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SECTION 4 REACTOR COOLANT SYSTEM

4.1 SUMMARY DESCRIPTION

The Reactor Coolant Systems of the two nuclear power units are essentially identical, and do not share any components. The following description applies to either unit.

The Reactor Coolant System, shown in the Flow Diagrams, Figure 4.1-1A for Unit 1 and Figure 4.1-1B for Unit 2, consists of two identical heat transfer loops connected in parallel to the reactor vessel. Each loop contains a circulating pump and a steam generator. The system also includes a pressurizer, pressurizer relief tank, connecting piping, and instrumentation necessary for operational control. The pressurizer surge line is connected to one of the loops.

The containment boundary shown on the flow diagrams indicates those major components which are located inside containment. The intersection of a process line with this boundary indicates a containment penetration.

Reactor Coolant System and components design data are listed in Tables 4.1-1 through 4.1-7.

The Reactor Coolant System transfers the heat generated in the core to the steam generators where steam is generated to drive the turbine generator. Borated demineralized water is circulated at the flow rate and temperature consistent with achieving the reactor core thermal hydraulic performance presented in Section 3. The water also acts as a neutron moderator and reflector, and as a solvent for the neutron absorber used in chemical shim control.

The Reactor Coolant System provides a boundary for containing the coolant under operating temperature and pressure conditions. It serves to confine radioactive material and limits to acceptable values any release of radioactive material to the secondary system and to other parts of the plant under conditions of either normal or abnormal reactor operation. During transient operation the system's heat capacity attenuates thermal transients. The Reactor Coolant System accommodates coolant volume changes within the bounds of the protection system criteria.

By appropriate selection of the inertia of the reactor coolant pumps, the thermal hydraulic effects which result from a loss-of-flow situation are reduced to a safe level during the pump coastdown. The layout of the system assures the natural circulation capability following a loss of flow to permit plant cooldown without overheating the core.

Pressure in the system is controlled by the pressurizer, where water and steam pressure is maintained through the use of electrical heaters and sprays. Steam can either be formed by the heaters, or condensed by pressurizer spray to minimize pressure variations due to contraction and expansion of the coolant. Instrumentation used in the pressurizer pressure control system is described in Section 7. Spring-loaded steam safety valves and power-operated relief valves are connected to the pressurizer and discharge to the pressurizer relief tank, where the discharged steam is condensed and cooled by mixing with water.

Maximum Heating and Cooling Rates

The reactor system operating cycles used for design purposes are given in Table 4.1-8 and described in Section 4.1.4. The normal system heatup rate limit is 100°F/hr and the cooldown rate is 100°F/hr. These limits are discussed in the Pressure-Temperature Limits Report (PTLR). The pressurizer heatup rate will not exceed 100°F/hr and the pressurizer cooldown rate will not exceed 200°F/hr. The original capacity of the pressurizer heaters permitted a heat up rate of 55°F/hr, starting with a minimum water level. This rate takes into account the small continuous bypass spray flow provided around the pressurizer spray valves to maintain the pressurizer liquid boron concentration homogeneous with that in the reactor coolant. The capacity of the heaters may be reduced below the original design; which translates into a reduced heat up rate.

The spray is not used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F. The fastest cooldown rates which result from the hypothetical case of a break of a main steam line are addressed in Section 4.1.4.11.

In January 1990, it was determined that the pressurizer cooldown rate limit of 200°F/hr and the temperature difference limit of 320°F between the pressurizer and the spray fluid was exceeded during the cooldown for the Unit 1 refueling outage that month. Subsequent investigation revealed that the cooldown rate limit had been previously exceeded during unit cooldowns due to procedural inadequacy. Westinghouse performed an analysis to determine the effects of exceeding these limits. The analysis (Reference 91) concluded that the transients did not compromise the structural integrity of the pressurizer. Measures have been taken to ensure that the limits will not be exceeded in the future (References 92 and 97).

Materials and Design Control

Each of the materials used in the Reactor Coolant System was selected for the expected environment and service conditions. The major component materials are listed in Table 4.1-1.

The safety of the reactor vessel and all other Reactor Coolant System pressure containing components and piping is dependent on several major factors including design and stress analysis, material selection and fabrication, quality control and operations control.

The phenomena of stress corrosion cracking and corrosion fatigue are not encountered unless a specific combination of conditions is present. The necessary conditions are a susceptible alloy, an aggressive environment, a stress, and time.

Stress analyses of the Reactor Coolant System components which reflect consideration of all design loadings detailed in the design specifications have been prepared by the component designers. The analyses show that the reactor vessel, steam generator, pump casing and pressurizer comply with the stress limits of Section III of the ASME Code. A similar analysis of the piping shows that it complies with the stress limits of the applicable USAS Code.

As part of the design control on materials, Charpy V-notch toughness tests were run on all ferritic material used in fabricating pressure parts of the reactor vessel, steam generator and pressurizer to provide assurance for hydrotesting and operation in the ductile region at all times. In addition, dropweight tests and Charpy V-notch transition temperature curves were performed on the reactor vessel materials.

As an assurance of system integrity, all components in the system were hydrotested at a nominal test pressure of 3107 psig prior to initial operation.

All Reactor Coolant System materials which are exposed to the coolant are corrosion resistant. They consist of stainless steels and Inconel, and they are chosen for specific purposes at various locations within the system for their superior compatibility with the reactor coolant.

Water Chemistry

The water chemistry is selected to provide the necessary boron content for reactivity control and to minimize corrosion of Reactor Coolant System surfaces. Maintenance of the water quality to minimize corrosion is accomplished using the Chemical and Volume Control System and Sampling System which are described in Section 10.

Typical reactor coolant chemistry compositions are given in Table 4.1-9. This subject is also discussed in the Technical Requirements Manual.

A typical condition of operation could include any combination of chemical elements, as long as none of the specified limits are exceeded.

Galvanic Corrosion

The only types of materials which are in contact with each other in borated water are stainless steels, Inconel, Stellite valve materials and Zircaloy fuel element coating. These materials have been shown (Reference 121) to exhibit only an insignificant degree of galvanic corrosion when coupled to each other.

For example, the galvanic corrosion of Inconel versus 304 stainless steel resulting from high temperature tests (575°F) in lithiated, boric acid solution was found to be less than -20.9 mg/dm^2 for the test period of 9 days. Further galvanic corrosion would be trivial since the cell currents at the conclusion of the tests were approaching polarization. Zircaloy versus 304 stainless steel was shown to polarize in 140°F boric acid solution in less than 8 days with a total galvanic attack of -3.0 mg/dm^2. Stellite versus 304 stainless steel was polarized in 7 days at 575°F in lithiated, boric acid solution. The total galvanic corrosion for this couple was -0.97 mg/dm^2.

As can be seen from the tests, the effects of galvanic corrosion are insignificant to systems containing borated water.

ZIRLO material properties are essentially identical to the Zircaloy alloy; therefore, the effect of galvanic corrosion on this new zirconium based fuel rod clad and guide thimble tube alloy is insignificant. (Reference 122)

Protection Against Proliferation of Dynamic Effects

Engineered Safety Features and associated systems are protected from loss of function due to dynamic effects and missiles which might result from a loss-of-coolant accident. Protection is provided by missile shielding and/or separation of redundant components. Further discussion of missile protection is given in Sections 6 and 12.

The Reactor Coolant System is surrounded by concrete shield walls. These walls provide shielding to permit access into the containment during full power operation for inspection and maintenance of miscellaneous equipment. These shielding walls also provide missile protection for the reactor containment building.

The steam generator is provided with hydraulic shock suppressors as part of the upper lateral support near the center of gravity to resist lateral loads, including those resulting from seismic forces and pipe rupture forces. Additional bracing is also provided at a lower elevation to resist pipe rupture loads.

Missile protection afforded by the arrangement of the Reactor Coolant System is discussed in Section 12.2.6.

4.1.1 General Design Basis

4.1.1.1 Quality Standards

Criterion: Those systems and components of reactor facilities which are essential to the prevention, or the mitigation of the consequences, of nuclear accidents which could cause undue risk to the health and safety of the public shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes and standards pertaining to design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance criteria to be used shall be identified. An indication of the applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance criteria used is required. Where such items are not covered by applicable codes and standards, a showing of adequacy is required. (GDC 1)

The Reactor Coolant System is of primary importance with respect to its safety function in protecting the health and safety of the public.

Quality standards of material selection, design, fabrication and inspection conform to the applicable provisions of recognized codes and good nuclear practice (Section 4.1.6). Details of the quality assurance programs, test procedures and inspection acceptance levels are given in Section 4.7. Particular emphasis is placed on the assurance of quality of the reactor vessel to obtain material whose properties are uniformly within code specifications.

4.1.1.2 Performance Standards

Criterion: Those systems and components of reactor facilities which are essential to the prevention or to the mitigation of the consequences of nuclear accidents which could cause undue risk to the health and safety of the public shall be designed, fabricated, and erected to performance standards that will enable such systems and components to withstand, without undue risk to the health and safety of the public the forces that might reasonably be imposed by the occurrence of an extraordinary natural phenomenon such as earthquake, tornado, flooding condition, high wind or heavy ice. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been officially recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design. (GDC 2)

All piping, components and supporting structures of the Reactor Coolant System, except for those components provided for in paragraph (c)(2) of Section 50.55a of 10CFR50, are designed as Class I equipment, i.e., they are capable of withstanding:

a. The Operational Basis Earthquake (OBE) ground acceleration within allowable working stresses.

b. The Design Basis Earthquake (DBE) ground acceleration acting in the horizontal and vertical direction simultaneously with no loss of function.

Allowable limits for the above are given in Section 12.

The Reactor Coolant System is located in the containment building whose design, in addition to being a Class I structure, also considers accidents or other applicable natural phenomena. Details of the containment design are given in Section 12.

4.1.1.3 Records Requirements

Criterion: The reactor licensee shall be responsible for assuring the maintenance throughout the life of the reactor of records of the design, fabrication, and construction of major components of the plant essential to avoid undue risk to the health and safety of the public. (GDC 5)

Records of the design, fabrication, and construction of the major Reactor Coolant System components and the related engineered safety features components are maintained at Prairie Island and will be retained there throughout the life of the plant.

4.1.1.4 Missile Protection

Criterion: Adequate protection for those engineered safety features, the failures of which could cause an undue risk to the health and safety of the public, shall be provided against dynamic effects and missiles that might result from plant equipment failures. (GDC 40)

The dynamic effects during blowdown following a loss-of-coolant accident are evaluated in the detailed layout and design of the high pressure equipment and barriers which afford missile protection. Fluid and mechanical driving forces are calculated, and consideration is given to possible damage due to fluid jets and secondary missiles which might be produced.

The steam generators are supported, guided and restrained in a manner which prevents rupture of the steam side of a generator, the steam lines and the feedwater piping as a result of forces created by a Reactor Coolant System pipe rupture. These supports, guides and restraints also prevent rupture of the primary side of a steam generator as a result of forces created by a steam or feedwater line rupture.

The mechanical consequences of a pipe rupture are restricted by design such that the functional capability of the engineered safety features is not impaired.

A further discussion of missile protection is given in Section 12.

4.1.2 Principal Design Basis

4.1.2.1 Reactor Coolant Pressure Boundary

Criterion: The reactor coolant pressure boundary shall be designed, fabricated and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime. (GDC 9)

The Reactor Coolant System in conjunction with its control and protective provisions is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions, and maintain the stresses within applicable code stress limits.

Fabrication of the components which constitute the pressure boundary of the Reactor Coolant System is carried out in strict accordance with the applicable codes. In addition there are areas where equipment specifications for Reactor Coolant System components go beyond the applicable codes. Details are given in Section 4.7.1.

The materials of construction of the pressure boundary of the Reactor Coolant System are protected by control of coolant chemistry from corrosion phenomena which might otherwise reduce the system structural integrity during its service lifetime.

System conditions resulting from anticipated transients or malfunctions are monitored and appropriate action is automatically initiated to maintain the required cooling capability and to limit system conditions to a safe level.

The system is protected from overpressure by means of pressure relieving devices, as required by Section III of the ASME Boiler and Pressure Vessel Code.

Isolable sections of the system containing components designed in conformance with Section III of the ASME Boiler and Pressure Vessel Code are provided with overpressure relieving devices discharging to closed systems such that the system code allowable relief pressure within the protected section is not exceeded.

For definition of Reactor Coolant Pressure Boundary, see Section 1.2.6.

4.1.2.2 Monitoring Reactor Coolant Leakage

Criterion: Means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary. (GDC 16)

Positive indications of leakage of coolant from the Reactor Coolant System to the containment are provided by equipment which permits continuous monitoring of containment air radioactivity and humidity in the control room and runoff from the condensate collecting pans under the cooling coils of the containment air cooling units in the auxiliary building. This equipment provides indication of normal background which is indicative of a basic level of leakage from Reactor Coolant System pressure boundary. Any increase in the observed parameters is an indication of change within the containment, and the equipment provided is capable of monitoring this change. The basic design criterion is the detection of deviations from normal containment environmental conditions including air particulate radioactivity, radiogas activity, humidity, condensate runoff and in addition, in the case of gross leakage, the liquid inventory in the process systems and containment sump.

Further details are supplied in Section 6.5. The maximum permitted reactor coolant leakage rates for uncontrolled sources are stated in the Technical Specifications.

4.1.2.3 Reactor Coolant Pressure Boundary Capability

Criterion: The reactor coolant pressure boundary shall be capable of accommodating without rupture the static and dynamic loads imposed on any boundary component as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition. (GDC 33)

The reactor coolant boundary is shown to be capable of accommodating without rupture, the static and dynamic loads imposed as a result of a sudden reactivity insertion such as a rod ejection. Details of this analysis are provided in Section 14.

The operation of the reactor is such that the severity of a rod ejection accident is inherently limited. Since rod cluster control assemblies (RCCA) are used to control load variations only and boron dilution is used to compensate for core depletion, only the RCCA in the controlling groups are inserted in the core at power, and at full power these rods are only partially inserted. A rod insertion limit monitor is provided as an administrative aid to insure that this condition is met.

By using the flexibility in the selection of control rod groupings, radial locations and position as a function of load, the design limits the maximum fuel temperature for the highest worth ejected control rod accident to a value which precludes any resultant damage to the primary system pressure boundary due to excessive pressure surges.

4.1.2.4 Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention

Criterion: The reactor coolant pressure boundary shall be designed and operated to reduce to an acceptable level the probability of rapidly propagating type failure. Consideration is given (a) to the provisions for control over service temperature and irradiation effects which may require operational restrictions, (b) to the design and construction of the reactor pressure vessel in accordance with applicable codes, including those which establish requirements for absorption of energy within the elastic strain energy range and for absorption of energy by plastic deformation and (c) to the design and construction of reactor coolant pressure boundary piping and equipment in accordance with applicable codes. (GDC 34)

The ability of the large steel pressure vessel containing the reactor core and its primary coolant to resist fracture constitutes an important factor in insuring safety in the nuclear industry. The beltline region of the reactor pressure vessel is the most critical region of the vessel because it is subjected to significant fast neutron bombardment. The overall effects of fast neutron irradiation on the mechanical properties of low alloy ferritic pressure vessel steels such as SA508 Class 3 (base material of the Prairie Island Units 1 and 2 reactor pressure vessel beltlines) are well documented in the literature. Generally, low alloy ferritic materials show an increase in hardness and tensile properties and decrease in ductility and toughness under certain conditions of irradiation.

The Reactor Vessel Material Surveillance Program, described in Section 4.7.2, monitors the effects of radiation on reactor vessel materials, and establishes operating limits to assure that brittle fracture of the reactor vessel will not occur. The program is in accordance with ASTM-E-185.

The special case of low temperature overpressurizations has been addressed by installing the Low Temperature Overpressure Protection System (OPPS) described in Section 4.4.3.3. The design criteria for this system is detailed in References 89 and 107.

4.1.3 Design Characteristics

4.1.3.1 Design Pressure

The Reactor Coolant System design and operating pressure together with the safety, power relief and pressurizer spray valve set points, and the protection system set point pressures are listed in Table 4.1-2. The design pressure allows for operating transient pressure changes. The selected design margin considers core thermal lag, coolant transport times and pressure drops, instrumentation and control response characteristics, and system relief valve characteristics. The design pressures and data for the respective system components are listed in Tables 4.1-3 through 4.1-7. Table 4.1-10 gives the design pressure drop of the system components.

4.1.3.2 Design Temperature

The design temperature for each component is selected to be above the maximum coolant temperature in that component under all normal and anticipated transient load conditions. The design and operating temperatures of the system components are listed in Tables 4.1-3 through 4.1-7.

4.1.3.3 Seismic Loads

The seismic loading conditions are established by the Operational Basis Earthquake (OBE) and the Design Basis Earthquake (DBE). The former is selected to be typical of the largest probable ground motion based on the site seismic history. The latter is selected to be the largest potential ground motion at the site based on seismic and geological factors and their uncertainties.

For the OBE loading condition, the nuclear steam supply system is designed to be capable of continued safe operation. Therefore, for this loading condition, critical structures and equipment needed for this purpose are required to operate within normal design limits. The seismic design for the DBE is intended to provide a margin in design that assures capability to shut down and maintain the nuclear facility in a safe condition. In this case, it is only necessary to ensure that the Reactor Coolant System components do not lose their capability to perform their safety function. This has come to be referred to as the “no-loss-of-function” criteria and the loading condition as the “no-loss-of-function earthquake” loading condition.

For the combination of normal plus OBE loadings, the stresses in the support structures are kept within the limits of the applicable codes.

For the combination of normal plus DBE loadings, the stresses in the support structures are limited to values necessary to ensure their integrity, and to keep the stresses in the Reactor Coolant System components within the allowable limits as given in Section 12. Shock suppressors are installed on the RCS system to prevent the unrestrained motion of the RCS pipes and components under dynamic loads such as earthquakes and other severe transients. Shock suppressors do not restrain the normal thermal movements during startup and shutdown.

4.1.4 Cyclic Loads

To provide the necessary high degree of integrity for the components in the Reactor Coolant System designed to ASME Section III, transient conditions are selected for fatigue evaluation based on a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from normal operation, normal and abnormal load transients and accident conditions. To a large extent, the specific transient operating conditions to be considered for equipment fatigue analyses are based upon engineering judgment and experience. Those transients are chosen which are representative of transients to be expected during plant operation and which are sufficiently severe or frequent to be of possible significance to component cyclic behavior. The number of thermal and loading cycles used in the fatigue evaluations for Reactor Coolant System components designed to ASME Section III, are given in Table 4.1- 8.

Clearly it is difficult to discuss in absolute terms the transients that the plant will actually experience during the 60 years operating life. For clarity, however, each transient condition is discussed in order to make clear the nature and basis for the various transients. These transient conditions are conservative estimates for equipment design purposes and are not intended to be accurate representations of actual transients or to reflect actual operating procedures.

Note that the analyses for the replacement RV Head and CRDM Housings utilize and envelop of the Prairie Island and Point Beach design transients. The analyses for the CETNAs utilize generic Westinghouse design transients. Both the Point Beach and Westinghouse generic design transients include additional transients beyond those required by Table 4.1-8. Both design transients also include larger differential temperatures and differential pressures than is required by the current Prairie Island operating conditions. Thus the stress results and cumulative usage factor are conservative relative to the requirements for Prairie Island (References 138 and 139).

A License renewal evaluation determined that the pre-license renewal limits for the total number of occurrences for each design transient for 40-year plant design life remain valid for the 60-year plant life extension. The evaluation (1) determined the actual numbers of occurrences experienced as of September 2006 for each transient at Prairie Island Units 1 and 2 based on the available plant data; (2) compared the number of actual plant transient occurrences with the Design Basis quantity of transient occurrences; and (3) estimated the anticipated number of transient occurrences for 60 years of operation based on actual transient occurrences through September 2006.

4.1.4.1 Heatup and Cooldown

The normal heatup or cooldown cases are conservatively represented by a continuous operation performed at a uniform temperature rate of 100°F per hour (except for a pressurizer cooldown rate of 200°F per hour).

For these cases, the heatup occurs from ambient to the no-load temperature and pressure condition and the cooldown represents the reverse situation. In actual practice, the rate of temperature change of 100°F per hour will not be attained because of other limitations such as:

a. Slower initial heatup rates when using pumping energy only.

b. Interruptions in the heatup and cooldown cycles due to such factors as drawing a pressurizer steam bubble, rod withdrawal, sampling, water chemistry and gas adjustments.

4.1.4.2 Unit Loading and Unloading

The unit loading and unloading cases are conservatively represented by a continuous and uniform ramp power change of 5% of nominal full load per minute between 15% and 100% of nominal full load, subject to possible xenon limitations. This load swing is the maximum possible without reactor trip consistent with operation with automatic reactor control. The reactor coolant temperature will vary with load as prescribed by the temperature control system.

4.1.4.3 Step Load Increase and Decrease of 10%

The 10% of nominal full load step change, increase or reduction, in load demand is a control transient which is assumed to be a change in turbine control valve opening which might be occasioned by disturbances in the electrical network into which the plant output is tied. The Reactor Control System is designed to restore plant equilibrium without reactor trip following a 10% of nominal full load step change, increase or decrease, in turbine load demand initiated from nuclear plant equilibrium conditions in the range between 15% and 100% of nominal full load, the range for automatic reactor control, subject to possible xenon limitations. In effect, during load change conditions, the Reactor Control System attempts to match turbine and reactor outputs in such a manner that peak reactor coolant temperature is minimized and reactor coolant temperature is restored to its programmed set point at a sufficiently slow rate to prevent excessive pressurizer pressure decrease.

Following a step decrease in turbine load, the secondary side steam pressure and temperature initially increase since the decrease in nuclear power lags behind the step decrease in turbine load. During the same increment of time, the Reactor Coolant System average temperature and pressurizer pressure also initially increase. Because of the power mismatch between the turbine and reactor and the increase in reactor coolant temperature, the control system automatically inserts the control rods to reduce core power. With load decrease, the reactor coolant temperature will be ultimately reduced from its peak value to a value below its initial equilibrium value at the inception of the transient. The reactor coolant average temperature set point change is made as a function of turbine-generator load as determined by first stage turbine pressure measurement. The pressurizer pressure will also decrease from its peak pressure value and follow the reactor coolant decreasing temperature trend. At some point during the decreasing pressure transient, the saturated water in the pressurizer begins to flash which reduces the rate of pressure decrease. Subsequently, the pressurizer heaters come on to restore the plant pressure to its normal value.

Following a step load increase in turbine load, the reverse situation occurs, i.e., the secondary side steam pressure and temperature initially decrease and the reactor coolant average temperature and pressure initially decrease. The control system automatically withdraws the control rods to increase core power. The decreasing primary pressure transient is reversed by actuation of the pressurizer heaters and eventually the system pressure is restored to its normal value. The reactor coolant average temperature will be raised to a value above its initial equilibrium value at the beginning of the transient.

4.1.4.4 Large Step Decrease in Load

This transient applies to a step decrease in turbine load from full power of such magnitude that the resultant rapid increase in reactor coolant average temperature and secondary side steam pressure and temperature will automatically initiate a secondary side steam dump system that will prevent a reactor shutdown or lifting of steam generator safety valves. The plant is designed to accept a step decrease of 40.0% of nominal full load with the combined operation of the Reactor Rod Control System and the Steam Dump System. If a steam dump system was not provided to cope with this transient, there would be such a strong mismatch between turbine and reactor power that a reactor trip and lifting of steam generator safety valves would occur.

4.1.4.5 Loss of Load

This transient applies to a step decrease in turbine load from full power brought about by a loss of turbine load without immediately initiating a reactor trip and represents the most severe transient on the Reactor Coolant System. The reactor and turbine eventually trip as a consequence of a high pressurizer level trip initiated by the Reactor Protection System.

Since redundant means of tripping the reactor upon turbine trip are provided as part of the Reactor Protection System, transients of this nature are not expected.

4.1.4.6 Loss of Offsite Power

This transient applies to the loss of outside electrical power to the station and a reactor and turbine trip, on low reactor coolant flow, culminating in a complete loss of plant AC electrical power. Under these circumstances, the emergency diesel generators are started, the reactor coolant pumps are de-energized and following the coastdown of the reactor coolant pumps, natural circulation builds up in the system to some equilibrium value. This condition permits removal of core residual heat through the steam generators which at this time are receiving feedwater from the Auxiliary Feedwater System operating from diesel generator power or steam driven auxiliary feedwater pumps. Steam is removed for reactor cooldown through atmospheric power operated relief valves provided for this purpose.

4.1.4.7 Loss of Flow

This transient applies to a partial loss of flow accident from full power in which a reactor coolant pump is tripped out of service as a result of a loss of power to that pump. The consequences of such an accident at high power level are a reactor and turbine trip, on low reactor coolant flow, followed by automatic opening of the steam dump system and flow reversal in the affected loop. The flow reversal results in reactor coolant at cold leg temperature, being passed through the steam generator and cooled still further. This cooler water then passes through the hot leg piping and enters the reactor vessel outlet nozzles. The net result of the flow reversal is a sizeable reduction in the hot leg coolant temperature of the affected loop.

4.1.4.8 Reactor Trip From Full Power

A reactor trip from full power may occur for a variety of causes resulting in temperature and pressure transients in the Reactor Coolant System and in the secondary side of the steam generator. This is the result of continued heat transfer from the reactor coolant in the steam generator. The transient continues until the reactor coolant and steam generator secondary side temperatures are in equilibrium at zero power conditions. A continued supply of feedwater and controlled dumping of secondary steam remove the core residual heat and prevent the steam generator safety valves from lifting. The reactor coolant temperature and pressure undergo a rapid decrease from full power values as the Reactor Protection System causes the control rods to move into the core.

4.1.4.9 Turbine Roll Test

This transient is imposed upon a plant during the hot functional test period for turbine cycle checkout. Reactor coolant pump power is used to heat the reactor coolant to operating temperature and the steam generated is used to perform a turbine roll test. However, the plant cooldown during this test may exceed the normal 100°F per hour maximum rate.

The number of such test cycles is specified at 10 times to be performed at the beginning of plant operating life prior to irradiation. Two such cycles were performed at Prairie Island on Unit 1; none, on Unit 2.

4.1.4.10 Hydrostatic Test Conditions

The pressure tests are outlined below:

a. Primary Side Hydrostatic Test Before Initial Startup at 3107 psig

The pressure tests covered by this section included both shop and field hydrostatic tests which occurred as a result of component or system testing. This hydro test was performed at a water temperature which was compatible with reactor vessel material design transition temperature (DTT) requirements which shift with lifetime and a maximum test pressure of 3107 psig. In this test, the primary side of the steam generator was pressurized to 3107 psig coincident with the secondary side pressure of 0 psig. The Reactor Coolant System was analyzed for 5 cycles of this hydro test. The Unit 1 and Unit 2 SGs were designed for 15 primary side hydro tests.

b. Secondary Side Hydrostatic Test Before Initial Startup

The secondary side of the steam generators were pressurized to 1357 psig for Unit 1 and Unit 2 with a minimum water temperature of 70°F coincident with the primary side at 0 psig.

The steam generators were analyzed for 5 cycles of this test. One test was made on Unit 1 and one on Unit 2 after installation.

c. Primary Side Leak Test

Subsequent to each time the primary system is opened, a leak test will be performed. During this test the primary system pressure, for design purposes, is assumed to be raised to 2,330 psig for Unit 1 and 2,500 psia for Unit 2, with the system temperature above Design Transition Temperature, while the system is checked for leaks.

For design purposes it was assumed that the primary side experienced 50 cycles of this test during the 60-year design life of the plant. In actual practice, the primary system is pressurized to the nominal operating pressure associated with 100% rated reactor power with the test pressure and temperature attained at a rate in accordance with DTT considerations.

During this leak test, the secondary side of the steam generator must be pressurized so that the pressure differential across the tube sheet does not exceed 1600 psi. This is accomplished by closing off the steam lines.

4.1.4.11 Accident Conditions

The effect of the accident loading was evaluated in combination with normal loads to demonstrate the adequacy to meet the stated plant safety criteria.

A brief description of each accident transient which was considered is listed below.

a. Reactor Coolant Pipe Break

This accident involves the rupture of a Reactor Coolant System pipe resulting in a loss of primary coolant. It is conservatively assumed that the system pressure and temperature are reduced rapidly and the Safety Injection System is initiated to introduce 70°F water into the Reactor Coolant System. The safety injection signal results in a turbine and reactor trip. Because of the rapid blowdown of coolant from the system and the comparatively large heat capacity of the metal sections of the components, it is likely that the metal is still at no-load temperature conditions when the 70°F safety injection water is introduced into the system.

b. Steam Line Break

For Reactor Coolant System component evaluation, the following conservative conditions were considered:

1. The reactor is initially in a hot, zero-load, just critical condition assuming all rods in except the most reactive rod which is assumed to be stuck in its fully withdrawn position.

2. A steam line break occurs inside the containment resulting in a reactor trip.

3. Subsequent to the break, there is no return to power and the reactor coolant temperature cools down to 212°F.

4. The Safety Injection System pumps restore the reactor coolant pressure.

The above conditions result in the most severe temperature and pressure variations which the Reactor Coolant System components will encounter during a steam break accident.

c. Steam Generator Tube Rupture

This accident postulates the double-ended rupture of a steam generator tube resulting in a decrease in pressurizer level and reactor coolant pressure. Reactor trip will occur due to low pressurizer pressure. Shortly after this, a low pressurizer pressure safety injection will occur. This safety injection signal will close the feedwater regulating valves. After the rupture, the primary system pressure is reduced below the secondary system design pressure (1100 psia). The planned procedure for recovery from this accident calls for isolation of the steam line leading from the affected steam generator at this time. Therefore, this accident will result in a transient which is no more severe than that associated with a reactor trip. For this reason, it requires no special treatment in so far as fatigue evaluation is concerned, so no occurrences have been evaluated.

4.1.4.12 Classification of RCS Transients

Transients shown in Table 4.1-8 are classified by the following conditions:

Normal Condition	-	Items 1-4, 12
Upset Condition	-	Items 5-8
Test Condition	-	Items 9-10
Faulted Condition	-	Items 11a,b,c

4.1.4.13 Pressurizer Surge and Spray Line Connections

The surge and spray nozzle connections at the pressurizer vessel are subject to cyclic temperature changes resulting from the transient conditions described previously. The various transients are characterized by variations in reactor coolant temperature which in turn result in water surges into or out of the pressurizer. The surges manifest themselves as changes in system pressure which, depending upon whether an increase or decrease in pressure occurs, result in introducing spray water into the pressurizer to reduce pressure or actuating the pressurizer heaters to increase pressure to the equilibrium value. To illustrate a load change cycle as it affects the pressurizer, consider a design step increase in load. The pressurizer initially experiences an outsurge with a drop in system pressure which actuates the pressurizer heaters to restore system pressure. As the Reactor Control System reacts, the reactor coolant temperature is increased which causes an insurge into the pressurizer raising system pressure. As pressure is increased, the heaters go off and at some pressure setpoint, the spray valves open to limit the pressure rise and restore system pressure. Thus the pressurizer surge nozzle is subjected to a temperature increase on the outsurge followed by a temperature decrease on the insurge during this load transient. The pressurizer spray nozzle is subjected to a temperature decrease when the spray valve opens to admit reactor coolant cold leg water into the pressurizer. The pressurizer experiences a reverse situation during a load decrease transient, i.e., an insurge followed by an outsurge. It is assumed that the spray valve opens to admit spray water into the pressurizer once at the design flowrate for each design step change in plant load. Thus the number of occurrences for the spray nozzle corresponds to that shown for the step changes in plant load in Table 4.1-8.

During plant cooldown, spray water is introduced into the pressurizer to cool down the pressurizer and to remove gas from the reactor coolant. The maximum pressurizer cooldown rate is specified at 200°F per hour which is twice the rate specified for the other Reactor Coolant System components.

4.1.5 Service Life

The service life of Reactor Coolant System pressure components depends upon the end-of-life material radiation damage, unit operational thermal cycles, quality manufacturing standards, environmental protection, and adherence to established operating procedures.

The reactor vessel is the only component of the Reactor Coolant System which is exposed to a significant level of neutron irradiation and it is therefore the only component which is subject to any appreciable material radiation damage effects.

The nil ductility transition temperature (NDTT) shift of the vessel material and welds, during service due to radiation damage effects is monitored by a radiation damage surveillance program which conforms with ASTM E185 and Appendix H of 10CFR50.

Reactor vessel design is based on the transition temperature method of evaluating the possibility of brittle fracture of the vessel material, as a result of operations such as leak testing and plant heatup and cooldown.

To establish the service life of the Reactor Coolant System components as required by the ASME (Part III), Boiler and Pressure Vessel Code for Class “A” Vessels, the unit operating conditions have been established for an enveloping 60 year design life. These operating conditions include the cyclic application of pressure loadings and thermal transients.

The number of thermal and loading cycles used for design purposes are listed in Table 4.1-8.

4.1.6 Codes and Classifications

All pressure-containing components of the Reactor Coolant System are designed, fabricated, inspected and tested in conformance with the applicable codes listed in Table 4.1-11.

The Reactor Coolant System is classified as Class I as detailed in Section 12, except for those components provided for in paragraph (c)(2) of Section 50.55a of 10CFR50.

4.1.7 Materials of Construction

All core structural load bearing members were made from annealed type 304 stainless steel, so there is no possibility of sensitization, with the exception of the core barrel itself, which required stress relief during manufacturing at temperatures over 750°F. The stress relieving operation was conducted in a manner to minimize the possibility of severe sensitization, while maintaining the necessary conditions for relieving residual fabrication stresses. This consisted of heating to 1650°F, holding at this temperature for several hours, then cooling very slowly in the furnace. This treatment results in massive carbide precipitation at the grain boundaries, and agglomeration of the carbides, instead of the formation of detrimental continuous carbide films. Further, the long times at high temperatures cause diffusion of chromium into the grain boundary areas that were depleted in chromium by the precipitation of chromium carbides. This combination of formation of massive carbides, plus diffusion of chromium back into the depleted zone is referred to as “desensitization”, and is commonly used to prevent severe sensitization of parts requiring heat treatments that otherwise would cause severe sensitization of the material. Strauss tests run according to ASTM A393 were performed on core barrel material given this heat treatment, and results verified that severe sensitization is prevented.

It is characteristic of stress corrosion that combinations of alloy and environment which result in cracking are usually quite specific. Environments which have been shown to cause stress corrosion cracking of stainless steels are free alkalinity in the presence of a concentrating mechanism and the presence of chlorides, fluorides, and free oxygen. With regard to the former, experience has shown that deposition of chemicals on the surface of tubes can occur in a steam blanketed area within a steam generator.

In the presence of this environment under very specific conditions, stress-corrosion cracking can occur in stainless steels having the nominal residual stresses resulting from normal manufacturing procedures. However, the steam generator contains Inconel tubes. Testing to investigate the susceptibility of heat exchanger construction materials to stress corrosion in caustic and chloride aqueous solutions has indicated that Inconel Alloy has excellent resistance to general and pitting-type corrosion in severe operating water conditions.

The use of lead in the materials of the secondary side of this plant has been minimized to the practical limit of that occurring as trace elements in metallurgical alloys and as such is insignificant.

All external insulation of Reactor Coolant System components is compatible with the component materials. The cylindrical shell exterior and closure flanges to the reactor vessel, the reactor vessel closure head, and all other external corrosion resistant surfaces in the Reactor Coolant System are insulated with metallic reflective insulation as required.

The nil ductility transition (NDT) temperature of the reactor vessel material opposite the core is established at a Charpy V-notch impact energy of 30 ft-lb or greater. The material is tested to verify conformity to specified requirements and to determine the actual NDT temperature value. In addition, this material is 100 percent volumetrically inspected by ultrasonic test using both straight beam and angle beam methods.

The remaining material in the reactor vessel, and other Reactor Coolant System components, meets the appropriate design code requirements and specific component function.

The reactor vessel material is heat-treated specifically to obtain good Charpy V-notch ductility which ensures a low NDT temperature and thereby gives assurance that the finished vessel can be initially hydrostatically tested and operated as near to room temperature as possible without restrictions. The stress limits established for the reactor vessel are dependent upon the temperatures at which the stresses are applied. As a result of fast neutron irradiation in the region of the core, the material properties will change, including an increase in the NDT temperature. A nominal maximum value of NDT temperature was established during fabrication.

The shift of the NDT is affected by neutron fluence. The methodology used to provide the best estimate neutron exposure evaluation of the vessel wall is based upon a technique where an analytical model of the irradiation capsule exposure is compared with measured data producing a bias. This bias is projected into the analytical model of exposure in the vessel wall. The techniques used to measure and predict the integrated fast neutron ($E > 1$ Mev) fluxes at the sample locations are described in References 105 and 106. The analytical methods used to obtain the maximum neutron ($E > 1$ Mev) exposure of the reactor vessel are described in Reference 140.

The maximum integrated fast neutron (E > 1 MeV) exposure has been updated to account for the MUR power uprate and license renewal to 60 years of operation. To 01451781 more accurately predict the effects of the MUR power uprate and life extension on reactor vessel integrity, the updated neutron exposure was calculated using the methodology of Reference 140 using cycle-specific core power distributions. Values were computed at 54 Effective Full Power Years. The computed maximum exposure at 01451781 the reactor vessel clad / base metal interface occurs in the intermediate shell course at both Unit 1 (5.162×10^{19} n/cm^2) and Unit 2 (5.196×10^{19} n/cm^2). The bounding Unit 1 RT_{PTS} value calculated at 54 EFPY is 157°F for circumferential weld seam W2, which joins the nozzle shell forging (upper shell course) to the intermediate shell forging. The bounding Unit 2 RT_{PTS} value calculated at 54 EFPY is 136°F for circumferential weld seam W2, which joins the nozzle shell forging (upper shell course) to the intermediate shell forging (Reference 141).

To evaluate the NDT temperature shift of welds, heat affected zones and base material for the vessel, test coupons of these material types have been included in the reactor vessel surveillance program described in Section 4.7.2. The methods used to measure the initial NDT temperature of the reactor vessel base plate material are also given in Section 4.7.2.

4.1.7.1 Effect of Aging on Cast Stainless Steel

As a result of investigations (References 118 and 119) conducted both by Westinghouse NES in the United States and by a Westinghouse Licensee in France, it was found that long-time thermal service could severely degrade the Charpy V-notch impact properties of cast AISI 316 stainless steels. Since the Charpy test has long been used as a measure of structural performance for carbon and low alloy steels, this degradation was cause for some concern as to the integrity of PWR primary coolant piping and some reactor internals components made from this type stainless steel.

The Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program manages loss of fracture toughness due to thermal aging embrittlement of CASS components, other than pump casings and valve bodies, that are exposed to reactor coolant operating temperatures. The program determines the susceptibility of CASS components to loss of fracture toughness due to thermal aging embrittlement based on the casting method, molybdenum content, and percent ferrite. For components determined to be potentially susceptible to thermal aging embrittlement, the program provides for enhanced volumetric examination or component-specific flaw tolerance 01451781 evaluations. For pump casings and valve bodies, screening for susceptibility to thermal aging embrittlement is not required since a bounding integrity analysis has shown they are resistant to failure.

4.1.8 Reliance on Interconnected Systems

The principal heat removal systems which are interconnected with the Reactor Coolant System are the Steam and Feedwater Systems and the Safety Injection and Residual Heat Removal Systems. The Reactor Coolant System is dependent upon the steam generators, and the steam, feedwater, and condensate systems for decay heat removal from normal operating conditions to a reactor coolant temperature of approximately 350°F. The layout of the system ensures the natural circulation capability to permit plant cooldown following a loss of all reactor coolant pumps.

The flow diagrams of the Steam and Power Conversion System are shown in Figures 11.1-1 through 11.1-8. In the event that the condensers are not available to receive the steam generated by residual heat, the water stored in the condensate and feedwater system may be pumped into the steam generator and the resultant steam vented to the atmosphere. The Auxiliary Feedwater System will supply water to the steam generators in the event that the main feedwater pumps are inoperative. Indication of auxiliary feedwater flow to each steam generator is provided in the control room. The system is described in Section 11.9.

The Safety Injection System is described in Section 6. The Residual Heat Removal System is described in Section 10.

4.2 REACTOR VESSEL

4.2.1 Design Basis

The design basis for the reactor vessel are discussed in Section 4.1.

4.2.2 Description

The reactor vessel is cylindrical in shape with a hemispherical bottom and a flanged and gasketed removable upper head. Figure 4.2-1 is a schematic of the reactor vessel. The materials of construction of the reactor vessel are given in Table 4.1-1.

The safe ends for the primary and safety injection nozzles of the reactor vessel are made of stainless steel with less than 0.02% carbon, to avoid partial or local severe sensitization of austenitic stainless steel.

Coolant enters the reactor vessel through inlet nozzles in a plane just below the vessel flange and above the core. The coolant flows downward through the annular space between the vessel wall and the core barrel into a plenum at the bottom of the vessel where it reverses direction. A small percentage also flows downward between the core baffle plates and core barrel to provide additional cooling of the barrel. Approximately ninety-five per cent of the total coolant flow is effective for heat removal from the core. The remainder of the flow includes the flow through the RCC guide thimbles, the leakage across the fuel assembly outlet nozzles, and the flow deflected into the head of the vessel for cooling the upper flange. All the coolant is united and mixed in the upper plenum, and the mixed coolant stream then flows out of the vessel through exit nozzles located on the same plane as the inlet nozzles.

A one-piece thermal shield, concentric with the reactor core, is located between the core barrel and the reactor vessel. The shield is bolted and welded to the top of the core barrel. The shield, which is cooled by the coolant on its downward pass, protects the vessel by attenuating much of the gamma radiation and some of the fast neutrons which escape from the core. This shield minimizes thermal stresses in the vessel which result from heat generated by the absorption of gamma energy. It is illustrated in Figure 3.6-1 and is further described in Section 3.6.

Thirty-six core instrumentation nozzles are located on the lower head.

The reactor closure head and the reactor vessel flange are joined by 48 - 6 in. diameter studs. Two metallic O-rings seal the reactor vessel when the reactor closure head is bolted in place. A leakoff connection is provided between the two O-rings to monitor leakage across the inner O-ring. In addition, a leak-off connection is also provided beyond the outer O-ring seal.

