## SECTION 6

## ENGINEERED SAFETY FEATURES

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# SECTION 6 ENGINEERED SAFETY FEATURES

## 6.1 SUMMARY DESCRIPTION

### 6.1.1 Introduction

Independent and separate Engineered Safety Features are provided for each Unit. The description which is contained herein is applicable to either Unit.

The central safety objective in reactor design and operation is control of reactor fission products. The methods used to assure this objective are:

a. Core design to preclude release of fission products from the fuel (Section 3).

b. Retention of fission products by the reactor coolant system boundary for whatever leakage occurs (Sections 4 and 6).

c. Retention of fission products by the containment for operational and accidental releases beyond the reactor coolant boundary (Sections 5 and 6).

d. Limit fission product release to minimize population exposure. (Sections 2, 7 and 10).

The Engineered Safety Features are the provisions in the plant which implement methods c and d (above) to prevent the occurrence or to minimize the effects of serious accidents.

The Engineered Safety Features in this plant are the Containment Systems, detailed in Section 5; the core Safety Injection System, detailed in Section 6.2; the Containment Air Cooling System, detailed in Section 6.3; the Containment Vessel Internal Spray System, detailed in Section 6.4, the Auxiliary Feedwater System is detailed in Section 11.9, Special Zone Ventilation System described in Section 10.3 and Section 5, the Diesel Generators described in Section 8, and the Station Batteries described in Section 8.

Evaluation of techniques and equipment used to accomplish the central objective including accident cases are detailed in Sections 5, 6 and 14.

10 CFR Part 50.72(b)(2)(ii) requires that any event that results in manual or automatic actuation of any Engineered Safety Feature, including the Reactor Protection System, is reportable, except for an actuation that resulted from or was part of a pre-planned sequence during testing or reactor operation. Table 6.1-1 contains the list of Prairie Island systems which are considered Engineered Safety Features for the purposes of reporting under the requirements of 10 CFR Part 50.72(b)(2)(ii). Further guidance on the reportability of engineered safety feature actuations is provided in the plant’s administrative control procedures.

### 6.1.2 Engineered Safety Features Criteria

#### 6.1.2.1 Engineered Safety Features Basis for Design

Criterion: Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends. (GDC 37)

The design, fabrication, testing and inspection of the core, the reactor coolant system pressure boundary and their protection systems give assurance of safe and reliable operation under all anticipated normal, transient, and accident conditions. However, engineered safety features are provided in the facility to back up the reliability provided by these components. These engineered safety features have been designed to cope with any size reactor coolant pipe break up to and including the circumferential rupture of any pipe assuming unobstructed discharge from both ends, and to cope with any steam or feedwater line break up to and including the main steam or feedwater headers.

The release of fission products from the reactor fuel is limited by the Safety Injection System which, by cooling the core, keeps the fuel in place and substantially intact and limits the metal water reaction to an insignificant amount.

The Safety Injection System consists of high and low head centrifugal pumps driven by electric motors, and passive accumulator tanks which are self energized and which act independently of any actuation signal or power source.

The release of fission products from the containment is limited in the following ways:

a. Blocking the potential leakage paths from the containment. This is accomplished by:

    1. A steel shell containment.

    2. Isolation of process lines by the double barriers in each line that penetrates the containment.

    3. A shield building surrounding the containment vessel with an associated ventilation system containing charcoal filters.

b. Reducing the fission product concentration in the containment atmosphere by spraying borated water buffered by a 9 wt. % to 11 wt. % NaOH solution.

c. Reducing the containment pressure and thereby limiting the driving potential for fission product leakage by cooling the containment atmosphere using the following systems:

   1. Containment Vessel Internal Spray System using a portion of the stored refueling water

   2. Containment Air Cooling System

d. A special zone ventilation system, collecting leakage from the auxiliary building, and discharging it through the charcoal filters.

### 6.1.2.2 Reliability and Testability of Engineered Safety Features

Criterion: All engineered safety features shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for a proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the systems, including engineered safety features, will be influenced by the known and the demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the plant. (GDC 38).

A comprehensive program of plant testing is formulated for all equipment systems and system control vital to the functioning of engineered safety features. The program consists of performance tests of individual pieces of equipment in the manufacturer's shop, integrated tests of the system as a whole, and periodic tests of the actuation circuitry and mechanical components to assure reliable performance, upon demand, throughout the plant lifetime.

The initial tests of individual components and the integrated tests of the system as a whole complement each other to assure performance of the system as designed and to prove proper operation of the actuation circuitry.

The engineered safety features components are designed to provide for routine periodic testing.

In general, Engineered Safety Features master relays (actuated by the logic) are tested at power while the slave relays are blocked. Actuating coils of the slave relays are checked for continuity. However, not all ESF circuits are designed and tested in this manner.

During plant shutdown, the master relays are actuated, actuating the slaves, which are allowed to start the safety injection sequence. The test is terminated when associated valves are properly aligned and associated pumps are started.

### 6.1.2.3 Missile Protection

Criterion: Protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures. (GDC 40)

A loss-of-coolant accident or other plant equipment failure might result in dynamic effects or missiles. For engineered safety features which are required to ensure safety in the event of such an accident or equipment failure, protection is provided primarily by the provisions which are taken in the design to prevent the generation of missiles. In addition, protection is also provided by the layout of the plant equipment or by missile barriers in certain cases. Refer to Section 12.2.6 for a discussion of missile protection.

Injection paths leading to unbroken reactor coolant loops are protected against damage as a result of the maximum reactor coolant pipe rupture by layout and structural design considerations. Injection lines penetrate the main missile barrier, which is the loop compartments wall, and the injection headers are located in the missile-protected area between the loop compartments wall and the containment wall. Individual injection lines, connected to the injection header, pass through the barrier and then connect to the loops. Separation of the individual injection lines is provided to the maximum extent practicable and restrained where necessary to prevent interaction with other systems. Movement of the injection line, associated with rupture of a reactor coolant loop, is accommodated by line flexibility and by the design of the pipe supports such that no damage outside the loop compartments is possible.

The containment structure is capable of withstanding the effects of missiles originating outside the containment and which might be directed toward it so that no loss-of-coolant accident can result.

All hangers, stops, restraints and anchors are designed in accordance with the applicable codes and design criteria set forth in Section 12 which provide minimum requirements on material, design and fabrication with ample safety margin for both dead and dynamic loads over the life of the plant.

### 6.1.2.4 Engineered Safety Features Performance Capability

Criterion: Engineered safety features such as emergency core cooling and containment heat removal systems shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single active component. (GDC 41)

All engineered safety features provide sufficient performance capability to accommodate any single failure of an active component and still function in a manner to avoid undue risk to the health and safety of the public.

The containment and associated engineered safety features have been designed on the basis of the fission product sources specified in Safety Guide 4. New radiological dose consequence analyses were performed consistent with Regulatory Guide 1.183. The extreme upper limits of public exposures are taken as the levels and in two periods outlined in 10CFR50.67. The accident condition considered is the hypothetical case of a release of fission products per Appendix D. Also, the total loss of all outside power is assumed concurrently with this accident. However, operation of the Safety Injection System, considering the single failure criterion limits the release of fission products from the core to only the gap activity between the fuel pellet and clad.

