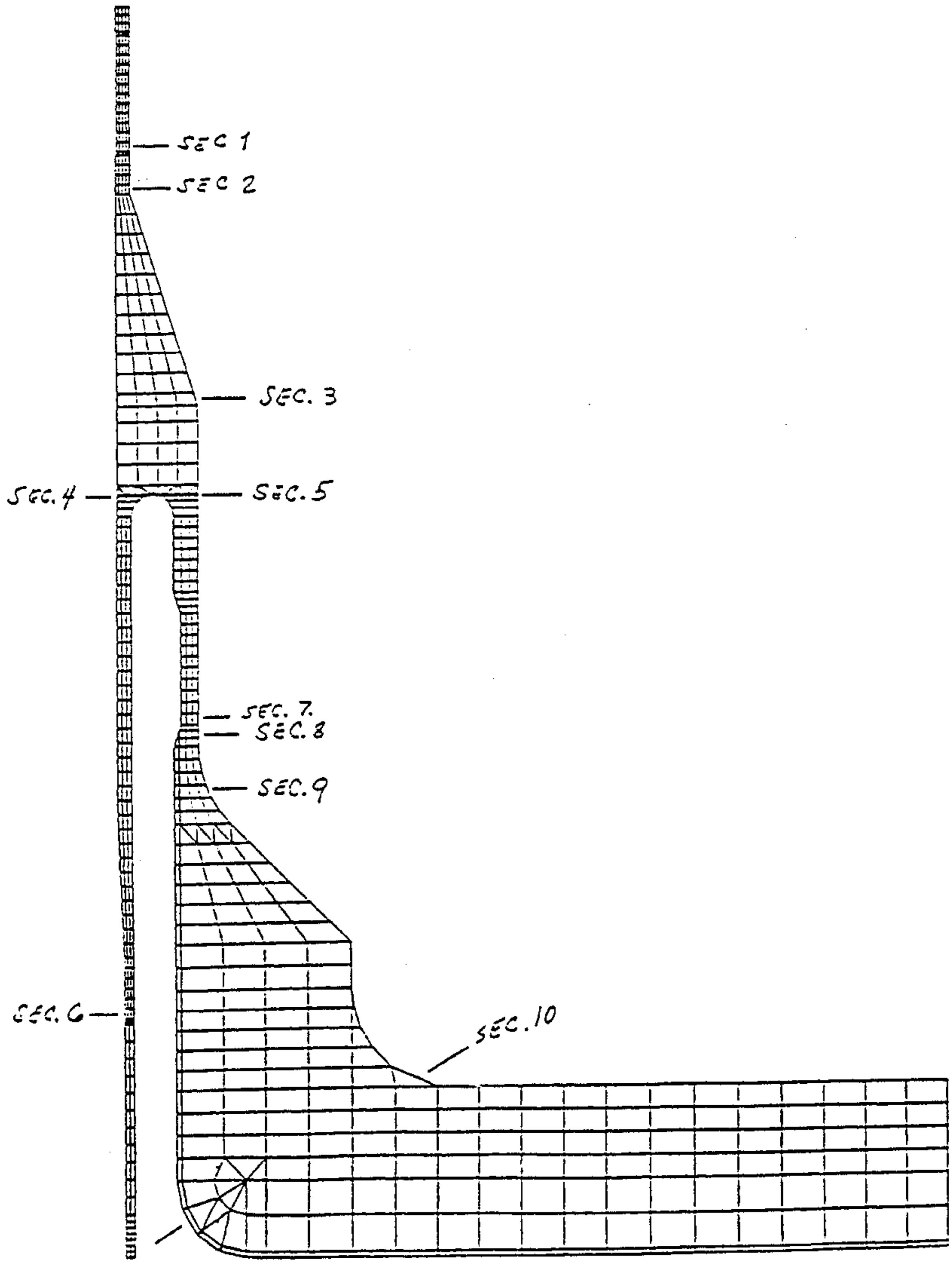


Figure 4.3-1 Reactor Coolant Recirculation System Isometric

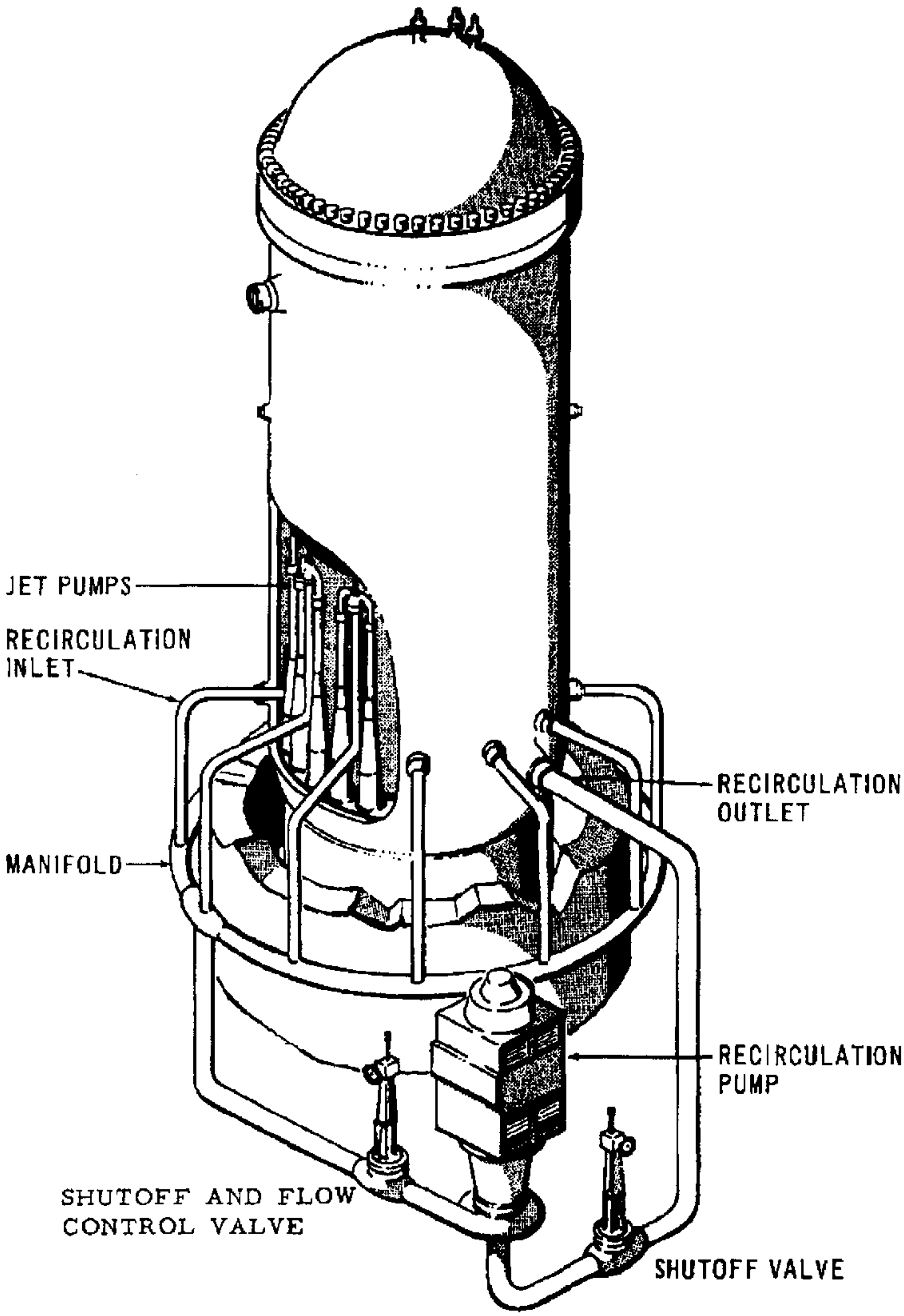


Figure 4.3-2 Jet Pump Isometric