The reactor vessel insulation is of the reflective type, supported from the nozzles and consisting of inner and outer sheets of stainless steel sheet metal with multilayer stainless steel foil as the insulating agent. The clearance to the reactor vessel is 1/2 inch. Insulation sheets are also provided for the reactor closure head, and are supported on the refueling seal ledge and vent shroud support rings.

The reactor vessel contains the core support assembly, upper plenum assembly, fuel assemblies, rod cluster control assemblies, surveillance specimens, and in-core instrumentation. The reactor vessel internals are designed to direct the coolant flow, support the reactor core, and guide the control rods.

Ring forgings have been used in the following areas of the reactor vessel:

1. Vessel Flange
2. Upper Shell Course
3. Four Primary and Two Safety Injection Nozzles
4. Intermediate Shell Course
5. Lower Shell Course
6. Bottom Head Transition Ring

Westinghouse has evaluated the use of sensitized stainless steel and reactor components in pressurized water reactors. The results of this evaluation are summarized in Reference 3 which covers the nature of sensitization, conditions leading to stress corrosion and associated problems with both sensitized and non-sensitized stainless steel. The results of extensive testing and service experience that justify the use of stainless steel in the sensitized condition for components in Westinghouse PWR Systems is presented in Reference 3.

Surveillance specimens made from reactor vessel core region shell forging low alloy steel material are located between the reactor vessel wall and the thermal shield. These specimens are examined at selected intervals to evaluate reactor vessel material NDTT changes as described in Section 4.7.2.

The reactor internals are described in detail in Section 3.6 and the general arrangement of the reactor vessel and internals is shown in Figure 3.1-2.

Reactor vessel design data are listed in Table 4.1-3.

The designer/manufacturer of the reactor vessels for Prairie Island is Creusot-Loire (formerly called SFAC) whose manufacturing facility is located in Southern France near the city of Creusot.

They were surveyed and approved as a qualified vessel supplier by the U.S. Atomic Energy Commission and the State of Minnesota (with assistance from the National Board) for the Prairie Island units.

Creusot-Loire had met all the latest code requirements for qualification with the one exception being that they were unable to obtain an ASME Code stamp since ASME did not have provisions for issuance of Code Stamps to foreign suppliers. However, their quality assurance program was in strict conformance with ASME Section III requirements.

The replacement reactor vessel closure heads and CRDM pressure housings were manufactured by Mitsubishi Heavy Industries Ltd (MHI) in Futami, Japan under their ASME Section III code program.

The Reactor Support Structure consists of six structural steel columns extending downward to the bottom of the reactor cavity. The top of each column (Reactor Vessel Ventilated Support Pad discussed in Section 5.2.2.3.1.3) is anchored by structural members imbedded in the surrounding steel reinforced concrete.

The reactor vessel has six supports, four pads, one at each nozzle, and two brackets. Each support bears on a ventilated support pad, which is fastened to the support structure (discussed in Section 12.2.5). The support shoe is a structural member that transmits the support loads to the supporting structure. The support pad is designed to restrain lateral and rotational movement of the reactor vessel, but allows for thermal growth by permitting radial sliding at each support on bearing plates.

4.2.3 Design Evaluation

4.2.3.1 General

The reactor vessel has a 132″ ID and is within size limits for which good experience exists. A stress evaluation of the reactor vessel was carried out in accordance with the rules of Section III of the ASME Nuclear Vessel Code. The evaluation demonstrates that stress levels are within the stress limits of the Code. Table 4.2-1 presents a summary of fatigue usage factors for components of the reactor vessel and results of the stress evaluation.

The cycles specified for the fatigue analysis are the results of an evaluation of the expected plant operation coupled with experience from nuclear power plants now in service. The conservatism of the design fatigue curves used in the fatigue analysis has been demonstrated by the Pressure Vessel Research Committee (PVRC) in a series of cyclic pressurization tests of model vessels fabricated to the Code. The results of the PVRC tests showed that no crack initiation was detected at any stress level below the code allowable fatigue curve and that no crack progressed through a vessel wall in less than three times the allowable number of cycles. Similarly fatigue tests have been performed on irradiated pressure vessel steels with comparable results (Reference 4).

The vessel design pressure is 2485 psig while the normal operating pressure is 2235 psig. The resulting operating membrane stress is therefore amply below the code allowable membrane stress to account for operating pressure transients.

The stress allowed in the vessel in relation to operation below NDT temperature and DTT (NDT temperature plus 60°F) to preclude the possibility of brittle failure are:

a. At DTT; a maximum stress of 20% yield

b. From DTT to DTT minus 200°F a maximum stress decreasing from 20% to 10% yield

c. Below DTT minus 200°F; a maximum stress of 10% yield.

These limits are based on a conservative interpretation of the Fracture Analysis Diagram developed at the Naval Research Laboratory (References 5, 6 and 7) after many years of research and confirmed by extensive correlations with service failures. There have been no known service failures under conditions permitted by these limits. The Fracture Analysis Diagram is the most widely known and generally accepted criterion for brittle fracture prevention and includes linear elastic fracture mechanics concepts. These limits established by the Fracture Analysis Diagram have been correlated with linear elastic fracture mechanics insofar as possible (Reference 8) and are conservative in providing protection against brittle fractures. The stress limits are maintained by prescribing operating procedures which rely upon administrative pressure and temperature control during heatup and cooldown as described in Reference 9.

The actual shift in NDT temperature can be established periodically during plant operation by testing of vessel material samples which are irradiated cumulatively by securing them near the inside wall of the vessel in the core area. To compensate for any increase in the NDT temperature caused by irradiation, the limits given in the plant operating manual on the pressure-temperature relationship are periodically changed to stay within the stress limits, which are stated above during heatup and cooldown.

The vessel closure contains forty-eight 6-inch studs. The stud material is ASTM A-540 and code case 1335-4 which has a minimum yield strength of 104,400 psi at design temperature. The membrane stress in the studs when they are at the steady state operational condition is less than half this value. This means that about half of the forty-eight studs have the capability of withstanding the hydrostatic end load on vessel head without the membrane stress exceeding yield strength of the stud material at design temperature.

The emphasis on conservative operation, in setting up the temperature-pressure relationship, is placed on heatup and cooldown because the normal operating temperature always exceeds even the highest anticipated DTT during the life of the plant. The emphasis on conservatism is required for heatup and cooldown rates because long term irradiation of the vessel raises the DTT and thereby limits the heatup or cooldown rates. The conservatism in the limits stated above are:

a. Use of a stress concentration factor of 4 on assumed flaws in calculating the stresses.

b. Use of nominal yield of material instead of actual yield.

c. Neglecting the increase in yield strength resulting from radiation effects.

The factor of four is not an actual stress concentration factor such as described in Article 4 Design of Section III but is a margin of conservatism based on the Fracture Analysis Diagram in ASTM E208 as well as the stress limits maintained by the prescribed operating procedures which rely upon administrative pressure and temperature control during heatup and cooldown (Reference 9). At the DTT, the stresses are 20% of the yield strength versus a prescribed upper limit of 80% of the yield strength; therefore, at this point there is a margin of four (80%/20%).

Since the Fracture Analysis Diagram is based on a plot of nominal stress versus temperature and different size flaws (cracks) are assumed, the use of actual stress concentration factors do not apply.

4.2.3.2 Asymmetric Blowdown Loads

The consequences of asymmetric blowdown loads on the primary system resulting from a pipe break at a vessel nozzle were evaluated (References 11 and 12). Based on these evaluations, restraints were installed to limit the maximum break area at the reactor vessel nozzles to less than one square foot. Structural analyses were performed and submitted to NRC to demonstrate the adequacy of the restraints in mitigating the consequences of the asymmetric LOCA loadings (References 13 and 120).

In 1984, Westinghouse completed analyses for both Prairie Island units which provided the technical bases for eliminating large primary loop pipe rupture as the structural design basis for the plant design. In 1986, these analyses were reviewed and approved by the NRC. Based on the NRC Safety Evaluation, the reactor vessel nozzle restraints were removed in 1989. Refer to Section 4.6.2.3 for additional discussion of this analysis.

4.2.3.3 Protection Against Nonductile Failure

Two Westinghouse analyses (References 38 and 39) were performed to confirm that the reactor vessels for Units 1 and 2 are adequately protected against nonductile failure. The analyses followed the procedure of Appendix G of the ASME B&PV Code, Section III, which is based on the principles of linear elastic fracture mechanics.

Normal, upset and test conditions were analyzed at eleven critical locations in the reactor vessel:

- Inlet nozzle and vessel wall junction,
- Inlet nozzle corner region,
- Inlet nozzle bore,
- Safety injection nozzle and vessel junction,
- Safety injection nozzle corner region,
- Outlet nozzle and vessel wall junction,
- Outlet nozzle corner region,
- Outlet nozzle bore,
- Top closure head junction,
- Belt line of vessel, and
- Lower head and shell junction.

Per Appendix G of the Code, analysis is not required for nozzle portions up to 2.5 inches thick. Since its thickness is less than 2.5 inches, the safety injection nozzle bore was not analyzed.

The reference analyses showed that, for both of the Prairie Island Units:

- The maximum combined stress intensity factor (K_I) values of the belt line of the vessel are less than the reference fracture toughness, and
- The maximum combined K_I values at the other ten critical locations are below the upper shelf limit of 200 ksi (in) $^{1/2}$.

As such, the requirements of Appendix G were fully satisfied and adequate protection against nonductile failure was demonstrated.

4.2.3.4 Reactor Vessel Nozzle Base Material Evaluation

As a result of underclad cracking detected in the heat-affected zone of reactor vessel nozzle base material in another Westinghouse reactor, analyses were performed by Westinghouse to evaluate the structural integrity of the Prairie Island reactor vessel nozzles.

These fracture-mechanics-based evaluations were performed on the:

- Outlet nozzles (Reference 40),

- Safety injection nozzles (Reference 41), and

- Inlet nozzles (Reference 42).

A related report on detection of underclad cracking by immersion ultrasonic techniques was also prepared by Westinghouse (Reference 43).

Based on the evaluations performed, including critical flaw size determination and crack growth calculations, the following conclusions were reached:

- Postulated base metal flaws as large as 1/2 inch deep and 3 inches long, and considered to be continually exposed to the PWR water environment, will not become critical during the design life of the plant under any realistic loading conditions, including loss-of-coolant accident (LOCA) and large steam line break (LSB) loadings, and

- The postulated 1/2 inch base metal flaw meets the ASME Section XI integrity criteria with respect to flaw instability.

The nozzle regions considered should therefore maintain structural integrity throughout their service life, since the capability for detection of underclad cracks smaller than ½ inch is documented (Reference 43).

4.2.3.5 Other Reactor Vessel Integrity Analyses

Additional analyses of the integrity of relevant areas of the reactor vessel were performed by Westinghouse as part of an integrated reactor vessel integrity program.

In one analysis (Reference 44), a fracture mechanics evaluation of the reactor vessel belt line was performed. Two postulated faulted conditions were considered in the study, a loss-of-coolant accident (LOCA) and a large steam line break (LSB). The effects of neutron embrittlement were also considered. This report concluded that vessel integrity criteria in the belt line region were met, since the minimum critical flaws were determined to be significantly larger than 1 inch for both postulated conditions.

Another analysis (Reference 45) evaluated fatigue crack growth in the critical sections of the top head, bottom head and belt line regions using the principles of linear elastic fracture mechanics. Conclusions drawn from this report were that fatigue crack growth over the life of the plant is small (maximum 15%), and the acceptance criteria for vessel integrity are satisfied.

4.2.3.6 Pressurized Thermal Shock

The issue of pressurized thermal shock damage to reactor pressure vessels was first raised for licensees of Babcock and Wilcox operating reactors following the accident at Three Mile Island. It was subsequently incorporated into the NRC's TMI Action Plan as NUREG-0737, Item II.K.2.13, and made applicable to all plants. A summary report on reactor vessel integrity for Westinghouse operating plants was submitted to the NRC for review in response to this item (Reference 10).

On July 23, 1985 the Commission issued a final rule on analysis of potential pressurized thermal shock events. This rule added a new section, Section 50.61, to 10 CFR Part 50 which required each pressurized water reactor licensee to compute projected values of pressurized thermal shock reference temperature using equations provided in the rule and compare these values with screening criteria of 270°F for plates, forgings, and axial weld materials and 300°F for circumferential weld materials. An NRC sponsored probabilistic fracture mechanics model (Reference 48) shows that the probability of a through-wall crack is sufficiently low within the context of the rule. If a licensee does not exceed the screening criteria in the rule, sufficient margin exists so that postulated severe reactor system overcooling accidents will not result in vessel failure.

Projected values of the Prairie Island Units 1 and 2 PTS reference temperatures were originally submitted in compliance with the rule by Reference 52. Those original projected values of the PTS reference temperatures were found acceptable by the NRC in References 71 and 72.

The original rule also required that the projected assessment of the PTS reference temperature must be updated whenever changes in core loadings, surveillance measurements or other information indicate a significant change in the projected values. In response to that requirement, the NRC PTS safety evaluations for Prairie Island (References 71 and 72) requested that a reevaluation of the PTS reference temperature and comparison of the projected value be submitted with any future pressure-temperature submittals which are required by 10 CFR 50, Appendix G. A re-assessment of the PTS reference temperatures for both units was subsequently included with the License Amendment Request (Reference 73) which updated the Technical Specification heatup and cooldown curves to 15 EFPY. The NRC found the revised projected PTS reference temperature values acceptable in Reference 74.

The NRC amended 10 CFR Part 50, Section 50.61, effective June 14, 1991. That revision to 10 CFR Part 50, Section 50.61, changed the procedure for calculating the amount of radiation embrittlement that a reactor vessel receives to make it consistent with the procedure utilized in Regulatory Guide 1.99, Revision 2, published in May 1988. The PTS screening criteria, described above, were not changed by this revision to 10 CFR Part 50, Section 50.61. The results of the PTS reference calculations utilizing the revised procedures had shown that both units were well within the screening criteria.

The PTS reference temperature was reevaluated after the removal of vessel radiation
capsules S and P (References 108 and 109) and the corresponding License
Amendment (Reference 112) update for the heatup/cooldown curves to 35 EFPY. PTS
Reference temperatures were again recalculated as part of the analysis supporting the
MUR power uprate for an operating term of 54 EFPY (Reference 141). The results are 01229941
shown in Table 4.2-2. Both units continue to have a wide margin to the screening
criteria at vessel fluence levels equivalent to 54 effective full power years. Pressurized 01229941
thermal shock is therefore not a concern for either Prairie Island reactor vessel.

4.2.4 Japan-U.S. Treaty Notification Regarding PI Replacement Reactor Vessel Head and CRDMs Foreign Obligations

The replacement reactor vessel closure head and control rod drive mechanisms for Prairie Island Nuclear Generation Plant were manufactured in Japan. Consequently, as stated in the letters from the NRC dated January 13 and 24, 2005, (Reference 134 and 135) use of this equipment obligates Prairie Island to comply with certain peaceful use commitments and material tracking obligations specified in the U.S.-Japan Agreement for Peaceful Nuclear Cooperation.

First, the U.S. has agreed that this equipment will not be used for any purpose that would result in any nuclear explosive device (e.g., producing tritium for weapons program). Second, the U.S. Government has agreed that to export this equipment to a country other than Japan will require similar peaceful use assurances from the proposed recipient country.

Finally, all nuclear material used in, or produced through, the use of the reactor with this equipment will also become obligated to Japan so long as that equipment is in use. All nuclear material transactions and status reports must be adjusted accordingly.

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4.3 STEAM GENERATORS AND REACTOR COOLANT PUMPS

4.3.1 Design Basis

The design bases for the Steam Generators and Reactor Coolant Pumps are discussed in Section 4.1.

4.3.2 Steam Generators

4.3.2.1 General Description

4.3.2.1.1 Steam Generators

Each loop of the Reactor Coolant System contains a vertical shell and U-tube steam generator. A steam generator of this type is shown in Figure 4.3-1. Principal (original) design parameters are listed in Table 4.1-5. Current information with respect to tube plugging is tracked by the steam generator program.

Reactor coolant enters the inlet side of the channel head at the bottom of the steam generator through the inlet nozzle, flows through the U-tubes to an outlet channel and leaves the generator through another bottom nozzle.

The inlet and outlet channels are separated by a partition. Manways are provided to permit access to the U-tubes and moisture separating equipment.

Feedwater to the steam generator enters just above the top of the U-tubes through a feedwater ring. The water flows downward through an annulus between the tube wrapper and the shell and then upward through the tube bundle where part of it is converted to steam.

The steam-water mixture from the tube bundle passes through a steam swirl vane assembly which imparts a centrifugal motion to the mixture and separates the water droplets from the steam. The water spills over the edge of the swirl vane housing and combines with the feedwater for another pass through the tube bundle.

A moisture carryover content less than or equal to 0.10% at the steam generator outlet is achieved with two moisture separation stages: one stage of primary separators (cyclones) and one stage of dryers. The SG is fed through an internal feedwater distribution ring welded onto the feedwater nozzle and equipped with J-tubes and an anti-stratification device (helix) in order to preclude water hammer and to mitigate the thermal stratification effects. A high circulation ratio and a high blowdown capacity help in cleansing of suspended particles and reduce the potential for crud build up.

The steam generator is constructed primarily of low alloy steel. The heat transfer tubes are Inconel (Alloy 690). The interior surfaces of the channel heads and nozzles are clad with austenitic stainless steel, and the side of the tube sheet in contact with the reactor coolant is clad with Inconel. The tube to tube sheet joint is welded.

4.3.2.1.2 Steam Generator Support Structure

The steam generator is supported on a structural system consisting of four vertical columns fitted at the top and bottom with a double clevis and pin assembly. The vertical column clevis base plates are bolted to the steam generator support feet and permit movement in the horizontal plane to accommodate reactor coolant pipe thermal expansion. Horizontal restraint is accomplished at two locations. The lower lateral support is located at the support feet and the upper lateral support is located near the center of gravity below the transition cone. This combination of upper and lower supports and included stops and hydraulic shock suppressors limit and control horizontal movement for pipe rupture and seismic effects. The steam generator support structures are further described in Section 12 and shown in Figure 12.2-26.

4.3.2.2 Performance Evaluation

4.3.2.2.1 Steam Generators

Calculations confirmed that the steam generator tube sheet withstands the loading (which is a quasi-static rather than a shock loading) by loss of reactor coolant. The maximum primary membrane plus primary bending stress in the tube sheet under these conditions is 66.82 KSI for Unit 1 and 66.92 KSI for Unit 2. This is well below ASME Section III criterion 1.05 S_u = 94.5 KSI. Because the pressure in the primary channel head drops to zero under the condition postulated, no damage results to the channel head.

The rupture of primary or secondary piping has been assumed to impose a maximum pressure differential of 2,485 psig across the tubes and tubesheet from the primary side or a maximum pressure differential of 1,005 psi across the tubes and tubesheet from the secondary side, respectively.

To meet the ASME criterion, it has been established that, under the postulated accident conditions where a primary to secondary side differential pressure of 2,485 psig exists, the primary membrane stresses in the tubesheet ligaments, averaged across the ligament and through the tube sheet thickness do no exceed 0.7 S_u at 607°F (where S_u is the tensile strength in conformity with the ASME code). Furthermore, the primary membrane plus primary bending stress in the tubesheet ligaments, averaged across the ligament width at the tubesheet surface location giving maximum stress, do not exceed 1.05 S_u at 607°F. An examination of stresses under these conditions show that for the case of a 2,485 psig maximum tubesheet pressure differential the stresses are within acceptable limits. These stresses together with the corresponding stress limits are given in Table 4.3-15.

In the case of a primary pressure loss accident, the secondary-primary pressure differential can reach 1,005 psi. Table 4.3-15 shows that the criteria are met for the tubesheet.

The tubes have been designed to the requirements (including stress limitation) of Section III for normal operation, assuming 2,485 psig as the normal operation pressure differential.

Hence, the secondary pressure loss accident condition imposes non extraordinary stress on the tubes beyond that normally expected and considered in Section III requirements.

For the tubes, the risk of elastoplastic instability has been verified under an external pressure: the secondary/primary differential pressure.

The criterion, in that case, is the allowable external pressure calculated according to the paragraph NB-3133 of the ASME code and to the document Collapse of ductile heat exchange tubes with ovality under external pressure. (A. LOHMEIER, N. C. SMALL, H. J. BONIN, B. GONNET - 2nd SMIRT - vo. II part E-F BERLIN sept. 73).

For design condition, the allowable external pressure is equal to 979 psi, higher than the maximum secondary/primary differential pressure 670 psi.

For hydrotest conditions, the maximum external 1,357 psi pressure does not exceed 80% of the lowest bound collapse pressure 2,414 psi.

In faulted conditions, the following loadings have been considered to verify the criteria in conformity with the ASME code for the tube bundle:

- the impulse loads due to rarefaction waves during blowdown,
- loads due to fluid friction from mass fluid accelerations,
- loads due to the centrifugal force on U-bend regions caused by high velocity fluid motion,
- loads due to the dynamic structural response of the steam generator components and supports,
- seismic loads
- transients pressure load differentials.

The structural analysis and evaluation of the lower assembly (composed of the channel head, the tubesheet, the lower secondary shell, the partition plate and the support pads) have been performed in accordance with the requirements of the Design Specification for the loads, and in accordance with the rules of the ASME code for the criteria.

This assembly is analyzed using a three-dimensional finite element model.

The model represents the perforated region of the tubesheet as an equivalent continuous isotropic material. This simulated isotropy leads to equivalent elastic constants E* (Young's modulus), and v* (Poisson's coefficient).

The equivalent elastic constants are a function of the geometry of the holes pattern and depend on whether or not the tubes are taken into account. This analysis ignores the influence of the tubes on the total stiffness or on local stress distribution. This assumption is conservative as it leads to increased stresses in the tubesheet ligaments under the pressure loadings.

These constants are found using the document O'Donnel - A study of perforated plates with square penetration patterns - WRC Bulletin n° 124 sept. 1967.

This reference is used by the CETIM. It is a French institute which generates the curve E*/E and v* as they are in the ASME code appendix A-8000 but for a square pitch.

To determine these curves, the methodology described in the ASME code appendix A-8000 was used.

The heat which passes through the tube walls has a considerable effect on the thermal behavior of the tubesheet. In order to take this into account, a volumetric heat transfer coefficient is applied to the elements representing the perforated part of the tubesheet and surface heat transfer coefficients are applied to the elements at the surface in contact with the secondary and primary fluid. The primary transients are applied to the channel head, the partition plate and the tubesheet. The secondary transients including auxiliary or main feedwater injection with or without water recirculation, are applied to the tubesheet on the secondary face and to the low secondary shell.

To perform the primary stresses analysis (design, faulted, test conditions), the primary and secondary stresses analysis (3 S_m analysis) and the fatigue analysis in the perforated tubesheet, appropriate stress correction factors taking into account the tubesheet hole pattern have been applied to the stresses:

- stresses in the tubesheet are multiplied by the ratio p/h to take into account the lack of metal (p: the distance between two holes, h the ligament)

- for the fatigue analysis stresses are also multiplied by stress concentration factors depending on the position (the angle) around the hole. These stress concentration factors have been calculated using elementary finite element models.

The tube bundle is analyzed in accordance with the ASME code:

To cover the whole tube bundle, the straight part, the curved part of the tube and the tube/tubesheet connection have been analyzed.

For both straight and curved parts, ovality tolerances have been taken into account.

The loads considered for all service conditions are:

- primary/secondary differential pressure,
- secondary/primary differential pressure,
- hot leg temperature and primary pressure,
- secondary temperature and secondary pressure,
- steam temperature.
- displacement induced by the tubesheet,
- bending stresses due the interaction between the tubes and the tube support plates.

The analysis has been performed in accordance with the following paragraphs of the ASME code:

- NB-3221 for design conditions
- NB-3222, NB-3223 for the normal and upset conditions
- F-1331.1 for faulted conditions
- NB-3226 for test conditions

Furthermore, the risk of elastoplastic instability is verified under an external pressure: the secondary/primary differential pressure.

For the straight part, the interaction between the tube bundle and the tube support plates is analyzed. It means that the reduction of the broached hole diameter under the various loads considered on the tube support plate is found lower than the threshold clearance and to the threshold rotation.

Tabulations of significant results of the tubesheet complex are shown in Tables 4.3-13 through 4.3-16 and Figures 4.3-12 through 4.3-14. Figure 4.3-15 denotes the primary - secondary boundary component locations.

In all cases evaluated, the tubesheet complex met the stress limitations and fatigue criteria specified in the ASME code.

4.3.2.2.2 DELETED

4.3.2.3 DELETED

4.3.2.4 DELETED

4.3.2.5 DELETED

4.3.2.6 DELETED

4.3.3 Reactor Coolant Pumps

4.3.3.1 General Description

4.3.3.1.1 Reactor Coolant Pumps

Each reactor coolant loop contains a vertical single stage centrifugal pump which employs a controlled leakage seal assembly. A view of a controlled leakage pump is shown in Figure 4.3-2 and the principal design parameters for the pumps are listed in Table 4.1-6. The reactor coolant pump performance and NPSH characteristics are shown in Figure 4.3-3. The performance characteristic is common to all of the higher specific speed centrifugal pumps and the 'knee' at about 45% design flow introduces no operational restrictions since the pumps operate at full flow.

Both reactor coolant pumps will be in operation when the reactor is critical (except during low power physics test) to provide core cooling in the event that a loss of flow occurs. Cladding damage and release of fission products to the reactor coolant will not occur in the event of loss of both pumps from 100% power since the minimum calculated DNBR remains above the applicable limit (see Section 14.4.8). At power above 10%, an automatic reactor trip will occur if flow from either pump is lost. Below 10% power, a shutdown under administrative control will be made if flow from either pump is lost.

All the pressure bearing parts of the reactor coolant pump were analyzed in accordance with Article 4 of the ASME B&PV Code, Section III. This included the casing, the main flange and the main flange bolts. The analysis included pressure, thermal and cyclic stresses, and these were compared with the allowable stresses in the Code.

Mathematical models of the parts were prepared and used in the analysis which proceeds in two phases.

a. In the first phase, the design was checked against the design criteria of the ASME Code, with stress calculations using the allowable stress at design temperature. By this procedure, the shells were profiled to attain optimum metal distribution with stress levels adequate to meet the more exacting requirements of the second phase.

b. In the second phase, the interacting forces needed to maintain geometric capability between the various components were determined, and applied to the components along with the external load, to determine the final stress state of the components. This stress was also used in the fatigue analyses. These results were finally compared with the Code allowable values.

There were no other sections of the Code which were specified as areas of compliance, but where Code methods, allowable stresses, fabrication methods, etc., were applicable to a particular component, these were used to give a rigorous analysis and conservative design.

Stress Analysis Reports were prepared on these components as described in Section 4.1. These reports include the calculation of stress intensities and a summary of fatigue usage factors. These reports are a part of the plant documentation on file with the Licensee.

Reactor coolant is pumped by the impeller attached to the bottom of the rotor shaft. The coolant is drawn up through the impeller, discharged through passages in the diffuser and out through a discharge nozzle in the side of the casing. The rotor-impeller can be removed from the casing for maintenance or inspection without removing the casing from the piping. All parts of the pump in contact with the reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials.

Each pump employs a three stage hydrodynamic mechanical seal cartridge assembly 01450457
(Flowserve model N-9000) to restrict leakage along the pump shaft, thereby minimizing the leakage of water and vapor from the pump into the containment atmosphere. Although each seal stage is capable of withstanding full reactor coolant system pressure, an orifice allows flow around each of the first three stages so that during normal operation, the differential pressure across each stage is one-third of the total pressure. The flow through these orifices (termed pressure breakdown devices or PBDs) is designated controlled bleed off (CBO) or seal return flow. In addition to equalizing the pressure across the seal stages, a second purpose of CBO flow is to remove frictional heat from the rotating seal parts. CBO from the third stage orifice normally is routed back to the volume control tank (VCT), and leakage across the third stage seal face flows directly to the reactor coolant drain tank (RCDT).

The 11 and 22 Reactor Coolant Pump seals have slotted carbon stationary seal faces and flat tungsten carbide rotating seal faces. The 12 and 21 Reactor Coolant Pump
seals have Mayer Groove tungsten carbide rotating seal faces and stationary flat carbon 01450457 01497849
seal faces, installed per EC 25405 and EC 26309 respectively, which are more resistant to foreign material intrusion. The remainder of the seal assembly for all four pumps is identical.

In addition to the three primary seal stages, the N-9000 seal cartridge includes a diverse and redundant sealing device referred to as an abeyance seal. This device will limit leakage from the seal package if excessive flow from the mechanical face seals occurs due to a catastrophic failure of all three seal stages.

A portion of the high pressure water flow from the charging pumps is injected into the reactor coolant pump (RCP) between the impeller and the shaft seals described above. Part of the flow enters the Reactor Coolant System through a labyrinth seal in the lower pump shaft to serve as a buffer to keep reactor coolant from entering the upper portion of the pump. The remainder of the injection water flows along the drive shaft, through
the shaft seals described above and finally out of the pump. 01450457

Component cooling water is supplied to the RCP thermal barrier heat exchanger and the motor bearing cooler. Reference 156 states that RCP bearing and (OEM) shaft mechanical seal operating temperature limits may be exceeded during certain loss of seal injection events if the Component Cooling water flow to the RCP thermal barrier is at its design conditions of maximum temperature and minimum flow (e.g. 105 °F, 35 gpm) coincident with (OEM) No. 1 seal leakage flow <2.5 gpm. Although operation with the replacement (Flowserve N-9000) mechanical shaft seals installed is not specifically addressed, the normal operating temperature limit of the replacement seals bounds the OEM seal temperature limit and the same conclusions apply for replacement (Flowserve N-9000) seal return flow <2.5 gpm. Reference 15 concludes that RCP operation is permitted with loss of seal injection for a period of at least one hour following the onset of a loss of seal injection event provided Component Cooling water is available to the RCP Thermal Barrier Heat Exchanger. 01450457

The squirrel cage induction motor driving the pump is air cooled and has oil lubricated thrust and radial bearings. A water lubricated bearing provides radial support for the pump shaft. In addition, pump vibration is monitored as a means of early detection of mechanical abnormalities.

4.3.3.1.2 Pump Support Structure

The reactor coolant pump is supported by a structural system consisting of three vertical columns fitted at the top and bottom with double clevis and pin assembly and a system of stops. The vertical column clevis base plates are bolted to the pump support feet and permit movement in the horizontal plane to accommodate reactor coolant pipe thermal expansion. Horizontal restraint is accomplished by a combination of the tie rods and stops which limit horizontal movement for pipe rupture and seismic effects. The reactor coolant pump support structures are further described in Section 12.

The reactor coolant pumps and other components are bolted down to foundations by means of high strength bolts and nuts. Double nuts or lock nuts are furnished to prevent loosening of the nuts due to vibration.

Pinned or bolted parts of support components that are subject to pivotal action or articulation due to temperature movements are designed as non-loosening devices.

The reactor coolant pump is mounted and anchored at the three pump casing support brackets to the support column pedestal by means of high strength threaded rods at each support point.

All clevis pins are held in place by means of retainer plates bolted to each end of each pin with four high strength bolts to prevent the pins from becoming dislodged.

With the positive bolting devices provided, procedures for the surveillance of loose bolts during normal operation is not required.

4.3.3.2 Performance Evaluation

4.3.3.2.1 Reactor Coolant Pump

Extensive test programs have been conducted to develop the shaft seals described above for pressurized water reactor applications. Long term tests have been conducted on less than full scale prototype seals as well as full size cartridge seal assemblies. 01450457

For conditions of the loss-of-coolant accident it is not considered likely that the reactor coolant pump will accelerate significantly. A program to determine with certainty the behavior of reactor coolant pumps for breaks in either the suction or discharge piping near the pumps was conducted by Westinghouse in cooperation with Purdue University. Westinghouse's analytical study program was confirmed by evaluating obtained data from the test program conducted at Purdue University.

The test program established pump head loss and torque under locked rotor, free spinning and reverse flow conditions. The test was conducted with a scale model of the 93A pump with air as fluid at a pressure of 15 to 60 psia.

Flow, simulating blowdown conditions range from 500% to -100% of normal. The results of the analytical study and test program was discussed with the NRC on a generic basis.

Precautionary measures, taken to preclude missile formation from reactor coolant pump components, assure that the pumps will not produce missiles under any anticipated accident condition.

The reactor coolant pumps run at about 1200 rpm and may operate briefly at overspeeds of 109% during loss of load. For conservatism the motors were designed in accordance with NEMA standards for operation at a maximum speed of 125% of rated speed.

Each component of the reactor coolant pump motors was analyzed for missile generation. Any fragments would be contained by the heavy stator. The same conclusion applies to the pump impeller because the small fragments that might be ejected would be contained by the heavy casing.

The reactor coolant pump flywheels are shown in Figure 4.3-4. As for the pump motors, the most adverse operating condition of the flywheels is visualized to be the loss-of-load situation. The following conservative design-operation conditions preclude missile production by the pump flywheels. The wheels are fabricated from rolled, vacuum-degassed, ASTM A-533 Grade B Class 1 steel plates. Flywheel blanks are flame-cut from the plate, with allowance for exclusion of flame-affected metal. A minimum of 3 Charpy tests are made from each plate parallel (RW, longitudinal) and normal (WR, transverse) to the rolling direction. An NDTT less than + 10°F is specified. Westinghouse has a great deal of experience and data in determining fracture toughness of A533 Grade B Class 1 steel utilizing fracture mechanics specimens as well as Charpy-V specimens. Fracture mechanics specimens up to 12-inches in thickness have been tested to characterize A533 Grade B material. From Westinghouse's experience and those of others found in the literature, an empirical relationship can be established for Charpy-V data and fracture toughness data. The finished flywheels are subjected to 100% volumetric ultrasonic inspection. The finished machined bores are also subjected to magnetic particle, or liquid penetrant examination.

Acceptability of flywheel material for NSP, in comparison to Safety Guide 14 toughness criteria, can be determined by the following two steps:

a. Establish a reference curve describing the lower bound fracture toughness behavior for the material in question.

b. Use Charpy impact energy values obtained in certification tests at 10°F to fix position of the heat in question on the reference curve.

The following supplier certification data shows the Charpy V-notch test results at +10°F for NSP flywheels:

5 in. Thick Plates
Heat No. 06458

	1	2	3
Slab 2C Transverse direction (ft-lbs)	44	44	50
(2 plates) Longitudinal direction (ft-lbs)	69	65	53
Slab 2E Transverse direction (ft-lbs)	74	74	62
(2 plates) Longitudinal direction (ft-lbs)	80	83	77

8 in. Thick Plates
Heat No. 07090 Slab No. 3 (one plate)

	1	2	3
Transverse direction (ft-lbs)	65	35	44
Longitudinal direction (ft-lbs)	53	57	69

Heat No. 7442 Slab No. 1 (3 plates)

	1	2	3
Transverse direction (ft-lbs)	53	58	52
Longitudinal direction (ft-lbs)	93	79	71

A lower bound fracture toughness reference curve (see Figure 4.3-5) was constructed from dynamic fracture toughness data generated by Westinghouse (Reference 15) on A-533 Grade B Class I steel. All data points are plotted on the temperature scale relative to the NDT temperature. The construction of the lower bound below which no single test point falls, combined with the use of dynamic data when flywheel loading is essentially static, together represent a large degree of conservatism.

The applicability of a 30 ft-lb Charpy energy reference value was derived from sections on Special Mechanical Property Requirements and Tests in Article 3, Section III of the ASME Boiler and Pressure Vessel Code. The implication is that the test temperature lies a safe margin above NDTT. The NSP flywheel plates exhibit an average value greater than 30 ft-lbs in the weak direction and, therefore, met the specific requirement C.1.a stated in Safety Guide 14 that NDTT must be no higher than 10°F. Making the conservative assumption that all materials in compliance with the Code requirements are characterized by an NDT temperature of 10°F, one is able to reassign the “zero” reference temperature position in Figure 4.3-5 a value of 10°F

Flywheel operating temperature at the surface is 120°F. The lower bound toughness curve indicates a value of 116 ksi-in$^{1/2}$ at the (NDTT + 110) position corresponding to operating temperature. Safety Guide 14 requirement C.1.c is fulfilled with considerable margin for safety.

By assuming a minimum toughness at operating temperature in excess of 100 ksi- in$^{1/2}$, it can be seen by examination of the Corten and Sailors correlation in Figure 4.3-6 (Reference 117) that the Cy upper shelf energy must be in excess of 50 ft-lb, therefore, the Safety Guide 14 requirement C.1.b, that the upper shelf energy must be at least 50 ft-lb, is satisfied.

Based on the above discussion, the flywheel materials meet the Safety Guide 14 toughness criteria on the basis of supplier certification data.

Justification for the 125% overspeed has been given above. The overspeed test was conducted in accordance with the NEMA Standards Publication for Motors and Generators, Part 20, Paragraph MG 1-20.44 with the flywheel installed on the motor.

These design-fabrication techniques yield flywheels with primary stress at operating speed (shown in Figure 4.3-7) less than 50% of the minimum specified material yield strength at room temperature (100 to 150°F). The stress resulting from the press fit of the flywheel on the shaft is less than 2000 psi at zero speed, but this stress becomes zero at approximately 600 rpm because of radial expansion of the hub. Bursting speed of the flywheels has been calculated on the basis of Griffith-Irwin's results (References 16 and 17) to be 3900 rpm, more than three times the operating speed.

RCP flywheel crack growth evaluations documented in Reference 136 expanded upon previous evaluations in order to provide a basis for extending the required flywheel inspection interval from ten to twenty years. The conclusion of the evaluation was that predicted RCP flywheel crack growth is negligible (0.08″) over a 60 year life of the flywheel, even when a large initial crack length (3.26″) is assumed. This evaluation considered the following assumptions:

a. Maximum tangential stress at an assumed overspeed of 125%.

b. A crack through the thickness of the flywheel at the bore.

c. 6000 cycles of RCP starts and stops for a 60 year plant life.

The requirements for the augmented inservice inspection of the reactor coolant pump flywheels was developed per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, dated August 1975.

In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT or PT) of exposed surfaces of the removed flywheels may be conducted at 20-year intervals.

Examination procedure and acceptance criteria should be in conformance with the requirements specified in Regulatory Guide 1.14, Revision 1.

To perform an inspection of this nature requires the removal of the motor cover plate and the four plugs from the flywheel. This would permit the partial ultrasonic examination of the keyways for evidence of cracking developing at the corners by the insertion of a special ultrasonic search unit into four holes drilled through the flywheel.

An ultrasonic inspection capable of detecting at least 1/2″ deep cracks from the ends of the flywheel is more than adequate as part of a plant surveillance program. The inservice inspection program of the flywheel is given in the Inservice Inspection Program.

Installation of New RCP Internals in Unit 2 #21 RCP Casing

The #21 Reactor Coolant Pump internals were replaced with internals obtained from D. C. Cook Nuclear Plant. The D. C. Cook Nuclear Plant RCP has the same style as the original #21 RCP but with slightly different design than the original and therefore had slightly different flow characteristics.

Westinghouse has determined the potential flow imbalance by replacing the #21 RCP with a spare from D. C. Cook Nuclear Plant. It was found that no significant flow imbalance will occur as a result of this pump replacement. Before replacement, the Loop A (#21 RCP) pump flow was approximately 0.6% below Loop B (#22 RCP). The D. C. Cook spare will shift flow in Loop A to approximately 1.0% greater than Loop B.

RCS Flow Measurement Testing was performed to verify that the new #21 RCP could meet the design requirements for flow and for flow coastdown following a Reactor Coolant Pump trip (See Section 14.4.8). All the data collected from the test is summarized in Table 4.3-11. The flow results collected compared very closely with the Westinghouse predicted flows for the replacement pump. The data clearly shows the change in the flows from previous measurements (Table 4.3-12) taken in previous years. The data was also checked for difference in flow from Loop A to Loop B at the total core flow. In both cases the acceptance criteria were met. The increase in core pressure drive, which is the driving force for baffle jetting, is on the order of 0.2 psi. This increase is on a total pressure drop of 24.6 psi so that any increase in baffle jetting could be considered small.

The use of the D. C. Cook impeller in the #21 RCP has no effect on the LOCA analysis input parameters.

4.3.3.2.2 Trip of Reactor Coolant Pumps During LOCA

In response to NUREG-0737, ITEM II.K.3.5, Westinghouse, in support of the Westinghouse Owners' Group, has performed 1) an analysis of delayed reactor coolant pump trip during small-break LOCA's and 2) test predictions of LOFT experiments L3-1 and L3-6. This analysis and test predictions are documented in References 18, 19, 20 and 21. Based on the Westinghouse analysis, the prediction of the LOFT experiment L3-6 results using the Westinghouse analytical model, and Westinghouse simulator data related to operator response time, the Westinghouse and NSP position is that automatic reactor coolant pump trip is not necessary since sufficient time is available for manual tripping of the pumps.

Generic Letters 83-10 c and d contained NRC staff requirements for resolution of NUREG-0737, Item II.K.3.5. Two Westinghouse Owners Group letters OG-117 dated March 12, 1984 entitled "Justification of Manual RCP Trip for Small Break LOCA Events," and OG-110 dated December 1, 1983 entitled "Evaluation of Alternate RCP Trip Criteria," fulfilled the requirements of the Generic Letters. This methodology was approved by the NRC (Reference 53). Revision 1 to the WOG Emergency Response Guidelines contains associated procedure revisions which have been incorporated into Prairie Island procedures.

Procedures based on the Westinghouse Owners Group Emergency Response Guidelines have been implemented at Prairie Island (see Section 13.7). The RCP trip criteria adopted in the Prairie Island procedures not only assures RCP trip for all losses of primary coolant for which trip is considered necessary but also permits RCP operation to continue during most non-LOCA accidents, including steam generator tube rupture events up to the design basis double-ended steam generator tube rupture. The RCP trip criteria is based on Reactor Coolant System Pressure. Two setpoints have been determined, one for normal containment conditions and one for adverse containment conditions. The use of two setpoints permits the setpoint to be lower (less transmitter uncertainty) for the accidents that don't affect the containment environment. This assures that RCPs will not need to be tripped for non-LOCA events like steam generator tube ruptures, which don't require a RCP trip but will be tripped for small break LOCA's requiring a RCP trip (References 53 and 54).