Under the above accident condition, the Containment Air Cooling System and the Containment Vessel Internal Spray System are designed to supply the necessary post-accident cooling capacity to rapidly reduce the containment pressure following blowdown and cooling of the core by safety injection. The Spray System provides for iodine removal by washing action of the spray.

#### 6.1.2.5 Engineered Safety Features Components Capability

Criterion: Engineered safety features shall be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss-of-coolant accident. (GDC 42)

All active components of the Safety Injection System (with the exception of injection line isolation valves) and the Containment Spray System are located outside the containment and not subject to containment accident conditions. The accumulators are located in a missile shielded area.

Instrumentation, motors, cables and penetrations located inside the containment are selected to meet the most adverse accident conditions to which they may be subjected. These items are either protected from containment accident conditions or are designed to withstand, without failure, exposure to the worst combination of temperature, pressure, and humidity expected during the required operational period.

The Safety Injection System pipes serving each loop are restrained at the missile barrier in each loop area to restrict potential accident damage to the portion of piping beyond this point. The restraints are designed to withstand, without failure, the thrust force of any branch line severed from the reactor coolant pipe and discharging fluid to the atmosphere, and to withstand a bending moment equal to the ultimate strength of the pipe or equivalent to that which produces failure of the piping under the action of free end discharge to atmosphere or motion of the broken reactor coolant pipe to which the emergency core cooling pipes are connected. This prevents possible failure at any point upstream from the missile barrier including the branch line connection into the piping header.

### 6.1.2.6 Accident Aggravation Prevention

Criterion: Engineered safety features shall be designed so that any action of the engineered safety features which might accentuate the adverse after-effects of the loss of normal cooling is avoided. (GDC 43)

The reactor is maintained subcritical following a primary system pipe rupture accident. Introduction of borated cooling water into the core results in a net negative reactivity addition. The control rods insert and remain inserted.

The delivery of cold safety injection water to the reactor vessel following accidental expulsion of reactor coolant does not cause further loss of integrity of the Reactor Coolant System boundary.

### 6.1.2.7 Sharing of Systems

Criterion: Reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing. (GDC 4)

The residual heat removal pumps and heat exchangers serve dual functions. The residual heat removal exchangers and residual heat removal pumps are relied upon during periods of reactor shutdown, however during other plant operating modes, this equipment is aligned to perform the low head safety injection function.

During the injection phase, the safety injection pumps do not depend on any portion of other systems with the exception of the component cooling and cooling water systems. During the recirculation phase, if reactor coolant system pressure stays high due to a small pipe break accident, suction to the safety injection pumps is provided by the residual heat removal pumps.

The Containment Air Cooling System also serves the dual function of containment cooling during normal operation and containment cooling after an accident. Since the method of operation for both cooling functions is essentially the same, the dual aspect of this system does not affect its function as an engineered safety feature.

### 6.1.2.8 Engineered Safety Features Protection from Internal Flooding

Internal flooding which could be postulated to adversely affect the performance of Engineered Safety Features was a part of the original plant design criteria. Provisions have been made for sumps and sump pumps in the Screenhouse, Turbine building, Auxiliary building and the Reactor building, and sumps with level annunciation in the D5/D6 Diesel Generator Building. These provisions are intended to protect vital equipment from equipment leakage which occurs during normal plant operation. They were not originally designed as protection against large internal flooding conditions resulting from major failures of systems having relatively large capacities of water (rupture of flexible connections in the circulating water lines, cooling water header rupture or main feedwater break).

These buildings do however have the capacity to accommodate large internal floods since it takes time to increase water levels to an elevation where nuclear safety related equipment is located. It has been shown by various studies that the operating staff has enough time to isolate the cause of the flooding before safety related equipment function would be lost. (Reference 12 and 30).

In support of original plant licensing, PINGP was required to review the effects of flooding for two types of pipe failure events. These event types are: 1) breaks and leakage cracks in high energy piping systems, and 2) leakage cracks in non-high energy, non-Class I systems that are capable of providing high flooding rates or which have an unlimited water supply.

The results of flooding reviews for HELB events is located in USAR Appendix I.

#### 6.1.2.8.1 Flooding Review Basis for non-High Energy, non-Class I Piping Failures

**Atomic Energy Commission Requirement**

By letter dated August 3, 1972 (Skovholt letter) (Reference 55), the Atomic Energy Commission (AEC) requested that Northern States Power (NSP), the plant licensee at the time, review the design of its facilities in light of industry operating experience (OE). The industry OE was failure of an expansion bellows at Quad Cities Unit 1 that resulted in an internal flooding event. NSP was specifically asked to (1) review whether failure of any equipment that does not meet the criteria of Class I seismic construction, particularly the circulating water system, could cause flooding sufficient to adversely affect the performance of an engineered safety system, and (2) whether failure of any equipment could cause flooding such that common mode failure of redundant safety related equipment would result.

Additionally, in a letter dated September 26, 1972 (DeYoung letter) (Reference 56), the aforementioned review was reiterated. The following is an excerpt from the letter.

> "You are requested to review PINGP, Unit 1 and 2 to determine whether the failure of any non-Category I (seismic) equipment, particularly in the circulating water and fire protection system, could result in a condition, such as flooding or the release of chemicals, that might potentially adversely affect the performance of safety-related equipment required for safe shutdown of the facilities or to limit the consequences of an accident."
>
> "The integrity of barriers to protect critical equipment from potential damaging conditions should be assumed only when the barrier has been specifically designed for such conditions."

This review was requested within 30 days from the date of issuance.

**PINGP Response**

In a letter dated October 23, 1972 (Reference 57), NSP responded to the AEC request. The following is an excerpt from the letter.

> "We have reviewed the PI design for Units 1 and 2 and conclude that where the potential of flooding engineered safety features exist, the operator is provided sufficient information and means to take corrective action in a timely manner."
>
> "Each turbine building condenser/condensate pump pit is equipped with a sump and two sump pumps, one of which starts on a high water level (674′ m.s.l.) signal in the sump. At 678′6″ m.s.l. a high-high water level signal starts a second sump and finally a high-high-high water level (678′9″ m.s.l) annunciates in the control room. Since all safety related electrical/mechanical components are on or above floor elevation 695′, when the high-high-high water level annunciates there will still be ample time for visual operator inspection of the situation and initiation of corrective action (such as manual shutdown of the circulating water pumps). There is no further danger of loss of safeguards by flooding after shutdown of the circulating water pumps because normal river elevation is 674.5′ m.s.l."