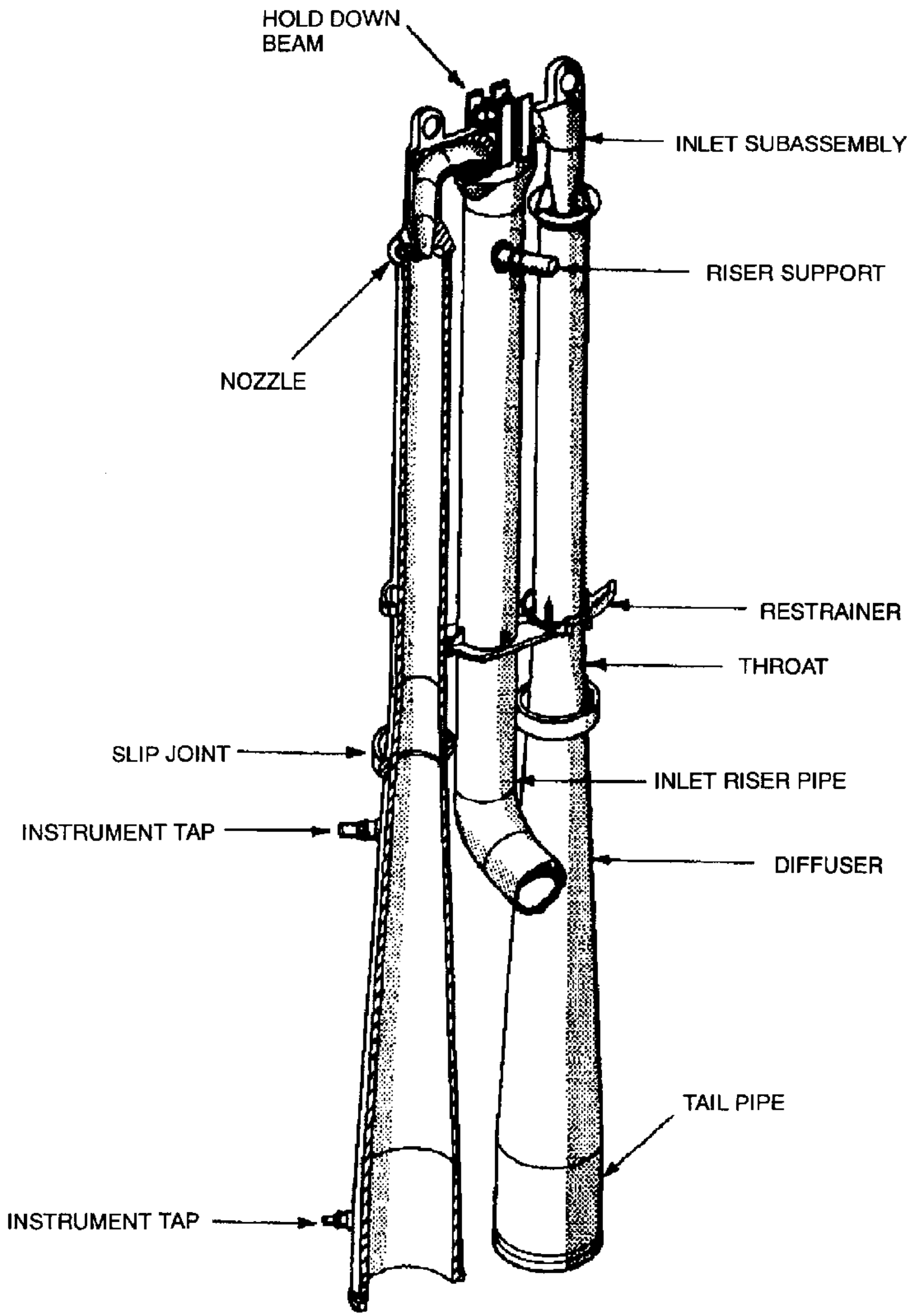


Figure 4.3-3 Jet Pump Efficiency versus Flow Rate

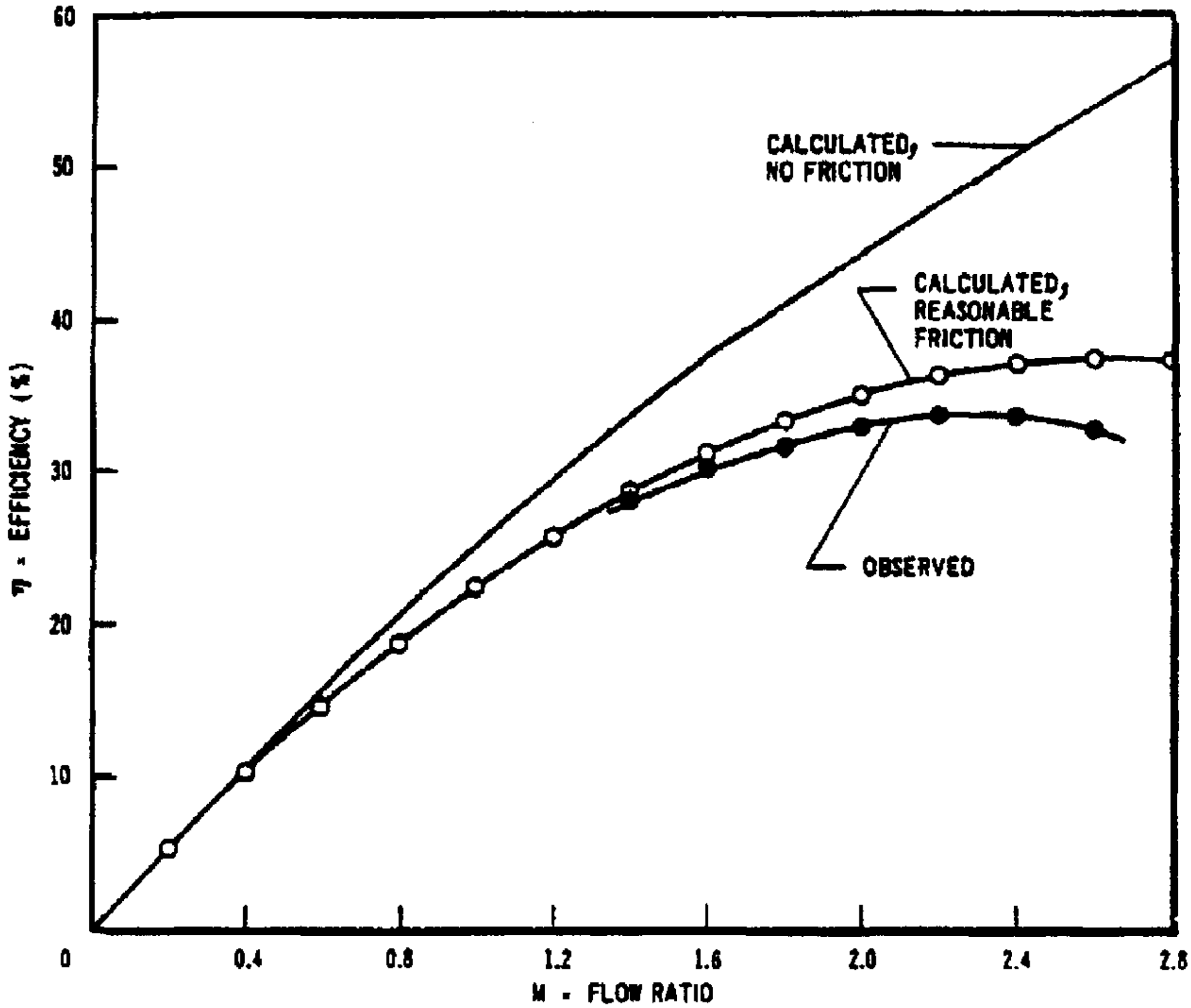


Figure 4.3-4 Jet Pump Characteristic Curve