The NRC staff found the treatment of the RCP trip criteria to be acceptable in a Safety Evaluation Report (SER) dated October 8, 1986. The SER discussed the uncertainties associated with the setpoint selection and operator training including recommendations in detail.

4.3.3.2.3 Effect of Loss of AC Power on Pump Seals

During normal operation, seal injection flow from the chemical and volume control system is provided to cool the RCP seals and the component cooling water system provides flow to the thermal barrier heat exchanger to limit the heat transfer from the reactor coolant to the RCP internals. In the event of loss of offsite power (LOOP), the RCP motor is deenergized and both of these cooling supplies are terminated. However, the diesel generators are automatically started and component cooling water to the thermal barrier heat exchanger is automatically restored. The load rejection and delayed reapplication logic will restart the component cooling water pumps within 15 seconds after the loading sequence starts. Charging pumps will be manually restarted. The NRC has concluded that this arrangement, as described in the submittal of December 29, 1981, and subsequent telecom of June 11, 1982, adequately responds to the requirements of NUREG-0737, Item II.K.3.25 (Reference 33).

The original Flowserve seal cartridge tests were run in two parts: a design verification 01450457
test of 1,458 hours and a qualification test which was run for 5,186 hours, for a total running time of 6,644 hours for the laboratory test program. This testing was performed on the cartridge stage design which is consistent with the stage design to be used at PINGP.

After the design verification test, the seal cartridge was removed, dismantled and the seal components were inspected and wear measurements were taken. The seal cartridge was then reassembled without any modification or replacement of parts, and reinstalled in the test stand for the qualification test. In addition to steady-state operating conditions a number of operational transients were run, which are typical of reactor coolant pump operations. The controlled variables in the transients were: supply temperature, shaft axial and radial position, and system pressure.

In the 6,644 hours of testing, a total of 590 transients were run. This number is comparable to the expected total number of transients that would be experienced during the 50,000 hour seal life (Note: The 50,000 hours is the original seal life, the PINGP Design Specification requires approximately 78,000 hours for the new N-9000 seals) (Reference 152).

Failures of RCP seals that could result in an uncontrolled loss of reactor coolant inventory have been a concern for the power industry and the Nuclear Regulatory Commission (NRC). Specifically, the condition when on-site emergency power failure occurs simultaneously with a failure of off-site electric power.

Flowserve completed an eight (8) hour confirmatory testing of the N-Seal due to a simulated station blackout event. The Loss-Of-Seal Coolant (LOSC) test procedure was based on two bracketing scenarios of station blackout event: one with a RCS leakage of 30 gpm, the second without RCS leakage. The test procedure combined the worst effects of both and the scenarios consisted of the following events:

- Reactor Trip
- Turbine Trip
- Coastdown of RCPs
- Failure of Auxiliary Diesel Generators
- AC Power Loss for Eight (8) Hours
- Loss of Seal Injection
- Loss of CCW
- Loss of Auxiliary Feedwater
- Controlled Bleed-Off / Seal Return Valve Open for 30 Minutes

An N-Seal cartridge designed to survive a loss-of-cooling transient, age-conditioned over 5,000 hours, successfully withstood simulated station blackout conditions which bracketed the worst features of two scenarios, for more than eight (8) hours. Its failure mode and leakage was accurately predicted and the leakage rate was approximately the same as the Controlled Bleed-Off / Seal Return flow in normal operation. (Reference 153)

The Flowserve N-Seal incorporates an abeyance seal that addresses the post-Fukushima need for zero leakage. The Abeyance Seal (shutdown seal) is a passive self-actuated device and does not rely on any additional complex sub-assemblies with small springs, pistons, etc. It is directly actuated by the pressure induced from increased leakage flow and will only come into play when there is a failure of the primary high pressure seals. In normal operation it is non–contacting and suffers no wear or degradation. Based on the abeyance seal configuration and associated testing, it is concluded the abeyance seal is capable of withstanding full temperature and pressure for up to 264 hours, with qualification test programs under real world scenarios holding for over 106 hours continuously. After the successful zero leakage condition, during pump depressurization seal leakage remained below 0.3 gpm for each pump. (Reference 154)

4.4 REACTOR PRESSURE RELIEF SYSTEM

4.4.1 Design Basis

The design bases for the reactor pressure relief system are discussed in Section 4.1.

4.4.2 Description

4.4.2.1 Pressurizer

The general arrangement of the pressurizer is shown in Figure 4.4-1, and the design data are listed in Table 4.1-4.

The pressurizer maintains the required reactor coolant pressure during steady-state operation, limits the pressure changes caused by coolant thermal expansion and contraction during normal load transients, and prevents the pressure in the Reactor Coolant System from exceeding the design pressure.

The pressurizer vessel contains replaceable direct immersion heaters, multiple safety and relief valves, a spray nozzle and interconnecting piping, valves and instrumentation. The electric heaters located in the lower section of the vessel pressurize and maintain the pressure of the Reactor Coolant System by keeping the water and steam in the pressurizer at the system saturation temperature. The original design for the pressurizer heaters (based on 1000 kw capacity) was to be capable of increasing the temperature of the pressurizer and contents at approximately 55°F/hr during startup of the reactor. It is acceptable to operate with a reduced pressurizer heater capability; which translates into a reduced heat up rate of the pressurizer and contents. The minimum requirements for the pressurizer heaters are as follows:

1. 100 KW of heater capacity is required to maintain subcooling during natural circulation operation. To accommodate a single active failure, 100 KW is required to be available in both Group A and Group B.

2. To prevent reaching a reactor trip setpoint during design operational transients (described in Section 7.2.3), 100 KW above the minimum capacity energized to support maintaining normal operation is required to be available.

Pressurizer Support Structure

The pressurizer is supported on a heavy concrete slab spanning between the concrete shield walls of its compartment. The pressurizer is a bottom-skirt supported vessel.

4.4.2.2 Pressurizer Relief Tank

Principal design parameters of the pressurizer relief tank are given in Table 4.1-4.

Steam discharged from the power relief and safety valves passes to the pressurizer relief tank which is partially filled with water at or near ambient containment conditions. The tank normally contains water in a predominantly nitrogen atmosphere. Steam is discharged under the water level to condense and cool by mixing with the water. The tank is equipped with a spray and drain which are operated to cool the tank following a discharge.

Pressure, temperature and level of the Pressurizer Relief Tank are indicated in the control room.

4.4.2.3 Pressure Relieving Devices

4.4.2.3.1 Devices Within Reactor Coolant Pressure Boundary

The Reactor Coolant System is protected against overpressure by control and protective circuits such as the high pressure trip and by code relief valves connected to the top head of the pressurizer. The relief valves discharge into the pressurizer relief tank which condenses and collects the valve effluent. The schematic arrangement of the relief devices is shown in Figures 4.1-1A and 4.1-1B, and the valve design parameters are given in Table 4.1-4. Valve sizes are determined as indicated in Section 4.4.3.2.

Power-operated relief valves (PORVs) and code safety valves are provided to protect against pressure surges which are beyond the pressure limiting capacity of the pressurizer spray.

The PORVs are spring-loaded-closed. Air is required to open the valves, which are supplied by a control air source. To assure ability of the valves to open upon loss of control air, a backup air supply is provided. The backup air supply consists of a Seismic Category I passive air accumulator for each PORV, which is supplemented by temporary compressed air bottles when the Over Pressure Protection System (OPPS) is in service. Check valves in the existing air lines prevent depressurizing the accumulators in the event of a ruptured air line. A relief valve installed on the backup air supply piping prevents overpressurization of the PORV actuator and accumulator. 01497161

The PORVs have direct position indication in the control room. An acoustic monitoring system was also installed to provide clear indication of valve position and redundancy to ensure reliable indication (see section 4.4.2.4).

4.4.2.3.2 Design and Installation Criteria for Mounting of Devices

The design and installation criteria for the mounting of the pressure-relieving devices within the reactor coolant pressure boundary and on the main steam lines outside of containment is in accordance with the applicable portions of the following codes:

ASME Section I	Power Boilers
ANSI B31.1.0	Power Piping Code
ASME Section III	Nuclear Power Plant Components

In designing the proper support systems for the pressure-relieving devices, a complete thermal, dead weight and dynamic analysis was performed and the stresses are within the code allowable values as set forth in Section 12. Calculations to determine the discharge forces assumed that all of those pressure-relieving devices which are connected to the same header (or have any significant influence by discharging simultaneously) would discharge concurrently at their maximum relieving capacity. The full discharge loads and all resulting forces, bending moments and torsional moments imposed on the connecting piping were analyzed. The resulting stress levels were maintained below the code allowable values by reinforcing and restraining the pipe as required.

4.4.2.3.3 Main Steam Line Safety and Relief Valves

In evaluating the design of the main steam line safety valves and headers, the only concern is for the moment induced on the valve and hence the nozzle by the discharge force. The manufacturer of the valve assures that the valve body and flange connection will satisfactorily withstand this loading.

The main steam safety valves were analyzed by two different methods:

a. A static analysis was performed to check the header and nozzle connection using loads which included a dynamic load factor of 2 on the discharge force and combined the resultant stress with pressure, weight and OBE seismic stresses.

b. A dynamic analysis was also performed which took into consideration the stiffness of the system and its response to the valve discharge load assuming a valve life time of 50 milliseconds. The resultant dynamic load factor was applied to the valve discharge force and the resultant stresses were combined with the appropriate weight, pressure and OBE seismic stresses.

The results of these analyses led to the design of the branch to header connection utilizing forced insert nozzles shown in Figure 4.4-2. These special fittings were designed specifically for this type of service. They are internally reinforced and designed to develop maximum sectional strength in the high stress transitional area. In addition, the fittings are of the contoured insert type of design and as such have the lowest possible code stress intensification factor for this type of connection.

With these fittings and applying conservative dynamic load and valve flow rate factors of 0.2 and 1.25 respectively, the calculated stresses in the branch to header connection are well below code allowable values.

In evaluating the design of the main steam line power-operated relief valve (PORV) piping, it can be seen from the sketch Figure 4.4-3 that the discharge loads of the PORVs are not transmitted back to the main steam header. Instead, they are transmitted directly to the floor slab through a snubber which is attached to the discharge elbow. The snubbers have been designed to allow for the free expansion of the piping, but provide a positive locking action for the continuous discharge loads of the relief valves.

4.4.2.4 Acoustic Monitoring System

The acoustic monitoring system indicates the position of the pressurizer safety valves and the PORVs. It provides a rapid means of detecting flow through the safety valves and the PORVs. The acoustic monitors are installed on the common discharge of the safety valves and the inlets for each PORV. The system has outputs to individual status lights and common annunciator alarm on the control room, and to the plant computer system. The acoustic monitoring system instrumentation is seismically qualified consistent with the component or system to which it is attached.

4.4.3 Performance Analysis

4.4.3.1 Pressurizer

The pressurizer is designed to accommodate positive and negative surges caused by load transients. The surge line which is attached to the bottom of the pressurizer connects the pressurizer to the hot leg of a reactor coolant loop. During a positive surge, caused by a decrease in plant load, the spray system, which is fed from the cold leg of the coolant loops, condenses steam in the pressurizer vessel to prevent the pressure from reaching the set point of the power operated relief valves. Power operated spray valves on the pressurizer spray line limit the pressure during load transients. In addition the spray valves can be operated manually from the control room. A small continuous spray flow is provided to assure that the pressurizer liquid is homogeneous with the coolant and to prevent excess cooling of the spray piping.

During a negative pressure surge, caused by an increase in plant load, flashing of water to steam and generation of steam by automatic actuation of the heaters keep the pressure above the minimum allowable limit. Heaters are also energized on high water level during positive surges to heat the subcooled surge water entering the pressurizer from the reactor coolant loop.

It has been demonstrated by test that one group of pressurizer heaters is more than sufficient to maintain natural circulation upon loss of reactor coolant pumps. Evaluation has shown that one group of pressurizer heaters with at least 100 kw of capacity is sufficient to maintain natural circulation. Backup Group “A” Pressurizer Heaters are connected to the 480V Train A Safeguards Buses and are sequenced on by the voltage restoration scheme after loss of offsite power. The Backup Group “B” Pressurizer Heaters are normally energized by the “B” Train Safeguards bus but may be transferred to a Non-Safeguards bus by a manual transfer switch located in the Rod Drive Power Supply Room which is located within 1 minute of the control room. The pressurizer heater groups are connected to the emergency 480 volt buses through safety grade circuit breakers.

The pressurizer is constructed primarily of low alloy steel with internal surfaces clad with austenitic stainless steel. The safe ends of the pressurizer were cut off in the field and replaced with type 316L material to avoid partial or local severe sensitization of austenitic stainless steel. The heaters are sheathed in austenitic stainless steel. 01450457

The pressurizer vessel surge nozzle is protected from thermal shock by a thermal sleeve. A thermal sleeve also protects the pressurizer spray nozzle connection.

The pressurizer relief tank size is based on the requirement to condense and cool a discharge from the pressurizer equivalent to 110% of the steam in the pressurizer at full power.

The pressurizer relief tank is protected against a discharge exceeding the design value by a rupture disc which discharges into the reactor containment. The rupture disc relief conditions are given in Table 4.1-4. The rupture disc on the relief tank has a relief capacity at least equal to the combined capacity of the pressurizer safety valves. The tank design pressure (and the rupture disc setting) is twice the calculated pressure resulting from the maximum safety valve discharge described above. The noncondensible gases in the pressurizer may raise the pressure in the tank enough to rupture the disc if the valves are open for a significant period of time. The tank and rupture disc holder are also designed for full vacuum to prevent tank collapse if the tank contents cool without nitrogen being added.

The discharge piping from the safety and relief valves to the relief tank is sufficiently large to prevent backpressure at the safety valves from exceeding 20 percent of the set point pressure at full flow.

In cases where the pressurizer safety valves open, the power relief valves could also be open. The pressurizer relief tank would not be overpressurized if the safety and relief valves simultaneously discharged into the tank. The pressurizer relief tank is not capable of accepting continuous discharge of steam from the pressurizer without rupture of the tank disc. The rupture disc is sized to relieve the maximum possible steam flow rate resulting from the maximum surge rate into the pressurizer. This results in the disc's capacity being equivalent to that of the safety valves.

The tank is constructed of carbon steel and lined with corrosion resistant coating.

4.4.3.2 Pressure Relief

The Reactor Coolant System is protected against overpressure by safety valves located on the top of the pressurizer. The safety valves on the pressurizer are sized to prevent system pressure from exceeding the design pressure by more than 10 percent, in accordance with Section III of the ASME Boiler and Pressure Vessel with the 1968 Winter Addendum. The capacity of the pressurizer safety valves is determined from consideration of: (1) the reactor protection system; and (2) accident or transient conditions which may potentially cause overpressure.

The pressurizer safety valve Technical Specification originally required lift setpoints to be within ±1% of the specified setpoint. The Specification was difficult to meet when test instrument error and repeatability were considered. A License Amendment Request justified increasing the as-found setpoint tolerance to ±3%, provided the setpoint was returned to ±1% following test. License Amendments 123 and 116 approving the request were issued May 21, 1996.

The combined capacity of the safety valves is equal to or greater than the maximum surge rate resulting from complete loss of load without a direct reactor trip or any other control, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve setting. Reference 27 provides a complete analysis of the protection against overpressurization of either the Reactor Coolant System or the Steam System. This report shows that, on a generic basis, following a complete loss of heat sink, i.e. steam flow to the turbine, protection against overpressure is afforded by the pressurizer and steam generator safety valves along with any of a number of reactor trip functions. The Reference 27 analysis was performed for a 4-loop PWR with justification provided for how it bounds a 2-loop PWR such as Prairie Island.

4.4.3.2.1 Probability of a Small-Break LOCA Caused by a Stuck-Open PORV

The probability and consequences of a small break LOCA caused by a transient-related PORV opening and failure to reclose were analyzed by Westinghouse in support of the Westinghouse Owners Group. The analysis in response to NUREG-0737, Item II.K.3.2, is documented in WCAP-9804 (Reference 28).

Data concerning the operation of pressurizer PORVs and safety valves in Westinghouse NSSS plants as well as data concerning modifications made to reduce the likelihood of PORV LOCA was collected and compiled in this report. A probabilistic analysis was performed utilizing representative initiating event frequency data and probability data for transient characteristics, equipment reliability and human reliability to show the effect of the post-TMI modifications made to reduce the likelihood of a PORV LOCA. An evaluation of a conceptual automatic PORV isolation system was also performed. The analysis included sensitivity studies to accommodate differences among plants including PORV block probability, safety injection system design, and operator reliability, rendering the probabilistic analysis generically applicable to all Westinghouse NSSS plants.

The results of the analysis and the included sensitivity studies showed that, generically, an approximate 80% reduction in PORV LOCA frequency was observed as the result of post-TMI modifications. Also, it was shown that neither the PORV nor the safety valve LOCA were significant contributors to the overall small break LOCA frequency.

The NRC Staff Safety Evaluation report closing out this item (Reference 46) found that the requirements of NUREG-0737 Item II.K.3.2 are met with the existing PORV, safety valve and reactor high-pressure trip setpoints and that an automatic PORV isolation system is not required for Prairie Island.

4.4.3.2.2 Relief, Safety and Block Valves and Associated Piping

NUREG-0737, Item II.D.1, required licensees to conduct testing to qualify the performance of reactor coolant system relief, safety and block valves and associated piping under expected operating conditions for design-basis transients and accidents.

Full flow tests of safety and relief valves were performed by the Electric Power Research Institute (EPRI) under the direction of the Westinghouse Owner’s Group. The results of those tests were evaluated and formed the basis for determinations of the adequacy of relief and safety valves installed in Westinghouse plants.

An interim, plant-specific report was submitted to the NRC on July 1, 1982 (Reference 34). This report described the relief and safety valve installation at Prairie Island, and the completed EPRI generic test program documentation and test data. Documentation resolving the issue of utilization and performance of PORV block valves was also described. In addition, the methodology to be used in the plant-specific piping and support evaluation and a summary of that project’s status was presented.

The completed Westinghouse plant-specific report on safety and relief valve performance was submitted to the NRC on July 19, 1982 (Reference 35). A number of piping and support modifications were completed on both Unit 1 and 2. The Westinghouse report demonstrated that the valves, piping arrangements (as modified), and fluid inlet conditions at Prairie Island were bounded by those test parameters in the EPRI generic test program. As such, the EPRI tests confirmed the ability of the safety and relief valves to open and close under the expected operating fluid conditions.

Additional information was requested by the NRC related to the analyses performed for Prairie Island. This information was provided in Reference 55. By letter dated January 15, 1987 (Reference 67), final resolution of NUREG-0737, Item II.D.1 was issued by the NRC. For resolution of this item procedural controls were implemented to assure that the safety valves are inspected and any necessary maintenance performed after each valve lift while installed in the system.

4.4.3.3 Low Temperature Overpressurization Mitigation

A Low Temperature Overpressure Protection System (OPPS) has been installed on Prairie Island Units 1 and 2 (Reference 75). The purpose of the system is to ensure the reactor coolant system pressure/temperature limits are not exceeded during pressure transient events when the reactor coolant system is at low temperatures and more subject to brittle fracture. The system consists of modified actuation circuitry for the pressurizer power operated relief valves to provide a low pressure setpoint during plant startup and shutdown conditions. When the reactor vessel is at low temperature, the pressurizer power operated relief valves are placed in the low pressure set point mode. Pressure transients will be terminated by relief valve operation before exceeding 110% of the pressure/temperature limits established per the requirements of 10 CFR Part 50, Appendix G. A keylock switch is used to enable and disable the low pressure set point of the pressurizer power operated relief valves. Backup air supply accumulators, supplemented by compressed air bottles when OPPS is in service, have also been provided for each pressurizer power operated relief valve.

The Low Temperature Overpressure Protection System is described in Reference 75. The system consists of a control board enable/disable keylock switch, a control room annunciator for improper conditions, backup air accumulators for each pressurizer power operated relief valve including air check valves to assure they do not depressurize on an upstream airline break, and pressure switches/gauges to monitor accumulator condition. The electrical portions of the Low Temperature Overpressure Protection System have been designed to IEEE 279-1971 and seismic category I criteria with the exception of the position alarm on the enable/disable switch. Two independent trains with full testability are provided. Each valve's air accumulator is designed to hold approximately 36 cubic feet of air. When supplemented by compressed air bottles, the backup air system allows at least 174 full strokes of each 01404424
PORV in a ten minute period without crediting station air (Reference 131). This number of strokes is based on the requirements during the design basis low temperature overpressure mass injection event with one charging pump and one SI pump injecting and letdown isolated. The backup air supply is Seismic Category I.

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The pressurizer power operated relief valve low pressure setpoints are enabled by actuating a keylock switch on the control board. A status light is provided on the main control board that informs the control room operator that the Low Temperature Overpressure Protection System is enabled. The plant process computer generates an alarm on a control room annunciator panel (OPPS Improper Alarm) to alert the control room operator that the Low Temperature Overpressure Protection System may be in a condition that would prevent it from operating within its design basis. The specific conditions that will result in the OPPS Improper alarm on the main control board are:

a. The control board keylock switch in the enable position with either of the pressurizer power operated relief valve motor operated isolation valves not fully open.

b. The control board keylock switch in the enable position with reactor coolant system pressure greater than 450 psig.

c. The control board switch in the disable position with the reactor coolant system temperature less than 310°F.

d. Either pressurizer power operated relief valve air accumulator pressure less than 65 psig.

The pressurizer power operated relief valve low pressure setpoint is enabled whenever the reactor coolant system temperature is below the Over Pressure Protection System (OPPS) Enable Temperature. The OPPS Enable Temperature is the RCS water temperature corresponding to a vessel metal temperature at the controlling beltline location such that the metal condition is well up the transition curve from brittle to ductile properties. This provides high assurance that brittle fracture will not occur. The analytical method used to determine the value is contained in Reference 107. The OPPS Enable Temperature is determined each time the reactor coolant system heatup and cooldown curves are evaluated per 10 CFR Part 50 Appendix G.

Above the OPPS Enable Temperature the pressurizer safety valves limit reactor coolant system overpressure events to <10% above design pressure. Below the OPPS Enable Temperature, another method of relieving reactor coolant system pressure must be utilized to ensure that the 10 CFR Part 50 Appendix G pressure/temperature limits are not exceeded. This is accomplished by lowering the pressurizer power operated relief valve setpoint. Like the OPPS Enable Temperature, the pressurizer power operated relief valve low pressure setpoint is also determined each time the heatup and cooldown curves are evaluated per 10 CFR Part 50 Appendix G. The current OPPS Enable Temperature of 310°F and pressurizer power operated relief valve low pressure setpoint of 500 psig are the result of an analysis utilizing the current 10 CFR Part 50 Appendix G pressure/temperature limitation curves. See section 4.7.2 for further discussion regarding the pressure/temperature curves. 01451781 01404424

The current Low Temperature Overpressure Protection System setpoint analyses were completed in Reference 132 using the methodology described in Reference 107. The 01404424 methodology specifically addressed two pressure transient events, a Mass Input Event and a Heat Input Event. The mass input event is a reactor coolant system overpressurization resulting from the injection of water into a water solid reactor coolant system. The heat input event is the addition of heat to a water solid reactor coolant system which results in water expansion and a simultaneous increase in reactor coolant system pressure. The analysis methodology described in Reference 107 was used to calculate the peak reactor coolant system pressure for each of these types of events for a specific OPPS system setpoint. The analysis specifically calculated the reactor coolant system pressure overshoot above the low pressure setpoint of 500 psig for limiting heat input and mass input events using plant specific information, including instrument uncertainties (Reference 132). The resulting maximum calculated reactor coolant system pressures were then verified to be less than 110% of the current (54 01404424 EFPY), most limiting steady state curve (Unit 1), 10 CFR Part 50 Appendix G 01451781 pressure/temperature limitations or less than 800 psig at the PORV discharge piping, whichever is more limiting. These comparisons showed that the 500 psig setpoint is acceptable.

The OPPS setpoint analysis applied a 10% relaxation of the Appendix G limitations based upon that allowed by ASME Code Case N-514 (Reference 111). The 800 psig pressure limit for the PORV piping is based upon bounding water hammer conditions due to rapid opening/closing of the valves in water solid conditions (Reference 107).

Technical Specification, the PTLR and procedural restrictions cause the Mass Input analysis to be completed in three distinct temperature ranges. The Heat Input event is presented as a continuous function since it is a function of initial reactor coolant system temperature at the start of the event. In both cases, the limiting reactor coolant system pressure/temperature limitation curve is the steady state curve due to the fact that overpressure events most likely occur during isothermal conditions (Reference 107). The basic assumptions and reactor coolant system (RCS) temperature ranges for each of the mass input and heat input events are listed below:

I. MASS INPUT EVENT

a. RCS < 200°F

Below 218°F*, the Technical Specifications in conjunction with the PTLR require both SI pumps to be incapable of injecting into the RCS. Therefore, the limiting case is a letdown and RHR isolation with all three charging pumps in service delivering the maximum flow the system is capable of.

b. 200°F ≤ RCS ≤ 310°F 01404424

Below 310°F, the Technical Specifications in conjunction with PTLR require at least one SI pump to be incapable of injecting into the RCS. Therefore, between 218°F*, and 310°F, the limiting case is an unplanned start of one SI pump and all three charging pumps simultaneously. This condition would also be in conjunction with an isolation of both letdown and RHR.

c. RCS > 310°F

There are no restrictions because the pressurizer safety valves would lift to prevent reactor coolant system overpressurization.

* These ranges include instrument uncertainty margin as described in the PTLR.

II. HEAT INPUT EVENT

The limiting credible heat input event is the starting of a reactor coolant pump in an idle reactor coolant system loop where the steam generator is at a higher temperature than the reactor coolant system. Technical Specifications only allow a reactor coolant pump to be started at reactor coolant system temperatures less than 310°F if there is a steam or gas bubble in the pressurizer or if the temperature difference between the reactor coolant system and the steam generator in the affected loop is less than 50°F. Therefore, the limiting heat input case is a 50°F temperature difference between a steam generator and the reactor coolant system.

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4.5 REACTOR COOLANT GAS VENT SYSTEM

4.5.1 Design Basis

The Reactor Coolant Gas Vent System (RCGVS) is designed to be used to remotely vent non-condensible gases from the reactor vessel head and pressurizer steam space during post-accident situations when large quantities of non-condensible gases may collect in these high points. The purpose of venting is to prevent possible interference with core cooling. Pressure instrumentation is included in the design to monitor system leakage during normal plant operation. The RCGVS satisfies the requirements of NUREG-0737, Item II.B.1 (See References 36, 37 and 47).

When Instrument Air and the supported normal and excess letdown flowpaths are unavailable (e.g. during a seismic event or fire), the RCGVS may be used to maintain RCS inventory control by providing an alternate letdown flowpath. Under these conditions, the RCGVS supports the plant's ability to achieve and maintain safe shutdown.

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Although designed for accident conditions, the system may be used to aid in the pre- or post-refueling venting of the reactor coolant system. Pressurizer and reactor vessel venting can be accomplished with the system if desired. Vent flow would be directed to the PRT or waste gas system for this operation to prevent inadvertent release of radioactive fluid to the containment. However, venting of the individual CRDMs and RCPs, if necessary, cannot be accomplished with the system.

Portions of the RCGVS system are designed to ASME B & PVC Section III Division I, Code Class 1 & 2.

4.5.2 Description

The system, consisting of six remotely operated solenoid valves, is designed to permit the operator to vent the reactor vessel head or pressurizer steam space from the control room under post-accident conditions, and is operable following all design basis events except those requiring evacuation of the control room. The vent path from either the pressurizer or reactor vessel head is single active failure proof with active components powered from emergency power sources. Parallel valves powered off alternate power sources are provided at both vent sources to assure a vent path exists in the event of a single failure of either a valve or the power source. The system provides a redundant vent path either to the containment directly or to the PRT. The PRT route allows removal of the gas from the RCS without the need to release the highly radioactive fluid into containment. Use of the PRT provides a discharge location which can be used to store small quantities of gas without influencing containment hydrogen concentration levels. However, venting large quantities of gas to the PRT will result in rupture of the PRT rupture disc providing a second path to containment for vented gas.

Cooling of gas vented to the PRT is provided by introducing the gas below the PRT liquid volume. The direct vent path is located to take advantage of mixing and cooling in the containment.

As shown in Figure 4.1-1A and 4.1-1B, non-condensible gases are removed from either the pressurizer or reactor vessel through the flow restricting orifice and one of the parallel isolation valves and delivered to the PRT or containment via their isolation valves.

System Parameters

Flow	>100 scfm H_2, dependent upon RCS pressure and temperature
Design Temperature	700°F
Design Pressure	2500 psia
Line Size	1″

Piping and Valves

All piping and valves used in the RCGVS are either type 304 or type 316, austenitic stainless steel or equivalent. Socket welded connections are used throughout except where disassembly for maintenance, particularly refueling operations, is required. The system is designed for 2500 psia and 700°F and compatible with superheated steam, steam/water mixtures, fission gas, helium, nitrogen and hydrogen. A 7/32″ x 1″ stainless steel flow restriction orifice is provided in each vent path to limit reactor coolant leakage to less than one charging pump in the event of a line break or inadvertent operation.

Instrumentation

Pressure indication in the vent line downstream of the reactor vessel head and pressurizer isolation valves is provided to detect leakage past any of these valves during normal power operations. This instrument is not required to function during post-accident conditions and is therefore not provided with emergency power.

Open/close position indication for all remotely operated solenoid valves is provided in the control room.

4.5.3 Performance Evaluation

The RCGVS may be required to operate during post-accident situations to remove non-condensible gases from the RCS. To assure operability under those conditions, the components of the system required to perform venting operations have been designed to operate under post-accident environmental conditions. They are provided with emergency power sources. Since the system is allowed to vent to containment there is no requirement to maintain system integrity following a LOCA. Thus the structural analysis of the system does not include LOCA loads for a Design Class 1 component per USAR Table 12.2-11. Parallel valves assure a vent flow path to containment in the event of single active failure. 01491095

The RCGVS is not required to operate during normal power operation. Operator training and normal administrative controls minimize the possibility of an inadvertent actuation of the system. Should inadvertent operation occur, a flow limiting orifice is provided to limit mass loss from the RCS to less than a single charging pump. Also pressure indication provided in the control room enables the operator to recognize any discharge of reactor coolant through the vent system.

Post-Accident System Operation

In the unlikely event that an accident results in the generation of significant quantities of non-condensible gases within the RCS, the RCGVS can be used to remove the gases from the RCS. Plant operating guidelines provide detailed instructions concerning the detection of the presence of non-condensible gas and the determination of the duration of venting.

Failure Modes and Effects

Table 4.5-1 provides the failure modes and effects of those failures upon the RCGVS.

4.5.4 Reactor Coolant Gas Vent System Jumper

A RCGVS jumper line between the Reactor Vessel head and the Pressurizer is installed to provide a continuous vent path to prevent gas accumulation in the RV Head during Mode 5, Cold Shutdown, and to aid filling and venting during startup, without using the RCGVS Solenoid Valves. The jumper was added to eliminate Mode 5, Cold Shutdown cycling and operations that maintain the RCGVS Solenoid valves energized, open, and dry for long periods of time. This permanently installed jumper tubing utilizes a portion of RCGVS piping and is illustrated in Figures 4.5-4 and 4.5-5. Operating Procedures connect the jumper to the RCGVS system during Mode 5, Cold Shutdown, and disconnect it after fill and vent, prior to leaving Mode 5, Cold Shutdown.

Jumper design condition (800 psig @ 450°F) exceeds RCS Mode 5, Cold Shutdown, fill and vent with RCP jogs, and postulated RCS Natural Circulation from draindown loss of cooling, conditions. Jumper component allowable pressure at ambient and Mode 5, Cold Shutdown RCS temperatures is consistent with RCS design pressure; however, temperature increases derate the components to the design conditions specified above. Additionally, the Low Temperature Overpressure Protection System is in service whenever the jumper is connected.

Design, materials and installation comply with ANSI B31.1. Austenitic Stainless Steel tubing and Swagelok fittings are used which are suitable for contact with RCS fluid. Materials are QA Type III with Design Class 1* mounting and analysis to alleviate seismic concerns.

4.6 PIPING, INSTRUMENTATION AND VALVES

4.6.1 Description

4.6.1.1 Piping

The general arrangement of the Reactor Coolant System piping is shown on the plant layout drawings in Section 1. Piping design data are presented in Table 4.1-7.

The reactor coolant piping layout is designed on the basis of providing ‘‘floating” supports for the steam generator and reactor coolant pump in order to absorb the thermal expansion from the fixed or anchored reactor vessel.

The austenitic stainless steel reactor coolant piping and fittings which make up the loops are 29 in. ID in the hot legs, 27-1/2 in. ID in the cold legs and 31 in. ID between each loop’s steam generator outlet and its reactor coolant pump suction. Unit 2 piping is centrifugally cast ASTM A351 CF8M material. The Unit 1 reactor coolant loop pipe material is seamless, forged, ASTM A376 Type 316. Smaller piping, including the pressurizer spray and relief lines, drains and connections to other systems are austenitic stainless steel.

Unisolable sections of piping connected to the Reactor Coolant System have been evaluated for potential temperature distributions or oscillations which could cause unacceptable thermal stresses. This evaluation was initially performed in response to Bulletin 88-08 (Reference 94, 101, 80 and 83). Subsequent industry evaluation of the associated phenomena has resulted in the publication of updated guidance. Prairie Island has committed to implementation of a program to provide continuing assurance that unisolable sections of all piping connected to the RCS will not be subjected to combined cyclic and static thermal and other stresses that could cause fatigue failure during the remaining life of the unit. This program is aligned with industry NEI 03-08, “Guideline for the Management of Materials Issues”, “Needed” elements for management of thermal fatigue in normally stagnant non-isolable reactor coolant system branch lines.

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All joints and connections are welded except for stainless steel flange connections to the pressurizer relief tank and the connections at the safety valves.

Thermal sleeves are installed at the following locations where high thermal stresses could otherwise develop due to rapid changes in fluid temperature during normal operational transients:

a. Return line from the residual heat removal loop.

b. Both ends of the pressurizer surge line.

c. Pressurizer spray line connection to the pressurizer.

d. Charging line connections.

4.6.1.2 Valves

All valve surfaces in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials. Connections to stainless steel piping are welded.

The potential for valve stem leakage is reduced by monitoring valves for leakage, early detection, and repacking without leakoff per current maintenance standards. Post maintenance testing, inservice system leak tests, periodic walkdown surveillances, radiation monitoring, fluid inventory surveillances, and leakage control programs address early leak detection and repair.

The only valves manufactured outside the United States were valves RC-2-1 (2RC2-1)/8001A and RC-2-2 (2RC-2-2)/8001B on the Bypass Manifold. These were fabricated by Velan Engineering Company, Montreal, Canada.

The manufacturer's qualifications are as follows:

a. Velan's capabilities are evaluated in the same manner as domestic plants to assure that they are able to manufacture quality materials.

b. All specifications used in procurement are identical to the specifications utilized for domestic procurement.

c. Extensive quality assurance coverage is maintained to assure compliance with specifications. Valves are manufactured within the quality assurance program as established by Velan to the satisfaction of Westinghouse.

d. Velan supplies valves for nuclear power plant suppliers other than Westinghouse.

e. Velan has supplied valves for other Westinghouse units and has an "N" stamp.

f. Velan obtains materials from domestic (U.S.) suppliers.

4.6.1.2.1 Pressure Isolation Valves (PIV)

RCS PIVs are two normally closed valves in series within the reactor coolant pressure boundary, which separate the high pressure RCS from an attached low pressure system. The purpose of the PIVs is to prevent overpressure failure of the low pressure system. To assure that this purpose is met, the leakage through the PIVs is limited by the Technical Specifications. The following valves are the PIVs required by Technical Specifications:

<u>RHR to Loop B accumulator injection line</u>

SI-6-2 (2SI-6-2)

<u>SI to Upper Plenum</u>

SI-9-3 (2SI-9-3)
SI-9-4 (2SI-9-4)
SI-9-5 (2SI-9-5)
SI-9-6 (2SI-9-6)

Response to Generic Letter (GL) 87-06 expanded the list based on the criteria described in the GL 87-06. Per Commitment Change 96-03, this response was rescinded. However, these additional valves that have been evaluated and found to meet the definition of Pressure Isolation Valves using the guidance in GL 87-06 are listed below. These valves are also leak tested in accordance with the IST Program and Technical Specification Section 3.4.15. 01472364

<u>SI Accumulator Injection Line</u>

SI-6-1 (2SI-6-1)

SI-6-3 (2SI-6-3)

SI-6-4 (2SI-6-4)

<u>RHR Injection line to RCS</u>

MV-32066 (MV-32169)

<u>RHR Suction Line</u> 01444254

MV-32164 (MV-32192)

MV-32165 (MV-32193)

MV-32230 (MV-32232)

MV-32231 (MV-32233)

4.6.1.3 Reactor Coolant Flow Measurements

Elbow taps are used in the Reactor Coolant System as an instrument device that indicates the status of the reactor coolant flow. The basic function of this device is to provide information as to whether or not a reduction in flow rate has occurred. The correlation between flow reduction and elbow tap read out has been well established by the following equation (Reference 29):

$$\frac{\Delta P}{\Delta P_0} = \left[\frac{w}{w_0}\right]^2$$

where ΔP_0 is the pressure differential with the corresponding referenced flow rate w_0 and ΔP is the pressure differential with the corresponding flow rate w. The full flow reference point is established during initial plant startup. The low flow trip point is then established by extrapolating along the correlation curve. The technique has been well established in providing core protection against low coolant flow in Westinghouse PWR plants. The expected absolute accuracy of the channel is within ±10% and field results have shown the repeatability of the trip point to be within ±1%. As a result of the calibration techniques used, the absolute accuracy of the coolant flow measurement is not relevant. As indicated in Section 14, the limiting trip setpoint assumed for analysis was 87% loop flow. This represents a 4% of flow allowance below the setpoint of ≥91% which is specified in the Technical Specifications. Since the trip point is calibrated as a function of full flow output of the instrument and since the flow rate of the reactor is verified during startup testing to be equal or greater than the design flow rate listed in Table 4.1-2 which is the initial flow used for the safety analysis, the actual trip point would be 90% based on the 1% repeatability. Westinghouse has concluded that a more accurate measurement of Reactor Coolant System flow is not required for either plant operation or safety.

Startup tests provided a means for verifying that reactor coolant flow is equal to or greater than the design flow rate. The core flow rate was verified with an accuracy better than 10% by correlating a secondary system heat balance and the inlet and outlet core temperatures. In addition measurements of pump input power and loop ΔP were made at hot shutdown condition for various configurations of running pumps (A pump running, B pump running and both pumps running), the absolute flow rate of each pump is verified to be greater than the design flow.

A blocked or plugged common instrument line to the three redundant reactor coolant flow instruments will produce a low flow indication on the control board for the affected RCS loop. If reactor power is above permissive eight (P-8), that is, 10% full power, a reactor trip will occur. If the power level is below P-8, the operator will bring the plant into Mode 3, Hot Standby as per stipulated in Table 3.3.1-1 of the Technical Specifications. No safety problem occurs as a result of this condition; the core is maintained well within all operational safety limits.

A rupture of the common instrument line to the reactor coolant loop flow instrument will also be indicated as a low flow condition in the affected loop. Reactor trip will occur if power level is above 10%; manual shutdown below 10% will be provided exactly as in the case of a blocked line. A ruptured instrument line is a type of loss-of-coolant accident discussed in Section 14.7.

Reactor coolant system flow anomalies resulting in simultaneous changes to other reactor coolant system parameters and nuclear instrumentation system parameters have been reported at other Westinghouse plants. These anomalies were first observed in November, 1986. Westinghouse, in conjunction with owners of several operating plants, took test data to determine the nature and magnitude of these flow disturbances. Analysis of the data resulted in the conclusion that these flow disturbances do not occur in the Prairie Island units. Westinghouse prepared an investigation report on this subject (Reference 79).

4.6.1.4 Pump Power-Differential Pressure

This procedure has been used experimentally in an existing plant. The results have produced calculated flowrates in close agreement with the analytically predicted most probable flow and consistent flowrates to within $\pm$.3% for a number of pumps. It is a refinement of the pump power method that utilizes a procedure to establish the actual operating curve from its known shape, determined from model tests, by interrelating pump input power and a relative change in system pressure drop under conditions of one and two pumps running. This procedure reduces the uncertainties associated with the absolute relation of the pump input power curve and flow. This procedure is described in more detail than the more familiar mentioned previously. Figure 4.6-1 is an example of a typical pump input power curve and is included to describe the procedure which is as follows:

a. With the reactor coolant system pressurized, all pumps are started. The flow within the loop to be measured is assumed to be equal to the design (represented by line 1 on Figure 4.6-1) and pump power (represented by line 2 on Figure 4.6-1) and a reference differential pressure is measured. The intersection of lines 1 and 2 establishes a point on the assumed pump power input curve. This allows construction of the assumed curve by shifting the model test curve vertically until it intersects this point.

b. The other pump is stopped. The flow within the active loop increases because of the reduced flow through the reactor vessel. This increased flow above the assumed design flow is determined from the relative increase in the measured differential pressure.

c. This increased flow is then plotted on Figure 4.6-1 (line 3). Its intersection with the previously assumed pump curve will yield the amount of anticipated input power (line 4). If the anticipated input power equals the measured input power with one pump running, the originally assumed flowrate was correct.

The above procedure is all that is necessary to establish whether actual flow is less than, equal to, or greater than design flow. The sense of the difference between anticipated one loop operation input power and measured one loop input power will indicate this. If anticipated power is greater than measured power, the actual flow rate was greater than design. (This can be seen by following the construction of lines 5, 6 and 7 on Figure 4.6-1.) If it is desired to know the actual flowrate, the flow with all pumps operating must again be assumed and the construction of the lines repeated until anticipated one loop power equals measured one loop input power.

This procedure makes use of elbow tap (or steam generator) differential pressure readings. These readings are not used as absolute quantities but only in reference to each other in order to determine the magnitude of the change in flow from one point to another. Therefore, calibration or accurate knowledge of elbow characteristics and dimensions are not required.