> "Furthermore, a study was made in the Class I areas of the Auxiliary Building for any non-Class I pipes whose failure might constitute a flooding problem. Only those systems that have access to large volumes of water and/or potentially large flow rates were considered. The results are that only various size fire protection and feedwater lines are potentially problematic. However, at most, the feedwater can only raise water to 6.4 inches. Failure of a fire protection line would raise the water level only 1 inch in 15.5 minutes (pumping at 2,000 gpm). Numerous sump level alarms would notify the operator of an abnormal condition and provide him with adequate time to terminate the incident prior to damaging any safeguards equipment."
>
> "Hence, it can be concluded that any potential failure of non-Class I equipment does not pose a threat to the overall plant safety, either by impeding safeguard performance or by causing common mode failure of redundant safeguard related equipment."

The Turbine Building condenser pits, in which no safety related equipment is installed, are each capable of containing greater than 750,000 gallons. Each provides a large volume for water collection prior to flood waters reaching safety related equipment located on elevation 695'. (Reference 58)

A plant modification (Reference 59) added level switches in the condenser pits at approximately 685' m.s.l. (approximately 1/3 of the condenser pit volume). Actuation of these switches, on a two-out-of-three logic, will automatically trip the main circulating water pumps and will alarm in the Control Room. The modification eliminated the need for operator action and provided an improvement on the original configuration.

FSAR Amendment 31, Section I.4-4, (Reference 60) discusses flooding in the Auxiliary Building, but does not specifically reference the Turbine Building. It states that the seismic Class 1 areas of the Auxiliary Building were surveyed and the main feedwater and fire protection systems were considered the only non-seismic flooding sources with access to large water volumes. The expected water flow rates and/or volumes would not endanger any equipment required for safe reactor shut down.

Minutes of a HELB meeting with the AEC on January 4, 1973, were transmitted by AEC letter dated January 11, 1973 (Reference 61). This was the first time crack size criterion appeared in regulatory correspondence. NRC Branch Technical Position MEB 3-1, attached to NRC Generic Letter 87-11, Relaxation in Arbitrary Intermediate Pipe Rupture Requirements (Ref. 62), includes in Section B.3.c(3) the same high energy pipe leakage crack size criterion as originally provided in Reference 61.

MEB 3-1 Section B.3.c, Subparagraph (3) states, "Fluid flow from a leakage crack should be based on a circular opening of area equal to that of a rectangle one-half pipe diameter in length and one-half pipe wall thickness in width."

The MEB 3-1 criteria of Sections B.3.c(3) is presented here to quantify a break size that would be reasonable, from a more current licensing viewpoint, for studying flooding vulnerability.

**Conclusion:**

The AEC regulatory requirements for evaluating non-seismic Class 1 systems for the potential to damage safety related equipment from flooding is defined in the Skovholt and DeYoung letters (References 55 and 56). An evaluation and response was completed by PINGP demonstrating compliance with these regulatory requirements. PINGP continues to be in compliance with the initial licensing basis requirements. The NSPM interpretation of the above licensing requirement is that safety related equipment functionality must not be adversely affected by a single postulated failure of a non-seismic class 1 pipe, with the failure size being equal to a round opening with area equal to a rectangle ½ the inside pipe diameter in length by ½ the pipe wall thickness in width.

### 6.1.2.8.2 Auxiliary Building Flooding Review Results

The Design Class I area of the Auxiliary Building was reviewed for the effects of flooding due to non-high energy, non-Class I piping system failures. Failures in containment & auxiliary building chilled water and fire protection lines were selected for analysis. The analysis determined the time period after which flood level would reach critical flood levels in the Auxiliary Building for each of the postulated pipe failures.

Due to its large size and unlimited water supply if lined up to the cooling water system, a leakage crack in the containment & auxiliary building chilled water system yielded the shortest required response time for any non-high energy, non-Class I piping system failure; 296 minutes to detect, identify and isolate the leak. (Ref. 64) The required response times from failures in any other non-Class I piping system are bounded by these results.

The cooling water supply to the containment & auxiliary building chilled water system is through 14″ piping located in the component cooling heat exchanger area.

In addition, the auxiliary building was evaluated for potential damage to required equipment due to cascading water as it passes through floor openings (stairwells, pipe chases, floor drains, etc.) on its way to elevation 695′ and for damage due to water spray. Most of the required equipment is located on elevation 695', which is a steam exclusion area; all penetrations from elevation 715' are sealed. Water from the upper elevations would be directed through the floor drain system and not cascade on required equipment located on elevation 695'. Due to the physical separation of the opposite-trained equipment, water spray from any leakage crack can only affect the operability of one train of any required equipment. This satisfies the required review criteria of the August 3, 1972 letter (Ref. 55).

### 6.1.2.8.3 Turbine Building Flooding Review Results

The Turbine Building was reviewed for the effects of flooding due to non-high energy, non-Class I piping system failures.

The evaluation of the failure of the largest non-seismic pipe in the Turbine Building determined that piping cracks could not result in sufficient water flow rates to cause engineered safety system failure due to the large volume of the condenser pit available to contain water. The evaluation showed that it would take over 27 hours before water filled the condenser pit to elevation 695', the elevation at which safety related equipment is located. (Reference 63) Water spray effects were determined to be not applicable because all engineered safety system components in the Turbine Building are enclosed in rooms in the Design Class 1 area. This satisfies the required review criteria of the August 3, 1972 letter (Ref. 55).

### 6.1.2.9 Gas Accumulation Management

The potential for gas accumulating in safety significant systems is a concern to the operability of the systems. Gas accumulation has the potential to air bind the pump, cause a loss of discharge pressure and flow capacity, affect core cooling flow, and cause significant pressure transients downstream of the system pumps. NRC Generic Letter 2008-01 (Reference 54) “requests that each addressee evaluate its ECCS, DHR [Decay Heat Removal] system, and containment spray system licensing basis, design, testing and corrective actions to ensure that gas accumulation is maintained less than the amount that challenges operability of these systems, and that appropriate action is taken when conditions adverse to quality are identified.”

Acceptability of gas voiding can be determined via analysis. Such analyses determine 01453043
specific void sizes per specific susceptible locations that will be considered acceptable. Discovered voids that are less than the analyzed allowable void volume will be considered acceptable and not challenge the operability of the given system.

Loading of piping and supports due to voids on pump discharge piping will be evaluated via pipe stress calculations per the methodology outlined in USAR Section 12. Voiding on pump suction piping will be evaluated for void transport to the pumps. Void transport 01453043
calculations will employ computer software to determine the void fraction at pump inlets during pump starts. Acceptance of the void fractions is based on the criteria listed below. An alternative approach to computer software can be to use the method where the maximum acceptable void volume anywhere on the system's suction tubing is equal the maximum percentage of void versus water that may pass through the pump for a given time interval. The void percentage and time interval are based on the acceptance criteria below. Acceptance criterion for the transport of voids in suction piping is given within industry guidance NEI 09-10, Rev 1a-A which provides acceptance criteria for 01453043
void transport in suction piping.

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# 6.2 SAFETY INJECTION SYSTEM

## 6.2.1 Design Basis

### 6.2.1.1 Emergency Core Cooling System Capability

Criterion: At least two emergency core cooling systems, preferably of different design principles, each with a capability for accomplishing abundant emergency core cooling, shall be provided. Each emergency core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe. The performance of each emergency core cooling system shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that (a) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, (b) failure of the shared feature or component does not initiate a loss-of-coolant accident, and (c) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss-of-coolant accident and is not lost during the entire period this function is required following the accident. (GDC 44)

Adequate emergency core cooling is provided by the Safety Injection System (which is included in the Emergency Core Cooling System) whose components operate in three modes. These modes are delineated as passive accumulator injection, active safety injection and residual heat removal recirculation.