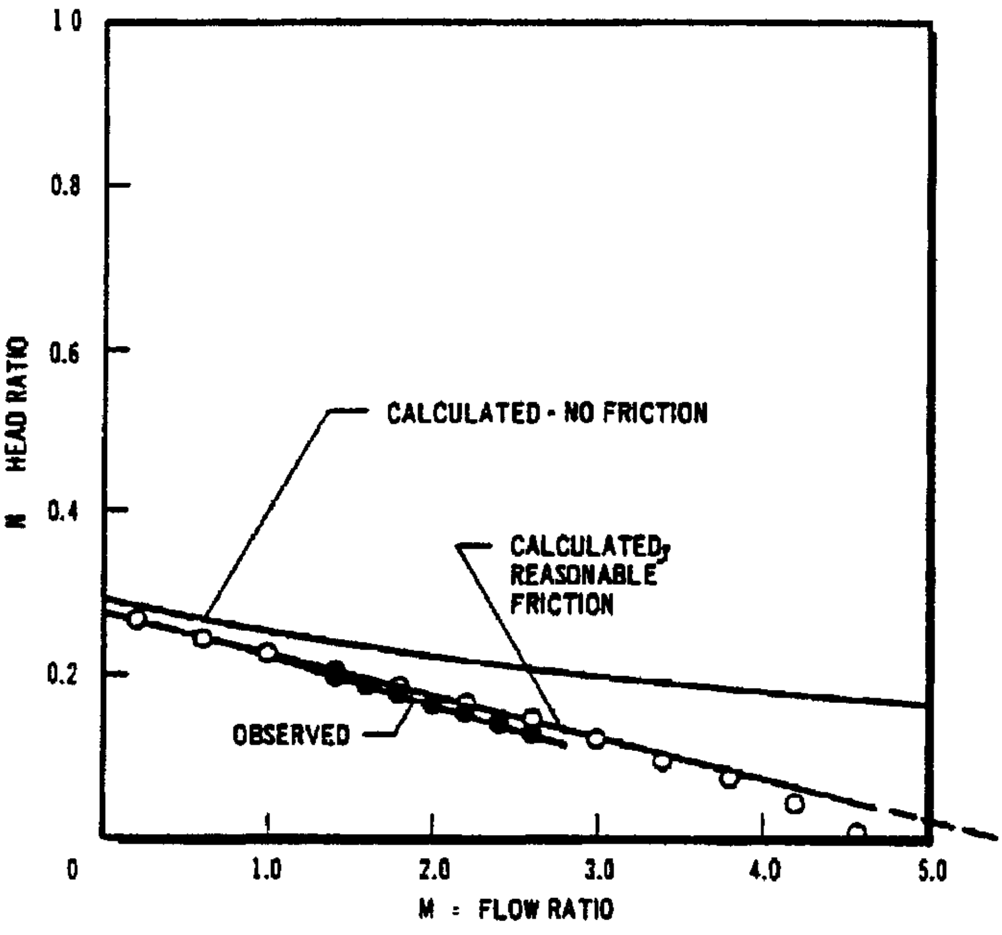


Figure 4.3-5 Jet Pump Head Ratio versus Area Ratio

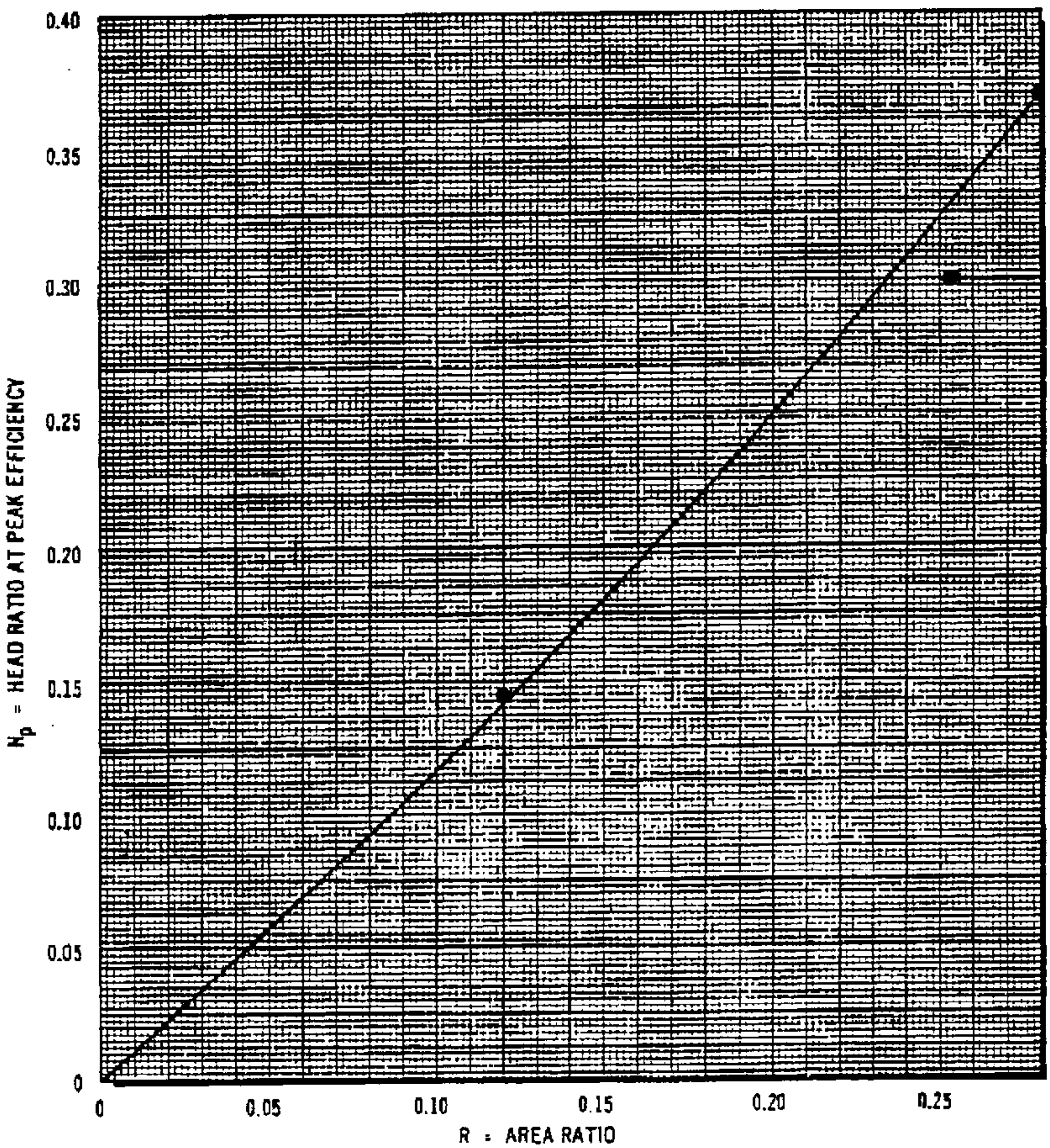


Figure 4.3-6 Jet Pump Flow Ratio versus Area Ratio