The accuracy of this procedure is affected by the accuracy of measured input power, the accuracy of determining the relative change in flow, and the accuracy of the shape of the input power curve. From a review of data from full scale tests of smaller earlier model pumps and the accuracies associated with model tests and hydraulic scaling theory it has been judged that an accuracy of .5% is a conservative tolerance to apply to the accuracy of the shape of the curve. The relative change in flow between the two pump running condition and the one pump running condition can be determined to an accuracy of .5% by the use of pre-test deadweight tester calibrated differential pressure cells and a digital voltmeter. Pump input power can be measured to an accuracy of 0.5% by use of procedures and instrumentation available from a test organization at the Westinghouse Large Rotating Apparatus Division. Typical instrumentation that would be used consists of a wattmeter, and volt and ammeters. These accuracies result in an expected total flowrate measurement accuracy of ±2.5%.

4.6.1.5 Reactor Coolant System Temperature Measurements

Resistance Temperature Detectors (RTD’s) are located in bypass loops for each hot and cold leg to develop signals used as part of the reactor control and protection systems. The RTD bypass design improves the capability to perform maintenance without sacrificing accuracy.

In addition to the bypass loop RTD’s, one well type RTD is located in each hot and cold leg to provide loop temperature signals independent of the bypass loops. However, these temperature signals are not used in the control or protection of the reactor. Figures 4.1-1A and 4.1-1B show the various RTD locations.

The hot and cold leg RTD's are inserted into reactor coolant bypass loops. A bypass loop from upstream of the steam generator to downstream of the steam generator is used for the hot leg RTD's and a bypass loop from downstream of the reactor coolant pump to upstream of the pump is used for the cold leg RTD's. The RTD's are located in manifolds within the containment and are directly inserted into the reactor coolant bypass loop without thermowells. Direct immersion in the RCS piping is not used in order to keep the detector thermal lag small. Also the bypass arrangement permits replacement of defective temperature elements while the plant is in Mode 3, Hot Standby without draining or depressurizing the reactor coolant loops.

To obtain a representative hot leg temperature, three sampling probe connections are installed 120° apart on the same cross-sectional plane of the Reactor Coolant System piping and extend into the Reactor Coolant System pipe. The hot leg RTD bypass flow from the three connections joins a common line upstream of the hot leg bypass loop isolation valves.

Each of the sampling probes, which extend several inches into the hot leg coolant stream, contains five inlet orifices distributed along its length. In this way a total of fifteen locations in the hot leg stream are sampled providing a representative coolant temperature measurement. The two inch diameter pipe leading to the manifold containing the temperature measuring elements (RTD's) provides mixing of the samples to give an accurate temperature measurement.

Care has been taken to distribute the flow evenly among the five orifices of each probe by effectively restricting the flow through the orifices. This has been done by designing a smaller overall flow area than that of the common flow channel within the probe.

This arrangement has also been applied to the flow transition from the three probe flow channels to the pipe leading to the temperature element manifold. The total flow area of the three probe channels has therefore been designed to be less than that of the two inch pipe connecting the probes to the manifold.

Flow for the cold leg RTD bypass originates downstream of the reactor coolant pump discharge. Because of the mixing action of the pump, only one connection is required for the cold leg bypass. This connection is located in the same relative position for each loop.

The accuracy of the RTD bypass loop temperature measurements was demonstrated during plant startup tests by comparing temperature measurements from all bypass loop RTD's with one another as well as with the temperature measurements obtained from the RTD's located in the hot leg and cold leg piping of each loop. The comparisons are done with the Reactor Coolant System in an isothermal condition. The linearity of the ΔT measurements obtained from the hot leg and cold leg bypass loop RTD's as a function of plant power was also checked during plant startup tests. As part of the plant startup tests, the loop RTD signals were compared with the core-exit thermocouple signals.

Low flow is to be avoided since it could result in an overall time delay in the temperature measurement greater than that assumed in the safety analysis (see Section 14.3). Loss of flow or reduced flow in a single bypass loop would result in an increase in the time response of the coolant loop temperature measurement.

An alarm occurs if the flow in a bypass loop is reduced below the full power flow by 10% or more. If redundancy conditions on the ∆T trips are not met with the reactor at power, the Technical Specifications require proceeding to Mode 3, Hot Standby. However, bypass flow is not a direct input quantity to either the protection or control systems.

The use of more flow instruments in each bypass loop does not enhance the plant safety design. Failure of the flow instrument in a bypass line does not by itself result in any adverse behavior or loss of either protection or control system function associated with the RTD's.

An actual occurrence of reduced flow or loss of flow will tend to cause the RTD to read a lower T_{avg} for the affected loop. Bypass loop low flow will not cause control system behavior requiring protection system countermeasure action. Low flow in one bypass loop, even if undetected as a result of a faulted instrument would not negate the capability of the protection system to function properly. However, the coincidence of a low flow condition with failure of the flow instrument is considered by Westinghouse to be an extremely unlikely situation. Further, aberrant readings and inconsistencies in expected behavior of RTD's in a bypass line will provide additional indication of reduced flow. Periodic inspection of the bypass loop flow indicators in accordance with Technical Specifications is performed to check against malfunctions.

Sufficient alarms, indicators and recorders are available on the control board for the operator to monitor the status of both RCS loops with regard to all operating variables and reactor trips, including RCS pump operation, flow, ∆T, Tavg, pressurizer pressure and water level.

Reactor coolant system pressure and temperature are continuously recorded on both units 1 and 2 by permanently installed recorders in the control room.

4.6.2 Design Evaluation

4.6.2.1 System Incident Potential

The potential of the Reactor Coolant System as a cause of accidents is evaluated by investigating the consequences of certain credible types of components and control failures as discussed in Section 14. Reactor coolant pipe rupture is evaluated in Sections 14.6 and 14.7.

4.6.2.2 Blowdown Jet Forces & Pipe Whip

All piping systems were routed, barriers installed, or the piping otherwise restrained such that all vital equipment is shielded from potential damage due to pipe whip caused by jet reaction loads. Individual lines and components of all engineered safety features are separated to the maximum extent practicable and restrained where necessary to prevent interaction with redundant lines and components, as well as with other systems.

Pipe restraint design requirements are such that restraints were located such that plastic hinges were prevented, unless formation of the plastic hinge did not allow pipe whip to impair containment integrity or a safety system function. The pipe rupture restraints are designed with an allowable stress less than 0.9 times the yield strength of the restraint material.

Pipe rupture analyses were performed on all high pressure piping (including lines with diameters less than 3/4″). The analyses establish that the containment vessel and all essential equipment within and without the containment (system and equipment defined as Class I in Section 12) are adequately protected against the effects of potential pipe ruptures.

Method of Analysis:

Pipe ruptures were postulated in the portions of piping systems pressurized during normal plant operations, the resulting forces were determined in accordance with the criteria as specified below, and potential damage to the system under consideration and to other Class I systems and equipment was evaluated.

a. Ruptures were postulated in adjacent Class I, II and III high pressure piping and potential damage to Class I systems or equipment under consideration was evaluated.

b. Potential damage to Class I Systems and Equipment, including the containment vessel, resulting from pipe rupture was evaluated to assure that their minimum required performance is not reduced below that specified in the USAR. As part of the evaluation it is assumed that the failure of any single active component could occur coincident with the assumed pipe rupture.

Pipe rupture constitutes potential sources of damage as a result of:

a. Jet impingement - the force loading of the jet issuing from the break.

b. Pipe whip - the unrestrained movement of a length of pipe caused by the reaction loading at point of break.

The evaluation of damage propagation from these forces was based upon piping configuration, location of barriers and supports, locations of postulated breaks, separation of redundant parts of the system, and location of other systems and equipment in relation to the system under consideration. Location of breaks were assumed such that the most critical results would occur.

Rupture Forces:

The initial force at the point of rupture is

$$F = 1.2\ PA$$

where:

P = static pressure at point of rupture

A = flow area of the pipe

For breaks in compressible fluid systems and in liquid systems connected to reservoirs that are large relative to the pipe size, the pressure is the maximum normal operating pressure at the point of rupture. For breaks in liquid systems not connected directly to reservoirs, the pressure is based on the saturation pressure at the maximum normal operating temperature.

Break Size:

The area of any postulated rupture is assumed equal to the flow area of the ruptured pipe. A longitudinal break is assumed to be rectangular in shape with length equal to two times the inside diameter of the ruptured pipe.

Jet Impingement Load:

The jet impingement load is defined as the load on a component (piping or equipment) of the undeflected jet from an instantaneous circumferential or longitudinal break of an adjacent pipe.

At the point of rupture, the jet pressure is assumed equal to the rupture pressure (P), and the effective loading area is assumed to be the break area (A). As the flow progresses away from the point of rupture the jet is assumed to diverge at an inclined angle of 45°. Hence, the effective loading area at some distance from the point of a longitudinal break is,

$$A_i = [L_1 + 2L_3 \ \text{Tan } 22\text{-}1/2^\circ] \ [L_2 + 2L_3 \ \text{Tan } 22\text{-}1/2^\circ]$$

where:

L_1 = width of break

L_2 = length of break

L_3 = distance to target

and the jet pressure at some distant target object with the effective load area is,

$$P_i = P\left[\frac{A}{Ai}\right]$$

The effective load on a distant target is then,

$$F_e = f_i \, P \, A_e$$

where:

A_e = projected area of the target object

f = shape factor of the target object

Criteria For Pipe Whip:

The evaluation of the effects of pipe whip is based upon a review of the physical arrangement of the piping system under consideration in conjunction with the evaluation of the ultimate load carrying capability of the piping. At any point in the pipe where the load resulting from a rupture exceeds the ultimate load carrying capability, it is assumed that a plastic hinge is formed and that the pipe will rotate freely about this point unless restrained at another point.

Discussion and derivations of the ultimate loads, bending strengths and bending moments for carbon steel and stainless steel piping are given in Reference 30.

For conservatism, the lower limit value is used to determine the location of a plastic hinge in stainless steel piping.

If either of the two following conditions are found to exist, it is assumed that pipe whip would result:

a. A circumferential break such that a section of pipe is subjected to a cantilever type loading in a manner which produces a bending moment greater than the ultimate bending moment.

b. A longitudinal break resulting in bending moments greater than the ultimate bending moment at the point of break and at the restraint points on either side of the break.

A whipping section of pipe is assumed to move freely until striking an object capable of stopping it and no recurring plastic hinges are assumed to develop. Further, it is assumed that the pipe will neither rebound nor change directions.

It is assumed that a whipping pipe will not damage another pipe of equal or greater size and schedule.

Restraint/Anchor Loading:

Determination of the maximum loads on pipe rupture restraints and anchors is based upon the following assumptions:

a. Pipe rupture loads are “point” loads.

b. Restraints/anchors act as “fixed” supports.

c. A plastic hinge is not formed in stainless steel pipe until the upper limit value of the ultimate load carrying capability is exceeded.

d. Loads can originate from either a pipe rupture force in the piping system under consideration or a jet impingement force resulting from a break in adjacent Class I, II or III piping.

4.6.2.3 Elimination of Large Primary Loop Pipe Rupture as the Structural Design Basis

On February 1, 1984 the NRC issued Generic Letter 84-04, “Safety Evaluation of Westinghouse Topical Reports Dealing With Elimination of Postulated Pipe Breaks in PWR Primary Main Loops” (Reference 49). This safety evaluation was based on review of Westinghouse analyses which demonstrated on a generic basis that reactor coolant system primary loop pipe breaks are highly unlikely and should not be included in the structural design basis of Westinghouse plants. These analyses are referred to as “leak-before-break” (LBB).

In order to demonstrate the applicability of the generic Westinghouse evaluations to the Prairie Island units, Westinghouse performed a fracture mechanics evaluation, a determination of leak rates from a through-wall crack, a fatigue crack growth evaluation, and an assessment of margins for both Unit 1 and 2 (References 50, 51, 68, 69). These reports provided the basis for elimination of reactor coolant system primary loop pipe breaks from the design basis. Thermal aging and degradation of cast stainless steel was considered in these evaluations. Additional consideration of thermal aging effects was completed by the utilities in the Westinghouse Owners’ Group (Reference 76).

The analyses submitted by Northern States Power Company were accepted by the NRC as documented in a Safety Evaluation Report (References 70, 130). In this safety evaluation the NRC found that the criteria provided in Chapter 5.0 of NUREG-1061, Volume 3, for evaluation of compliance with General Design Criterion 4, (GDC 4) of Appendix A to 10CFR50 as revised were satisfied and concluded that “the probability or likelihood of large pipe breaks occurring in the primary coolant system loops of Prairie Island Units 1 and 2 is sufficiently low such that dynamic effects associated with postulated pipe breaks in these facilities need not be a design basis. Furthermore, the staff concludes that the licensee is in compliance with GDC 4, as revised.”

It should be noted that there are limitations regarding the use of the design basis change and the LBB technology. As stated in NUREG-1061, Volume 3, the dynamic effects which may be excluded are:

1. pipe whip and other pipe break reaction forces,

2. jet impingement forces,

3. vessel cavity or sub-compartment pressurization including asymmetric transient effects, and

4. pipe break-associated transient loadings in functional systems or portions thereof whose pressure-retaining integrity remains intact.

The exemption and LBB technology do not apply to ECCS, containment or other system design.

Finally, the NRC also based their acceptance of the LBB technology on the ability of the reactor coolant system leak detection system to detect leakage from the RCS at a factor of 10 more restrictive than the reference flaw size. The leak detection system at Prairie Island is consistent with the guidelines of Regulatory Guide 1.45, Rev 0 for detecting leakage of 1 gpm in one hour. Reference 103 quotes sensitivities of the leakage detection system in excess of those cited to meet the guidance of the regulatory guide. The original Westinghouse evaluation uses the regulatory guide values for comparison. Operating history at Prairie Island shows these values to be conservative when compared to the leaks that have been detected.

The pipe rupture restraints installed in the reactor vessel shield wall have been removed.

The pipe rupture restraints installed on the Unit 1 and Unit 2 RCS crossover leg piping have been removed.

Reference 70 lists the six criteria used as the bases for approval of the LBB submittal along with the responses that demonstrated that each of the criterion had been met. As part of the structural evaluation of the reactor coolant loop with the replacement steam generators installed, it was demonstrated that the response for each of the six criteria listed in the original LBB analysis remains applicable or bounding (Reference 133 and 155). Therefore the conclusions of reference 70 remain applicable to the reactor coolant loop piping with the replacement steam generators installed.

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4.6.2.4 Elimination of Pressurizer Surge Line Rupture as the Structural Design Basis for Prairie Island Unit 1

On December 20, 1988 the NRC issued NRC Bulletin 88-11, “Pressurizer Surge Line Thermal Stratification” which required a stress and fatigue analysis for pressurizer surge lines considering the effects of thermal stratification and cycling. Results of this analysis for Prairie Island Unit 1 showed that in order to keep stress levels below ASME limits, modification of pipe whip restraints was required to allow unrestrained pipe movement (Reference 98).

A leak-before-break (LBB) analysis was performed by Westinghouse consistent with the criteria in NUREG-1061, Volume 3 thereby complying with General Design Criterion-4 (GDC-4) of Appendix A to 10 CFR Part 50. The analysis concluded that the probability of large pipe breaks occurring in the surge line is sufficiently low such that dynamic effects associated with the postulated pipe breaks need not be a design basis. The LBB analysis was submitted by NSP for NRC review (Reference 99, Reference 95), and was approved as documented in a NRC Safety Evaluation Report (Reference 100). The NRC staff conclusion was conditioned on NSP’s commitment to remove the shims or modify the gaps of the whip restraints to allow the surge line to satisfy NRC Bulletin 88-11.

The pipe rupture restraint shim packs for the Prairie Island Unit 1 pressurizer surge line were removed during the Fall 1992 outage. The LBB analysis now eliminates the need for pipe rupture restraints in the design basis for the Unit 1 surge line.

4.6.2.5 Application of Leak-Before-Break Methodology to Piping Attached to the RCS, Units 1 and 2, and to the Pressurizer Surge Line, Unit 2

(The following description of the Leak-Before-Break licensing basis for Prairie Island includes locations where implementation remains in progress (as of September 25, 2012). The paragraphs that discuss such locations are denoted by a delta symbol within brackets; [Δ]. EC 19607 will implement this license amendment on these branch lines.)

In a letter dated October 27, 2011 (Reference 145), the NRC approved a license amendment to apply Leak-Before-Break (LBB) methodology to piping systems six-inches and larger connected to the reactor coolant system (RCS) in both Units 1 and 2, and to the pressurizer surge line and dissimilar metal weld overlay in Unit 2. This approval allowed excluding from the Licensing Basis the dynamic effects of postulated ruptures of the following lines:

1. 12-inch safety injection (SI) lines (loops A and B) for both units. These lines are connected to the SI accumulators. The loop B line also serves as the RHR return line. [Δ]

2. 8-inch residual heat removal (RHR) lines (loops A and B) for both units. These lines serve as the RHR system suction lines. This analysis also evaluates thermal stratification in the RHR suction lines for Units 1 and 2. [Δ]

3. 6-inch cold leg SI lines (loops A and B) for both units. These lines provide flow from the high pressure SI pumps. [Δ]

4. 6-inch reactor vessel SI lines (loops A and B) for both units. These lines are composed of 4-inch diameter lines from the reactor vessel nozzle connected to a shorter section of 6-inch diameter lines near the isolation valves. Only the 6-inch portions of these lines are evaluated. [Δ]

5. 6-inch RCS drain down line on the hot leg (loop A on Unit 1 and loop B on Unit 2). This line consists of a short section of the 6-inch diameter piping prior to reducing to 2-inch diameter at the isolation valve. Only the 6-inch portions of these lines are evaluated. [Δ]

6. 6-inch capped nozzle on the hot leg (Loop B on Unit 1 and Loop A on Unit 2). [Δ]

7. The PINGP Unit 2 pressurizer surge line (including the weld overlay application installed to mitigate the possibility of primary water stress corrosion cracking (PWSCC) in the pressurizer to surge line 82/182 nozzle-to-safe-end weld).

The supporting analyses (References 146, 147, 148) were included with NSPM’s license amendment request (Reference 149) as clarified in Reference 150.

The NRC Safety Evaluation (Reference 145) explained that the LBB concept is based on calculations and experimental data demonstrating that certain pipe material has sufficient fracture toughness (ductility) to prevent a small through-wall flaw from propagating rapidly and uncontrollably to catastrophic pipe rupture and to ensure that the probability of a pipe rupture is extremely low. The small leaking flaw is demonstrated to grow slowly and the limited leakage would be detected by the RCS leakage detection systems early enough that licensees can shut down the plant to repair the degraded pipe long before the potential catastrophic pipe rupture.

NSPM explained (Reference 145) that the subject LBB application includes piping where flaws would have slow growth rates and where water hammer, corrosion, creep, fatigue, erosion, environmental conditions, and indirect sources are remote causes for pipe rupture. Also, a deterministic fracture mechanics evaluation supported the conclusion that the probability of rupture of these lines is extremely low.

The RCS leakage detection systems ensure that small leakage will be detected early enough to allow repair long before a potential catastrophic rupture. As described in Section 6.5, the leak detection systems are sufficiently reliable, redundant, diverse, sensitive, and can detect small leaks with acceptable margins. The limiting leakage flaw from the RCS branch piping analysis (Reference 146) is 2.12 gpm, which provides a factor of at least 2 from the critical flaw size. The RCS leakage detection system is capable of detecting leaks of 0.2 gpm, which provides a margin of at least 10 from the leakage flaw. To determine the timeframe available to perform repairs to piping with detectable leakage, an analysis was performed on crack growth rates (Reference 150). This analysis determined that it would take approximately 95 days for a crack leaking at 2.0 gpm (10 times the 0.2 gpm RCS leakage detection capability) to grow to the 2.12 gpm leakage flaw, and another approximately five years to grow to critical size. This analysis demonstrates that sufficient time will be available to complete repairs long before the potential catastrophic pipe rupture.

Application of LBB methodology to the Unit 2 pressurizer surge line eliminates the need for pipe rupture restraints in the design basis for the Unit 2 surge line. The pipe rupture restraint shim packs and pipe saddles for the Prairie Island Unit 2 pressurizer surge line were removed during refueling outage 2R27.

4.6.2.6 Loss of Decay Heat Removal During Periods of Nonpower Operation

Generic Letter 88-17 (Reference 127) identified areas of weakness in the industry related to loss of decay heat removal during periods of nonpower operation. NSP responded in a January 6, 1989 letter (Reference 128) to the NRC. The focus of this letter was to describe changes to the facility, which would be made to better accommodate operation of the Reactor Coolant System in a reduced inventory condition. Below are the enhancements to the facility that were described:

1. Procedures and administrative controls are used to reasonably assure that containment closure can be achieved prior to the time at which a core uncovery could result from a loss of RHR coupled with an inability to initiate alternate cooling or addition of water to the RCS inventory. (NRC Commitment #0000516)

2. A minimum of two core exit thermocouples or two incore thermocouples (Reference 129) are used to provide continuous RCS temperature indication when the RCS is in a mid-loop condition. The indication is not designed to be independent as use of a common power supply is acceptable. (NRC Commitment #0000519)

3. Two continuous and independent RCS water level indications are provided to the control room whenever the RCS is in a reduced inventory condition. (NRC Commitment #0000594)

4. Procedures and administrative controls are used to avoid operations that deliberately or knowingly lead to perturbations of the RCS when in a reduced inventory condition. (NRC Commitment #0000583)

5. The charging system is required to be lined up to the RCS as the required boric acid flowpath. Additional makeup capabilities using the Safety Injection system via the reactor vessel upper plenum are also required when in a reduced inventory condition. (NRC Commitment #0000597, 0000584, 0000587, 0000599, 0000609)

6. Nozzle dams are required to be installed in cold legs prior to installation in the hot legs. Hot leg nozzle dams are required to be removed from the hot legs prior to removal of the cold leg nozzle dams. (NRC Commitment #0000588)

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4.7 INSPECTION AND TESTING

Criterion: Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance of critical areas by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with current applicable codes shall be provided. (GDC 36)

4.7.1 Reactor Coolant System Inspection and Testing

Non-Destructive Inspection of Material and Components

Table 4.7-1 summarizes the quality assurance program for all Reactor Coolant System components. In this table, all of the non-destructive tests and inspections which were required by Westinghouse (and/or Framatome-ANP) specifications on Reactor Coolant System components and materials were specified for each component. All tests required by the applicable codes were included in this table. Westinghouse requirements which were more stringent in some areas than those requirements specified in the applicable codes, were also included. The fabrication and quality control techniques used in the fabrication of the Reactor Coolant System were equivalent to those used for the reactor vessel. The reactor coolant system piping was installed, welded, inspected and non-destructively examined according to the requirements of the project specification and installers quality assurance procedures which are on file at the Prairie Island Site. The welding was performed in accordance with Section IX of the ASME Boiler and Pressure Vessel Code. Tests and inspections were documented to an extent which conforms to the requirements conveyed by the AEC to NSP on October 14, 1969.

Westinghouse required, as part of its reactor vessel specification, that certain special tests which are not specified by the applicable codes be performed. These tests are listed below:

a. Ultrasonic Testing - Westinghouse required that a 100% volumetric ultrasonic test of reactor vessel plate for shear wave be performed in addition to code requirements. This 100% volumetric ultrasonic test is a severe requirement, but it assures that the plate is of the highest quality.

b. Radiation Surveillance Program - In the surveillance programs, the evaluation of the radiation damage is based on pre-irradiation testing of Charpy V-notch, drop weight and tensile specimens and post-irradiation testing of Charpy V-notch and tensile specimens. These programs are directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the transition temperature approach and the fracture mechanics approach, and is in accordance with ASTM-E-185, “Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels” and 10CFR50 Appendix H. The surveillance program does not include thermal control specimens. These specimens are not required since the surveillance specimens are exposed to the combined neutron irradiation and temperature effects and the test results provide the maximum transition temperature shift. Thermal control specimens as considered in ASTM-E-185 would not provide any additional information on which the operational limits for the reactor vessel are set.

In addition to the inspections shown in Table 4.7-1, there are those which the equipment supplier performs to confirm the adequacy of material he received and those performed by the material manufacturer in producing the basic material. The inspections of reactor vessel, pressurizer, and steam generator were governed by ASME code requirements. The inspection procedures and acceptance standards required on pipe materials and piping fabrication are governed by USAS B31.1 and Westinghouse requirements and are equivalent to those performed on ASME coded vessels.

Procedures for performing the examinations were consistent with those established in ASME Code Section III and were reviewed by qualified engineers. These procedures have been developed to provide the highest assurance of quality material and fabrication. They consider not only the size of the flaws, but equally as important, how the material is fabricated, the orientation and type of possible flaws, and the areas of most severe service conditions. In addition, the accessible external surfaces of the primary Reactor Coolant System pressure containing segments receive a 100% surface inspection by Magnetic Particle or Liquid Penetrant Testing after hydrostatic test (See Table 4.7-1). All reactor vessel plate material is subjected to angle beam as well as straight beam ultrasonic testing to give maximum assurance of quality. All reactor vessel forgings received the same inspection. In addition, 100% of the material volume is covered in these tests as an added assurance over the grid basis required in the code.

Quality Control engineers (Westinghouse, Pioneer, and Northern States Power) monitored the supplier’s work, witnessing key inspections not only in the supplier’s shop but in the shops of subvendors of the major forgings and plate material. Normal surveillance included verification of records of material, physical and chemical properties, review of radiographs, performance of required tests, and qualification of supplier personnel.

Equipment specification for fabrication required that suppliers submit the manufacturing procedures (welding, heat treating, etc.) which were reviewed by qualified Westinghouse engineers. Field fabrication procedures were also reviewed to assure that installation field welds were of equal quality.

Section III of the ASME Code requires that nozzles carrying significant external loads are attached to the shell by full penetration welds. This requirement was carried out in the reactor coolant piping, where all auxiliary pipe connections to the reactor coolant loop were made using full penetration welds.

The Reactor Coolant System components were welded under procedures which required the use of both preheat and post-heat. Preheat requirements, not mandatory under Code rules, were performed on all weldments including P1 and P3 materials which are the materials of construction in the reactor vessel, pressurizer and steam generators. Preheat and post-heat of weldments both served a common purpose: the production of tough, ductile metallurgical structures in the completed weldment. Preheating produces tough ductile welds by minimizing the formation of hard zones whereas post-heating achieves this by tempering any hard zones which may have formed due to rapid cooling.

Electroslag Weld Quality Assurance

Piping Elbows

The 90° elbows were electroslag welded. The following were performed for quality assurance of the welding procedures:

a. The electroslag welding procedure employing one wire technique was qualified in accordance with the requirements of ASME B&PV Code Section IX and Code Case 1355 plus supplementary evaluations as requested by Westinghouse. The following test specimens were removed from a 5 inch thick weldment and successfully tested. They were:

   1. 6 Transverse Tensile Bars - as welded

   2. 6 Transverse Tensile Bars - 2050°F, H_20 Quench

   3. 6 Transverse Tensile Bars - 2050°F, H_2O Quench + 750°F stress relief heat treatment

   4. 6 Transverse Tensile Bars - 2050°F, H_2O Quench, tested at 650°F

   5. 12 Guided Side Bend Test Bars

b. The casting segments were surface conditioned for 100% radiographic and penetrant inspections. The acceptance standards were ASTM-E-186 severity level 2 except no category D or E defectiveness was permitted and USAS Code Case N-10, respectively.

c. The edges of the electroslag weld preparations were machined. These surfaces were penetrant inspected prior to welding. The acceptance standards were USAS Code Case N-10.

d. The completed electroslag weld surfaces were ground flush with the casting surface. Then, the electroslag weld and adjacent base material were 100% radiographed in accordance with ASME Code Case 1355. Also, the electroslag weld surfaces and adjacent base material were penetrant inspected in accordance with USAS Code Case N-10.

e. Weld metal and base metal chemical and physical analysis were determined and certified.

f. Heat treatment furnace charts were recorded and certified.

Reactor Coolant Pump Casings

a. The electroslag welding procedure employing two- and three-wire techniques was qualified in accordance with the requirements of the ASME B&PV Code Section IX and Code Case 1355 plus supplementary evaluations as requested by WNES-PWRSD. The following test specimens were removed from an 8-inch thick and from a 12-inch thick weldment made from a separate static casting and successfully tested for both the two-wire and three-wire techniques, respectively. They are:

   1. Two-wire electroslag process - 8″ thick weldment.

      a. 6 Transverse Tensile Bars - 750°F post weld stress relief

      b. 12 Guided Side Bend Test Bars

   2. Three-wire electroslag process - 12″ thick weldment

      a. 6 Transverse Tensile Bars - 750°F post weld stress relief

      b. 17 Guided Side Bend Test Bars

      c. 21 Charpy Vee Notch Specimens

      d. Full section macroexamination of weld and heat-affected zone

      e. Numerous microscopic examinations of specimens removed from the weld and heat-affected zone regions.

      f. Hardness survey across weld and heat-affected zone.

3. A separate weld test using the two-wire electroslag technique was performed to evaluate the effects of a stop and restart of welding by this process. This evaluation was performed to establish proper procedures and techniques. The following test specimens were removed from an eight inch thick weldment in the stop-restart-repaired region and tested. They are:

    a. 2 Transverse Tensile Bars - as welded

    b. 4 Guided Side Bend Test Bars

    c. Full section macroexamination of weld and heat-affected zone.

4. All of the weld test blocks in (1), (2), and (3) above were radiographed using a 24 Mev Betatron. The radiographic quality level, as defined by ASTM E-94, obtained is between one-half of 1% and 1% (1-1T). There were no discontinuities evident in any of the electroslag welds.

    a. The casting segments were surface conditioned for 100% radiographic and penetrant inspections. The radiographic acceptance standards were ASTM E-186 severity level 2 except no category D or E defect was permitted for section thickness up to 4-1/2 inches. The penetrant acceptance standards were ASME B&PV Code Section III, paragraph N-627.

    b. The edges of the electroslag weld preparations were machined. These surfaces were penetrant inspected prior to welding. The acceptance standards were ASME B&PV Code Section III, paragraph N-627.

    c. The completed electroslag weld surfaces were ground flush with the casting surface. Then the electroslag weld and adjacent base material were 100% radiographed in accordance with ASME Code Case 1355. Also, the electroslag weld surfaces and adjacent base material were penetrant tested in accordance with B&PV Code Section III, paragraph N-627.

    d. Weld metal and base metal chemical and physical analysis were determined and certified.

    e. Heat treatment furnace charts were recorded and certified.

In-Process Control of Variables

There are many variables that must be controlled in order to maintain desired quality welds. These, together with an explanation of their relative importance are as follows:

Heat Input vs Output

The heat input is determined by the product of volts times current and they are measured by voltmeters and ammeters which are considered accurate, as they are calibrated every 30 days. During any specific weld these meters are constantly monitored by the operators.

The ranges specified are 500-620 amperes and 44-50 volts. The amperage variation, even though it is less than ASME allows by Code Case 1355, is necessary for several reasons: 01450457

a. The thickness of the weld is in most cases the reason for changes.

b. The weld gap variation during the weld cycle also requires changes. For example, the procedure qualifications provide for welding thicknesses from 5″ to 11″ with two wires. The current and voltage are varied to accommodate this range.

c. Also, the weld gap is controlled by spacer blocks. These blocks must be removed as the weld progresses. Each time a spacer block is removed there is the chance of the weld pinching down to as much as 1″ or opening to perhaps as much as 1-1/2″. In either case, a change in current may be necessary.

d. The heat output is controlled by the heat sink of the section thickness and metered water flow through the water-cooled shoes. The nominal temperature of the discharged water is 100°F.

Weld Gap Configuration

As previously mentioned, the weld gap configuration is controlled by 1-1/4″ spacer blocks. As these blocks are removed there is the possibility of gap variation. It has been found that a variation from 1″ to 1-3/4″ is not detrimental to weld quality as long as the current is adjusted accordingly.

Flux Chemistry

The flux used for welding is Arcos BV-1 Vertomax. This is a neutral flux whose chemistry is specified by Arcos Corporation. The molten slag is kept at a nominal depth of 1-3/4″ and may vary in depth by plus or minus 3/8″ without affecting the weld. This is measured by a stainless steel dipstick.

Weld Cross Section Configuration

It is noted that the higher the current or heat input and the lower the heat output that the dilution of weld metal with base metal is greater, causing a more round barrel-shaped configuration as compared to welding with less heat input and higher heat output. This would cut the amount of dilution to provide a more narrow barrel-shaped configuration. This is also a function of section thicknesses; the thinner the section, the more round the pattern that is produced.

Welder Qualification

Welder qualification in accordance with ASME B&PV Code, Section IX rules is required, using transverse side bend test specimens per Table Q.24.1.

4.7.2 Reactor Vessel Material Surveillance Program

4.7.2.1 Program Basis

The ability of the reactor pressure vessel to resist fracture is an important factor in ensuring the safety of the plant. The beltline region of the reactor pressure vessel is the most critical region of the vessel because it is subjected to significant fast neutron bombardment. The overall effects of fast neutron irradiation on the mechanical properties of low alloy ferritic pressure vessel steels such as SA508 Class 3 (base material of both Prairie Island reactor pressure vessel beltlines) are well known and documented. Generally, low alloy ferritic materials show an increase in hardness and tensile properties and a decrease in ductility and toughness under certain conditions of irradiation.

A method for performing analyses to guard against fast fracture in reactor pressure vessels is presented in “Protection Against Non-ductile Failure”, Appendix G to Section III of the ASME Boiler and Pressure Vessel Code. The method utilizes fracture mechanics concepts and is based on the reference nil ductility temperature (RT_{NDT}). RT_{NDT} is defined as the greater of either the drop weight nil ductility transition temperature (NDTT per ASTM E-208) or the temperature 60°F less than the 50 ftlb (and 35-mil lateral expansion) temperature as determined from Charpy specimens oriented normal (transverse) to the major working direction of the material. The RT_{NDT} of a given material is used to determine allowable pressure-temperature (P-T) limits using either of the methodologies contained in WCAP-14040- NP-A, Revision 2 (Reference 107) or WCAP-14040-A, Revision 4 (Reference 140). Revision 2 to the WCAP uses the K_{Ia} reference stress intensity factor which is obtained from the reference fracture toughness curve, defined in Appendix G to Section XI of the ASME Code. Revision 4 to the WCAP uses the K_{Ic} stress intensity factor, which is a lower bound of static fracture toughness as defined in Appendix A to Section XI of the ASME Code. When a given material is indexed to the K_{Ia} or K_{Ic} curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Allowable operating limits can then be determined as a function of temperature.

RT_{NDT} and, in turn, the plant operating limits are adjusted to account for the effects of radiation on the reactor vessel material properties. The radiation embrittlement or changes in mechanical properties of the Prairie Island reactor pressure vessels is monitored by the Reactor Vessel Material Surveillance Program (References 60 and 61), in which surveillance capsules are periodically removed from each reactor and the encapsulated specimens are tested. The increase in the average Charpy V-notch 30 ftlb temperature (delta RT_{NDT}) due to irradiation is added to the original RT_{NDT} to adjust the RT_{NDT} for radiation induced embrittlement. This adjusted RT_{NDT} (RT_{NDT} initial + delta RT_{NDT}) is used to index the material to the K_{Ia} or K_{Ic} curve and, in turn, to set plant operating limits which take into account the effects of irradiation on the reactor vessel materials.

The current Pressurized Thermal Shock evaluations use the methodology of revision 4 to WCAP-14040-A (Reference 140) to calculate fast neutron fluence whereas the current setpoints and curves in the Pressure-Temperature Limits Report (PTLR) were derived using the methodologies of revision 2 to WCAP-14040-NP-A (Reference 107).

The fast neutron exposure and adjusted RT_{NDT} were re-calculated using the methodology of WCAP-14040-A revision 4 and cycle-specific core power distributions for 54 EFPY. An evaluation determined that the Pressure-Temperature Limits for 35 EFPY, based on revision 2 of WCAP-14040, conservatively bound the Pressure-Temperature Limits at 54 EFPY using the fluence determined using revision 4 of WCAP-14040. Therefore, the PTLR curves which were developed for 35 EFPY remain bounding for 54 EFPY. (References 156) 01451781 01514777

4.7.2.2 Reactor Pressure Vessel Materials

A summary of the initial (unirradiated) toughness data for the reactor vessel materials is provided in Tables 4.7-2 and 4.7-3. The location of the reactor vessel beltline region weld and forging material is shown in Figure 4.7-4. The identification, chemical composition and mechanical properties of the reactor vessel beltline materials is provided in Tables 4.7-4 through 4.7-9.

4.7.2.3 Surveillance Capsules

Six specimen capsules are located in each reactor between the thermal shield and the vessel wall and are vertically positioned opposite the center of the core. The capsules are in baskets attached to the thermal shield. Elevation and plan views showing the location and dimensional spacing of the capsules with relation to the core, thermal shield, and vessel weld seams are provided in Figures 4.7-1 and 4.7-2. The capsules can be removed and replaced when the vessel head and upper internals are removed.

The capsules contain test specimens made from material obtained from a 6.692 inch thick shell ring forging from the reactor vessel intermediate shell course (unit 1) and lower shell course (unit 2) adjacent to the core region, and also representative weld metal and heat-affected zone (HAZ) metal. The thermal history or heat treatment given these specimens is similar to the thermal history of the reactor vessel material, with the exception that the postweld heat treatment received by the specimens has been simulated. Correlation monitors made from fully documented specimens of A533 Grade B Class 1 material obtained through Subcommittee II of ASTM Committee E10 on Radioisotopes and Radiation Effects are also inserted in the capsules.

The six capsules in each unit contain Charpy V-notch impact specimens, tensile specimens, and Wedge Opening Loading specimens, from the shell ring forging of the reactor vessel and associated weld metal, and Charpy V-notch impact specimens of HAZ metal and the ASTM correlation monitor material. Dosimeters which are used to measure the integrated neutron flux at specific neutron energy levels and thermal monitors to more accurately define the maximum temperature attained by the test specimens during irradiation are also included in each capsule. The specimens are enclosed in a tight fitting stainless sheath to prevent corrosion and ensure good thermal conductivity. The complete capsules were helium leak tested. Vessel material sufficient for at least 2 capsules is kept in storage should the need arise for additional replacement test capsules in the program. The specific contents of the six surveillance capsules are provided in Table 4.7-10.

An additional surveillance capsule designated W was inserted into Unit 1 at the beginning of cycle 16. This capsule contains test specimens originally removed from the Monticello surveillance capsule located at the 30° azimuthal location of the Monticello Nuclear Generating Plant in November 1981. These specimens are Charpy V-notch impact specimens and tensile specimens machined from the Monticello beltline materials. This capsule was removed in June of 1994 at the end of cycle 16.

The present schedules for removal of the Unit 1 and 2 surveillance capsules are provided in Table 4.7-11. These schedules are based on recommendations from the most recent capsule reports References 105 and 106.

4.7.2.4 Specimen Mechanical Testing

The post irradiation mechanical testing of the most recently removed Charpy V-notch and tensile specimens was performed in accordance with 10 CFR 50, Appendix H and ASTM Specification E185-82. The specific details and results of the mechanical testing performed on the most recently removed capsules is described in References 105 and 106.

4.7.2.5 Radiation Analysis and Neutron Dosimetry

Knowledge of the neutron environment within the reactor pressure vessel/surveillance capsule geometry is required as an integral part of reactor pressure vessel surveillance programs for two reasons. First, in order to interpret the neutron radiation-induced material property changes observed in the test specimens, the neutron environment (energy spectrum, flux and fluence) to which the test specimens were exposed must be known. Second, in order to relate the changes observed in the test specimens to the present and future condition of the reactor vessel, a relationship must be established between the neutron environment at various positions within the reactor vessel and that experienced by the test specimens. The former requirement is normally met by employing a combination of rigorous analytical techniques and measurements obtained with passive neutron flux monitors contained in each of the surveillance capsules. The latter information is derived solely from analysis.

The neutron environment information obtained from the capsule neutron dosimeters and the neutron flux, fluence and energy spectra calculated for the reactor vessel and surveillance capsules, are used to develop lead factors for use in relating neutron exposure of the reactor vessel to that of the surveillance capsules. The details and results of these calculations for the most recently removed capsules are discussed in References 105 and 106.

Irradiation of the surveillance specimens is higher than the radiation of the adjacent vessel wall because they are closer to the core than to the vessel wall. Since these specimens experience higher irradiation and are actual samples from materials used in the reactor vessel, the NDTT measurements are representative of the vessel at a later time in life. The fluence values resulting from the analysis of the most recently removed capsules (References 105 and 106), are summarized in Tables 4.7-12 and 4.7-13. Current neutron fluence projections to 35 EFPY and 54 EFPY are summarized in Table 4.7-14 for the reactor vessel materials in Units 1 and 2.

4.7.2.6 Heatup and Cooldown Limit Curves

4.7.2.6.1 Most Limiting RT_{NDT}

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} which is determined as follows:

The highest RT_{NDT} of the material in the core beltline region of the reactor vessel is determined using the initial (unirradiated) values of RT_{NDT}, shown in Tables 4.7-2 and 4.7-3, the radiation induced delta RT_{NDT} and margin to account for uncertainties. The Delta RT_{NDT} is estimated using Regulatory Guide 1.99, Revision 2 methodology, accounting for the fast neutron (E greater than 1 Mev) fluence at the 1/4 T and 3/4 T vessel locations, material chemistry and actual surveillance data where available.

The most limiting RT_{NDT} estimated for the planned service period is factored into the heatup and cooldown curves provided. Values of the most limiting RT_{NDT} determined in this manner may be used until the results from the material surveillance program, when evaluated according to ASTM E185, indicate that they are inappropriate or until the planned service period is exceeded on either unit. At that time, the heatup and cooldown curves must be recalculated.

The heatup and cooldown curves in the Prairie Island Pressure/Temperature Limits Report, PTLR, are based on the Unit 1 weld metal Type-UM40, Heat-2269 with a copper content of 0.15 w/o and an initial generic RT_{NDT} of 0°F. This material is the Nozzle Course Forging to Intermediate Course Forging Circumferential Weld (W2). The material with the highest ART value at 54 EFPY is the Unit 1 Intermediate Shell to Lower Shell Weld (W3). This is weld metal Type-UM40, Heat-1752 with an initial measured RT_{NDT} of -13°F. 01451781

4.7.2.6.2 Pressure/Temperature Limits

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50. The Pressure/Temperature Limits have most recently been updated to 54 EFPY (Reference 112 and 156). The updating methods are discussed in detail in Reference 64 and 107 and the following paragraphs. 01451781

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code (Reference 65). The K_{IR} curve is given by the equation (Reference 107):

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \qquad (1)$$

where K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code (Reference 65) as follows:

$$C\ K_{IM} + K_{IT} \text{ less than } K_{IR} \qquad (2)$$

Where, K_{IM} is the stress intensity factor caused by membrane (pressure) stress.