The primary purpose of the Safety Injection System is to automatically deliver cooling water to the reactor core in the event of a loss-of-coolant accident. This limits the fuel clad temperature and ensures that the core will remain substantially intact and in place, with its heat transfer geometry preserved. This protection is afforded for:

a. All pipe sizes up to and including the hypothetical instantaneous circumferential rupture of a reactor coolant loop, assuming unobstructed discharge from both ends.

b. A loss-of-coolant associated with the rod ejection accident.

c. A steam generator tube rupture.

The performance of the emergency core cooling systems is determined through application of the 10CFR50 Appendix K evaluation models and then showing conformance to the acceptance criteria of 10CFR50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors”. The applicable acceptance criteria, extracted from 10CFR50.46 are as follows:

a. Peak cladding temperature. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

b. Maximum cladding oxidation. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

c. Maximum hydrogen generation. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

d. Coolable geometry. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

e. Long-term cooling. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

For any rupture of a steam pipe and the associated uncontrolled heat removal from the core, the Safety Injection System adds shutdown reactivity so that with a stuck rod, with or without offsite power and minimum engineered safety features, there is no consequential damage to the Reactor Coolant System and the core remains in place and intact.

Redundancy and segregation of instrumentation and components are incorporated to assure that postulated malfunctions will not impair the ability of the system to meet the design criteria. The system is effective in the event of loss of normal plant auxiliary power coincident with the loss of coolant, and can accommodate the failure of any single component or instrument channel to respond actively in the system. During the recirculation phase of a loss-of-coolant accident, the system can accommodate a loss of any part of the flow path since back up alternative flow path capability is provided.

The ability of the Safety Injection System to meet its design criteria is presented in Section 6.2.3. The analysis of the accidents is presented in Section 14.

### 6.2.1.2 Inspection of Emergency Core Cooling System

Criterion: Design provisions shall be made to facilitate physical inspection of all critical parts of the emergency core cooling systems, including reactor vessel internals and water injection nozzles. (GDC 45)

Design provisions are made to facilitate access to the critical parts of the reactor vessel internals, injection nozzles, pipes, valves and safety injection pumps for visual or boroscopic inspection for erosion, corrosion and vibration wear evidence, and for non-destructive inspection where such techniques are desirable and appropriate.

### 6.2.1.3 Testing of Emergency Core Cooling System Components

Criterion: Design provisions shall be made so that active components of the emergency core cooling systems, such as pumps and valves, can be tested periodically for operability and required functional performance. (GDC 46)

The design provides for periodic testing of active components of the Safety Injection System for operability and functional performance.

Power sources are arranged to permit individual actuation of each active component of the Safety Injection System.

The safety injection pumps can be tested periodically during plant operation using the minimum recirculation lines provided. During Mode 6, Refueling, with the reactor vessel head removed, the safety injection pumps and various associated components can be tested while directing flow to the core.

The residual heat removal pumps can also be tested during plant operation using the minimum flow recirculation lines. The pumps are used every time the residual heat removal loop is put into operation. Remote operated valves can be exercised and actuation circuits can be tested to the extent practical during routine plant operation.

### 6.2.1.4 Testing of Emergency Core Cooling System

Criterion: A capability shall be provided to test periodically the delivery capability of the emergency core cooling systems at a location as close to the core as is practical. (GDC 47)

ECCS flow tests are conducted during refueling outages and demonstrate the actual flow capability from the accumulators and the RWST to the reactor core. The high and low head injection pumps are operated as part of these tests.

The accumulator piping and the safety injection piping up to the final isolation valve before the Reactor Coolant System are filled with borated water while the plant is in operation. The accumulators and injection lines will be refilled with borated water as required by using the safety injection pumps to recirculate refueling water through the injection headers. A small bypass line and a return line are provided for this purpose.

Flow in each of the high-head safety injection header lines and in the main flow line for the residual heat removal pumps is monitored by a flow indicator. Pressure instrumentation is also provided for the main flow paths of the high head and residual heat removal pumps. Level and pressure instrumentation are provided for each accumulator tank.

### 6.2.1.5 Testing of Operational Sequence of Emergency Core Cooling System

Criterion: A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the emergency core cooling systems into action, including the transfer to alternate power sources. (GDC 48)

An integrated system test can be performed during plant shutdown. This test does not introduce flow into the Reactor Coolant System but does demonstrate the operation of the valves, pump circuit breakers, and automatic circuitry upon initiation of safety injection. The design provides for capability to test initially, to the extent practical, the full operational sequence up to the design conditions for the Safety Injection System to demonstrate the state of readiness and capability of the system. A summary of the operational sequence testing is presented in Section 6.2.4.4.

### 6.2.1.6 Codes and Classifications

Table 6.2-1 tabulates the codes and standards to which the safety injection system components are designed.

### 6.2.1.7 Service Life Under Accident Conditions

All portions of the system located within the containment are designed to operate without benefit of maintenance and without loss of functional performance for the duration of time the component is required following the accident.

## 6.2.2 System Design and Operation

### 6.2.2.1 System Description

Each of the units is provided with identical, independent facilities for emergency core cooling as described, in the following pages. Adequate emergency core cooling following a loss-of-coolant accident is provided by the Safety Injection System shown in Figures 6.2-1A and 6.2-1B (Figures 6.2-2A and 6.2-2B). The system components operate in the following possible modes:

a. Injection of borated water by the passive accumulators.

b. Injection by the safety injection pumps drawing borated water from the refueling water storage tank.

c. Injection by the residual heat removal pumps also drawing borated water from the refueling water storage tank.

d. Recirculation of spilled coolant, injected water, and Containment Vessel Internal Spray System drainage back to the reactor from the containment sump by the residual heat removal pumps.

The initiation signal for the core cooling by the safety injection pumps and the residual heat removal pumps is the Safety Injection Signal which is actuated by any of the following (See Section 7 for more details):

a. Low pressurizer pressure;

b. High containment pressure;

c. Low steam line pressure in either loop;

d. Manual Actuation.

#### 6.2.2.1.1 Injection Phase

The principal components of the Safety Injection System which provide emergency core cooling immediately following a loss of coolant are the accumulators (one for each loop), the two safety injection (high head) pumps and the two residual heat removal (low head) pumps.

The accumulators, which are passive components, discharge into the cold legs of the reactor coolant piping when pressure decreases below about 750 psig, thus rapidly assuring core cooling for large breaks. They are located inside the containment, but outside the shield wall; therefore, each is protected against possible missiles.

The safety injection signal starts the safety injection pumps and the residual heat removal pumps. The items on Figures 6.2-1A and 6.2-1B (Figures 6.2-2A and 6.2-2B) marked with an “S” receive the safety injection signal (refer also to Figure 7.4-15). The high head safety injection pumps take suction from the refueling water storage tank.