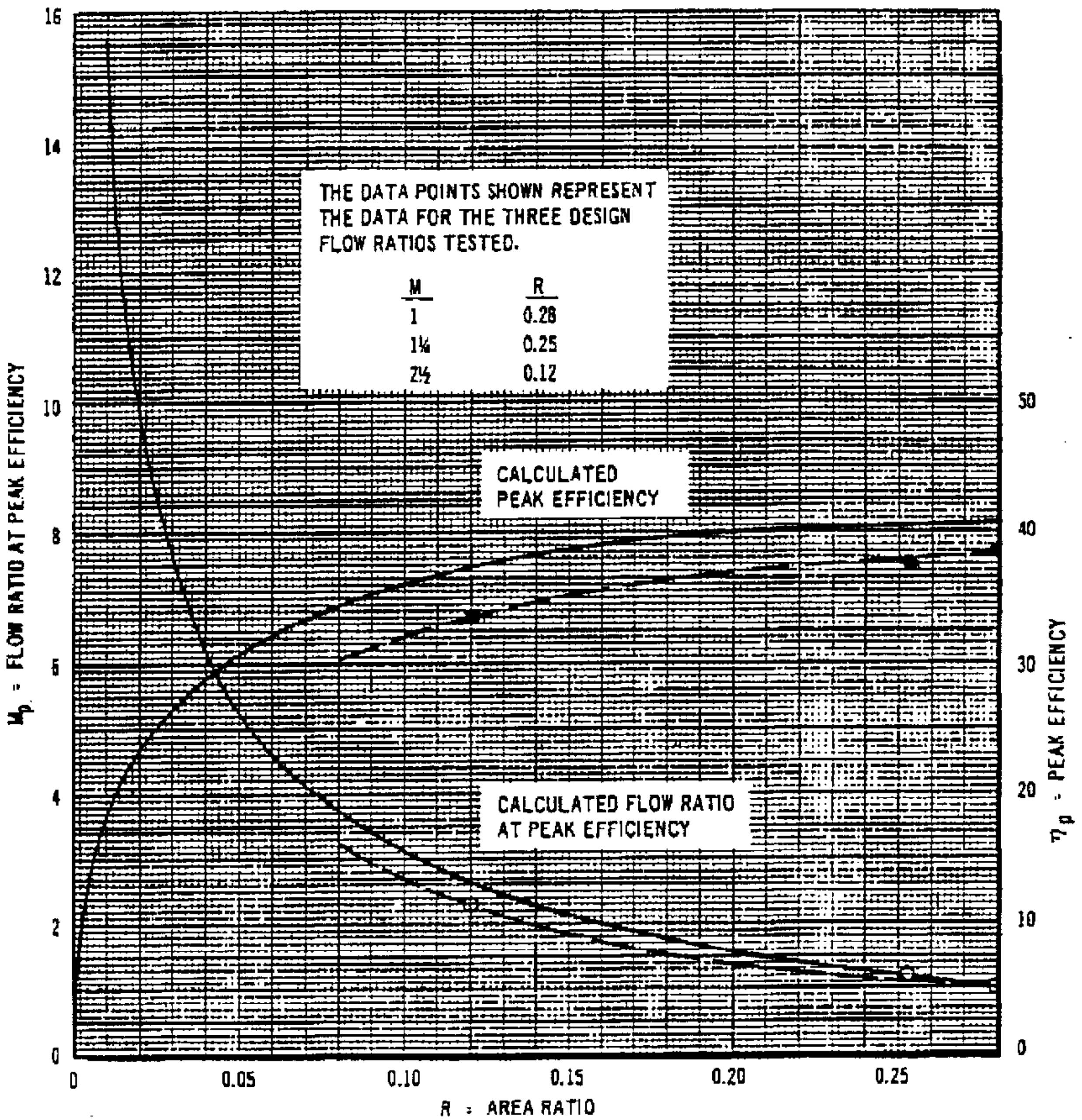


Figure 4.3-7 Typical Jet Pump Head Capacity Characteristics

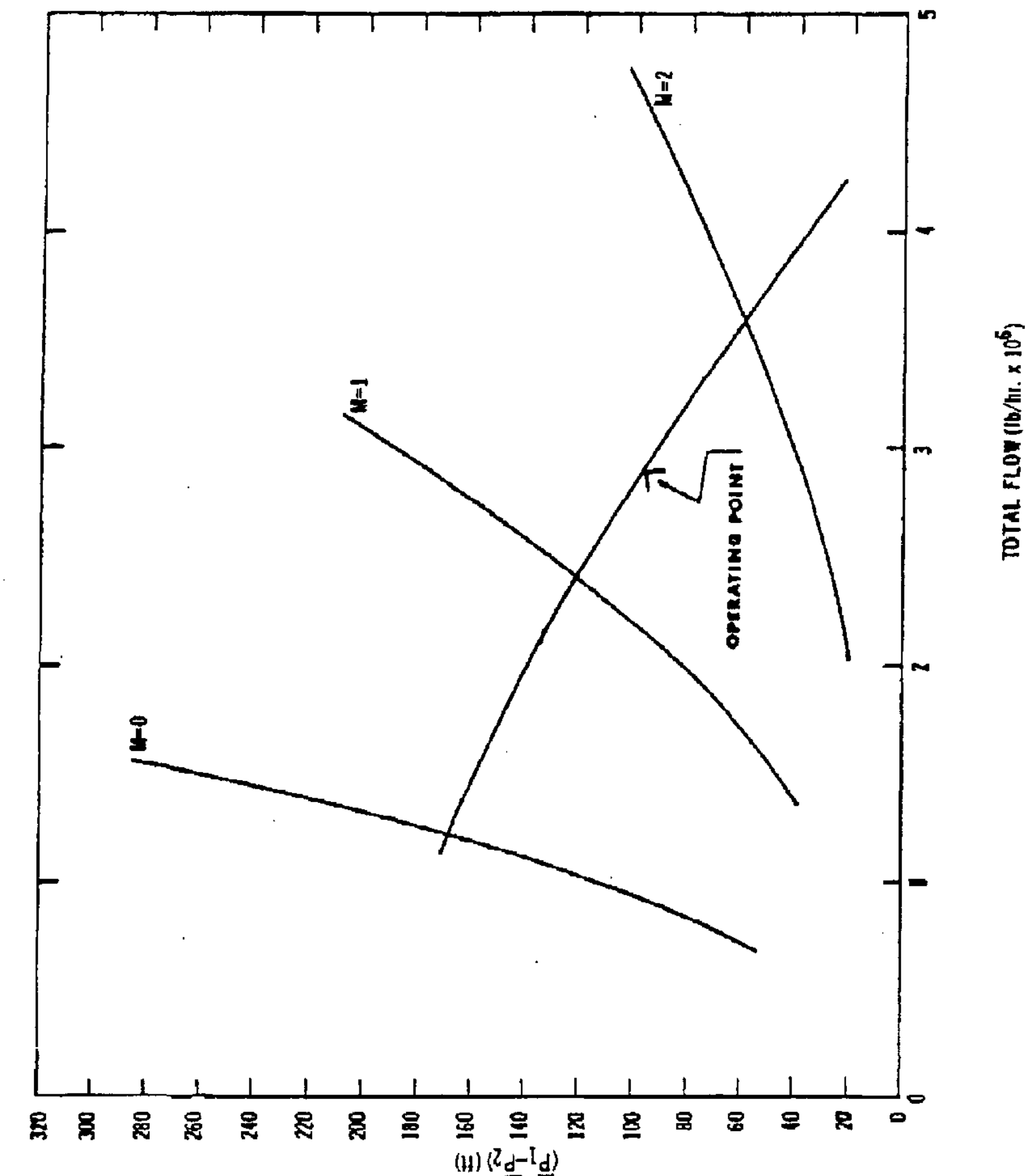