K_{IT} is the stress intensity factor caused by the thermal gradients.

K_{IR} is a function of temperature relative to the RT_{NDT} of the material.

C = 2.0 for level A and B service limits, and

C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{IT}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

During cooldown, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from tensile at the inner wall to compressive at the outer wall. The thermal induced tensile stresses at the inner wall increase with increasing cooldown rates and are additive to the pressure induced stresses which are already present. Therefore, for the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is always assumed to exist at the inside of the vessel wall. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4 T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the delta T developed during cooldown results in a higher value of K_{IR} at the 1/4 T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{IT}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4 T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4 T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4 T crack during heatup is lower than the K_{IR} for the 1/4 T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and lower K_{IR}'s do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4 T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4 T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses, at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

The composite curves for the heatup rate data and the cooldown rate data are then adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

4.7.2.6.3 Flange Regions

The requirements of 10 CFR 50, Appendix G, which address the metal temperature of the closure head flange and vessel flange regions, are considered in the development of the heatup and cooldown curves. These requirements specify that when the pressure exceeds 20 percent of the preservice hydrostatic test pressure, the temperature of the closure flange regions must exceed the RT_{NDT} of the material in those regions by a least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. These requirements have no impact on the limiting heatup and cooldown curves included in the Prairie Island PTLR.

When the head closure bolts are tensioned the flange material is prestressed. The heatup and cooldown curves have a minimum boltup temperature of 60°F to account for the flange condition. The development of this limit is discussed in reference 107.

4.7.2.6.4 Leak Test Limits

The leak test limit portion of the heatup and cooldown curves contained in the PTLR represents the minimum temperature requirements at the leak test pressure specified by applicable codes. The leak test limit curve was determined by methods of References 65, 66 and 107.

4.7.2.6.5 Criticality Limits

Appendix G of 10 CFR Part 50 requires that for a given pressure, the reactor must not be made critical unless the temperature of the reactor vessel is 40°F above the minimum permissible temperature specified on the heatup curve and above the minimum permissible temperature for the inservice hydrostatic pressure test. For Prairie Island the curves were prepared, requiring that criticality must occur above the maximum permissible temperature for the inservice hydrostatic pressure test.

4.7.2.6.6 ASME Code Section XI Inservice Test Limits

The pressure temperature limits for the ASME Code Section XI Inservice Test Limits (hydrostatic pressure test) are less restrictive than the heatup and cooldown curves to allow for the periodic inservice hydrostatic test. These limits are allowed to be less restrictive because the hydrostatic test is based on a 1.5 safety factor versus the 2.0 safety factor built into the heatup and cooldown curves and because the test is run at a constant temperature so the thermal stresses in the vessel are minimal.

4.7.3 In-Service-Inspection

4.7.3.1 Design Considerations

During the design phase of the Reactor Coolant System, careful consideration was given to provide access for both visual and non-destructive in-service inspection of primary loop components. If necessary, the following components and areas can be made available for 100% visual and 100% non-destructive inspection (except as noted):

a. Reactor Vessel - The entire inside surface

b. Reactor Vessel Nozzles - The entire inside surface

c. Closure Head - The entire inside and outside surface

d. Reactor Vessel Studs, Nuts and Washers

e. Field Welds between the Reactor Vessel, Steam Generators, and Reactor Coolant Pumps and the Reactor Coolant Piping

f. Reactor Internals

g. Reactor Vessel Flange Seal Surface

h. Fuel Assemblies

i. Rod Cluster Control Assemblies

j. Control Rod Drive Shafts

k. Control Rod Drive Mechanism Assemblies

l. Reactor Coolant Pipe External Surfaces (except for the five foot penetration of the primary shield)

m. Steam Generator - The external surface, the internal surfaces of the steam drum, and the channel head

n. Pressurizer - The internal and external surfaces

o. Reactor Coolant Pump - The internal and external surfaces, motor and impeller

The design considerations which have been incorporated into the primary system design to permit the above inspections are as follows:

a. All reactor internals are completely removable. The tools and storage space required to permit reactor internals removal for these inspections are provided.

b. The closure head is stored dry on an operating deck during refueling to facilitate direct visual inspection.

c. All reactor vessel studs, nuts and washers are removed to dry storage during refueling.

d. Removable plugs are provided in the primary shield just above the coolant nozzles, and the insulation covering the nozzle welds is readily removable.

e. Access holes are provided in the lower internals barrel flange to allow remote access to the reactor vessel internal surfaces between the flange and the nozzles without removal of the internals.

f. A removable plug is provided in the lower core support plate to allow access for inspection of the bottom head without removal of the lower internals.

g. The storage stands provided for storage of the internals allow for inspection access to both the inside and outside of the structures.

h. The station provided for changeout of control rod clusters from one fuel assembly to another is specially designed to allow inspection of both fuel assemblies and control rod clusters.

i. The control rod mechanism is specially designed to allow removal of the mechanism assembly from the reactor vessel head.

j. Manways are provided in the steam generator, steam drum and channel head to allow access for internal inspection.

k. A manway is provided in the pressurizer top head to allow access for internal inspection.

1. All insulation on primary system components (except the reactor vessel) and piping (except for the penetration in the primary shield) is removable.

4.7.3.2 Reactor Vessel

The use of conventional non-destructive, direct visual and remote visual test techniques can be applied to the inspection of all primary loop components except for the reactor vessel. The reactor vessel presents special problems because of the radiation levels and remote underwater accessibility to this component. Because of these limitations on access to the reactor vessel, several steps were incorporated into the design and manufacturing procedures in preparation for non-destructive test techniques.

These were:

a. Shop ultrasonic examinations were performed on all internally clad surfaces to an acceptance and repair standard to assure an adequate cladding bond to allow later ultrasonic testing of the base metal. Size of cladding bonding defect allowed was 3/4 inch.

b. The design of the reactor vessel shell in the core area is a clean, uncluttered cylindrical surface to permit positioning of test equipment without obstruction.

4.7.3.2.1 Ultrasonic Examination

An ultrasonic examination of the reactor vessel was performed by the reactor vessel manufacturer after the hydrostatic test and included the following:

a. All circumferential shell welds

b. Vessel-to-flange (and head to flange on the original RV head) circumferential welds

c. Primary coolant nozzle-to-shell welds

d. Safety injection nozzles to shell welds

e. Internal extensions of outlet primary coolant nozzle and safety injection nozzles

f. Primary coolant nozzle safe end welds

g. Safety injection nozzle safe end welds

h. Closure studs and nuts

i. Vessel stud hole ligaments

j. External support welds

As indicated above, the only sophisticated remote inspection equipment currently required is for inspection of the reactor vessel. The baseline inspection was performed by Westinghouse utilizing a remote reactor vessel ultrasonic inspection tool to perform the code required inspection of the circumferential welds and the ligaments between the flange holes, the nozzle to vessel safe end weld and the safe end to nozzle pipe welds. (It should be noted that the Prairie Island Unit 1 and Unit 2 vessels have no longitudinal welds.)

After system hydro but prior to placing the facility into commercial operation, baseline examinations of the Reactor Vessel and the Reactor Coolant System were performed for areas specified in Section XI of the ASME Boiler and Pressure Vessel Code. The objective of the examinations was detection, location and recording of indications to provide a baseline reference and record for future examinations.

4.7.3.2.2 Inspection Using Acoustic Monitoring System

In addition to the above mentioned baseline examinations, a baseline inspection of the reactor vessel was performed using an Acoustic Monitoring System (AMS). The AMS employed proven electronic recording techniques to provide baseline acoustic data during the initial primary system cold hydrostatic tests and hot functional testing.

The Acoustic Monitoring System (AMS) consisted of:

a. Transducer and coupling techniques to attach the transducer to the reactor vessel (20 transducers were installed at various locations).

b. A signal transmission system to couple the transducer electrical output to the AMS electronics console.

c. AMS electronics console consisting of a Westinghouse designed transducer-transmission line amplifier, discriminator, and monitoring system. The AMS electronics console was located outside the reactor containment structure.

d. Data analysis system which analyzed recorded acoustic data to provide a determination of the severity as well as the location of the flaw.

4.7.3.3 Acceptance Criteria

The in-service inspection program conforms to the requirements of 10CFR50, Section 50.55 a(g).

Fracture mechanics criteria from later editions of ASME Section XI are utilized during reactor vessel inservice inspections. Specifically, indications found during Reactor Vessel Inservice Inspection volumetric examinations will be characterized, and then be accepted or evaluated, in accordance with ASME Section XI, as delineated in the current Inservice Inspection Program.

4.7.4 Loose Parts Monitoring

A Digital Metal Impact Monitoring System is installed to detect the presence of metallic debris in the reactor coolant system when the debris impacts the RCS. This system monitors reactor vessel and steam generator primary and secondary areas, providing remote indication and alarming.

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TABLE 4.1-1 MATERIALS OF CONSTRUCTION OF THE REACTOR COOLANT SYSTEM COMPONENTS

Component	Section	Materials
Reactor Vessel	Pressure Plate	SA-533, Grade B, Class 1
	Replacement RV Head Forging	SA-508 Grade 3, Class 1
	Shell & Nozzle Forgings	SA-508 Class 3
	Cladding, Stainless Weld Rod	Type 304, 309L, 308L, SST or equivalent and Inconel
	Core Support	Inconel
	Thermal Shield and Internals	A-204, Type 304
	Insulation	Reflective Type (100% type 304 SS construction)
	Control Rod Housing	Inconel and Type 316
	Instrumentation Nozzles	Inconel
Steam Generators	Shell	SA-508 Gr3 Class 2
	Channel Head Castings	SA-508 Gr3 Class 2
	Channel Head Cladding	Type 308L & 309L
	Primary Nozzle Safe Ends	SA-182F 316LN
	Primary Divider Plate	Alloy 690
	Tubesheet	SA-508 Gr3 Class 2
	Tubes	Alloy 690 Thermally Treated
	Cladding for Tube Sheets	Alloy 82 & 182
	Primary & Secondary Separators Frames	SA515, Grade 60 or Equivalent
	Dryer Vanes	SA240, Type 304L

TABLE 4.1-1 MATERIALS OF CONSTRUCTION OF THE REACTOR COOLANT SYSTEM COMPONENTS
Page 2 of 2

Component	Section	Materials	
Pressurizer	Shell	SA-533, Grade A, Class 1	
	Heads and Nozzles	SA-216 Grade WCC	
	External Plate (Skirt)	SA-516, Grade 70	
	Cladding, Stainless	Type 304 or equivalent	
	Internal Plate	SA-240 Type 304	
	Internal Piping	SA-376 Type 316	
Pressurizer Relief Tank	Shell	A-285 GR C	01450457
	Heads	A-285 GR C	
Piping	Pipes	Unit 1: A-376 Type 316	
		Unit 2: A-351, CF8M	
	Fittings	A-351, CF8M	
	Nozzles	A-182 F316	
Pump	Shaft	Type 347	
	Impeller	A-351, CF8	
	Casing	A-351, CF8	
Valves	Pressure Containing Parts	A-351, CF8M and A-182 F316	

TABLE 4.1-2 REACTOR COOLANT SYSTEM DESIGN PARAMETERS AND PRESSURE SETTINGS

Total Primary (NSSS) Heat Output, MWt	1690	01186822
Total Primary (NSSS) Heat Output, Btu/hr	5767 x 10^6	
Number of Loops	2	
Coolant Volume (Liquid), including pressurizer volume, at full power, ft^3	6267 (Unit 1) 6191 (Unit 2)	
Total Reactor Coolant Flow, lb/hr	137.8 x 10^6 (1)	01193868
	Pressure, psig	
Design Pressure	2485	
Operating Pressure (at pressurizer)	2235	
Safety Valves	2485	
Power Relief Valves	2335	
Pressurizer Spray Valves (open)	2260	
High Pressure Trip	2370	
High Pressure Alarm	2335	
Low Pressure Trip	1900	
Hydrostatic Test Pressure (Cold), Initial hydro only	3107	

(1) The minimum RCS Flow required by safety analyses is 178,000 gpm total vessel flow. 01193868

TABLE 4.1-3 REACTOR VESSEL DESIGN DATA

Design/Operating Pressure, psig	2485/2235	
Hydrostatic Test Pressure, psig	3107	
Design Temperature, °F	650	
Overall height of Vessel and Closure Head, ft-in. (Bottom Head O.D. to top of Control Rod Mechanism Housing)	39 - 1.4	
Water Volume (with core and internals in place), ft^3	2473	
Thickness of Insulation, min., in.	3	
Number of Reactor Closure Head Studs	48	
Diameter of Reactor Closure Head Studs, in.	6	
Flange, ID, in.	123.8	
Flange, OD, in.	157.3	
ID at Shell, in.	132	
Inlet Nozzle ID, in.	27.5	
Outlet Nozzle ID, in.	29.0	
Clad Thickness, min., in.	0.125	
Lower Head Thickness, min., in.	4.252	
Vessel Belt-Line Thickness, min., in.	6.7	
Closure head Thickness, in.	5.51	
Reactor Coolant Inlet Temperature, °F[1]	532.2 / 527.4[2]	
Reactor Coolant Outlet Temperature, °F[1]	607.4/592.6[2]	01186822
Reactor Coolant Flow, lb/hr (2 loop total)	137.8×10^6	
Safety Injection Nozzle, number/size, in.	2/4	

1. The Inlet and Outlet Temperature used in the thermal stress and fatigue analysis for design of the Replacement Reactor Vessel closure head, CRDMs, and Core Exit Thermocouple Nozzle Assemblies are 527.4° and 611.3°F respectively.

2. The qualified inlet and outlet temperatures used in the original Unit 1 and Unit 2 reactor vessel stress analysis are 532.2°F and 607.4°, respectively (Reference 137). The original thermal stress intensities bound the revised thermal stresses at the revised inlet and outlet temperatures of 527.4° and 592.6°, respectively. The revised inlet and outlet temperatures of 527.4° and 592.6°, respectively, are used in the RPV fatigue transient analysis and in the analysis to develop the LOCA blowdown loads in the RPV. 01186822

TABLE 4.1-4 PRESSURIZER & PRESSURIZER RELIEF TANK DESIGN DATA
Page 1 of 2

Pressurizer	
Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure (cold), psig	3107
Design/Operating Temperature, °F	680/653
Water Volume, Full Power, ft^3 [(1)]	600
Steam Volume, Full Power, ft^3	400
Surge Line Nozzle Diameter, in./Pipe Schedule	14/140
Shell ID, in./Minimum Shell Thickness, in.	84/4.1
Minimum Clad Thickness, in.	0.188
Electric Heaters Capacity, kw (total)	1000[(2)]
Heatup rate of Pressurizer using Heaters only, °F/hr	55 (Approx.)[(2)]
Power Relief Valves	
Number	2
Set Pressure (open), psig	2335
Capacity, lb/hr Saturated steam/valve	179,000
Safety Valves	
Number	2
Set Pressure, psig	2485
Capacity (ASME rated flow) lb/hr/valve	345,000

(1) 60% of net internal volume (maximum calculated power)

(2) These are original design values. Due to operational and maintenance considerations, pressurizer heater capacity may be reduced below the original design value; which results in a slower heat up rate.

TABLE 4.1-4 PRESSURIZER & PRESSURIZER RELIEF TANK DESIGN DATA
Page 2 of 2

Pressurizer Relief Tank

Design pressure, psig	100	
Rupture Disc Release Pressure at 120°F, psig	99	00249
Design temperature, °F	340	
Normal water temperature, °F	120	
Total volume, ft^3	800	
Rupture Disc Relief Capacity, lb/hr	8.0×10^5	

TABLE 4.1-5 STEAM GENERATOR DESIGN DATA

	Unit 1 & 2
Number of Steam Generators Per Unit	2
Design Pressure, Reactor Coolant/Steam psig	2485/1085
Reactor Coolant Hydrostatic Test pressure (tube side-cold), psig	3107
Design Temperature, Reactor Coolant/Steam, °F	650/600
Reactor Coolant Flow, lb/hr/loop	38.4×10^6
Total Heat Transfer Surface Area, ft^2/S.G.	61,281
Heat Transferred, Btu/hr	2873×10^6
Steam Conditions at Full Load, Outlet Nozzle:	
Steam Flow, lb/hr/S.G., max. guaranteed (1690 MWt NSSS Power)	3.66×10^6
Steam Temperature, °F	518.4
Steam Pressure, psia	800.9
Feedwater Temperature, °F	436.2
Overall Height, ft-in	67-10
Shell OD, upper/lower, in.	175.1/134.9
Shell Thickness, upper/lower, in.	3.39/2.73
Number of U-tubes	4868
Stress Analysis	
RCS Pressure, Design/Normal, psig	2485 / 2235
SG Inlet Temperature, °F	599.1
SG Outlet Temperature, °F	535.5
Zero Power Average Temperature, °F	547.0
Fatigue Analysis	
SG Inlet Temperature, °F	592.6
SG Outlet Temperature, °F	527.4
Pipe Rupture Loads	
SG Inlet Temperature, °F	520.0
SG Outlet Temperature, °F	580.0
U-tube Diameter, OD, in.	0.75
Tube Wall Thickness, (average), in.	0.0429
Number of Manways/ID in. (2 primary and 2 secondary)	4/18
Number of handholes/ID, in.	6/8

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TABLE 4.1-5 STEAM GENERATOR DESIGN DATA
Page 2 of 2

	1684 MWt NSSS Power (max. guaranteed)	**Zero Power**
	Unit 1 & 2	**Unit 1 & 2**
Reactor Side Coolant Water Volume, ft^3	1140	1140
Primary Side Fluid Heat Content, Btu	29.5×10^6	29.2×10^6
Secondary Side Water Volume, ft^3	2141.9	3190.5
Secondary Side Steam Volume, ft^3	3525.3	2476.7
Secondary Side Fluid Heat Content, Btu	60.2×10^6	87.1×10^6

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Notes:

(1) The values defined in Table 4.1-5 envelop operation at 1684 MWt NSSS Power, with zero percent steam generator tube plugging and with an average Reactor Coolant temperature of 560°F.

TABLE 4.1-6 REACTOR COOLANT PUMPS DESIGN DATA
Page 1 of 2

Number of Pumps	2
Design Pressure/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure (cold), psig	3107
Design Temperature (casing), °F	650
RPM at Nameplate Rating	1189
Suction, Temperature, °F	544.5
Net Positive Suction Head, ft.	172
Developed Head, ft	259/277*
Capacity, gpm/pump	89,000/88,500*
Seal Water Injection, gpm/pump	8
Seal Water Return, gpm/pump	3
Pump Discharge Nozzle, ID, in.	27 - 1/2
Pump Suction Nozzle ID, in.	31
Overall Unit Height, ft.	27
Water Volume, ft^3/pump	56
Pump-Motor Moment of Inertia, $lb\text{-}ft^2$	80,000 (21 RCP is slightly higher)
Motor Data:	
Type	AC Induction Single Speed, Air Cooled
Voltage	4000

* For Unit 2 - #21 RCP only.

TABLE 4.1-6 REACTOR COOLANT PUMPS DESIGN DATA
Page 2 of 2

Insulation Class	B Thermalastic Epoxy
Phase	3
Frequency, Hz	60
Starting Current, maximum, amp	4800
Input (hot reactor coolant), kw, approx.	4600
Input (cold reactor coolant), kw, approx.	6000
Power, HP (nameplate)	6000

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TABLE 4.1-7 REACTOR COOLANT PIPING DESIGN DATA

	Unit 1	Unit 2
Design/Operating Pressure, psig	2485/2235	2485/2235
Hydrostatic Test Pressure, (cold) psig	3107	3107
Design Temperature, °F	650	650
Design Temperature, (pressurizer surge line), °F	680	680
Reactor Inlet Piping, ID, in.	27 - 1/2	27 - 1/2
Reactor Inlet Piping, minimum wall thickness, in.	2.215	2.56
Reactor Outlet Piping, ID, in.	29	29
Reactor Outlet Piping, minimum wall thickness, in.	2.335	2.70
Coolant Pump Suction Piping, ID, in.	31	31
Coolant Pump Suction Piping, minimum wall thickness, in.	2.495	2.88
Pressurizer Surge Line Piping, ID, in./Pipe Schedule*	10/140	10/140
Pressurizer Surge Line Piping, nominal thickness, in.	1	1
Water Volume, (2 loops) ft^3	565	565

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* Surge line fitted with a 14″/10″ adapter at the pressurizer

TABLE 4.1-8 REACTOR COOLANT SYSTEM OPERATING TRANSIENTS USED FOR DESIGN (60-YEAR PLANT LIFE)

	Operating Cycle	Occurrences+
1.	Heatup at 100°F/hr	200
	Cooldown at 100°F/hr (Pressurizer 200°F/hr)	200
2.	Plant Loading at 5% of nominal full load/min	18,300(2)
	Plant Unloading at 5% of nominal full load/min	18,300(2)
3.	Step Load Increase of 10% of nominal full load	2,000
	Step Load Decrease of 10% of nominal full load	2,000
4.	Large Step Decrease in Load (with steam dump)	200
5.	Loss of Load (without immediate turbine or reactor trip)	80
6.	Loss of Offsite Power (LOOP with natural circulation in Reactor Coolant System)	40
7.	Loss of Flow (partial loss of flow one pump only)	80
8.	Reactor Trip From Full Power	400
9.	Turbine Roll Test	10
10.	Hydrostatic Test Condition	
	a. Primary Side Hydrostatic Test Before Initial Startup at 3107 psig	5(1)
	b. Secondary Side Hydrostatic Test Before Initial Startup at 1357 psig for Unit 1 and 1356 for Unit 2	5
	c. Primary Side Leak Test at 2330 psig for Unit 1 and 2500 psia for Unit 2	50

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(1) The SGs were designed for 15 primary side hydrostatic tests

(2) The Reactor Vessel Internals Baffle-Former Bolts are restricted to no more than 1577 cycles (each for Plant Loading at 5% and Plant Unloading at 5%) to conservatively ensure that fatigue usage in the bolts remains less than 1.0.

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TABLE 4.1-8 REACTOR COOLANT SYSTEM OPERATING TRANSIENTS USED FOR DESIGN (60-YEAR PLANT LIFE)

Operating Cycle	Occurrences+
11. Accident Conditions	
a. Reactor Coolant Pipe Break	1
b. Steam Pipe Break	1
c. Steam Generator Tube Rupture	-
12. Steady state fluctuations - the reactor coolant average temperature for purposes of design is assumed to increase and decrease a maximum of 6°F in one minute. The corresponding reactor coolant pressure variation is less than 100 psi. It is assumed that an infinite number of such fluctuations will occur.	

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\+ Estimated for equipment design purposes (60-year life) and not intended to be an accurate representation of actual transients or to reflect actual operating experience.

TABLE 4.1-9 TYPICAL REACTOR COOLANT WATER CHEMISTRY

Electrical Conductivity	Determined by the concentration of boric acid and alkali present. Expected range is < 1 to 40 μ Mhos/cm at 25°C.	
Solution pH	Determined by the concentration of boric acid and alkali present. Expected values range between 4.2 (high boric acid concentration) to 10.5 (low boric acid concentration) at 25°C.	
Oxygen, ppm, max.	0.10	
Chloride, ppm, max.	0.15	
Fluoride, ppm, max.	0.15	
Hydrogen, cc (STP)/kg H_2O	25-50	05-034
Total Suspended Solids, ppm, max.	1.0	
pH Control Agent (Li^7 OH)	0.3×10^{-4} to 6.8×10^{-4} molal (equivalent to 0.22 to 5.0 ppm Li^7)	05-034
Boric Acid as ppm B	Variable from 0 to ~4000	

TABLE 4.1-10 REACTOR COOLANT SYSTEM DESIGN PRESSURE DROP

	Pressure Drop, psi
Across Pump Discharge Leg	1.61
Across Vessel, including nozzles	47.0
Across Hot Leg	1.71
Across Steam Generator	35.5*
Across Pump Suction Leg	3.31
Total Pressure Drop	89.13

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* Pressure drop across Steam Generators was calculated with loop flow of 38.4 x 10^6 lbm/Hr and zero percent steam generator tube plugging.

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TABLE 4.1-11 REACTOR COOLANT SYSTEM - CODE REQUIREMENTS

Component	Codes
Reactor Vessel	ASME III* Class A 1968 Edition Winter 1968 Addenda
Reactor Vessel Head	ASME III* Class 1 1998 Edition 2000 Addenda
Control Rod Drive Mechanism Housing	ASME III* Class 1 1998 Edition 2000 Addenda
Core Exit Thermocouple Nozzle Assemblies (CETNAs)	ASME III* Class 1 1998 Edition 2000 Addenda
Steam Generators	
Tube Side	ASME III* Class 1, 1995 Edition through 1996 Addenda 01406856
Shell Side***	ASME III* Class 2
Reactor Coolant Pump Casing	No Code (Design per ASME III-Article 4)
Pressurizer	ASME III* Class A Unit 1: 1965 Edition Summer 1966 Addenda Unit 2: 1965 Edition Winter 1966 Addenda
U2 Pressurizer Surge Nozzle Weld Overlay	ASME III* Class 1 1998 Edition 2000 Addenda Relief Request 2-RR-4-8 (Reference 151) 01404391
Pressurizer Relief Tank	ASME III* Class C
Pressurizer Safety Valves	ASME III* 1968 Edition
Reactor Coolant Piping	Unit 1: ASA B31.1 1955**** Unit 2: USAS-B31.1.0-1967**
Reactor Coolant Gas Vent System Piping	ASME III* Class 1 & 2 1983 Edition
Reactor Vessel Level Instrumentation piping (from head penetration to isolation valve RC-17-3 [2RC-17-3])	ASME III* Class 1 & 2 1998 Edition 2000 Addenda

* ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

** USA Standard Code for Pressure Piping - Power Piping.

*** The shell side of the steam generator conforms to the requirements for Class 1 (1995 Edition through 1996 Addenda) and is so stamped. 01406856

**** American Standard Code for Pressure Piping

TABLE 4.2-1 SUMMARY OF CUMULATIVE FATIGUE USAGE AND STRESS INTENSITY FOR COMPONENTS OF THE REACTOR VESSEL [2]

ITEM	USAGE[1] FACTOR	STRESS INTENSITY (PSI)	ALLOWABLE STRESS INTENSITY PSI (OPERATING TEMPERATURE)
CRDM Housing[3]	0.62	44,400	60,000
Head Flange	0.082	53,120	80,100
Vessel Flange	0.04	50,940	80,000
Closure Studs	0.56	87,070	110,200
Primary Nozzles			
• Inlet Nozzle	0.0165	48,200	80,000
• Outlet Nozzle	0.035	54,200	
Vessel Support	0.01	41,400	80,000
Core Support Pad	0.021	57,100	69,900
Bottom Head to Shell	0.006	44,710	80,100
Bottom Instrumentation Nozzles	0.102	39,600	70,000
Safety Injection Nozzle	0.57	49,500	80,000

1 As defined in ASME Section III code

2 Stress intensity and fatigue usage apply for the reactor vessel operating parameters defined in Table 4.1-3.

3 At the J-groove weld for the CRDM Head penetration.

TABLE 4.2-2 PRESSURIZED THERMAL SHOCK REFERENCE TEMPERATURE INPUT PARAMETERS AND CALCULATION RESULTS
Page 1 of 6

UNIT 1 - 35 EFPY

Component	Initial RT_{NDT} (*F)	Source of RT_{NDT}	M (*F)	Wt % Cu	WT % Ni	CF (*F)	Fluence [a] (n/cm^2)	RT_{PTS} (*F)
Nozzle Shell Forging (B)	-4	Measured	34	0.08	0.68	51	2.20E+19	92
Nozzle to Intermediate Shell Weld (W2)	0	Generic	66	0.15	0.15	79.5	2.20E+19	162
Intermediate Shell Forging	14	Measured	34	0.07	0.80	44	3.95E+19	107
Intermediate Shell Forging Using Surveillance Data	14	Measured	34	NA[b]	NA[b]	54.7	3.95E+19	122
Intermediate to Lower Shell Weld (W3)	-13	Measured	56	0.13	0.13	69.7	3.95E+19	137
Intermediate to Lower Shell Weld (W3) Using Surveillance Data	-13	Measured	56	NA[b]	NA[b]	81	3.95E+19	152
Lower Shell Forging	-4	Measured	34	0.07	0.66	44	3.95E+19	89

Note: The data in this table comes from Reference 108, with fluence determined by the methodology of Reference 107. This table is retained to document the fluence and **RT_{PTS}** values used to determine the current PTLR setpoints and limitations. 01451781

[a] Peak clad/base metal interface fluence.

[b] Value is derived from the material specific chemistry factors - elemental percentages not used.

TABLE 4.2-2 PRESSURIZED THERMAL SHOCK REFERENCE TEMPERATURE INPUT PARAMETERS AND CALCULATION RESULTS
Page 2 of 6

UNIT 1 - 54 EFPY

Component	Initial RT_{NDT} (*F)	Source of RT_{NDT}	M (*F)	Wt % Cu	WT % Ni	CF (*F)	Fluence [a] (n/cm^2)	RT_{PTS} (*F)
Nozzle Shell Forging	-4	Measured	34	0.08	0.68	51	1.770E+19	89
Nozzle to Intermediate Shell Weld (W2)	0	Generic	66	0.15	0.15	79.5	1.770E+19	157
Intermediate Shell Forging	14	Measured	34	0.07	0.80	44	5.162E+19	110
Intermediate Shell Forging Using Surveillance Data	14	Measured	34	NA[b]	NA[b]	54.7	5.162E+19	125
Intermediate to Lower Shell Weld (W3)	-13	Measured	56	0.13	0.13	69.7	4.969E+19	141
Intermediate to Lower Shell Weld (W3) Using Surveillance Data	-13	Measured	56	NA[b]	NA[b]	80.8	4.969E+19	156
Lower Shell Forging	-4	Measured	34	0.07	0.66	44	5.026E+19	92

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Note: The data in this table comes from References 141 and 142, with fluence determined by the methodology of Reference 140.

[a] Peak clad/base metal interface fluence.

[b] Value is derived from the material specific chemistry factors - elemental percentages not used.

TABLE 4.2-2 PRESSURIZED THERMAL SHOCK REFERENCE TEMPERATURE INPUT PARAMETERS AND CALCULATION RESULTS

Page 3 of 6

UNIT 2 - 35 EFPY

<u>Component</u>	Initial RT_{NDT} (*F)	Source of RT_{NDT}	M (*F)	Wt % Cu	WT % Ni	CF (*F)	Fluence [a] (n/cm^2)	RT_{PTS} (*F)	
Nozzle Shell Forging	-13	Measured	34	0.07	0.73	44	2.38E+19	75	01514777
Nozzle to Intermediate Shell Weld (W2)	-13	Measured	56	0.13	0.13	70	2.38E+19	129	
Nozzle to Intermediate[c] Shell Weld (W2) Using Surveillance Data	-13	Measured	56	NA[b]	NA[b]	81	2.38E+19	143	01514777
Intermediate Shell Forging	14	Measured	34	0.07	0.75	44	4.18E+19	108	
Intermediate to Lower Shell Weld (W3)	-31	Measured	56	0.09	0.11	52	4.18E+19	96	
Intermediate to Lower Shell Weld (W3) Using Surveillance Data	-31	Measured	28	NA[b]	NA[b]	80	4.18E+19	106	01514777
Lower Shell Forging	-4	Measured	34	0.08	0.67	51	4.18E+19	100	
Lower Shell Forging Using Surveillance Data	-4	Measured	34	NA[b]	NA[b]	60	4.18E+19	112	

Note: The data in this table comes from Reference 109, with fluence determined by the methodology of Reference 107. This table is retained to document the fluence and **RT_{PTS}** values used to determine the current PTLR setpoints and limitations. 01451781

[a] Peak clad/base metal interface fluence.

[b] Value is derived from the material specific chemistry factors - elemental percentages not used.

[c] Data from Unit 1 W3 surveillance weld (same weld wire heat #1752) with the exception of fluence.

TABLE 4.2-2 PRESSURIZED THERMAL SHOCK REFERENCE TEMPERATURE INPUT PARAMETERS AND CALCULATION RESULTS

UNIT 2 - 54 EFPY

<u>Component</u>	Initial RT_{NDT} (*F)	Source of RT_{NDT}	M (*F)	Wt % Cu	WT % Ni	CF (*F)	Fluence [a] (n/cm^2)	RT_{PTS} (*F)
Nozzle Shell Forging {B}	-13	Measured	34	0.07	0.73	44	1.743E+19	72
Nozzle to Intermediate Shell Weld (W2)	-13	Measured	56	0.13	0.13	69.7	1.743E+19	123
Nozzle to Intermediate[c] Shell Weld (W2) Using Surveillance Data	-13	Measured	56	NA[b]	NA[b]	80.8	1.743E+19	136
Intermediate Shell Forging	14	Measured	34	0.07	0.75	44	5.196E+19	110
Intermediate to Lower Shell Weld (W3)	-31	Measured	56	0.09	0.11	51.6	5.043E+19	97
Intermediate to Lower Shell Weld (W3) Using Surveillance Data	-31	Measured	28	NA[b]	NA[b]	80.2	5.043E+19	110
Lower Shell Forging	-4	Measured	34	0.08	0.67	51	5.112E+19	102
Lower Shell Forging Using Surveillance Data	-4	Measured	34	NA[b]	NA[b]	59.6	5.112E+19	114

Note: The data in this table comes from References 141 and 143, with fluence determined by the methodology of Reference 140.

[a] Peak clad/base metal interface fluence.

[b] Value is derived from the material specific chemistry factors - elemental percentages not used.

[c] Data from Unit 1 W3 surveillance weld (same weld wire heat #1752) with the exception of fluence.

TABLE 4.2-2 PRESSURIZED THERMAL SHOCK REFERENCE TEMPERATURE INPUT PARAMETERS AND CALCULATION RESULTS

NOTES:

$$RT_{PTS} = RT_{NDT(u)} + M + \Delta RT_{PTS}$$

and

$$M = 2\sqrt{\sigma_u^2 + \sigma_\Delta^2}$$

and

$$\Delta RT_{PTS} = CF * f^{(0.28 - 0.10 Log f)}$$

Where:

$RT_{NDT(u)}$	=	The initial reference temperature (RT_{NDT}) of the unirradiated material.
M	=	The margin to be added to cover uncertainties in the values of RT_{NDT}, copper and nickel contents, fluence and the calculational procedures. M is evaluated per the equation above.
σ_u	=	The standard deviation for $RT_{NDT(u)}$ and is 0°F for measured values and 17°F for generic values.
σ_Δ	=	The standard deviation for ΔRT_{PTS}. The value for forgings is 17°F when surveillance data is not used or is used but is not credible and 8.5°F when surveillance data is used and is credible. The value for weld material is 28°F when surveillance data is not used or is used but is not credible and is 14°F when surveillance data is used and is credible. (Where Credibility is defined in 50CFR50.61.)

TABLE 4.2-2 PRESSURIZED THERMAL SHOCK REFERENCE TEMPERATURE INPUT PARAMETERS AND CALCULATION RESULTS

NOTES:

ΔRT_{PTS} = The mean value of the adjustment in the Pressurized Thermal Shock reference temperature caused by irradiation.

CF = The chemistry factor, a function of copper and nickel content of the metal defined in Tables 1 and 2 of 10CFR50.61. Alternately, with credible surveillance data, the chemistry factor is defined by the following formula:

$$CF = \frac{\sum\left[A_i * f_i^{\,(0.28 - 0.10\log f_i)}\right]}{\sum\left[f_i^{\,0.56 - 0.20\log f_i}\right]}$$

, Where A_i is the measured value of ΔRT_{PTS} for each surveillance data point

f = The best estimate neutron fluence in units 10^{19} n/cm^2 (E greater than 1 MeV). The fluence values utilized in the RT_{PTS} calculations were conversely based on the calculated maximum fast neutron flux at the reactor vessel clad-base metal interface. The calculated maximum fast neutron flux was obtained from References 105 and 106.

TABLE 4.3-1 through TABLE 4.3-10, DELETED

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TABLE 4.3-11 JULY 1981 REACTOR COOLANT PUMP TEST DATA 98135

INITIAL TEST CONDITIONS					LOOP A RAW DATA		LOOP A CORRECTED		LOOP B RAW DATA		LOOP B CORRECTED		LOOP A ΔP ELBOW		LOOP B ΔP ELBOW	
Mode	**Temp**	**Press**	**Date**	**Time**	**CFS**	**GPM**	**CFS**	**GPM**	**CFS**	**GPM**	**CFS**	**GPM**	**MA**	**GPM**	**MA**	**GPM**
A 21	318 (°F)	370 (PSIG)	7/22	2023	237.5	106,550	235.5	105,671	-33.1	-14,854	-32.9	-14,827	.551	110,549	-	-
B 22	323 (°F)	360 (PSIG)	7/22	2005	-32.2	-14,450	-31.9	-14,372	235.3	105,608	233.9	105,013	-	-	.540	109,439
A+B 21 + 22	325 (°F)	360 (PSIG)	7/22	1947	219.8	98,652	218.0	97,857	217.4	97.560	216.2	97.015	.4860	103,818	.4730	102,425
A 21	548 (°F)	2260 (PSIG)	7/23	1630	239.4	107,418	239.3	107,379	-32.2	-14,435	-32.3	-14,479	.4795	103,120	-	-
B 22	548 (°F)	2260 (PSIG)	7/23	1630	-34.6	15,512	-34.6	-15,519	237.1	106,385	237.6	106,599	-	-	.4731	102,439
A+B 21 + 22	548 (°F)	2260 (PSIG)	7/23	1550	221.8	99,534	221.7	99,484	219.0	98,323	219.5	98,520	.4244	97,019	.4187	96,698
A 21	NO DATA TAKEN															
B 22	NO DATA TAKEN															
A+B 21 + 22	530.2 (°F)	2235 (PSIG)	7/28	1000	220.1	98.772	219.9	98,673	217.6	97,665	217.8	97,763	.4262	97,224	.4193	96,440

TABLE 4.3-12 JULY, 1981 RCS FLOW TESTING COMPARISON WITH PREVIOUS MEASUREMENTS

TEST (1981) CONDITIONS			LOOP A 1981	LOOP B 1981	LOOP A 1980	LOOP B 1980	LOOP A 1974	LOOP B 1974	COMMENTS	
Mode	**Temp**	**Press**	**GPM**	**GPM**	**GPM**	**GPM**	**GPM**	**GPM**		
A 21	318 (°F)	370 (PSIG)	105,671	-14,827	- - -	- - -	105,600 (-0.07%)	-15,400 (3.7%)	During Heatup	97117
B 22	323 (°F)	360 (PSIG)	-14,372	105,013	- - -	- - -	-15,600 (-7.9%)	106,300 (1.21%)	During Heatup	
A + B 21 + 22	325 (°F)	360 (PSIG)	97,857	97,015	97,112 (-0.76%)	97,399 (0.39%)	97,400 (-0.47%)	98,900 (0.90%)	During Heatup	97117
A 21	548 (°F)	2260 (PSIG)	107,379	-14,479	- - -	- - -	107,000 (-0.35%)	-14,700 (1.5%)	At Hot Shutdown	
B 22	548 (°F)	2260 (PSIG)	-15,519	106,599	-15,304 (-1.4%)	106,823 (0.31%)	-15,700 (1.15%)	108,200 (1.48%)	At Hot Shutdown	
A + B 21 + 22	548 (°F)	2260 (PSIG)	99,484	98,520	98,235 (-1.3%)	98,668 (0.15%)	98,900 (-0.59%)	99,700 (1.18%)	At Hot Shutdown	
A + B 21 + 22	530.2 (°F)	2235 (PSIG)	98,673	97,763	97,519 (-1.2%)	97.950 (0.19%)	- - -	- - -	At 100% Power	

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TABLE 4.3-13 PRIMARY-SECONDARY BOUNDARY COMPONENTS

CONDITION: FAULTED

interface primary/secondary = ΔP = 2.485 KSI Temp. = 607°F

ΔP = 1.005 KSI Temp. = 547°F

Location	Description*	Code Limit	Primary membrane Stress	
			Unit 1	**Unit 2**
4	Lower assembly			
4.1	Tubesheet	0.7 S_u = 63 KSI	22.56 KSI	24.83 KSI
4.2	Junction tubesheet/ secondary shell	0.7 S_u = 63 KSI	35.82 KSI	35.84 KSI
4.3	Low secondary shell	0.7 S_u = 63 KSI	45.34 KSI	45.46 KSI
4.4	Junction tubesheet/ primary bend	0.7 S_u = 63 KSI	22.56 KSI	22.6 KSI
4.5	Primary head	0.7 S_u = 63 KSI	29.83 KSI	29.87 KSI

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*See Figure 4.3-15

TABLE 4.3-14
STEAM GENERATOR USAGE FACTORS (INDIVIDUAL TRANSIENTS)
PRIMARY AND SECONDARY BOUNDARY COMPONENTS

Normal Conditions

Number	Transients	Cycles
1	plant heat-up	200
2	plant cooldown	200
3	plant loading (5% per minute)	18,300
4	plant unloading (5% per minute)	18,300
5	small step load increase	2,000
6	small step load decrease	2,000
7	large step load decrease	200
14	hot standby operations	18,300
13	turbine roll test	10
8	steady state fluctuations	infinite

Upset Conditions

Number	Transients	Cycles
9	loss of load	80
10	loss of power	40
11	loss of flow	80
12	reactor trip @ full power	400
	OBE	50

Test Conditions

Number	Transients	Cycles
EHP	primary side hydrostatic test	15
EHS	secondary side hydrostatic test	5
Leak PS	primary to secondary leak tests	50
Leak PP	secondary to primary leak tests	120

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TABLE 4.3-15
STEAM GENERATOR TUBE SHEET STRESS ANALYSIS RESULTS
Page 1 of 2

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	Perforated tubesheet				Non perforated tubesheet			
Conditions	**P_m (KSI)**	**Criteria (KSI)**	**$P_m + P_b$ (KSI)**	**Criteria (KSI)**	**P_m (KSI)**	**Criteria (KSI)**	**$P_m + P_b$ (KSI)**	**Criteria (KSI)**
Design								
2,485/885 psi 650°/600°F	15.52 Unit 1 15.70 Unit 2	S_m = 30	33.29 Unit 1 33.69 Unit 2	1.5 S_m = 45				
445/1,085 psi 650°/600°F					16.1 Unit 1 16.15 Unit 2	S_m = 30	42.33 Unit 1 42.39 Unit 2	1.5 S_m = 45
Primary hydrotest								
3,107/0 psi 70°/F	28.24 Unit 1 28.34 Unit 2	0.9 S_y = 55.62	56.88 Unit 1 57.03 Unit 2	1.35 S_y = 83.43	19.49 Unit 1 19.53 Unit 2	0.9 S_y = 55.62	45.17 Unit 1 45.22 Unit 2	1.35 S_y = 83.43
Secondary hydrotest								
0/1,357 psi 70°F	11.96 Unit 1 12.07 Unit 2	0.9 S_y = 55.62	20.08 Unit 1 20.26 Unit 2	1.35 S_y = 83.43	19.48 Unit 1 19.53 Unit 2	0.9 S_y = 55.62	54.79 Unit 1 54.85 Unit 2	1.35 S_y = 83.43
Faulted								
2,485/0 psi 607°F	22.56 Unit 1 24.83 Unit 2	0.7 S_y = 63	45.4 Unit 1 49.76 Unit 2	1.05 S_y = 94.5	35.46 Unit 1 35.56 Unit 2	0.7 S_y = 63	45.85 Unit 1 45.98 Unit 2	1.05 S_y = 94.5
0/1,005 psi 547°F	8.843 Unit 1 12.01 Unit 2	0.7 S_y = 63	14.82 Unit 1 21.66 Unit 2	1.05 S_y = 94.5	35.82 Unit 1 35.91 Unit 2	0.7 S_y = 63	66.82 Unit 1 66.92 Unit 2	1.05 S_y = 94.5

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TABLE 4.3-15
STEAM GENERATOR TUBE SHEET STRESS ANALYSIS RESULTS
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Conditions	Junction tubesheet/primary head: P_m (KSI)	Criteria (KSI)	$P_m + P_b$ (KSI)	Criteria (KSI)	Junction tubesheet/lower secondary shell: P_m (KSI)	Criteria (KSI)	$P_m + P_b$ (KSI)	Criteria (KSI)
Design								
2,484/885 psi 650°/600°F	16.17 Unit 1 16.21 Unit 2	$S_m = 30$	30.32 Unit 1 30.37 Unit 2	1.5 $S_m = 45$				
445/1,085 psi 650°/600°F					16.1 Unit 1 16.15 Unit 2	$S_m = 30$	23.05 Unit 1 23.11 Unit 2	1.5 $S_m = 45$
Primary hydrotest								
3,107/0 psi 70°F	19.75 Unit 1 19.79 Unit 2	0.9 $S_y = 55.62$	45.17 Unit 1 45.22 Unit 2	1.35 $S_y = 83.43$	11.53 Unit 1 11.58 Unit 2	0.9 $S_y = 55.62$	15.99 Unit 1 16.05 Unit 2	1.35 $S_y = 83.43$
Secondary hydrotest								
0/1,357 psi 70°F	7.08 Unit 1 7.103 Unit 2	0.9 $S_y = 55.62$	13.92 Unit 1 13.97 Unit 2	1.35 $S_y = 83.43$	19.48 Unit 1 19.53 Unit 2	0.9 $S_y = 55.62$	29.09 Unit 1 29.15 Unit 2	1.35 $S_y = 83.43$
Faulted								
2,485/0 psi 607°F	22.56 Unit 1 22.6 Unit 2	0.7 $S_y = 63$	44.83 Unit 1 44.9 Unit 2	1.05 $S_y = 94.5$	35.74 Unit 1 35.84 Unit 2	0.7 $S_y = 63$	46.13 Unit 1 46.26 Unit 2	1.05 $S_y = 94.5$
0/1,005 psi 547°F	17.01 Unit 1 17.05 Unit 2	0.7 $S_y = 63$	20.88 Unit 1 20.92 Unit 2	1.05 $S_y = 94.5$	35.98 Unit 1 36.07 Unit 2	0.7 $S_y = 63$	46.62 Unit 1 46.74 Unit 2	1.05 $S_y = 94.5$

Note 1: Unit 1 stress values are obtained from RSG stress and fatigue analysis report BUCPRI/NGV1734.
Note 2: Unit 2 stress values are obtained from RSG stress and fatigue analysis report BUCPRI/NGV2584.