The residual heat removal pumps deliver through two nozzles that penetrate the reactor vessel and the core barrel. The high head safety injection pumps can deliver into two separate headers; one header supplies the cold legs and the other header supplies the reactor vessel. The header to the reactor vessel divides into two separate injection lines which connect to the lines from the RHR pumps and supply the two reactor vessel nozzles. The header to the cold legs divides into two injection lines connected to the cold legs of the RCS. However, safety injection is lined up to deliver to the header which supplies the cold legs of the RCS.

For large breaks, the Reactor Coolant System is depressurized and voided of coolant rapidly (about 10 seconds for the largest break) and a high flow rate is required to quickly recover the exposed fuel rods and limit possible core damage. To achieve this objective, one residual heat removal pump (high flow, low head) is required to deliver borated water to the core. Two pumps are available for this purpose. Delivery from these pumps supplements the accumulator discharge.

On valve MV32083 (MV32186)/8809C power will be locked out at the motor control center via locking the three-phase breaker in the off position after valve MV32083 (MV32186) / 8809C has been closed.

On valves MV32081 (MV32184) / 8809A and MV32082 (MV32185) / 8809B (BAST to SI Pump suction valves), power will be locked out at the motor control center by locking the three-phase breaker in the off position after the valves have been closed.

On valves MV32079 (MV32182) / 8808A and MV32080 (MV32183) / 8808B (RWST to SI Pump suction valves), power will be locked out at the motor control center by locking the three-phase breaker in the off position after the valves have been opened.

The “SI Not Ready” panel status light in the control room provides the necessary indication to the operator regarding the position of the RWST valves. A non-lit light indicates that these valves are in their correct safeguards position.

The seismic and quality classification for the boric acid tanks and the piping and valves between the tanks and the emergency core cooling system is found in the site equipment database.

The residual heat removal pumps take suction from the refueling water storage tank.

Because the injection phase of the accident is terminated before the refueling water storage tank is completely emptied, all pipes are kept filled with water before recirculation is initiated. Water level indication and alarms on the refueling water storage tank give the operator ample warning to terminate the injection phase. Additional level indicators are provided in the containment sump which also gives back-up indication that injection can be terminated and recirculation initiated.

For all breaks, the Reactor Coolant System pressure will stabilize when break flow equals injection flow. For smaller breaks this will occur at a pressure well above the shutoff head of the residual heat removal pumps. In this case, auxiliary feedwater addition to the steam generator(s) and steam dump will be used to reduce Reactor Coolant System temperature. For larger breaks the Reactor Coolant System pressure will rapidly decrease such that the Residual Heat Removal System will inject into the RCS. This large break flow and injection flow will supply the necessary RCS cooling. Auxiliary feedwater and steam dump will be used to remove the residual heat in the steam generators and maintain water inventory in the steam generators.

For large breaks (6″ and larger) the reactor pressure drops below the steam side pressure quite rapidly. Before any gap activity could be released due to clad bursting, the Reactor Coolant System pressure becomes less than the steam generator pressure. If a small tube leak existed prior to the accident, the only activity that could be released during a steam dump would be the activity initially in the coolant. The activity released in this manner would be a fraction of that released for a full tube rupture.

### 6.2.2.1.2 Recirculation Phase

After the injection operation, coolant spilled from the break and water collected from the containment spray is cooled and returned to the Reactor Coolant System by the recirculation system.

When the break is large, depressurization occurs due to the large rate of mass and energy loss through the break to containment. The system is arranged so that the residual heat removal pumps take suction from the containment floor and deliver spilled reactor coolant and borated refueling water back to the core through the residual heat exchangers. The system is arranged to allow either or both of the residual heat removal pumps to take over the recirculation function.

Water is admitted to the recirculation sump through two passive, safety related Sure-Flow ® Strainer assemblies engineered and manufactured by Performance Contracting, Inc. (PCI). Both strainers connect to a common sump pit cover plate designed to form a suction chamber in the existing sump pit.

These strainer assemblies prevent most foreign matter from entering the recirculation system.

The recirculation sump lines comprise two independent lines which penetrate the containment. Each line has two remote motor operated valves located outside the containment. Each line is run independently to the suction of a residual heat removal pump. The recirculation sump lines pass through sleeves in the containment. The sleeves are welded to the steel containment and to the recirculation sump line with all welds pressure testable. Welds in the recirculation sump line are accessible for remote visual inspecting. Welds in the sleeve between the steel containment vessel shield and the shield building are not accessible for visual inspection. The first isolation valve is totally surrounded by a steel enclosure equal in design conditions to the reactor containment. The system permits long term recirculation in the event of a passive or active component failure.

Alternative flow paths are also provided from the discharge of the residual heat exchangers for both low and high head recirculation. This is evaluated in Section 6.2.3.

The design of the recirculation sump lines are shown in Figures 6.2-3 and 6.2-4. As illustrated, the containment building serves as a sump, after the stored water has been injected and the recirculation phase begins.

The high head recirculation flow path via the high head safety injection pumps is only required for the range of small break sizes for which the Reactor Coolant System pressure remains in excess of the shut-off head of the residual heat removal pumps at the end of the injection phase.

The operator initiates recirculation by transferring the first train to recirculation when the level decreases to 33% in the refueling water storage tank. If high head safety injection is in operation, then the operator must open, from the control room, the motor operated valve MV32206/8816A, and MV32207/8816B, (MV32208/8816A, MV32209/8816B) connecting the discharge from the residual heat removal heat exchanger to the suction of the corresponding safety injection pump.

The portions of the Safety Injection System located outside the Containment were designed to operate post-accident. The primary shielding design consideration was to maintain the operational levels at the shield surface to less than 2 mr/hr for plant operation with 1% fuel defects in the core. An analyses of the shielding adequacy with the gap release model showed that maintenance personnel would be able to spend eight hours in close proximity to the shielded equipment immediately following the LOCA without exceeding the annual exposure guideline of 10CFR20. Subsequent periods would involve much lower dose commitments. For the Safety Guide 4 release model, the analysis showed that the contribution to the operator from the shielded equipment would be less than 10% of the direct contribution from the shielded containment vessel. The design also insures that discharges from all the pressure relieving devices are contained. Means are also incorporated to limit radioactivity leakage to the environs, within the guidelines set forth in 10CFR100. Recirculation loop leak detection and isolation is discussed in Section 6.2.3 and Section 6.5.

Two redundant heat removal trains are provided. Each train is completely independent and physically separated from the other. The out of containment portion is largely housed in separated below grade vaults, each containing a residual heat removal pump, a heat exchanger and sump with associated sump pump. Supply and discharge valves are located in separate pipeways adjacent to the RHR pit. A 6′ high wall divides the pipeways.

The RHR pits are shielded concrete vaults that house the residual heat removal trains used to provide long term, post-accident core cooling. The vaults are of Class I, reinforced concrete construction, completely contained inside the Class I structure portion of the auxiliary building and located within the Category 1 Ventilation Zone (see Section 10.3). Each vault is completely below grade.