Figure 4.3-8 Core Flow Measurement System Schematic

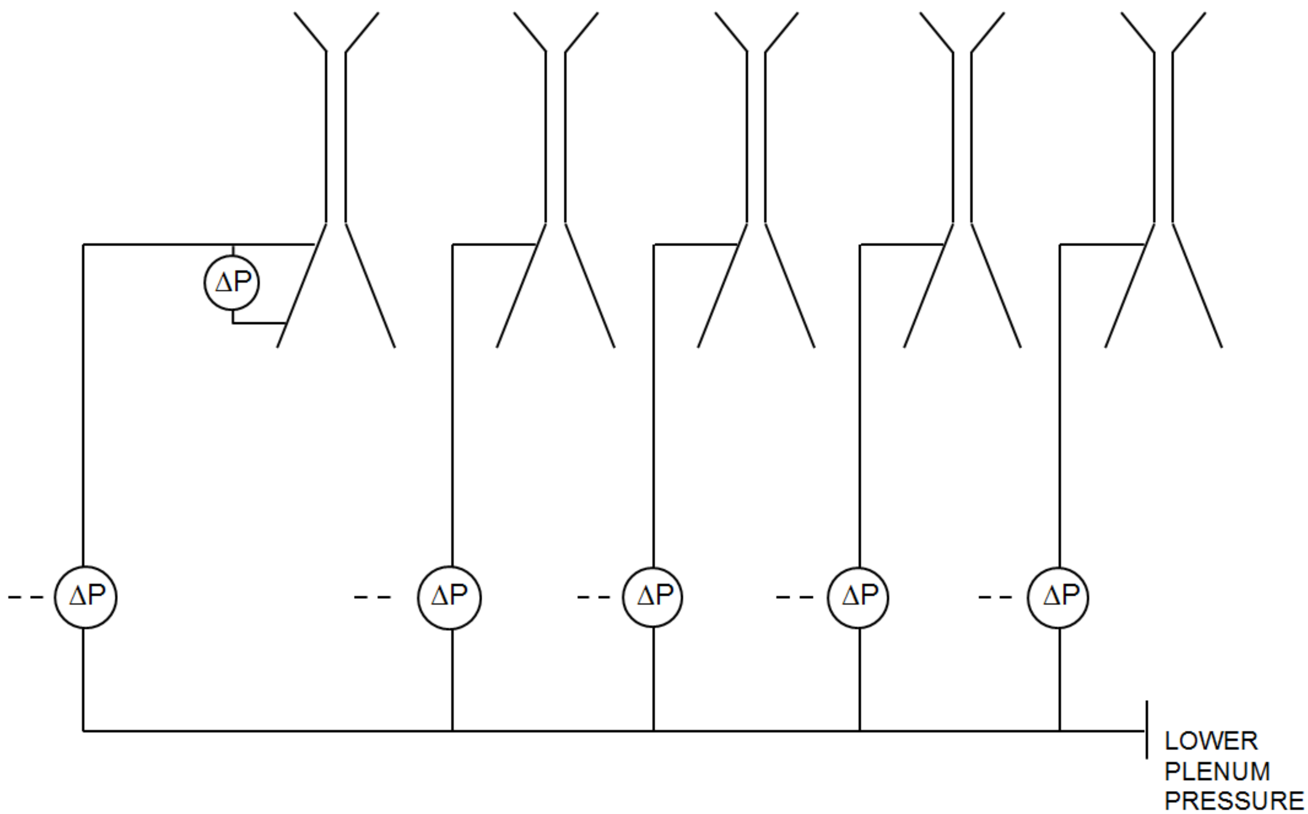


Figure 4.3-9 NPSH Available by Subcooling at Pump Inlet versus NPSH at Various Temperatures