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TABLE 4.3-16
STEAM GENERATOR
PRIMARY HEAD AND SECONDARY SHELL STRESS ANALYSIS RESULTS 01406856

Conditions	**Primary head** P_m (KSI)	**Criteria (KSI)**	$P_m + P_b$ (KSI)	**Criteria (KSI)**	**Lower secondary shell** P_m (KSI)	**Criteria (KSI)**	$P_m + P_b$ (KSI)	**Criteria (KSI)**
Design								
2,485/885 psi 650°/600°F	16.83 Unit 1 16.84 Unit 2	$S_m = 30$	29.51 Unit 1 29.51 Unit 2	$1.5\ S_m = 45$				
445/1,085 psi 650°/600°F					24.22 Unit 1 24.2 Unit 2	$S_m = 30$	24.7 Unit 1 24.68 Unit 2	$1.5\ S_m = 45$
Primary hydrotest								
3,107/0 psi 70°/F	21.78 Unit 1 21.77 Unit 2	$0.9\ S_y = 55.62$	42.1 Unit 1 42.15 Unit 2	$1.35\ S_y = 83.43$	11.23 Unit 1 11.28 Unit 2	$0.9\ S_y = 55.62$	14.98 Unit 1 15.04 Unit 2	$1.35\ S_y = 83.43$
Secondary hydrotest								
0/1,357 psi 70°F	6.949 Unit 1 6.971 Unit 2	$0.9\ S_y = 55.62$	13.28 Unit 1 13.33 Unit 2	$1.35\ S_y = 83.43$	29.92 Unit 1 29.92 Unit 2	$0.9\ S_y = 55.62$	30.65 Unit 1 30.65 Unit 2	$1.35\ S_y = 83.43$
Faulted								
2,485/0 psi 607°F	29.83 Unit 1 29.87 Unit 2	$0.7\ S_y = 63$	46.50 Unit 1 46.58 Unit 2	$1.05\ S_y = 94.5$	40.78 Unit 1 40.90 Unit 2	$0.7\ S_y = 63$	55.12 Unit 1 55.27 Unit 2	$1.05\ S_y = 94.5$
0/1,005 psi 547°F	18.96 Unit 1 19.01 Unit 2	$0.7\ S_y = 63$	31.02 Unit 1 31.13 Unit 2	$1.05\ S_y = 94.5$	45.34 Unit 1 45.46 Unit 2	$0.7\ S_y = 63$	60.57 Unit 1 60.72 Unit 2	$1.05\ S_y = 94.5$

Note 1: Unit 1 stress values are obtained from RSG stress and fatigue analysis report BUCPRI/NGV1734.
Note 2: Unit 2 stress values are obtained from RSG stress and fatigue analysis report BUCPRI/NGV2584.

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TABLE 4.5-1
FAILURE MODES EFFECTS ANALYSIS FOR THE REACTOR COOLANT GAS VENT SYSTEM
Page 1 of 4

No.	Name	Failure Mode	Cause	Symptoms and Local Effects Including Dependent Failures	Method of Detection	Inherent Compensating Provision	Remarks and other Effects
1	Pressure Indicator P-729	a. spurious high pressure indication/alarm	Electro-mechanical failure, setpoint drift	No impact on normal operation. Loss of ability to detect leakage into the vent system piping.	Valve position indication in the control room	None	
		b. spurious low pressure indication	Electro-mechanical failure, set point drift	No impact on normal operation. Loss of ability to detect leakage into the vent system piping.	Valve position indication in the control room	None	
2	Pressurizer Relief Tank (PRT) Isolation Valve SV-37039 [37095] *	a. Fails Open	Mechanical Binding, Seat Leakage	Inability to isolate the pressurizer relief from the reactor coolant gas vent system.	Valve position indication in the control room	None	Redundant isolation valves to the reactor vessel and pressurizer preclude uncontrolled venting to the PRT.
		b. Fails Closed	Mechanical Failure, Loss of Power	No impact on normal operation. Inability to vent pressurizer or reactor to PRT.	Valve position indication in the control room. Operator	None	Venting to the containment is possible, if necessary.
3	Pressure Instrument Isolation Valves RC-13-1 [2RC-13-1] RC-13-2 [2RC-13-2]*	a. Fails Open	Mechanical Binding, Seat Leakage	None	Operator	Redundant Valves	
		b. Fails Closed	Mechanical Failure	Loss of ability to detect seat leakage from the pressurizer and reactor isolation valves into the reactor coolant gas vent system piping.	Operator	None	Unlikely event since valve is normally open and has only a manual operator

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TABLE 4.5-1
FAILURE MODES EFFECTS ANALYSIS FOR THE REACTOR COOLANT GAS VENT SYSTEM
Page 2 of 4

No.	Name	Failure Mode	Cause	Symptoms and Local Effects Including Dependent Failures	Method of Detection	Inherent Compensating Provision	Remarks and other Effects
4	Containment Isolation Valve SV-37040 [37096] *	a. Fails Open	Mechanical Binding, Seat Leakage	Inability to isolate reactor coolant vent system from containment.	High containment pressure and humidity if venting is in progress. Valve position indication in the control room.	None	
		b. Fails Closed	Mechanical Failure, Loss of Power to the Valve	No impact on normal operation. Inability to vent pressurizer or reactor to containment.	Valve position indication in the control room. Operator	None	Venting to the PRT is possible if necessary.
5	Pressurizer Vent Isolation Valve SV-37035 [37091] SV-37036 [37092] *	a. Fails Open	Mechanical Binding, Seat Leakage	No impact on normal operation. Inability to vent the reactor vessel without also venting pressurizer	Valve position indication in control room. P-729 high pressure indication.	None	Redundant isolation valves to containment, SV-37040 [37096] *, and PRT, SV-37039 [37095] * precludes uncontrolled venting of the pressurizer.
		b. Fails Closed	Mechanical Failure, Loss of Power	Inability to vent the pressurizer.	Valve position in the control room. Operator.	Parallel redundant isolation valve.	Parallel isolation valve allows venting of the pressurizer.

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TABLE 4.5-1
FAILURE MODES EFFECTS ANALYSIS FOR THE REACTOR COOLANT GAS VENT SYSTEM
Page 3 of 4

No.	Name	Failure Mode	Cause	Symptoms and Local Effects Including Dependent Failures	Method of Detection	Inherent Compensating Provision	Remarks and other Effects
6	Reactor Vessel Vent Isolation Valve SV-37037 [37093] * SV-37038 [37094]*	a. Fails Open	Mechanical Binding, Seat Leakage	No impact on normal operation. Unable to vent pressurizer without also venting the reactor vessel.	Valve position indication in the control room. P-729 high pressure indication.	None	Redundant isolation valves to containment, SV-37040 [37096] *, and PRT, SV-37039 [37095] *, precludes uncontrolled venting of the reactor vessel.
		b. Fails Closed	Mechanical Failure, Loss of Power	Inability to vent the reactor vessel.	Valve position in the control room. Operator.	Parallel redundant isolation valve.	Parallel isolation valve allows venting of the reactor vessel.
7	Position Indicator for SV-37037 [37093] SV-37038 [37094] *	False indication of valve position	Electro-mechanical failure.	Loss of ability to detect valve position in reactor vessel vent line.	Pressure gauge (P-729) indication shows valve is opened.	None	
8	Position indicator for SV-37035 [37091] SV-37036 [37092] *	False indication of valve position	Electro-mechanical failure.	Loss of ability to detect valve position in pressurizer vent line.	Pressure gauge (P-729) indication shows valve is opened.	None	
9	Position indicator for SV-37039 [37095] *	False indication of valve position	Electro-mechanical failure	Loss of ability to detect valve position in pressurizer relief tank vent line.	PRT temperature and pressure verify valve position. Pressure gauge (P-729).	None	

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TABLE 4.5-1
FAILURE MODES EFFECTS ANALYSIS FOR THE REACTOR COOLANT GAS VENT SYSTEM
Page 4 of 4

No.	Name	Failure Mode	Cause	Symptoms and Local Effects Including Dependent Failures	Method of Detection	Inherent Compensating Provision	Remarks and other Effects
10	Position indicator for SV-37040 [37096] *	False indication of valve position.	Electro-mechanical failure	Loss of ability to detect valve position in containment vent line.	Containment pressure/ humidity/ radiation levels verify containment valve position. Pressure gauge (P-729).	None	
11	Drain valves RC-14-1 [2RC-14-1] RC-14-2 [2RC-14-2] RC-14-3 [2RC-14-3] *	a. Seat Leakage	Contamination, Mechanical damage	No impact on system operation.	None	Drain lines are blind flanges.	
		b. Fails Closed	Mechanical Binding	No impact on normal operations. Inability to drain affected line section or test isolation valves IAW ASME XI.	Operator	None	
12	Orifice Bypass Isolation Valve RC-8-33 [2RC-8-33]	a. Seat Leakage	Contamination, Mechanical damage	No impact on system Operation	None	None	
		b. Fails Closed	Mechanical Binding	No impact on normal operations. Inability to use orifice bypass to support reactor drain down.	Operator	None	

* Unit 2 valve numbers are bracketed.

TABLE 4.7-1
REACTOR COOLANT SYSTEM QUALITY ASSURANCE PROGRAM
Page 1 of 3

Component	RT*	UT*	PT*	MT*	ET*
1. Replacement Steam Generator					
1.1 Tubesheet					
1.1.1 Forging		yes		yes	
1.1.2 Cladding		yes	yes		
1.2 Channel head					
1.2.1 Forging		yes		yes	
1.2.2 Cladding		yes	yes		
1.3 Shells and upper head					
1.3.1 Forging		yes		yes	
1.4 Tubes		yes			yes
1.5 Nozzles (forgings),		yes		yes	
Safe ends		yes	yes		
Taps				yes	
1.6 Weldments					
1.6.1 Circumferential welds	yes	yes	yes		
1.6.2 Channel head - tubesheet joints: cladding restoration		yes	yes		
1.6.3 Manway, recirculation and feedwater nozzles to shell	yes	yes	yes		
1.6.4 Attachments and taps			yes		
1.6.5 Tube to tubesheet			yes		
1.6.6 Temporary attachments after removal			yes**	yes***	
1.6.7 After hydrostatic test (weld)			yes**	yes***	
1.6.8 Nozzle safe ends (circumferential weld deposit)	yes	yes	yes		

* RT – Radiographic
UT – Ultrasonic
PT – Dye Penetrant
MT – Magnetic Particle
ET – Eddy Current
(+) Flat Surfaces Only
(++) Weld Deposit Areas Only
(+++) UT of Clad Bond-to-Base Metal
(++++) Or a UT and ET
** In Shop
*** On-site
\# Replacement Reactor Vessel Head & CRDMs

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TABLE 4.7-1
REACTOR COOLANT SYSTEM QUALITY ASSURANCE PROGRAM

Component	RT*	UT*	PT*	MT*	ET*
2. Pressurizer					
2.1 Heads					
2.1.1 Casting	yes			yes	
2.1.2 Cladding			yes		
2.2 Shell					
2.2.1 Plates		yes		yes	
2.2.2 Cladding			yes		
2.3 Heaters					
2.3.1 Tubing (++++)		yes	yes		
2.3.2 Centering of element	yes				
2.4 Nozzle		yes	yes		
2.5 Weldments					
2.5.1 Shell, longitudinal	yes			yes	
2.5.2 Shell, circumferential	yes			yes	
2.5.3 Cladding			yes		
2.5.4 Nozzle Safe End (forging)	yes		yes		
2.5.5 Instrument Connections			yes		
2.5.6 Support Skirt				yes	
2.5.7 Temporary Attachments after removal				yes	
2.5.8 All welds and cast heads after hydrostatic test				yes	
2.6 Final Assembly					
2.6.1 All accessible surfaces after hydrostatic test				yes	
3. Piping					
3.1 Fittings and Pipe (Castings)	yes		yes		
3.2 Fittings and Pipe (Forgings)		yes	yes		
3.3 Weldments					
3.3.1 circumferential	yes		yes		
3.3.2 Nozzle to run pipe (No RT for nozzles less than 3 inches)	yes		yes		
3.3.3 Instrument connections		yes	yes		

TABLE 4.7-1
REACTOR COOLANT SYSTEM QUALITY ASSURANCE PROGRAM
Page 3 of 3

Component	RT*	UT*	PT*	MT*	ET*
4. Pumps					
4.1 Castings	yes		yes		
4.2 Forgings					
4.2.1 Main shaft		yes	yes		
4.2.2 Main studs		yes	yes		
4.2.3 Flywheel (Rolled plate)		yes			
4.3 Weldments					
4.3.1 Circumferential	yes		yes		
4.3.2 Instrument connections			yes		
5. Reactor Vessel					
5.1 Forgings					
5.1.1 Flanges		yes		yes	
5.1.2 Studs		yes		yes	
5.1.3 Head adaptors		yes	yes		
5.1.4 Head adaptor tube		yes	yes		yes #
5.1.5 Instrumentation tube		yes	yes		
5.1.6 Main nozzles		yes		yes	
5.1.7 Nozzle safe ends (If forging is employed)		yes	yes		
5.2 Plates		yes		yes	
5.3 Weldments					
5.3.1 Main Steam	yes			yes	
5.3.2 CRD head adaptor connection			yes		yes #
5.3.3 Instrumentation tube connection			yes		
5.3.4 Main nozzles	yes			yes	
5.3.5 Cladding		yes (++++)	yes		
5.3.6 Nozzle-safe ends (if forging)	yes		yes		
5.3.7 Nozzle safe ends (if weld deposit)	yes		yes		
5.3.8 CRDM and RV Head Full Penetration Weld	yes #	yes #	yes #		
5.3.9 All welds after hydrotest		yes #	yes #	yes	
6. Valves					
6.1 Castings	yes		yes		
6.2 Forgings (No UT for valves two inch and smaller)		yes	yes		

TABLE 4.7-2
REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)
UNIT NO. 1

Component	Material Type	Cu [d] (%)	Ni [d] (%)	P [d] (%)	S [d] (%)	NDTT (°F)	Bounding Transverse [a] 50 ft lb/35 mils Lateral Expansion Temp. (°F)	RT_{NDT} (°F)	Average Transverse [a] Upper Shelf Impact Energy (ft lb)
Vessel Flange	A508 Cl. 3					-4	41 [c]	-4	77.5 [c]
Injection Nozzles	A508 Cl. 3					-22	-114 [c]	-22	97 [c]
Inlet and Outlet Nozzle	A508 Cl. 3					5	39 [c]	5	92 [c]
Upper Shell (B Course Shell)	A508 Cl. 3	0.08	0.68	0.012	0.009	-4	39 [c]	-4	84 [c]
Upper to Inter. Shell Weld (W2)	UM40 Wire/UM89 Flux	0.15	0.15	0.017	0.012	0 [e]	10 [c]	0 [e]	84 [c]
Inter. Shell (C Course Shell) [b]	A508 Cl. 3	0.07	0.80	0.015	0.007	14	14	14	143
Inter. to Lower Shell Weld (W3) [b]	UM40 Wire/UM89 Flux	0.13	0.13	0.019	0.013	-13	10	-13	78.5
HAZ [b]	HAZ					-13	-125	-13	211
Lower Shell (D Course Shell)	A508 Cl. 3	0.07	0.66	0.013	0.007	-4	45[c]	-4	88 [c]
Tran. Ring	A508 Cl. 3					5	63 [c]	5	79 [c]
Bottom Head	A533 Gr. B, Cl. 1					-4	57 [c]	-3	68.5 [c]

a. Specimen oriented transverse (weak direction) to the major working direction

b. Based on data through the surveillance program

c. Estimated using ‘‘Pressure-Temperature Limits,” Section 5.3.2 of *Standard Review Plan*, NUREG-0800, from longitudinal data.

d. Best Estimate Chemistry

e. Estimated using “Pressure-Temperature Limits,” Section 5.3.2 of *Standard Review Plan*, NUREG-0800.

NOTE: The beltline material data in this table comes from Reference 126.

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TABLE 4.7-3
REACTOR VESSEL TOUGHNESS (UNIRRADIATED)
UNIT NO. 2

Component	Material Type	Cu [d] (%)	Ni [d] (%)	P [d] (%)	S [d] (%)	NDTT (°F)	Bounding Transverse 50 ft lb/35 mils Lateral Expansion Temp. (°F)	RTndt (°F)	Average Transverse [a] Upper Shelf Impact Energy (ft lb)
Vessel Flange	A508 Cl. 3					-22	18 [c]	-22	88 [c]
Injection Nozzles	A508 Cl. 3					-22	-114 [c]	-22	97 [c]
Inlet and Outlet Nozzle	A508 Cl. 3					-13	50 [c]	-10	89 [c]
Upper Shell (B Course Shell)	A508 Cl. 3	0.07	0.73	0.008	0.010	-13	41 [c]	-13	85 [c]
Upper to Inter. Shell Weld (W2)	UM40 Wire/UM89 Flux	0.13	0.13	0.019	0.013	-13	10	-13	78.5
Inter. Shell (C Course Shell)	A508 Cl. 3	0.07	0.75	0.010	0.013	14	56 [c]	14	112 [c]
Inter. to Lower Shell Weld (W3) [b]	UM40 Wire/UM89 Flux	0.09	0.11	0.017	0.015	-31	-6	-31	103
HAZ [b]	HAZ					-31	-86	-31	117
Lower Shell (D Course Shell) [b]	A508 Cl. 3	0.08	0.67	0.009	0.013	-4	54	-4	108
Tran. Ring	A508 Cl. 3					10	50	10	76 [c]
Bottom Head	A533 gr. B, Cl. 1					-13	56	-4	68 [c]

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a. Specimen oriented transverse (weak direction) to the major working direction.
b. Based on data through the surveillance program.
c. Estimated using “Pressure–Temperature Limits,” Section 5.3.2 of Standard Review Plan, NUREG-0800, from longitudinal data.
d. Best Estimate Chemistry.
e. Estimated using “Pressure–Temperature Limits,” Section 5.3.2 of the Standard Review Plan, NUREG-0800.

NOTE: The beltline material data in this table comes from Reference 126.

TABLE 4.7-4
IDENTIFICATION OF REACTOR VESSEL BELTLINE REGION WELD METAL

			Weld Wire		Flux		
Weld Location	**Welding Process**	**Weld Control No.**	**Type**	**Heat No.**	**Type**	**Lot No.**	**Post Weld Heat Treatment**
Unit 1:							
Nozzle Shell to Inter. Shell Circle Seam W2	Submerged Arc	PS-011	UM40 *UM40	2269 3049	UM89 UM89	1180 1180	1022°F ± 50°F - 25 HRS+ 1148°F ± 45°F - 20 HRS-FC
Inter. Shell to Lower Shell Circle Seam W3	Submerged Arc	PS-011	UM40 *UM40	1752 3049	UM89 UM89	1230 1230	1022°F ± 50°F - 25 HRS+ 1148°F ± 45°F - 20 HRS-FC
Surveillance Weld - Same as the Inter. Shell to Lower Shell Circle Seam							1022°F - 5 HRS + 1112°F - 7 HRS-FC
Unit 2:							
Nozzle Shell to Inter. Shell Circle Seam W2	Submerged Arc	PS-011	UM40 *UM40	1752 3049	UM89 UM89	1263 1263	1022°F ± 50°F - 25 HRS+ 1148°F ± 45°F - 20 HRS-FC
Inter. Shell to Lower Shell Circle Seam W3	Submerged Arc	PS-011	UM40 *UM40	2721 3049	UM89 UM89	1263 1263	1022°F ± 50°F - 25 HRS+ 1148°F ± 45°F - 20 HRS-FC
Surveillance Weld - Same as the Inter. Shell to Lower Shell Circle Seam							1022°F - 10 HRS + 1112°F - 7 HRS-FC

*Used Only in Root Area of the Weld

NOTE: The data in this table comes from Reference 126. 02-006

TABLE 4.7-5
REACTOR VESSEL BELTLINE REGION WELD METAL CHEMICAL COMPOSITION

Weld wire		Flux		Weight Percent									
Type	**Heat No.**	**Type**	**Lot No.**	**C**	**Mn**	**P**	**S**	**Si**	**Cr**	**Ni**	**Mo**	**Cu**	**Co**
Unit 1:													
UM40	2269	UM89	1180	.055	1.24	.016	.012	.435	.025	.150	.445	.170	.023
UM40	3049	UM89	1180	.055	1.32	.020	.011	.545	.035	.080	.445	.115	.020
UM40	1752	UM89	1230	.099	1.36	.016	.013	.360	.060	.170	.450	.140	.018
UM40	3049	UM89	1230	.054	1.44	.019	.010	.425	.050	.085	.480	.115	.019
Surveillance Weld [a]				.063	1.45	.021	.014	.333	.016	0.109	.586	.138	0.017
Unit 2:													
UM40	1752	UM89	1263	.060	1.39	.018	.014	.410	.050	.140	.480	.140	.020
UM40	3049	UM89	1263	.062	1.33	.021	.010	.510	.035	.130	.480	.190	.022
UM40	2721	UM89	1263	.050	1.36	.016	.013	.420	.030	.130	.440	.090	.033
Surveillance Weld [a]				.050	1.26	.018	.017	.376	.024	.083	.524	.082	.016

[a] The chemistry of the surveillance weld is an average of all available data through the surveillance program. 02-006

NOTE: The data in this table comes from References 126, 60, and 61.

TABLE 4.7-6
MECHANICAL PROPERTIES OF REACTOR VESSEL BELTLINE REGION WELD METAL

Weld Wire		Flux		T_{NDT}	Energy at 10°F	RT_{NDT}	Shelf Energy	YS	UTS	Elong.	RA
Type	Heat No.	Type	Lot No.	°F	ft-lbs	°F	ft-lbs	ksi	ksi	%	&
Unit 1:											
UM40	2269	UM89	1180	o*	95.5, 74.1, 82.2, 93.7, 91.4	0*	- - - -	66.1	80.6	29.8	- - - -
UM40	3049	UM89	1180	o*	78.1, 88.0, 61.3, 93.7, 68.3	0*	- - - -	69.4	84.0	32.0	- - - -
UM40	1752	UM89	1230	0*	83.9, 82.2, 83.9, 82.2, 88.0	0*	- - - -	65.5	80.6	29.8	- - - -
UM40	3049	UM89	1230	0*	57.3, 78.1, 68.3, 93.7, 82.2	0*	- - - -	74.2	88.4	29.0	- - - -
Surveillance Weld				-13	75, 50, 52	-13	78.5	70.9	86.3	26.6	69.8
Unit 2:											
UM40	1752	UM89	1263	o*	88.0, 82.2, 79.9	0*	- - - -	68.3	84.0	31.0	- - - -
UM40	3049	UM89	1263	o*	45.1, 43.4, 48.6	0*	- - - -	69.4	85.0	27.6	- - - -
UM40	2721	UM89	1263	o*	79.9, 79.9, 78.1, 82.2, 82.2	0*	- - - -	69.4	84.0	28.4	- - - -
Surveillance Weld				-31		-31	103	66.5	80.2	27.4	73.8

02-006

02-006

***Estimated Per NRC Standard Review Plan Section 5.3.2

NOTE: The data in this table comes from References 126, 60, and 61.

02-006

TABLE 4.7-7
IDENTIFICATION OF REACTOR VESSEL BELTLINE REGION FORGINGS

Component	Forging No.	Heat No.	Material Spec. No.	Supplier	Heat Treatment: Austenitize	Temper	Stress Relief
Unit 1:							
Upper Shell (Nozzle Shell)	B	21744/38384	A508 CL3	SFAC			
Inter. Shell	C	21918/38566	A508 CL3	SFAC	1652-1715°F-5 HR-WQ 1652-1724°F-5 1/2 HR-WQ	1175-1238°F-5 HR-FC 1202-1238°F-5 HR-FC	1022°F-8 HR + 1112°F-14 HR-FC
Lower Shell	D	21887/38530	A508 CL3	SFAC	1652-1715°5 HR-WQ 1652-1716-r HR-WQ	1175-1238°F-5 HR-FC 1202-1238°F-5 HR-FC	1022°F-13 HR + 1112°F-7 HR-FC
Surveillance Material - Same as Inter. Shell Forging C							
Unit 2:							
Upper Shell (Nozzle Shell)	B	22231/39088	A508 CL3	SFAC			
Inter. Shell	C	22829	A508 CL3	SFAC	1650-1745°F-5 HR-WQ 1565-1655°F-5 1/2 HR-WQ	1170-1260°F-5 HR-FC 1200-1260°F-5 HR-FC	1022°F-12 HR + 1130°F-14-1/4 HR-FC
Lower Shell	D	22642	A508 CL3	SFAC	1652-1715°F-5 HR-WQ 1652-1724-5-1/2 HR-WA	1175-1238°F-5 HR-FC 1202-1238°F-5 HR-FC	1022°F-11-1/2 HR + 1112°F-7 HR-FC
Surveillance Material - Same as Inter. Shell Forging D							

NOTE: The data in this table comes from Reference 126.

02-006

TABLE 4.7-8
CHEMICAL COMPOSITION OF REACTOR VESSEL BELTLINE REGION FORGINGS

	Weight Percent										
Forging No.	**C**	**Mn**	**P**	**S**	**Si**	**Ni**	**Mo**	**Cu**	**Co**	**V**	
Unit 1:											
B	.175	.490	.012	.009	.280	.683	.490	.075	.017	less than .005	
C	.168	1.410	.015	.007	.280	.799	.480	.066	.010	less than .002	02-006
D	.170	1.430	.013	.007	.280	.660	.530	.068	.015	less than .005	
Surveillance Material Forging C - Analysis as Reported by Fabricator											
Unit 2:											
B	.180	1.340	.008	.010	.303	.730	.523	.065	.025	less than .002	
C	.165	1.320	.010	.013	.260	.750	.465	.073	.029	less than .005	
D	.175	1.220	.009	.013	.280	.674	.443	.078	.026	less than .005	02-006
Surveillance Material Forging D - Analysis as Reported by Fabricator											

NOTE: The data in this table comes from References 126, 60, and 61. 02-006

TABLE 4.7-9
UNIRRADIATED MECHANICAL PROPERTIES OF REACTOR VESSEL BELTLINE REGION FORGINGS

Forging No.	T_{NDT} °F	RT_{NDT} °F	Upper Shelf ft–lb MWD	Upper Shelf ft–lb NMWD	YS ksi	UTS ksi	Elong. %	RA %	
Unit 1:									
B	-4	-4	129.5	84*	72.7	91.5	26.0	69.6	
C	14	14	132	86	67.4	86.7	28.5	72.6	
D	-4	-4	135	88*	68.8	87.3	27.6	73.0	
C	- -	14	158	143	66.2	85.9	26.0	69.4	(Surveillance Test Data)
Unit 2:									
B	-13	-13	131	85*	70.5	91.3	25	69.2	
C	14	14	134	87	60.4	80.6	31	73.4	
D	-4	-4	123	80	65.8	87.9	28	70.5	
D	- - -	-4	150	108	65.6	85.2	26.5	67.4	(Surveillance Test Data)

***** Estimated From Data in the Major Working Direction (MWD) Per NRC Standard Review Plan Section 5.3.2.

NOTE: The data in this table comes from References 126, 60, and 61.

02-006
02-006
02-006

TABLE 4.7-10
REACTOR VESSEL SURVEILLANCE CAPSULE CONTENTS

Material	No. Charpy	No. Tensile	No. WOL*
Unit 1:			
Forging C - (Axial) (21918/38566)	12	3	3
Forging C - (Tangential) (21918/38566)	12	3	3
Unit 2:			
Forging D- Axial (22642)	12	3	3
Forging D - Tangential (22642)	12	3	3
Both Units:			
Weld Metal	8	3	3
Weld HAZ	8	-	-
Correlation	8	-	-
Dosimeters	Pure Cu		
	Pure Fe		
	Pure Ni		
	CoAl (0.15% Co)		
	CoAl (Cadmium shielded)		
	U238		
	NP237		
Thermal Monitors:	97.5 Pb, 2.5 Ag (579°FMP**)		
	97.5 Pb, 1.75 Ag 0.75 An (590°F MP**)		

* WOL = Wedge Opening Loading

** MP = Melting Point

Note: The data in this table comes from References 60 and 61. 02-006

TABLE 4.7-11
REACTOR VESSEL SURVEILLANCE CAPSULE REMOVAL SCHEDULE

Capsule	Vessel Location (deg)	Lead[c] Factor	Removal Time[a]	Estimated[c] Fluence (n/cm^2)
Unit 1:				
V	77°	2.94	1.34 (removed)	5.630×10^{18}[b]
P	247°	1.72	4.60 (removed)	1.318×10^{19}[b]
R	257°	2.99	8.56 (removed)	4.478×10^{19}[b]
S	57°	1.77	18.12 (Removed)[e]	4.017×10^{19}[b]
T	67°	1.89	Standby	- - - - -
N	237°	1.77	Standby	- - - - -
W[d]	257°	- - - - -	(EOC cycle 16 Removed)	- - - - -
Unit 2:				
V	77°	2.95	1.39 (removed)	6.206×10^{18}[b]
T	67°	1.75	4.00 (removed)	1.199×10^{19}[b]
R	257°	2.99	8.81 (removed)	4.376×10^{19}[b]
P	247°	1.84	17.24 (removed)	4.165×10^{19}[b]
N	237°	1.72	standby	- - - - -
S	57°	1.72	standby	- - - - -

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a. Effective full power years from plant startup

b. Actual fluence

c. Updated in Capsule S (U1) or Capsule P (U2) dosimetry analysis (Ref. 105 & 106).

d. This capsule contained test specimens from the Monticello vessel surveillance program. It was irradiated for 1.5 Prairie Island effective full power years as part of the Monticello surveillance capsule program.

e. Capsule S removed in place of Capsule T.

TABLE 4.7-12
SUMMARY OF FAST NEUTRON FLUENCE RESULTS FROM UNIT 1 SURVEILLANCE CAPSULE S

	Fast (E greater than 1.0 MeV) Neutron Fluence [(a)] (n/cm^2)	
Location	**Measured**	**Calculated**
Capsule S	$4.017x10^{19}$	$4.338x10^{19}$
Vessel Ir[b]	- - - - - -	$2.45x10^{19}$
Vessel 1/4T	- - - - - -	$1.56x10^{19}$
Vessel 3/4T	- - - - - -	$4.81x10^{18}$

a) Fluences are based on operation at 1650 MWt for 18.12 EFPY

b) Reactor vessel clad/base metal interface.

Note: The data in this table comes from Reference 105.

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TABLE 4.7-13
SUMMARY OF FAST NEUTRON FLUENCE RESULTS
FROM UNIT 2 SURVEILLANCE CAPSULE P

	Fast (E greater than 1.0 MeV) Neutron Fluence [(a)] (n/cm^2)	
Location	**Measured**	**Calculated**
Capsule P	$4.165x10^{19}$	$4.393x10^{19}$
Vessel Ir[b]	- - - - - -	$2.44x10^{19}$
Vessel 1/4T	- - - - - -	$1.56x10^{19}$
Vessel 3/4T	- - - - - -	$4.78x10^{18}$

a) Fluences are based on operation at 1650 MWt for 17.24 EFPY

b) Reactor vessel clad/base metal interface.

Note: The data in this table comes from Reference 106.

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TABLE 4.7-14
SUMMARY OF CALCULATED MAXIMUM (0° AZIMUTH)
REACTOR PRESSURE VESSEL
NEUTRON FLUENCE EXPOSURE

Pressure Vessel Clad/Base Metal Interface
Neutron Fluence [E > 1.0 MeV]

Prairie Island Unit 1

Location	35 EFPY	54 EFPY
Nozzle Shell Outlet Nozzle Weld	< 1 E+17	< 1 E+17
Nozzle Shell	1.214E+19	1.770E+19
Nozzle / Intermediate Shell Weld	1.214E+19	1.770E+19
Intermediate Shell	3.611E+19	5.162E+19
Intermediate / Lower Shell Weld	3.500E+19	4.969E+19
Lower Shell	3.546E+19	5.026E+19
Lower Shell Lower Weld	< 1 E+17	< 1 E+17

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Prairie Island Unit 2

Location	35 EFPY	54 EFPY
Nozzle Shell Outlet Nozzle Weld	< 1 E+17	< 1 E+17
Nozzle Shell	1.199E+19	1.743E+19
Nozzle / Intermediate Shell Weld	1.199E+19	1.743E+19
Intermediate Shell	3.576E+19	5.196E+19
Intermediate / Lower Shell Weld	3.481E+19	5.043E+19
Lower Shell	3.532E+19	5.112E+19
Lower Shell Lower Weld	< 1 E+17	< 1 E+17

Note: The data in this table comes from Reference 141.