For the recirculation phase of the accident the reactor coolant water which eventually is located on the containment floor is recirculated through the line from the containment floor to the suction of the residual heat removal pump. Figures 6.2-3 and 6.2-4 show the piping from the containment floor to the suction side of the residual heat removal pumps. Each of the two 14 inch lines pass from the containment sump pit, as discussed earlier in this section, through the containment vessel shell into the auxiliary building. A 18″ guard pipe completely contains each pipe from the Containment Vessel, passing through the Containment Vessel and shield building foundation concrete, into the auxiliary building and continues into an enlarged enclosure which contains the first of two remotely operated valves such that the line and the valve can be isolated in the event of a passive failure. The second valve is located close to the first valve such that the line outside the containment can be isolated in the event of a passive failure. For post-DBA recirculation flow, both recirculation trains are started. However, only one recirculation train (One residual heat removal pump and its associated heat exchanger) is required. The flow goes from the discharge of the residual heat removal pump through the residual heat exchanger and then into the reactor via either a low-head injection path or a high head injection path via a safety injection pump. The high head injection paths are provided in the event of a small break in which the pressure in the Reactor Coolant System is higher than the shut-off head of the residual heat removal pump.

In the event of a failure in the operating train during long-term recirculation flow, the capability exists to switch to the other independent recirculation flow path.

Evaluation of the ECCS capability to provide long term cooling following a LOCA is provided in Section 14 of the USAR. Reference 35 provides an evaluation for the Component Cooling System configuration with valve stops on the CC HX CL outlet. This evaluation of long term cooling is based on one train of RHR, CC and CL operating and assumes minimum flow rates and heat transfer capabilities.

Extensive analysis of post-LOCA recirculation performance has been performed due to Generic Letter 2004-02 (Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized Water Reactors) (Reference 40). This analysis is divided into several key areas as follows:

- Debris Sources
- Debris Generation Analysis
- Debris Transport Analysis
- Programmatic Controls
- Strainer Head Loss and RHR pump NPSH
- Sump Structural Analysis
- Chemical Effects Analysis
- Downstream Effects Analysis
- Upstream Effects Analysis

Analysis inputs, assumptions, and conclusions related to Generic Letter 2004-02 are discussed in the NRC Audit of PINGP corrective actions for GL2004-02 (Reference 41) and the Supplemental Response to GL2004-02 (References 42 and 43), and the April 10, 2015 "Prairie Island Nuclear Generating Plant Units 1 and 2 – Closeout of Generic Letter 2004-02," which concludes that "NRC staff finds the licensee's responses to GL 2004-02 are adequate and considers GL 2004-02 closed for the Prairie Island Nuclear Generating Plant, Units 1 and 2" (Reference 70). 01482599

#### 6.2.2.1.2.1 Debris Sources

Containment walkdowns were performed to determine the types and locations of potential debris inside containment per NEI 02-01, Rev 1 as modified by the NRC safety evaluation.

Prior to the containment walkdowns, zones of destruction (ZOD) and zones of influence (ZOI) were defined based on potential LOCA scenarios. A zone of destruction was defined as the area susceptible to jet impingement or pipe whip due to a LOCA while a zone of influence was defined as an area subject to containment spray or submergence effects. The ZOD size is equivalent to a sphere around the postulated LOCA pipe break where the radius of the sphere is equal to a specified number of pipe diameters from the postulated pipe break.

The containment walkdown results were documented and used as inputs to the later debris generation and transport calculations. All insulation, coatings, foreign materials, and other potential debris sources were documented. Material specifications and installation techniques were evaluated to determine if the materials would be expected to fail in the post-LOCA environment. An evaluation was performed to document and summarize the walkdown results for each unit.

#### 6.2.2.1.2.2 Debris Generation Analysis

Debris generation determinations were performed for all potential break scenarios per NEI 04-07 as modified by the NRC safety evaluation. The debris generation evaluation was broken into three phases:

1. Selection of bounding break locations
2. Selection of bounding break locations
3. Quantification of debris by type for each break location

All RCS piping and attached energized piping was evaluated. Feedwater (FW) and main steam (MS) piping breaks are not considered since sump recirculation is not required for FW or MS breaks. Small-bore piping less than or equal to 2” was not evaluation because debris generation for small piping is bounded by the larger pipe breaks due to the much larger ZOD.

The bounding breaks determined in the analysis are as follows:

- Break S1: On the RCS hot leg at the inlet to steam generator 12 (22). This is the most limiting break because it affects the most reflective metallic insulation (RMI).
- Break S2: On the Unit 1 12” SI (Accumulator Injection) line at elevation 703’-6” or on the Unit 2 8” Train B RHR suction line at elevation 698’-6”. These breaks were chosen because they are the breaks with the largest debris generation in close proximity to the sump.
- Break S3: On the pressurizer surge line at the connection to the pressurizer. This break was chosen because it would affect all the pressurizer insulation and most other piping and equipment insulation in the vault.
- Break S4: On the RCS hot leg at the connection to the steam generator. This break is the same location as break S1 but was chosen based on the alternate methodology of NEI 04-07.

The following assumptions were used in the debris generation determination:

- Insulation debris quantities were based upon calculation of the total volume within a specified ZOD radius.
- Qualified coatings debris quantities were based upon calculation of the total volume within a specified ZOD radius.
- Qualified coatings debris quantities were based upon calculation of the total volume within a specified ZOD radius.
- All unqualified coatings and degraded qualified coatings become debris, even when outside the ZOD.
- All latent debris becomes debris, even when outside the ZOD.
- Foreign materials (labels, stickers, placards, etc.) expected to fail are accounted for by summing all the material within containment. The sump screen wetted flow area is reduced by 75% of the original single-sided area of the items.

#### 6.2.2.1.2.3 Debris Transport Analysis

A calculation was performed to determine how much debris generated for each of the bounding break locations will reach the sump strainer. The debris transport methodology is based on guidance from NEI 04-07 as modified by the NRC safety evaluation. In lieu of performing a rigorous analysis of debris transport, it was conservatively assumed that 100% of the generated debris transports to the sump strainers with the exception of latent debris. For latent debris, it is assumed that 15% of the latent debris present in containment will be trapped in inactive portions of the sump pool as the pool fills and debris is washed down by containment spray (Reference 44).

#### 6.2.2.1.2.4 Programmatic Controls

Programmatic controls were put in place to track the debris source term for containment coatings and other potential debris sources inside containment.

The safety related coatings program tracks the quantity of unqualified and degraded coatings inside each ZOI and the quantity of qualified coatings inside each ZOD. The program performs periodic inspections and ensures that the total quantity of unqualified and degraded coatings inside the ZOIs and qualified coatings inside the ZODs are maintained below the assumed quantities in the sump performance analysis (Reference 45).

The GSI-191 debris monitoring program tracks the quantity of latent debris inside containment by periodic sampling and ensures that the latent debris levels do not exceed the assumed quantities in the sump performance analysis. The program also provides reviews of modifications and other plant changes as necessary to ensure that the design basis debris levels are not adversely changed (Reference 46).