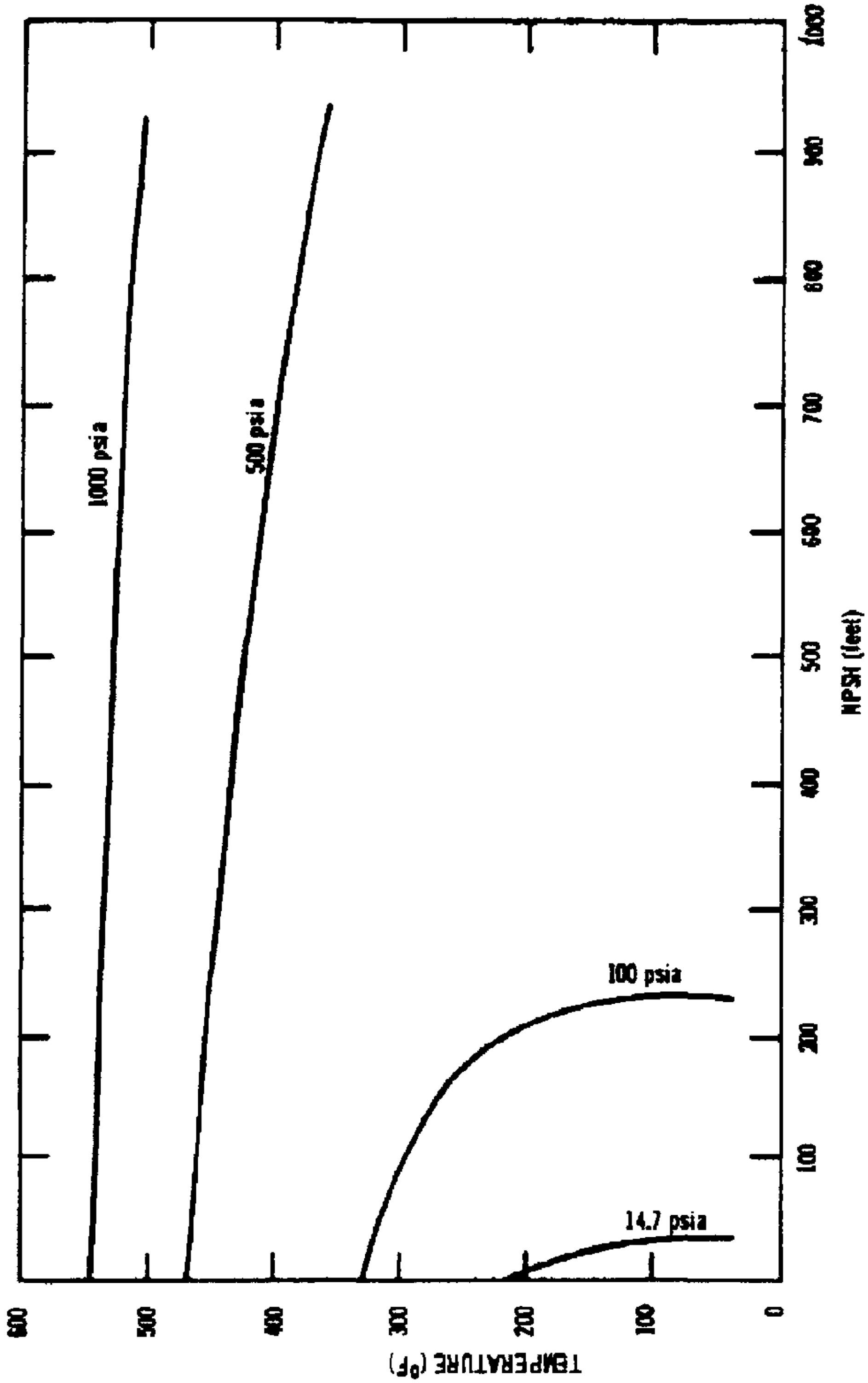


Figure 4.4-2 Dual Relief/Safety Valve - Valve Closed

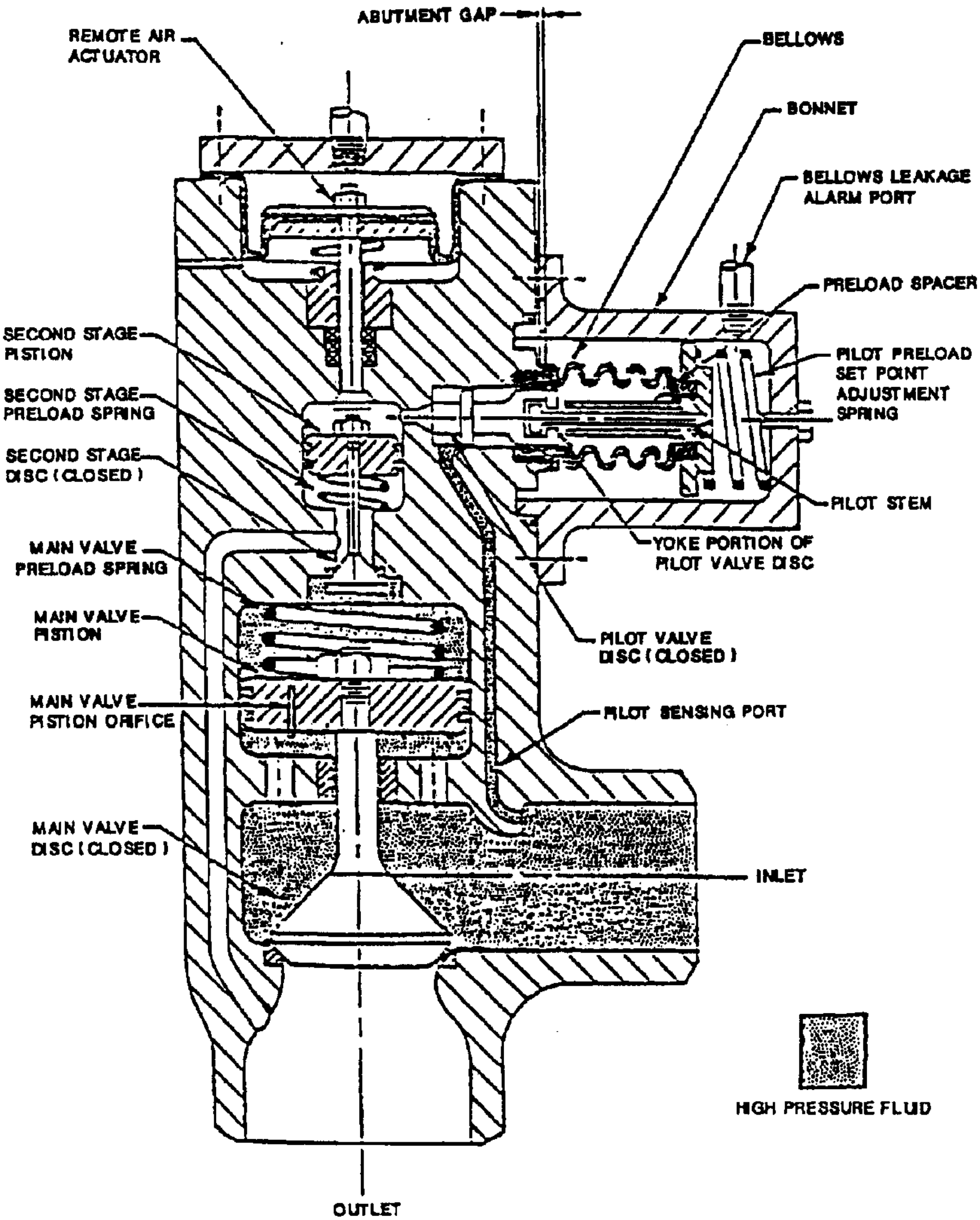


Figure 4.4-3 Dual Relief/Safety Valve - Valve Open