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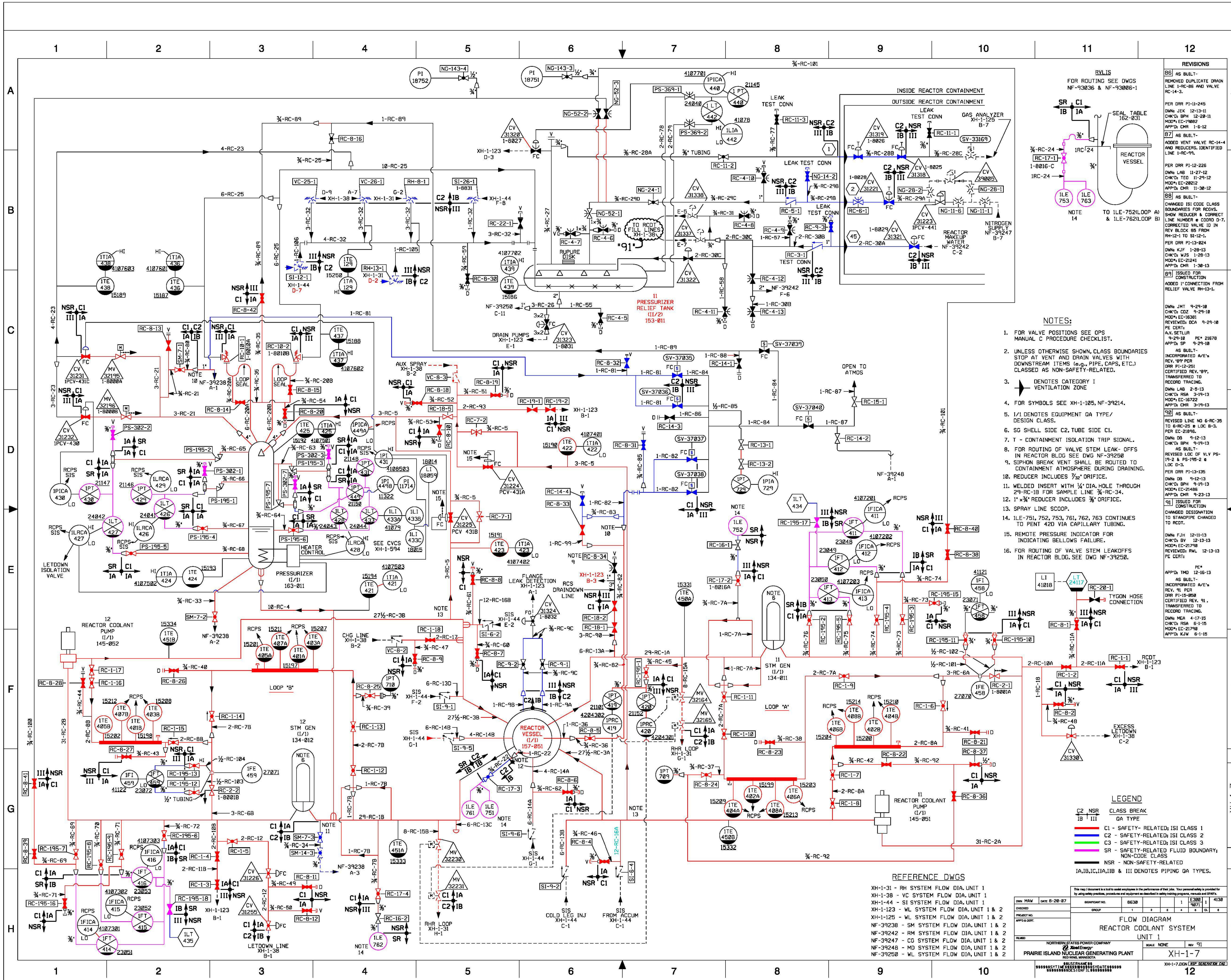


FIGURE 4.1-1A REV. 34

CAD FILE: U04101A.DGN

01516979

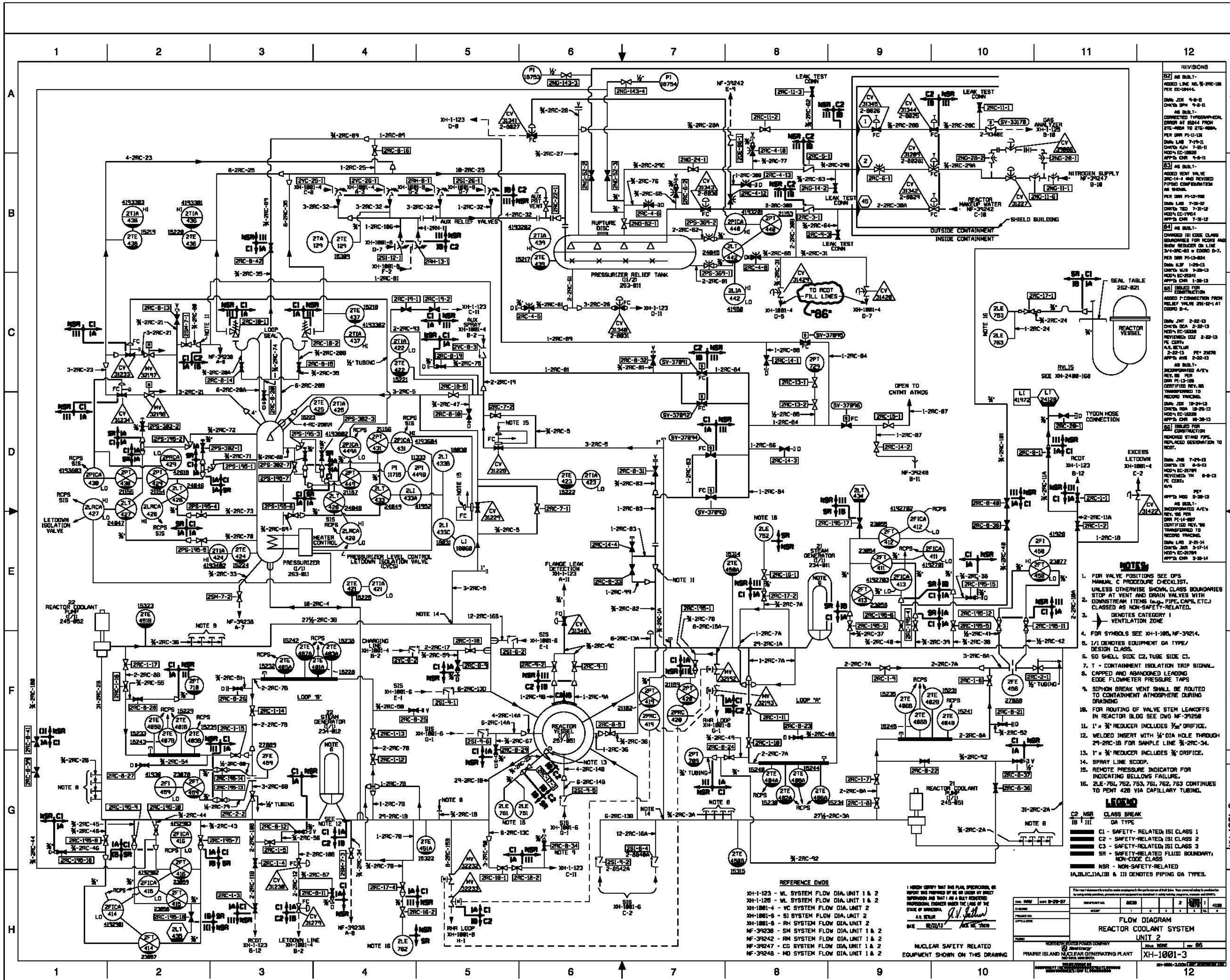
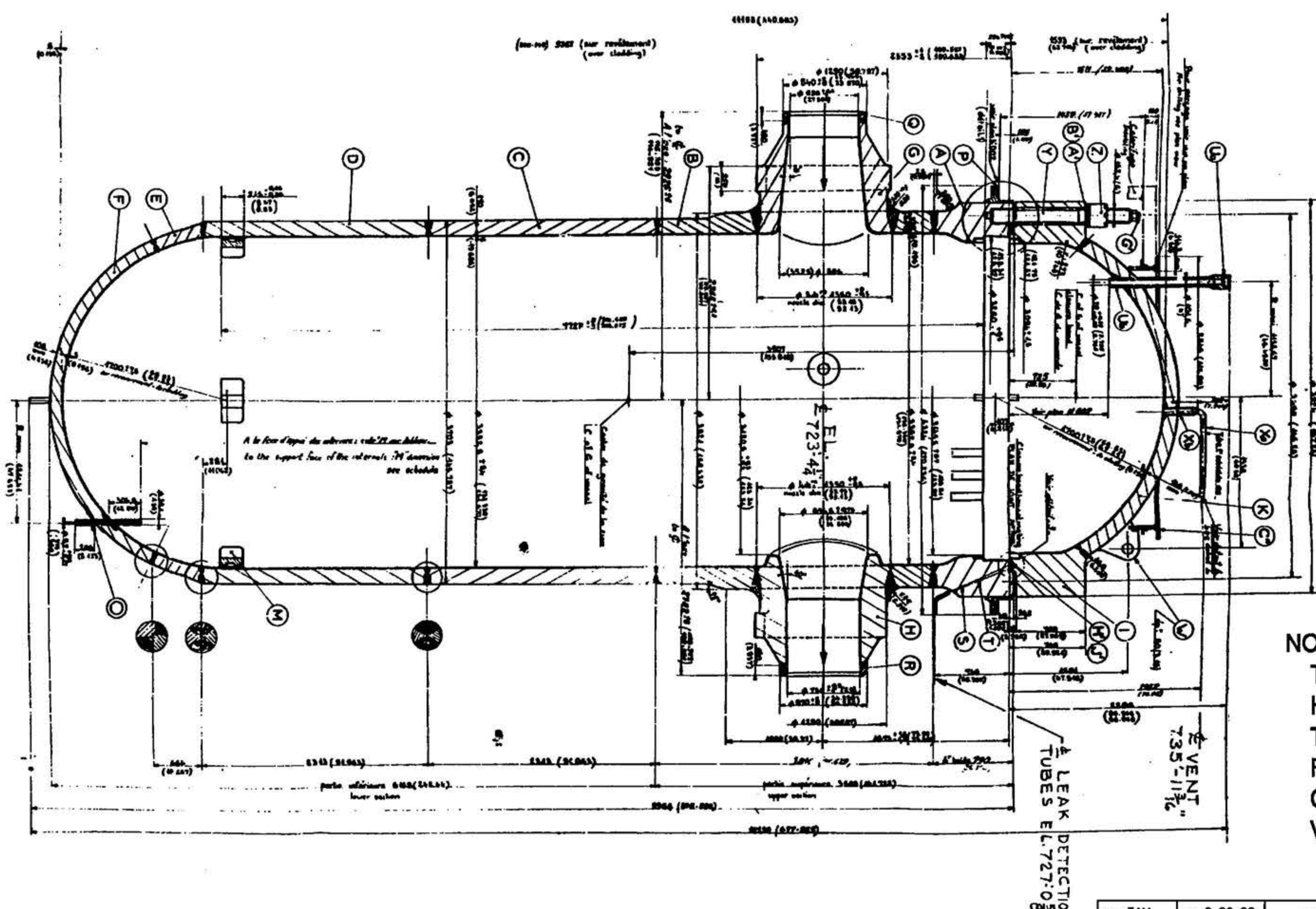


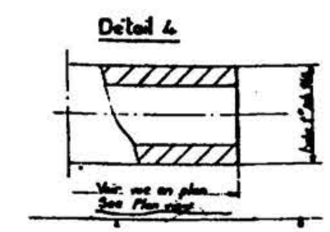
FIGURE 4.1-1B **REV. 33**

01429088

CAD FILE: U04101B.DGN

REVISIONS



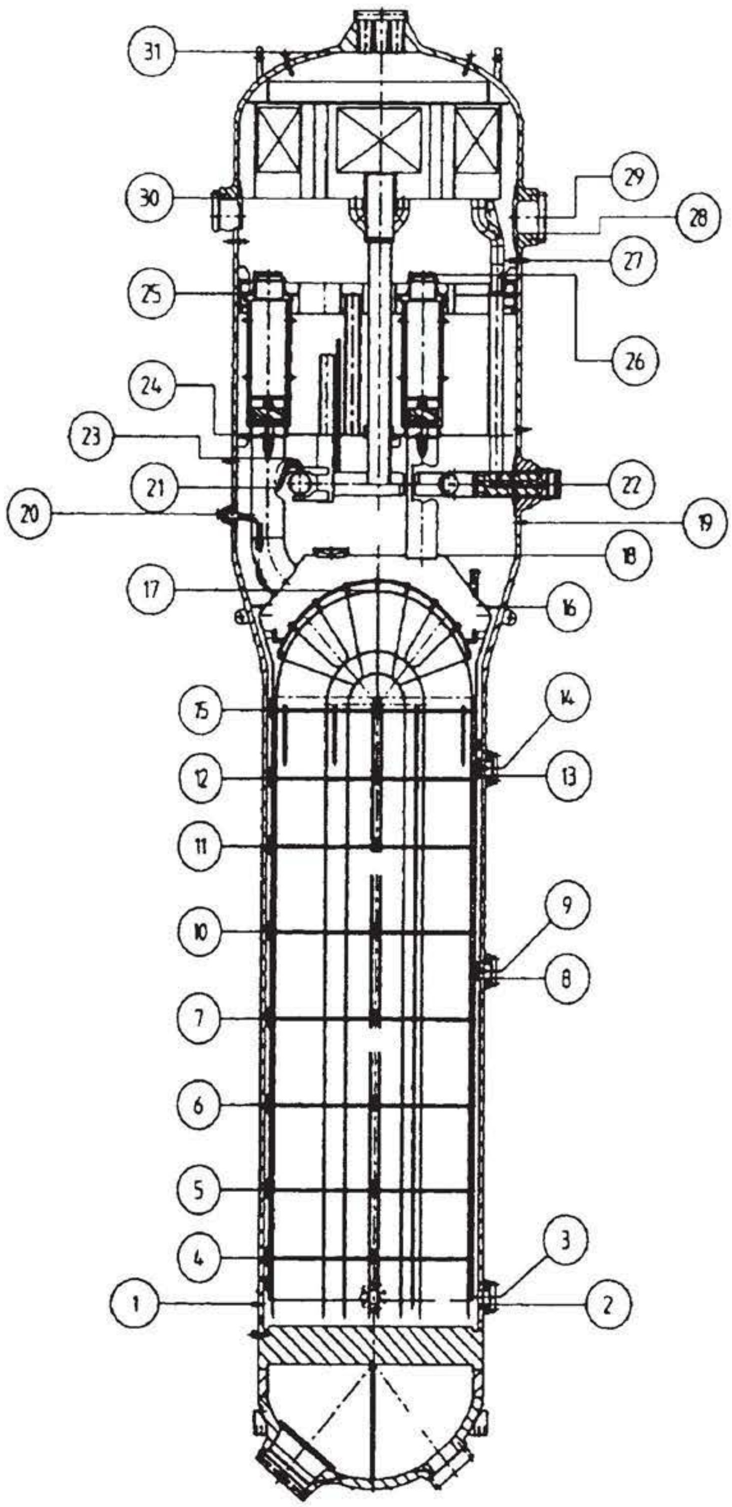


NOTE:
THIS DIAGRAM IS A HISTORICAL SCHEMATIC ONLY. THE REACTOR VESSEL HEAD WAS REPLACED IN DC 03RV05. SEE CURRENT PLANT REACTOR VESSEL DRAWINGS FOR DETAILS.

DWN TAM	DATE 6-23-99	SIGNIFICANT NO.
CHECKED		GROUP 1 2 3 4 5 CL 6
PROJECT NO. ETNSUR		REACTOR VESSEL SCHEMATIC
APP'D & CERT.		
CAD FILE: U04201.DGN		
NORTHERN STATES POWER COMPANY PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	SCALE NONE	REV
	FIGURE 4.2-1	REV. 29

01047869

FIGURE 4.2-1



ITEM	DESIGNATION
1	Lower WR level tap CL
2	Bottom of hand hole (ID)
3	Hand hole CL
4	TSP 1 CL
5	TSP 2 CL
6	TSP 3 CL
7	TSP 4 CL
8	Bottom of hand hole (ID)
9	Hand hole CL
10	TSP 5 CL
11	TSP 6 CL
12	TSP 7 CL
13	Bottom of hand hole (ID)
14	Hand hole CL
15	TSP 8 CL
16	Foreign object catcher CL
17	Top of tube bundle (Apex)
18	Top of wrapper roof
19	Lower NR level tap CL
20	Recirculation nozzle CL
21	Bottom of J-nozzles (outlet)
22	Feedwater nozzle CL
23	Top of J-nozzles (Apex)
24	Lower deck plate (Bottom)
25	Upper deck plate (Top)
26	Top of cyclones (Foreign object catcher CL)
27	Upper NR and WR level tap CL
28	Bottom of secondary manway (ID)
29	Secondary manway CL
30	Bottom of dryer block
31	Bottom of steam outlet nozzle

FRAMATOME - STEAM GENERATOR

DWN V. SHACKLETON	DATE 04-21-05	NORTHERN STATES POWER COMPANY PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U04301A.DGN		*FIGURE 4.3-1A REV. 27*

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FIGURE 4.3-1B, DELETED

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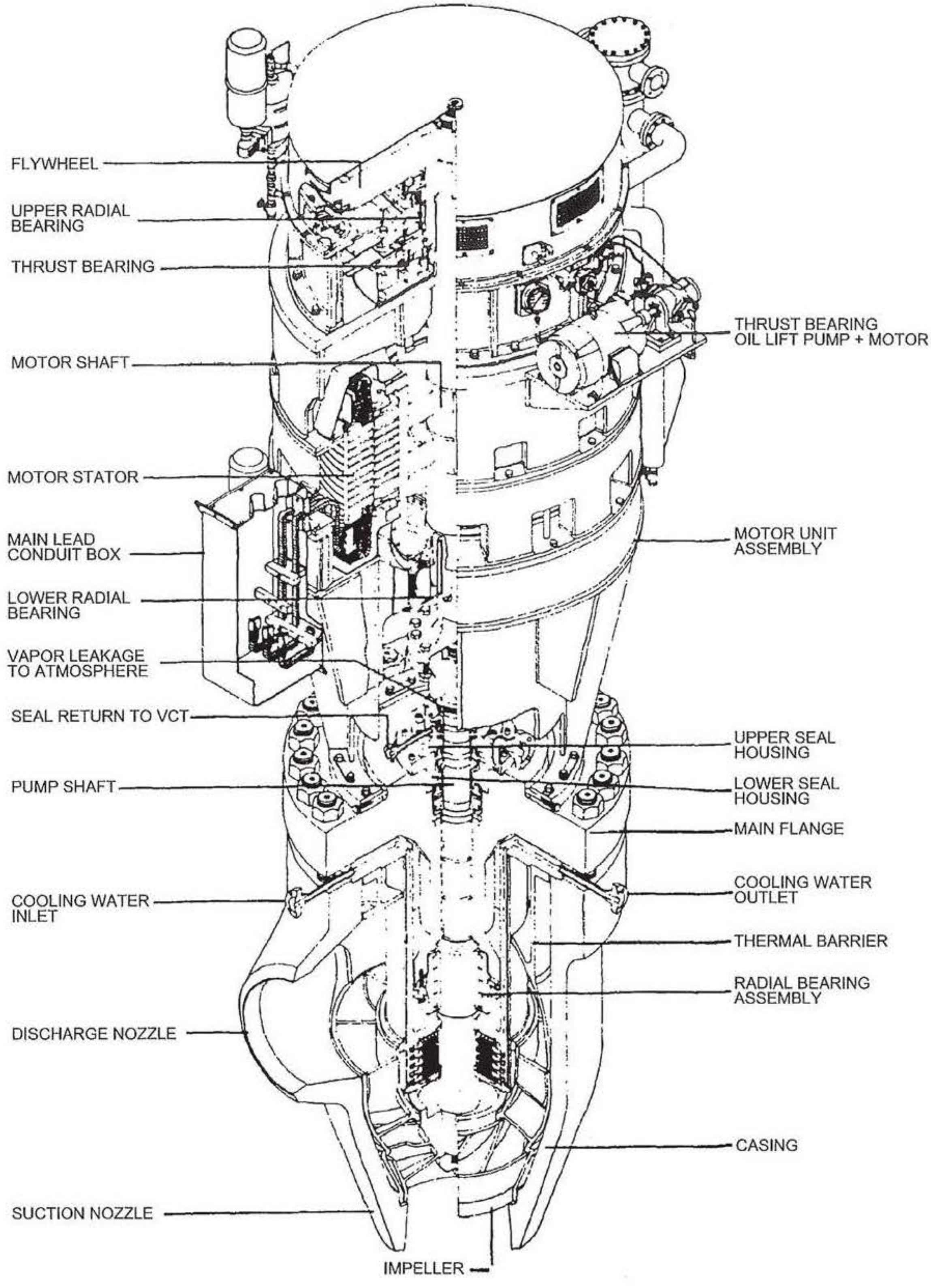
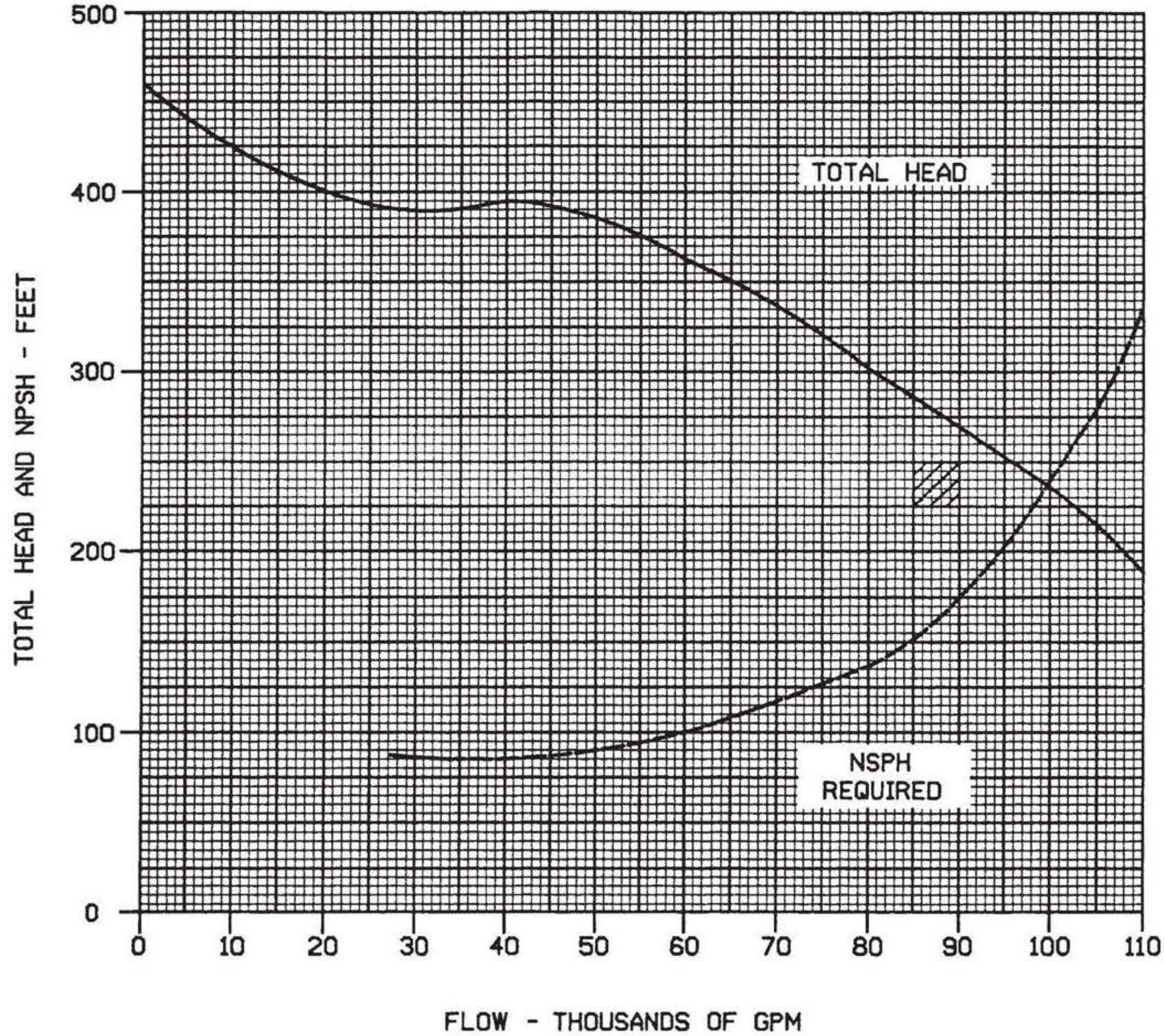
FIGURE



REACTOR COOLANT CONTROLLED LEAKAGE PUMP

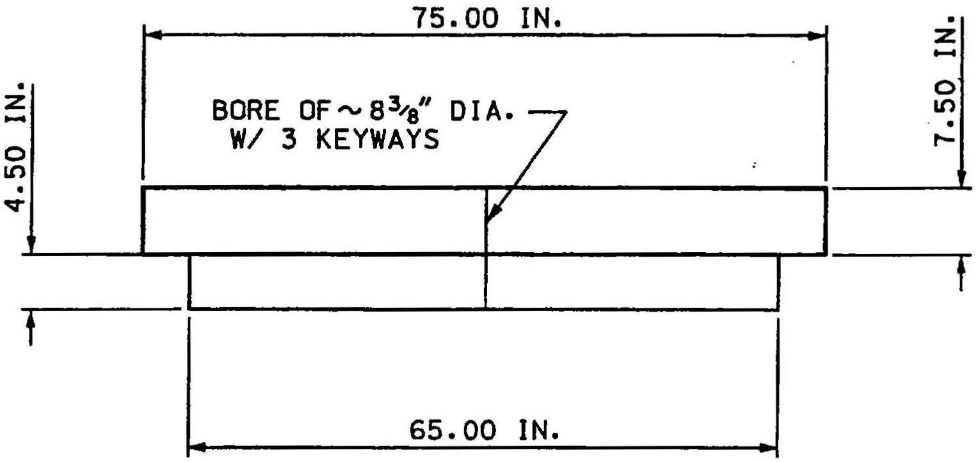
DWN: KJF	DATE: 2-5-14	NORTHERN STATES POWER COMPANY Xcel Energy	SCALE: NONE	
CHECKED:	CAD FILE: U04302.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	*FIGURE 4.3-2*	*REV. 33*

01396356



REACTOR COOLANT PUMP ESTIMATED
PERFORMANCE CHARACTERISTICS

DWN T. MILLER	DATE 6-23-99	NORTHERN STATES POWER COMPANY PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING MINNESOTA	SCALE: NONE	
CHECKED	CAD FILE U04303.DGN		**FIGURE 4.3-3**	**REV. 18**

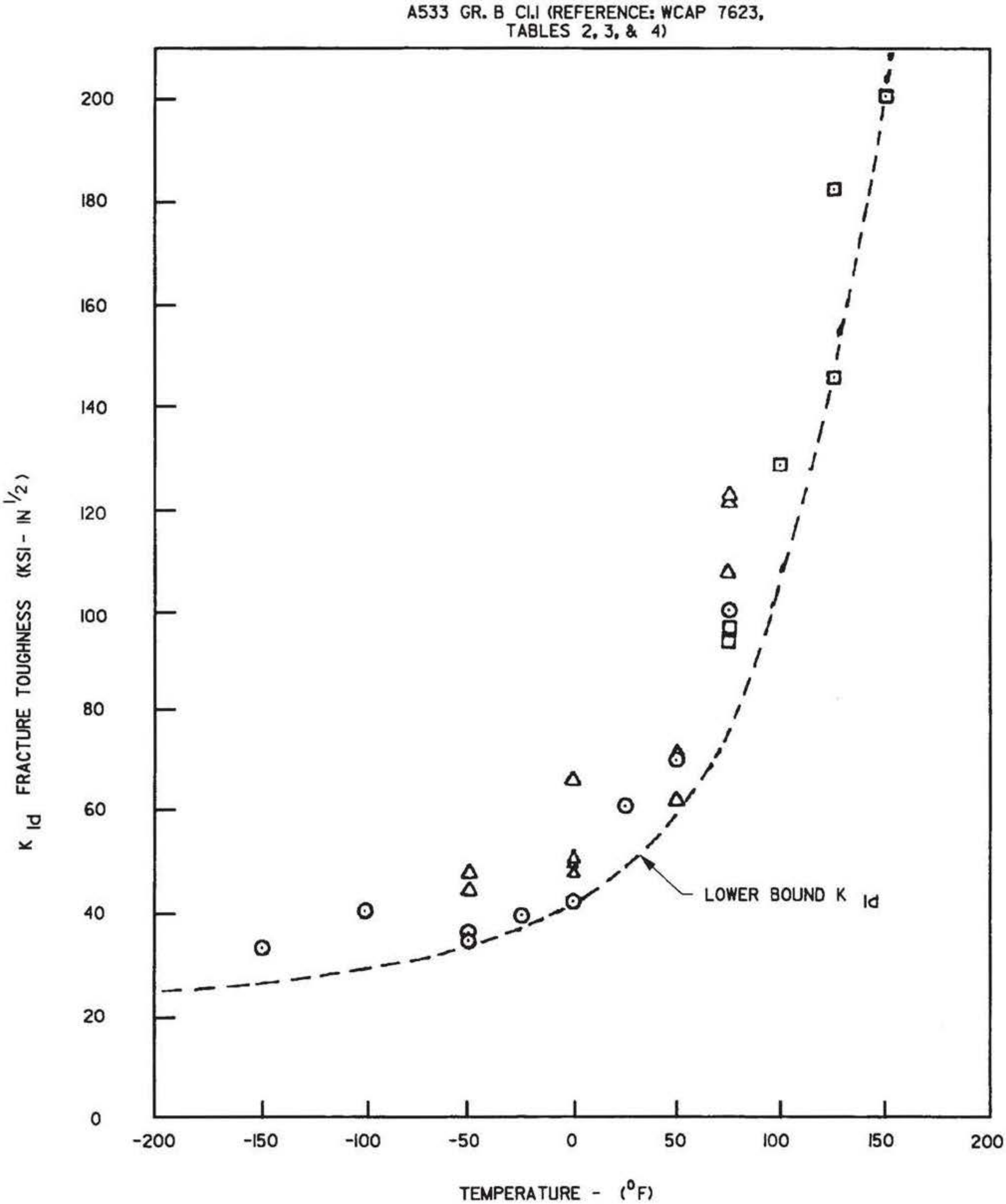


NOTE:

THE PLATES ARE BOLTED TOGETHER WITH THE BOLTS ALIGNED PERPENDICULAR TO THE PLANES OF THE PLATES

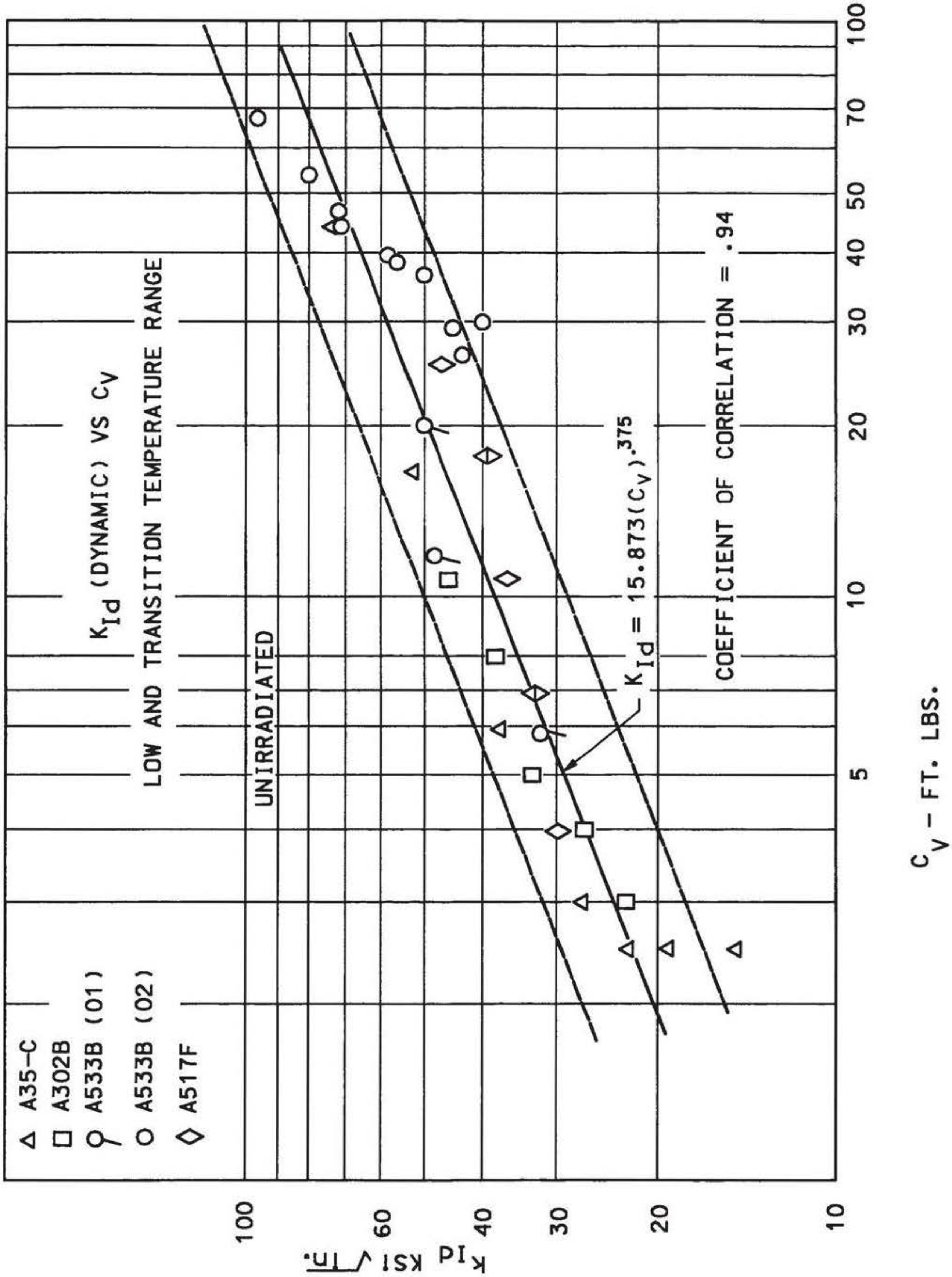
REACTOR COOLANT PUMP FLYWHEEL

DWN T. MILLER	DATE 6-23-99	NORTHERN STATES POWER COMPANY	SCALE: NONE	
CHECKED	CAD FILE U04304.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING MINNESOTA	FIGURE 4.3-4	REV. 18



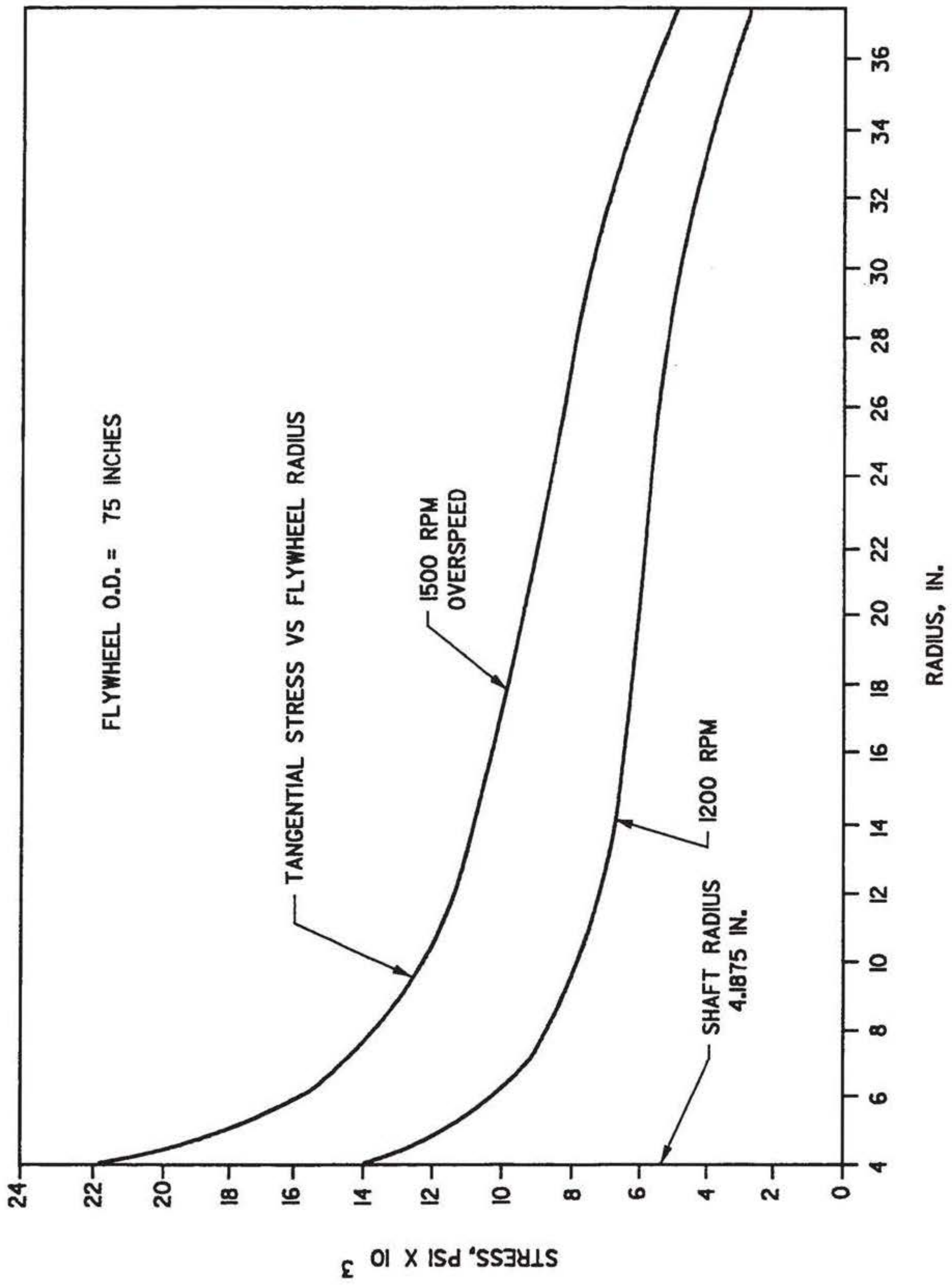
K_{Id} LOWER BOUND FRACTURE TOUGHNESS

DWN T. MILLER	DATE 6-23-99	NORTHERN STATES POWER COMPANY	SCALE: NONE	
CHECKED	CAD FILE U04305.DGN U04305.CIT	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING MINNESOTA	FIGURE 4.3-5	REV. 18



CORTEN AND SAILORS CORRELATION

DWN T. MILLER	DATE 6-23-99	NORTHERN STATES POWER COMPANY	SCALE: NONE
CHECKED	CAD FILE U04306.DGN USAR24.CIT	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING MINNESOTA	**FIGURE 4.3-6 REV. 18**



FLYWHEEL STRESS

DWN T. MILLER	DATE 6-23-99	NORTHERN STATES POWER COMPANY PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U04307.DGN U04307.CIT		**FIGURE 4.3-7 REV. 18**

FIGURE 4.3-8, DELETED

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FIGURE 4.3-9, DELETED

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FIGURE

FIGURE 4.3-10, DELETED

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FIGURE

FIGURE 4.3-11, DELETED

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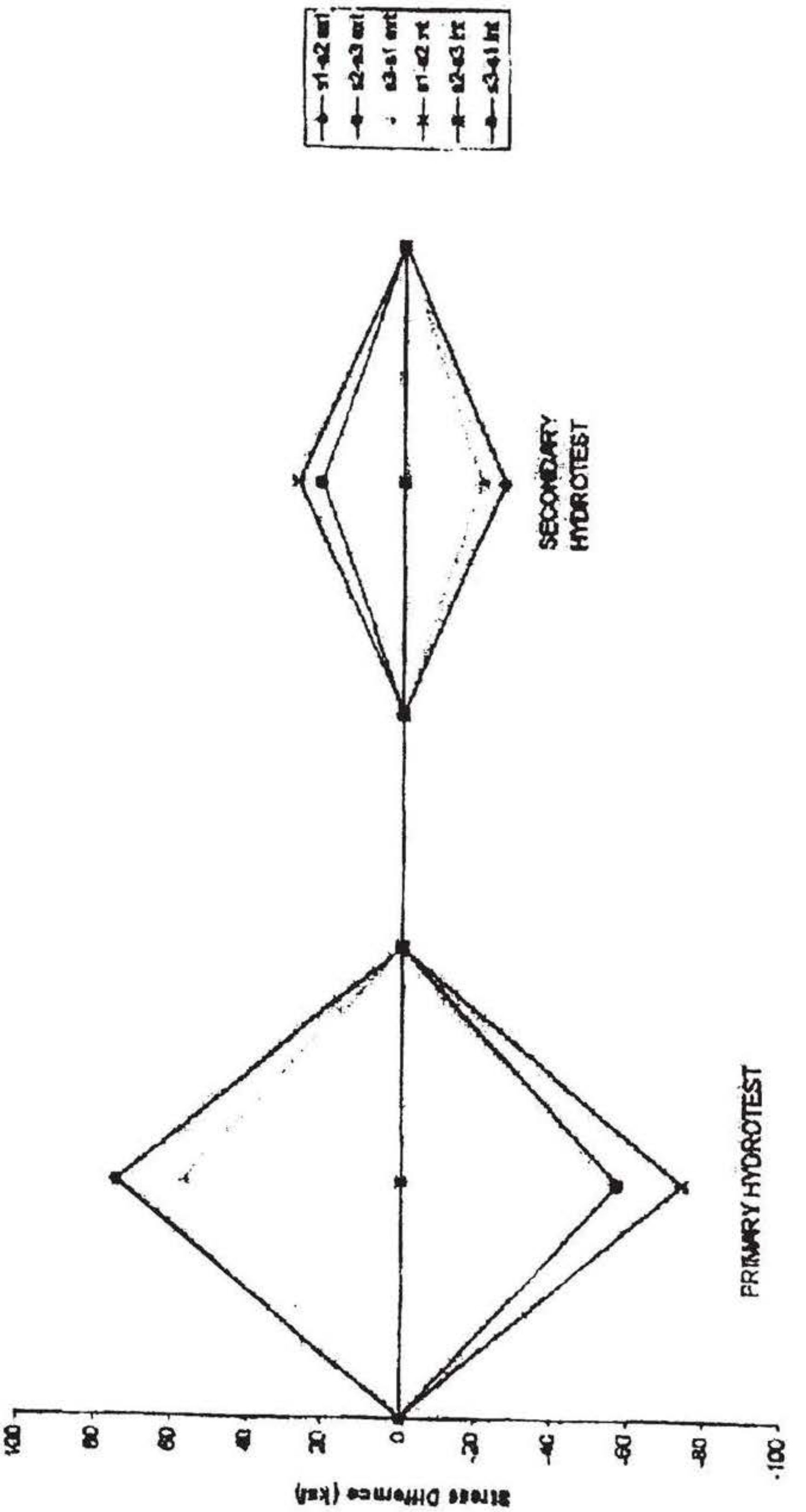
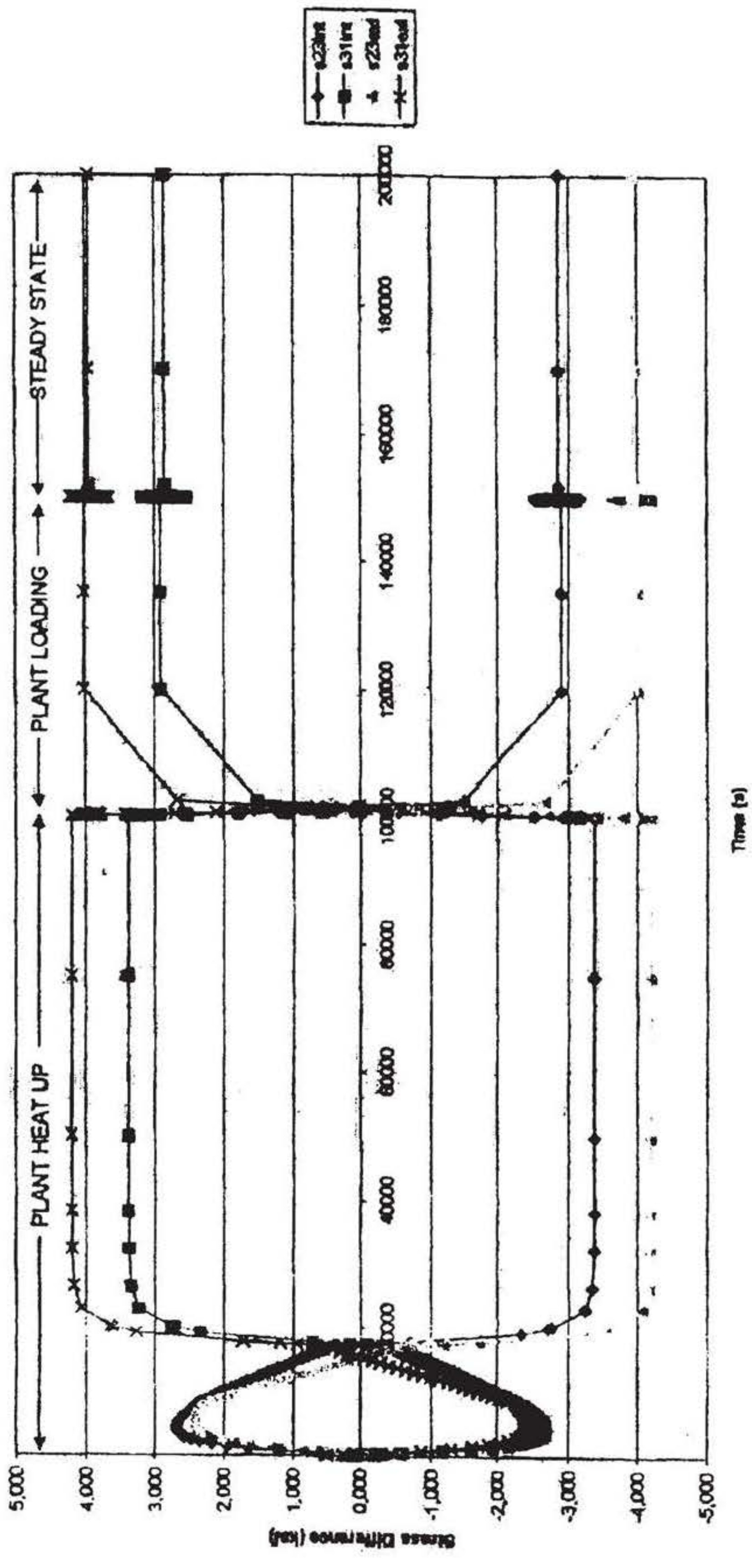
FIGURE