A foreign material exclusion program was put in place that tracks material being brought into and removed from containment at power. In addition to the at-power FME program, walkdowns are performed at the end of refueling outages to ensure potential debris sources are minimized (Reference 51).

#### 6.2.2.1.2.5 Strainer Head Loss and RHR pump NPSH

The strainers are sized for the calculated post-LOCA debris load to achieve the lowest practical head loss, thereby minimizing the Impact on the Residual Heat Removal (RHR) pump NPSH during recirculation. The effective surface area for each strainer assembly is 413.65 $ft_2$, for a total of 827.3 $ft_2$. (Reference 38).

The NUREG-CR/6224 correlation and the uniform debris bed assumption was used to calculation head loss across the strainer assembly as part of the initial strainer sizing and scoping analysis. A later analysis was performed by the strainer vendor to determine head loss through the clean strainer assembly. This analysis used classic hydraulic head loss equations to determine head loss through attached pipe and fittings and an experimentally derived head loss correlation for the strainer assembly head loss.

Subsequently, prototypical head loss testing was performed using a reduced-scale prototype module to assess debris loaded head loss. This testing was performed with prototypical debris loads that bound the debris source term in containment. Review of the test results showed that the head loss across the debris bed found during testing would bound the plant specific approach velocity and water viscosity.

A calculation was developed to determine total head loss across the strainer, strainer piping, and debris bed using the previously calculated clean strainer head loss correlation and the additional head loss due to debris found by testing. This calculation determined that total head loss under worst case debris and flow conditions was less than the NPSH margin for the RHR pumps (Reference 38).

#### 6.2.2.1.2.6 Sump Structural Analysis

Structural analysis of the sump strainer assemblies, sump cover, and sump was performed. Pressure drop across the strainer under debris loaded conditions was considered along with inertial effects of submerged piping. Hydrodynamic drag loads due to water sloshing in containment due to seismic forces was evaluated and determined to be negligible. The structural analysis concluded that interaction ratios for all components were less than 1 (References 47 and 48).

An evaluation of the potential for jet impingement on the sump strainer components was performed. The conclusion of the evaluation was that no break locations were present in containment that could result in direct steam jet impingement on the strainer components.

The detailed structural analysis of the strainer assemblies was performed using a combination of manual calculations and computer analysis. GTSTRUDL version 25 was used in the seismic response spectra analysis of the strainer modules. ANSYS version 5.7.1 was used in the analysis of the inner gap plate.

The detailed structural analysis of the sump cover, piping, and piping supports was performed using a combination of manual calculations and computer analysis. AutoPIPE version 8.5 was used to generate the piping analysis.

The strainer assemblies, cover plate, piping, and base plates were designed to meet the requirements, as applicable, of USAS (ANSI) B31.1 Power Piping 1967 Edition, AISC-1963 Edition, and ASME Boiler and Pressure Vessel Code Section III. Load combinations of USAR Table 12.2.4 were used in the analysis.

#### 6.2.2.1.2.7 Chemical Effects

An evaluation of the effect of chemical precipitants on sump strainer head loss was performed. Chemical precipitants can form due to corrosion products coming out of solution within the sump pool over time. These precipitants, when combined with a preexisting strainer debris bed, can cause excessive head losses across the combined debris bed.

The approach used at PINGP to evaluate the acceptability of chemical precipitants is to demonstrate that sufficient “bare” strainer area remains available under the most adverse debris loads while still providing sufficient NPSH to the RHR pumps. This bare strainer area will allow chemical precipitants to pass through the open portions of the sump strainer.

The minimum bare strainer surface area was calculated based on the head loss correlations developed during head loss testing, maximum RHR pump flow, and required NPSH.

The total quantity of fibrous material at PINGP is very low as previously described. This low fiber quantity combined with strainer and containment geometry was evaluated to show that sufficient bare strainer area is present to exceed the minimum bare strainer area required to meet NPSH requirements for the RHR pumps.

#### 6.2.2.1.2.8 Downstream Effects

Downstream effects due to debris ingestion into the recirculation system were evaluated. Each ECCS flowpath was evaluated to determine components that could be affected by debris. Wear of pumps, seals, heat exchangers, orifices, and valves was found to be acceptable for the applicable mission time. Clogging of instrumentation lines, valves, and reactor vessel internal passages were evaluated and found to be acceptable.

All debris less than 1.1 times the sump screen size is conservatively assumed to pass through the strainer openings. No settling of debris in the sump pool is credited to reduce debris levels. The only settling mechanism credited for debris is settling in the reactor vessel lower plenum.

Wear models were developed using the methodology described in WCAP-16406 to address both abrasive and erosive wear on ECCS components. Abrasive wear was evaluated using both free flow wear and packing (or Archard’s) type.

Pump wear at the end of mission time was evaluated to determine if adequate flow to the core was maintained. In addition, downstream system resistance changes due to orifice and valve wear was evaluated to verify pump runout did not occur. Mission times used in the downstream effects analysis were 30 days for RHR components and 10 hours for SI components. In both cases, performance was acceptable for both RHR and SI pumps at the end of mission time (Reference 50).

RHR pump seal leakage was evaluated based on a passive failure of the seal and subsequent wear due to debris laden water leaking out of the failed seal with the pump in operation. The mission time used in the RHR pump seal failure analysis was 30 minutes following a passive failure of the seal. Calculated leakage at the end of the seal mission time was less than the 50 gpm assumed leakage specified in USAR Section 6.2.3.9 (Reference 50).

In-vessel downstream effects were evaluated using the methodology described in WCAP-16793-NP-A revision 2 (Reference 71), and the NRC has approved (Reference 70) PINGP's evaluation (Reference 49). The PINGP evaluation (Reference 49) demonstrated that PINGP meets all the Limitations and Conditions defined in the NRC's Safety Evaluation (Reference 72) of WCAP-16793-NP-A revision 2. The most important of these Limitations and Conditions are the quantitative acceptance criteria: fibrous debris deposition less than 15 grams per Fuel Assembly; available hot-leg break driving head greater than 13 psid; and cladding temperature below 800°F (Reference 72). 01482599

Because the PINGP fibrous debris has been calculated at less than 15 grams/Fuel Assembly, sump debris and related chemical effects do not establish a boric acid precipitation concern (Reference 49 and Reference 72).

### 6.2.2.1.2.9 Upstream Effects

Upstream debris accumulation and the potential for water hold-up was evaluated to determine if significant quantities of water could be trapped in areas of containment away from the sump pool. Each flowpath from a postulated break location to the sump pool was evaluated. In addition, the evaluation included flowpaths from containment spray liquid to the sump pool.

Each flowpath determined from the initial evaluation was further examined to determine if debris generated by the break or containment latent debris could cause blockages and result in water being held-up.

The evaluation concluded that adequate drainage paths exist to ensure that significant quantities of water will not be held-up away from the sump pool. Debris generated by the break and containment latent debris will not result in blockages that will prevent water flow to the sump pool.