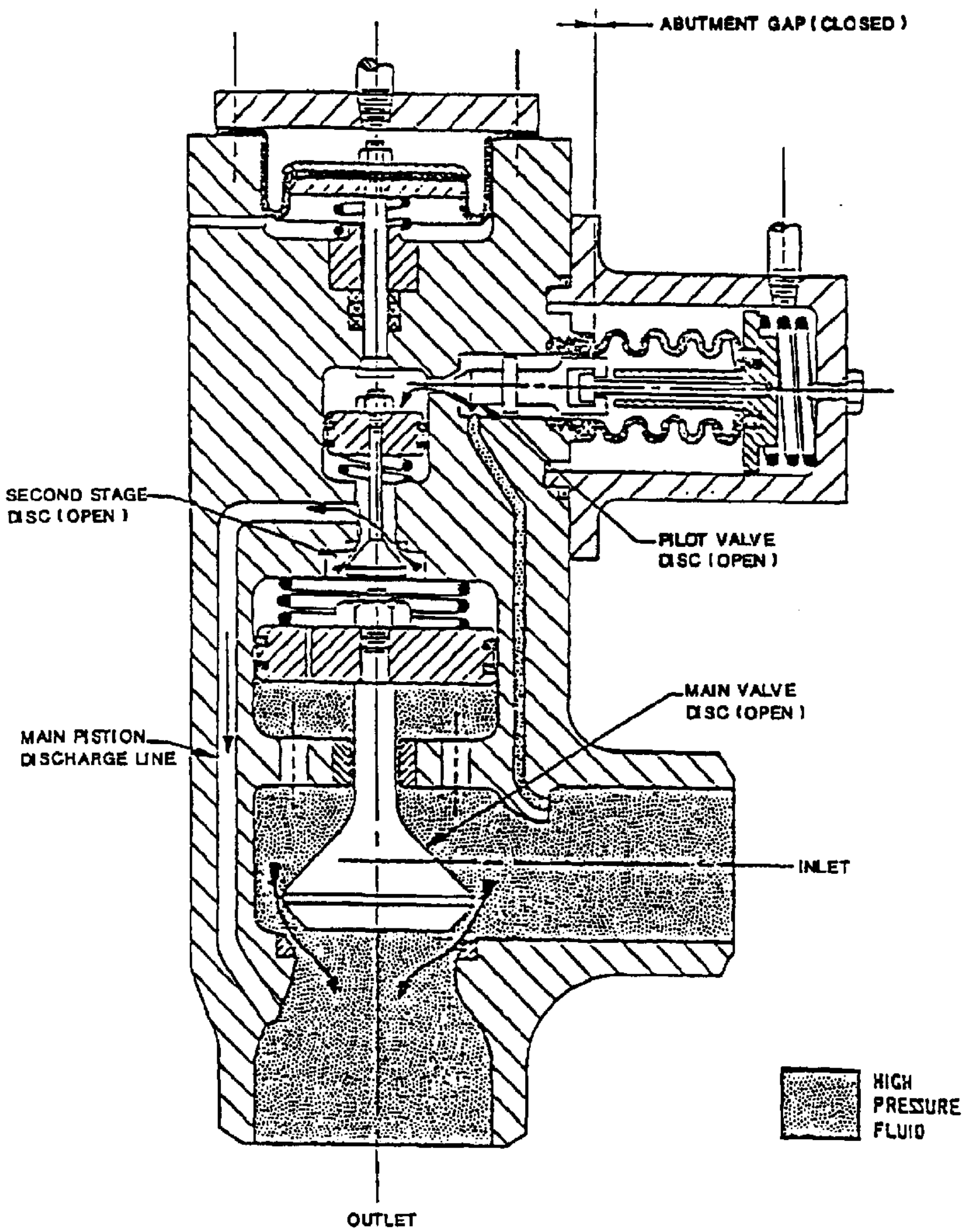
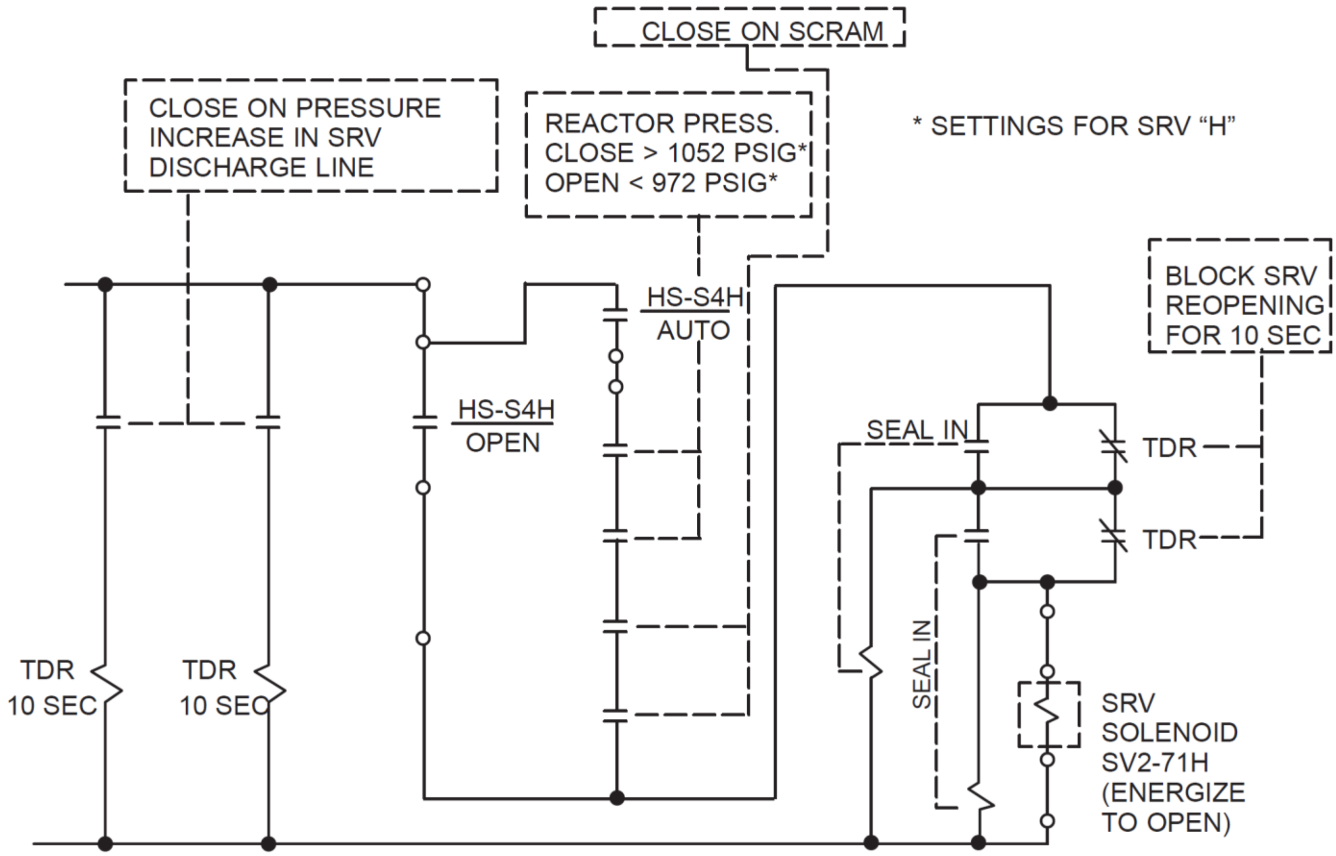
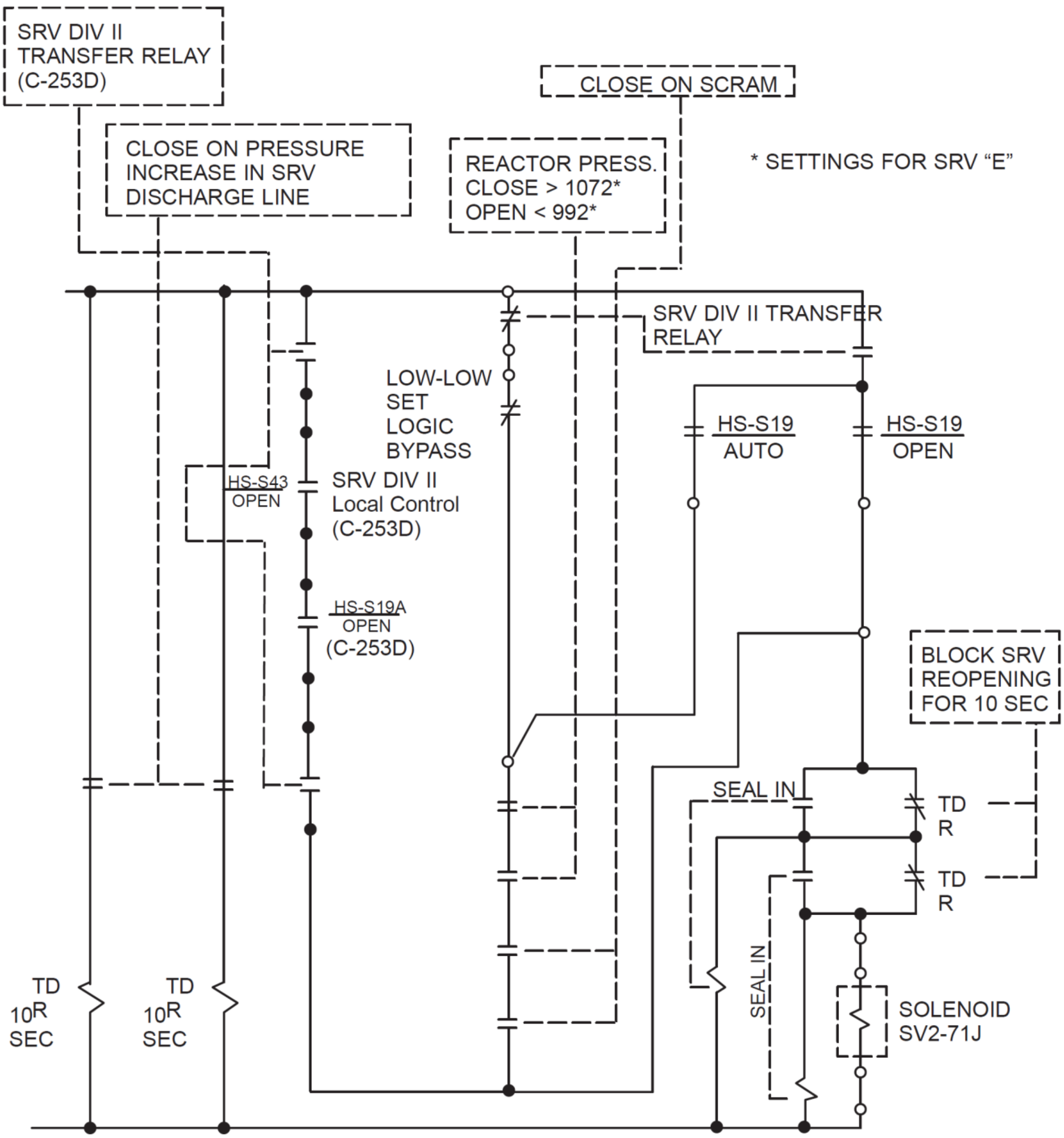


Figure 4.4-4 SRV Low-Low Set Solenoid Actuation Circuit for SRV "H"



NOTE: TYPICAL FOR DIVISION I SRVs "E", "G" & "H"

Figure 4.4-5 SRV Low-Low Set Solenoid Actuation Circuit for SRV "E"



NOTE: TYPICAL FOR DIVISION II SRVs "E", "G" & "H"