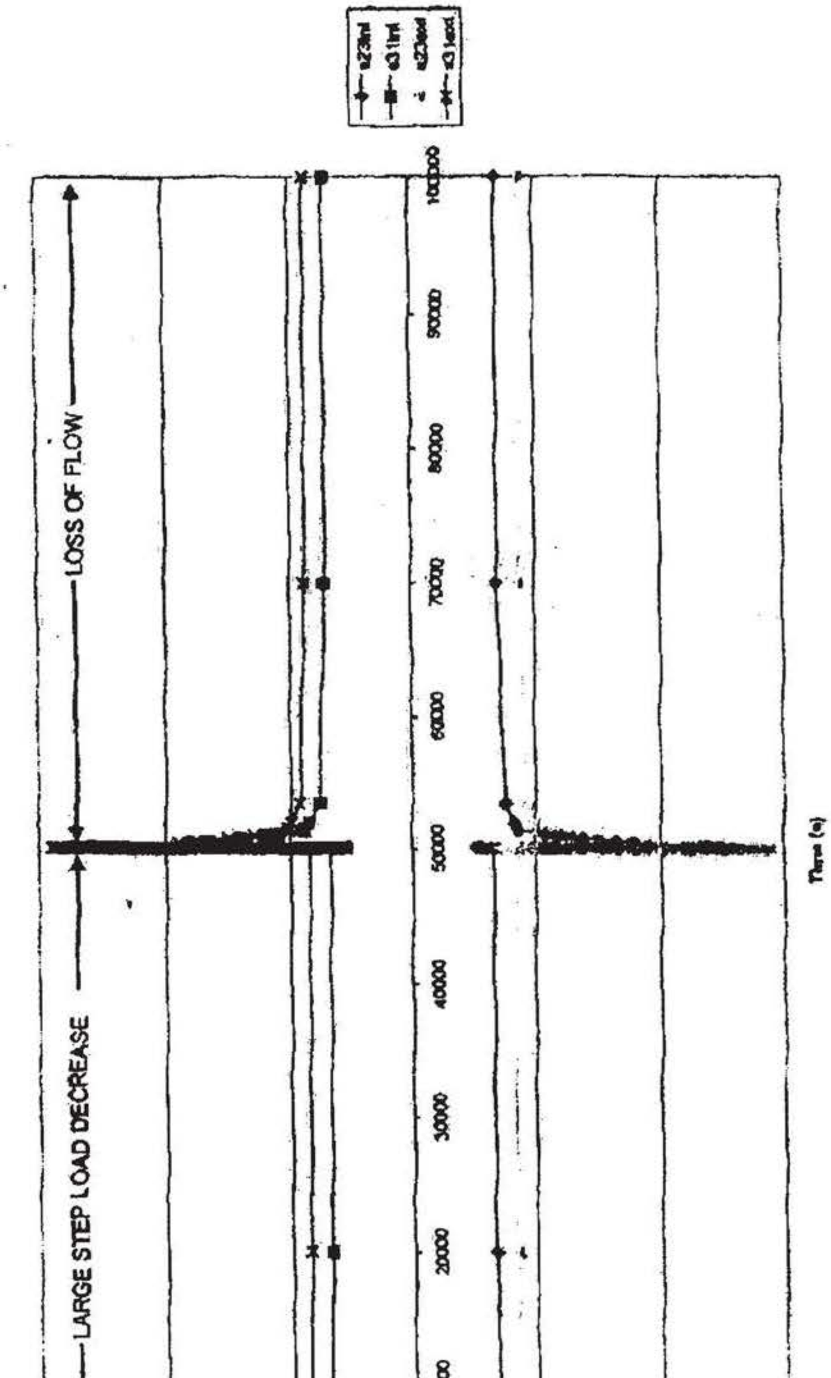
UNIT 1 - PRIMARY AND SECONDARY HYDROSTATIC TEST STRESS HISTORY AT PERIPHERAL TUBE

DWN V. SHACKLETON	DATE 04-22-05	NORTHERN STATES POWER COMPANY PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U04312.DGN U04312.CIT		**FIGURE 4.3-12 REV. 27**



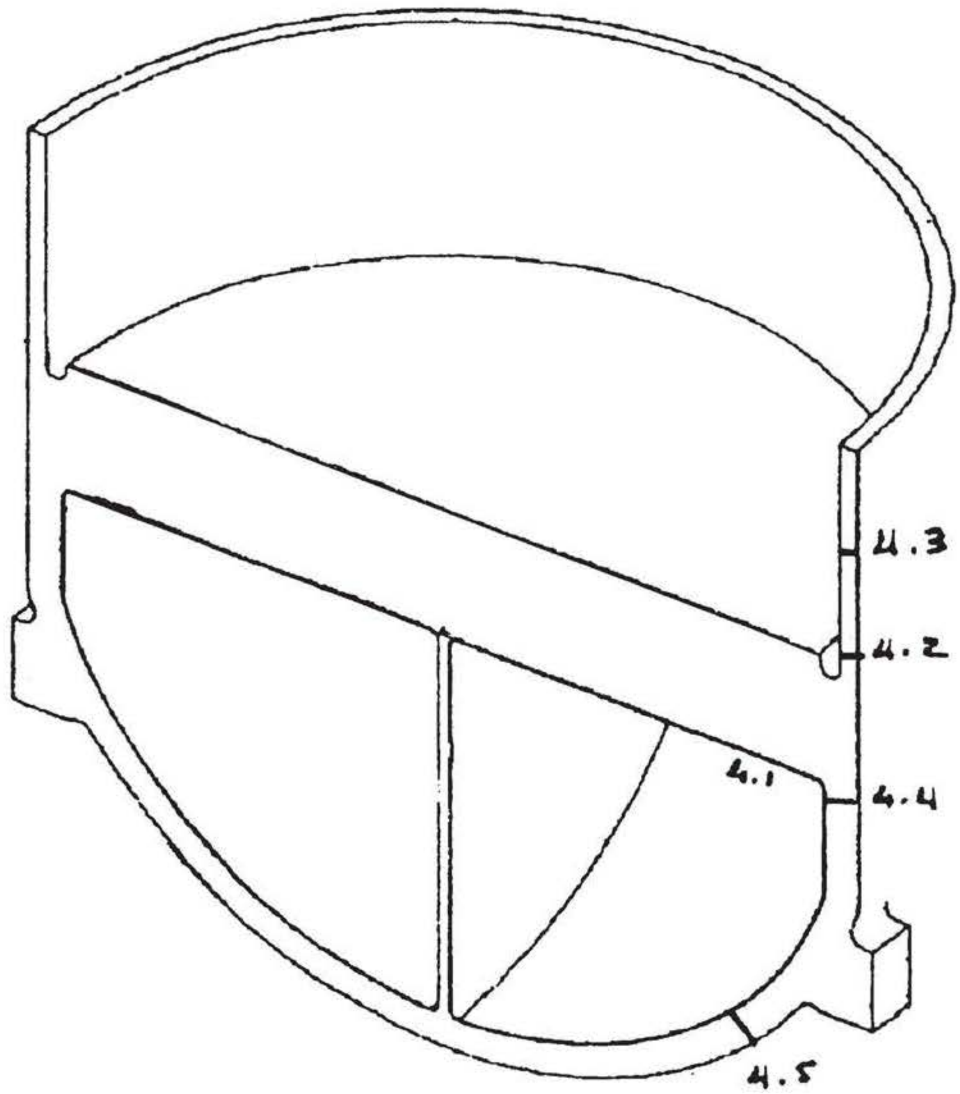
UNIT 1 - PLANT HEAT UP AND LOADING OPERATIONAL TRANSIENTS
(WITH STEADY-STATE PLATEAU) STRESS HISTORY
FOR THE HOT SIDE PERIPHERAL TUBE

DWN V. SHACKLETON	DATE 04-22-05	NORTHERN STATES POWER COMPANY PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U04313.DGN U04313.CIT		**FIGURE 4.3-13 REV. 27**



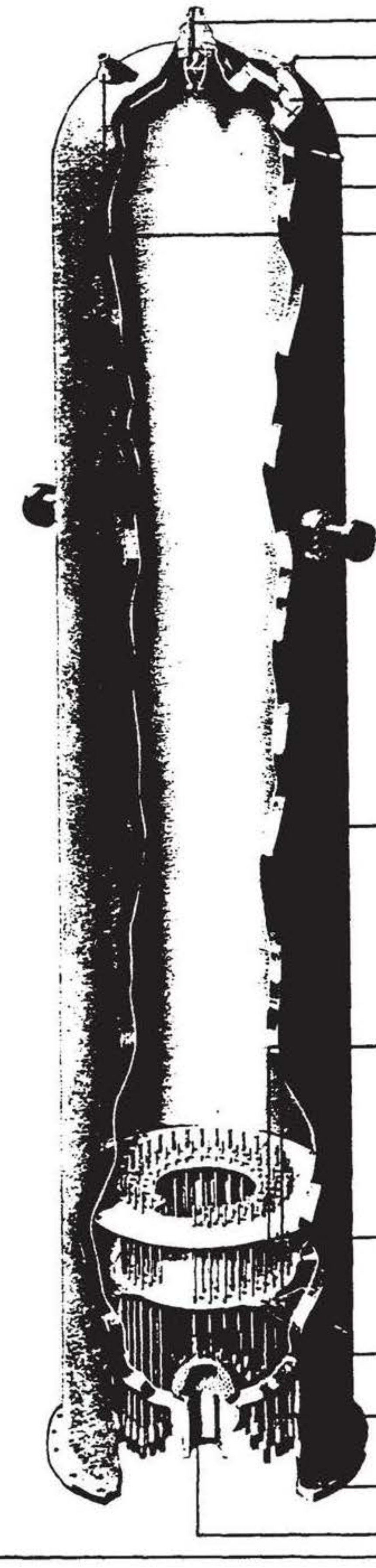
LARGE STEP LOAD DECREASE AND LOSS
OF FLOW STRESS HISTORY FOR THE HOT SIDE PERIPHERAL TUBE

DWN V SHACKLETON	DATE 04-22-05	NORTHERN STATES POWER COMPANY PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U04314.DGN U04314.CIT		**FIGURE 4.3-14 REV. 27**



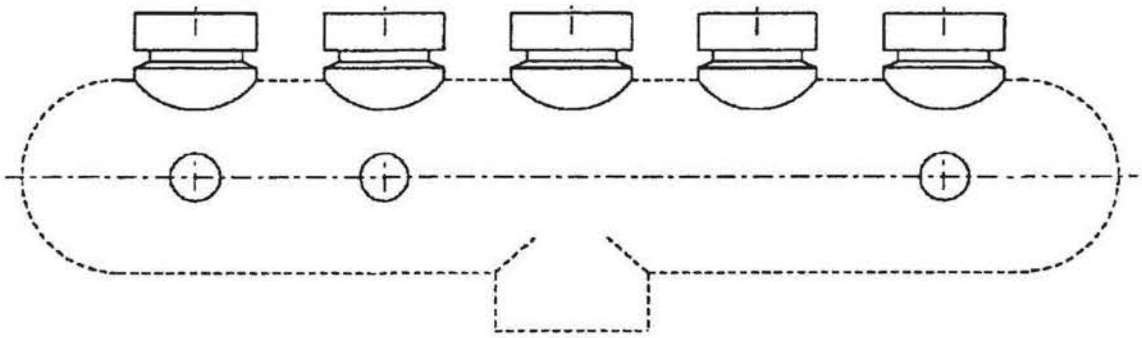
UNIT 1 PRIMARY-SECONDARY BOUNDARY COMPONENTS SHELL LOCATIONS OF STRESS INVESTIGATIONS

DWN V SHACKLETON	DATE 04-22-05	NORTHERN STATES POWER COMPANY PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U04315.DGN U04315.CIT		**FIGURE 4.3-15 REV. 27**

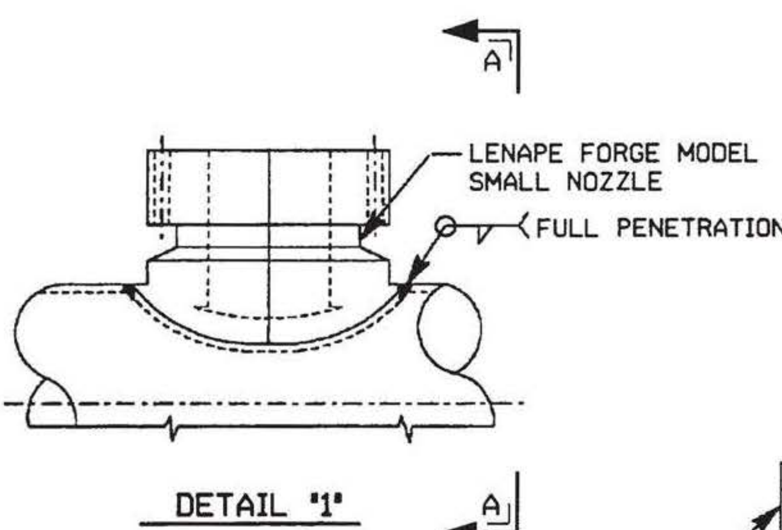


PRESSURIZER

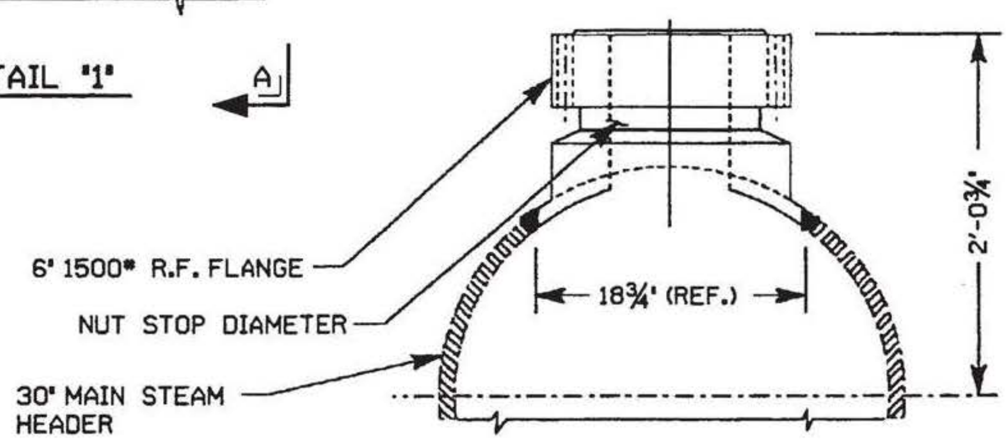
DWN T. MILLER	DATE 6-23-99	NORTHERN STATES POWER COMPANY PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING MINNESOTA	SCALE: NONE		
CHECKED	CAD FILE U04401.DGN		FIGURE 4.4-1	REV. 18	



ELEVATION VIEW



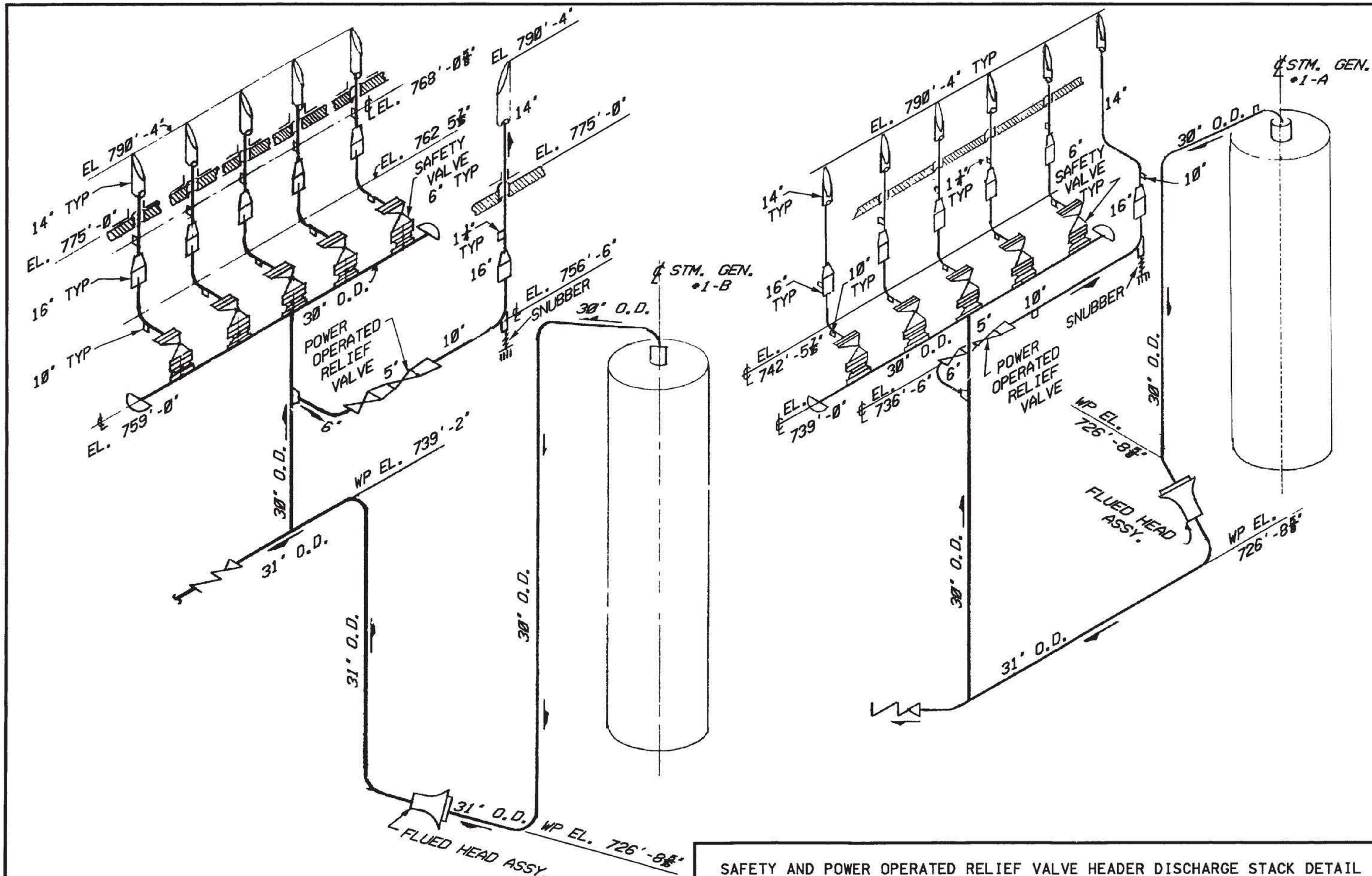
DETAIL "1"



SECTION A-A

SAFETY RELIEF VALVE HEADER DETAIL

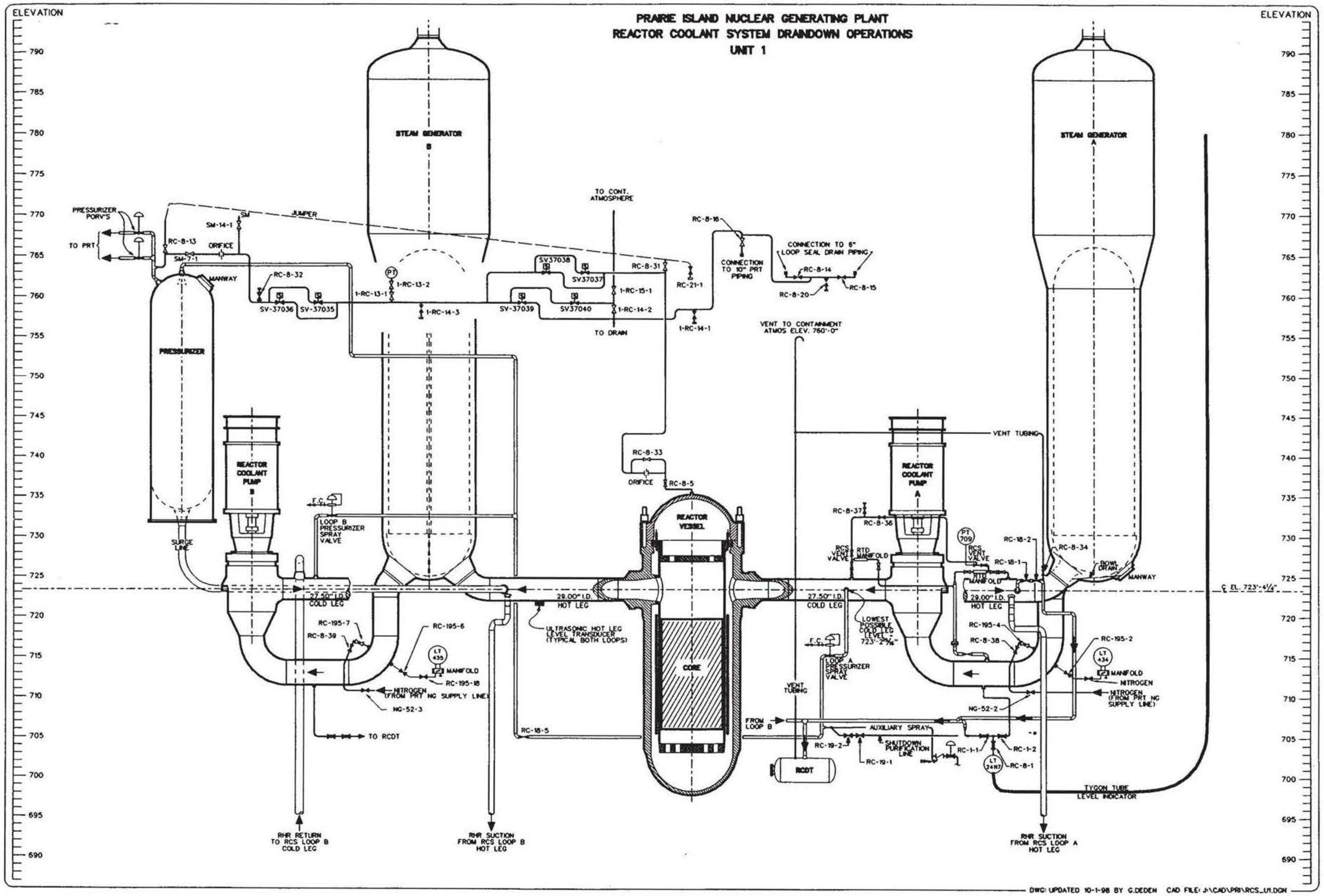
DWN T. MILLER	DATE 6-23-99	NORTHERN STATES POWER COMPANY PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING MINNESOTA	SCALE: NONE	
CHECKED	CAD FILE U04402.DGN		FIGURE 4.4-2	REV. 18



SAFETY AND POWER OPERATED RELIEF VALVE HEADER DISCHARGE STACK DETAIL

DWN T. MILLER	DATE 6-23-99	NORTHERN STATES POWER COMPANY PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING MINNESOTA	SCALE: NONE	
CHECKED	CAD FILE U04403.DGN U04403.CIT		FIGURE 4.4-3	REV. 18

REVISIONS



DWN TAM DATE 6-23-99
CHECKED
PROJECT NO. ETNSUR
APP'D & CERT.
CAD FILE: U04504.DGN

SIGNIFICANT NO.
GROUP 1 2 3 4 5 CL 6

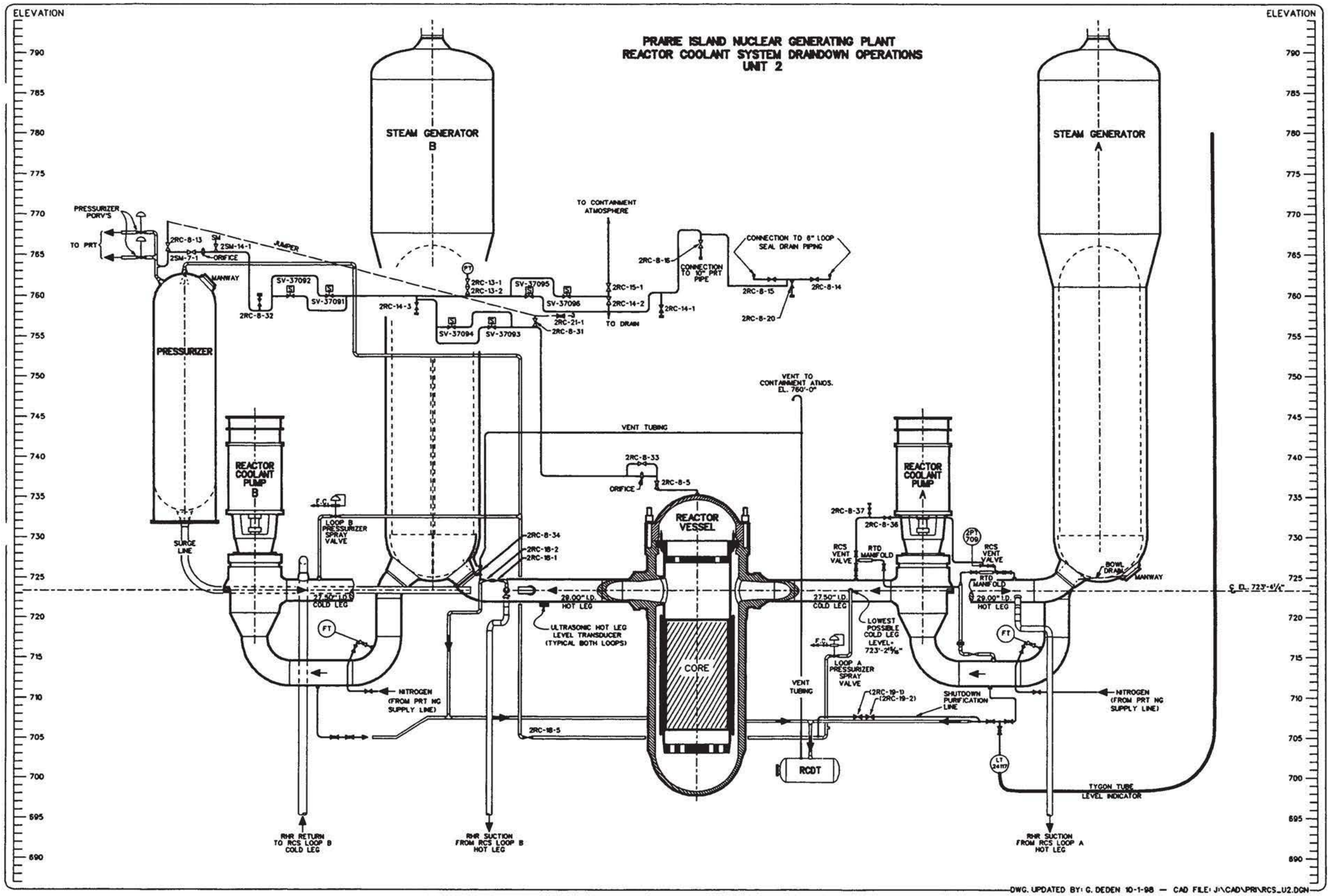
REACTOR COOLANT SYSTEM
UNIT 1
DRAINDOWN OPERATIONS AND VENT JUMPER

NORTHERN STATES POWER COMPANY
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RED WING, MINNESOTA

SCALE NONE REV

FIGURE 4.5-4 REV. 24

REVISIONS



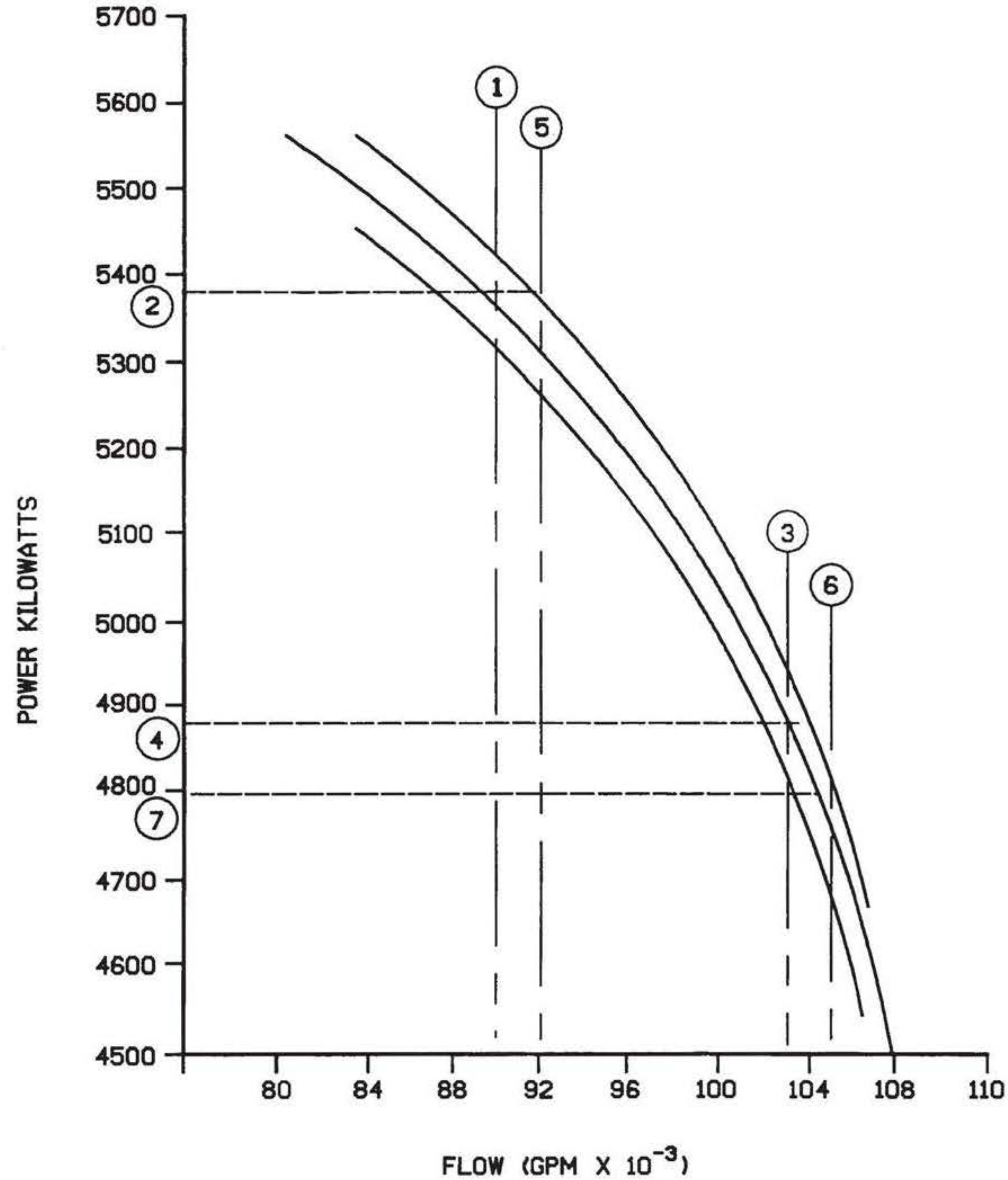
DWN TAM	DATE 6-23-99	SIGNIFICANT NO.							
CHECKED		GROUP	1	2	3	4	5	CL	6
PROJECT NO. ETNSUR									
APP'D & CERT.									
CAD FILE: U04505.DGN									

REACTOR COOLANT SYSTEM
UNIT 2
DRAINDOWN OPERATIONS AND VENT JUMPER

NORTHERN STATES POWER COMPANY
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RED WING, MINNESOTA

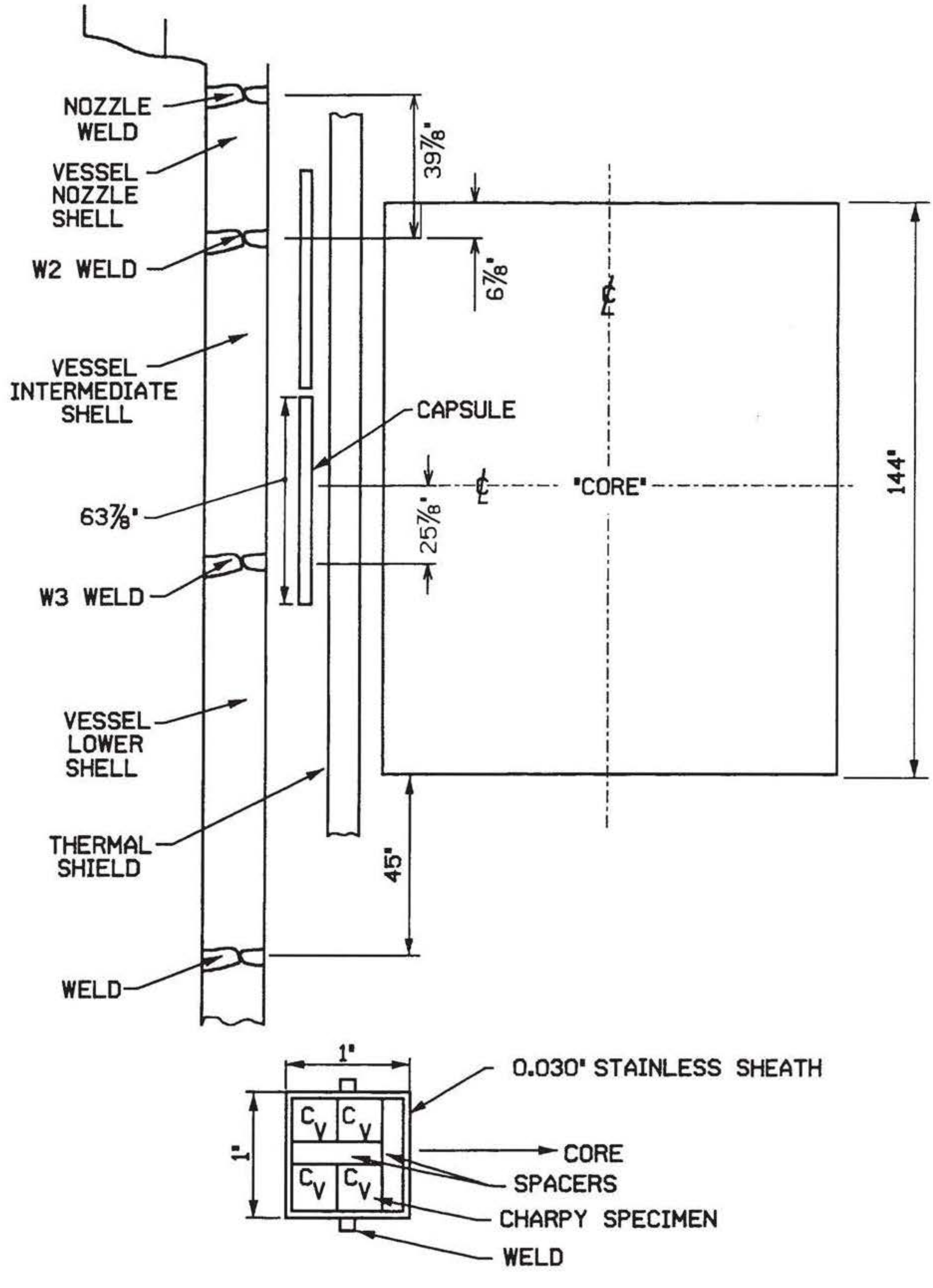
SCALE NONE | REV

FIGURE 4.5-5 REV. 23



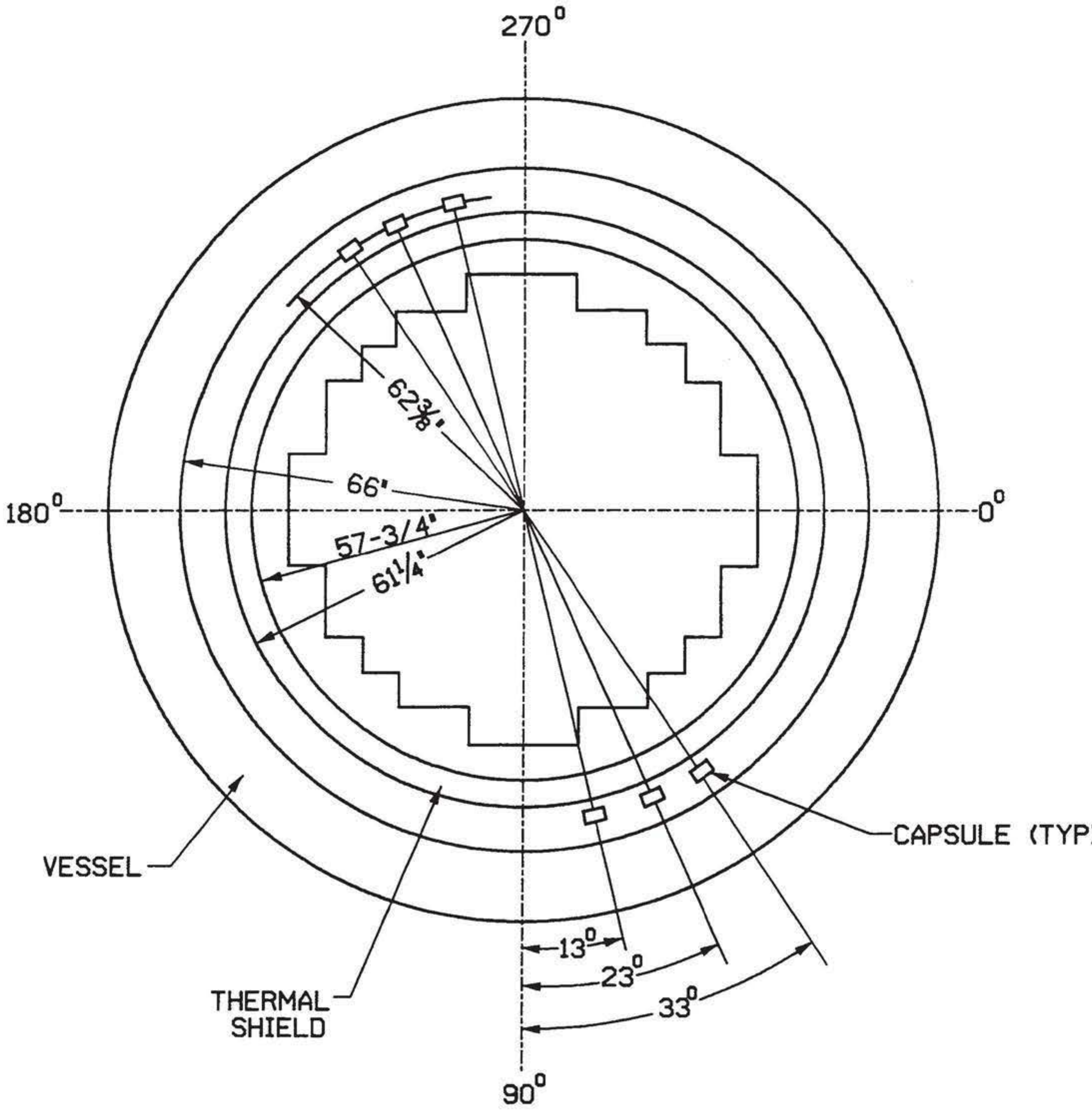
REACTOR COOLANT PUMP INPUT POWER VERSUS FLOW

DWN T. MILLER	DATE 6-23-99	NORTHERN STATES POWER COMPANY PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U04601.DGN		FIGURE 4.6-1 REV. 18



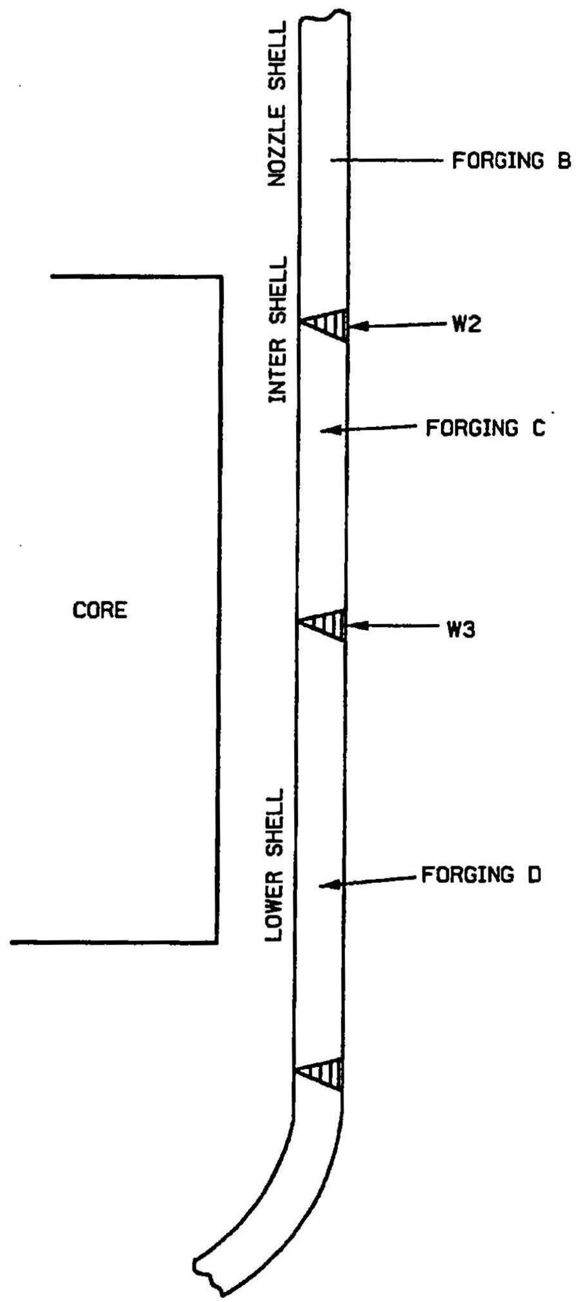
REACTOR VESSEL SURVEILLANCE CAPSULE (ELEVATION VIEW)

DWN T. MILLER	DATE 6-23-99	NORTHERN STATES POWER COMPANY PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING MINNESOTA	SCALE: NONE	
CHECKED	CAD FILE U04701.DGN		FIGURE 4.7-1	REV. 18



SURVEILLANCE CAPSULE PLAN VIEW

DWN T. MILLER	DATE 6-23-99	NORTHERN STATES POWER COMPANY PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U04702.DGN		FIGURE 4.7-2 REV. 18



LOCATION OF REACTOR VESSEL BELTLINE
REGION WELD AND FORGING MATERIAL

DWN T. MILLER	DATE 6-23-99	NORTHERN STATES POWER COMPANY	SCALE: NONE
CHECKED	CAD FILE U04704.DGN	PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING MINNESOTA	**FIGURE 4.7-4 REV. 18**