### 6.2.2.1.3 Cooling Water

During the recirculation mode, the Component Cooling System is used to cool the recirculation fluid as it passes through the residual heat exchanger.

The Cooling Water System is made up of five pumps feeding a ring header which distributes cooling water throughout the plant. The header can be automatically or manually separated into two redundant supply headers. The Component Cooling heat exchangers are supplied with Cooling Water from the Cooling Water System ring header. Flow can be directed to the Component Cooling heat exchangers from either side of the ring header. This assures a heat sink for the Component Cooling System should any portion of the Cooling Water ring header need to be isolated. For post-DBA recirculation flow, two of the three safeguards Cooling Water pumps (two diesel driven and one motor driven) are started. However, only one Cooling Water pump is required to operate during the recirculation phase to cool the recirculation flow and containment atmosphere in the unit suffering the accident and provide the necessary cooling for the other unit.

### 6.2.2.1.4 Change-Over from Injection Phase to Recirculation Phase

During the injection phase, the RHR pumps and the safety injection pumps take suction from the RWST through a common suction pipe. In addition, the spray pumps also take suction from the RWST through a separate suction line. As the water level is reduced in the refueling water storage tank the static head on the suction of the pumps decreases and the available NPSH approaches the required NPSH. A calculation determined that the NPSH required by each pump taking suction from the tank is satisfied as the level decreases during the injection phase. The RWST has both a low level and a low-low level alarm. As the RWST is drained during the injection phase, one train of SI, RHR and CS pumps are stopped upon reaching the low level alarm point. This action is taken to maintain core cooling during the transfer to recirculation and to slow the RWST depletion rate while the transfer to recirculation (aligning the RHR pumps to take suction from containment sump B) is made.

The sequence, from the time of the safety injection signal, for the changeover from the injection to the recirculation is as follows (assuming both trains of ECCS are available).

a. First, sufficient water flows out the break to the containment floor to provide the required net positive suction head (NPSH) of the residual heat removal pumps to change to recirculation.

b. Second, the low level alarm on the refueling water storage tank sounds. The operator, at this point, performs the switchover to the recirculation mode for one train of ECCS.

c. Finally the low-low level alarm on the refueling water storage tank sounds. At this time, the operator performs the switchover operation for the second train of ECCS.

The following operator action is required to transfer from the injection phase to high head recirculation for small break sizes:

a. Stop one residual heat removal pump (low head safety injection pump), one high head safety injection pump and one containment spray pump (all in the same train) when level in the refueling water storage tank reaches the low level alarm set point.

b. Close the RWST to the idle low head pump suction isolation valve.

c. Close at least one high head safety injection pump test line to RWST isolation motor operated valve.

d. Open the two motor operated valves from the containment sump and thereby align the suction of the idle low head pump with the containment sump. The pump side valve bonnet is manually vented prior to opening the valve.

e. Start the low head safety injection pump. (With system pressure above the shut off head of the pump, the pump will run on miniflow).

f. Close the idle high head safety injection pump suction isolation valve from the RWST.

g. Open the cross connect valve between the low head pump discharge and the high head pump suction. The valve breaker is manually closed prior to opening the valve.

h. Start the high head safety injection pump.

i. Stop/Realign the remaining pumps (opposite train) when the RWST level reaches the low-low level alarm set point.

Remote operated valves for the injection phase of the Safety Injection System (Figure 6.2-1A and 6.2-1B (Figures 6.2-2A and 6.2-2B) which are under manual control, (that is, valves which normally are in their ready position and do not receive a safety injection signal) have their positions indicated by lights on the ready status section of the control board. At any time during operation when one of these valves is not in the ready position for injection, it is shown visually on the control board.

The following remote operated valves are normally maintained in the open position (which is their safeguards position) and do not receive a signal to open.

a. The high head safety injection pump header isolation valves (near the containment penetration) MV32073(MV32176)/8806A, MV32074(MV32177)/8806B and the high head SI pump suction isolation valves MV32162(MV32190)/8807A, MV32163(MV32191)/8807B.

b. The RHR pump suction isolation valves MV32084(MV32187)/8810A, MV32085(MV32188)/8810B.

c. The RHR vessel injection valves MV32064(MV32167)/8803A, MV32065(MV32168)/8803B even though they receive an open signal on SI.

The RHR pump suction valves MV32084(MV32187)/8810A, MV32085(MV32188)/8810B and the high head safety injection pump suction isolation valves MV32162(MV32190)/8807A, MV32163(MV32191)/8807B must be closed by the operator in order to transfer from the injection phase to the recirculation phase following a LOCA.

However to ensure that these valves are maintained in their proper position safeguard monitor lights are provided on the MCB, to give the operator rapid intelligence as to the status of these valves during normal plant operation.

Valve MV32073(MV32176)/8806A is opened and power is locked out at the motor control center via locking the valve's three phase breaker in the off position. Tripping is indicated to the operator by both lights on the control board valve controllers being dark. No additional indication is provided.

The "SI Not Ready" panel status light in the control room provide the necessary indication to the operator regarding the position of these valves. A nonlit light indicates that these valves are in their correct safeguards position.

A single active failure cannot prevent effective operation of safety injection or of the Residual Heat Removal System. If either valve MV32162(MV32190)/8807A or MV32163(MV32191)/8807B is closed, the pump in the alternate line is able to supply the required amount of water to the cold leg injection lines. The crosstie containing normally open manual valves SI-14-1(2SI-14-1)/8814A, SI-14-2(2SI-14-2)/8814B assures this.

Similarly, either Residual Heat Removal loop is sufficient to meet the system functional requirements; hence, if either valve MV32084(MV32187)/8810A or MV32085(MV32188)/8810B is closed, system performance is adequate.

Since no safety function is impaired, no changes of the control and indication features provided for valves MV32162(MV32190)/8807A, MV32163(MV32191)/8807B and MV32084(MV32187)/8810A, and MV32085(MV32188)/8810B are incorporated. The red/green lights on the valve controllers and the white lights on the safeguard status panels provide adequate operator indication of valve position.

There are no MOV's inside containment which must be closed to isolate a spilling line following a LOCA. Two separate and independent low and high head injection systems are provided for long term recirculation. The MOV's in the low head system are normally open during recirculation conditions. The two MOV's in the high head system deliver into the Reactor Vessel upper plenums and are not required to satisfy the function of the system.

Power is locked out at the motor control center via locking the valves' three phase breaker in the off position after valves MV32070(MV32173)/8801A, MV32068(MV32171)/8801B have been opened. Loss of power to the breakers is indicated to the operator by both lights on the control board valve controllers being dark.

The "SI Not Ready" panel status light in the control room provides the necessary indication to the operator regarding the position of these valves. A non lit light indicates that these valves are in their correct safeguards position.

In the event that the hypothetical loss-of-coolant accident should occur, the operator would use all means available including all high head paths to ensure adequate redundancy of paths each of which is capable of providing adequate core cooling during the recirculation phase. Therefore any single active or passive failure could not prevent the system from fulfilling its intended function. Leak detection would be provided by sump level instrumentation and radiation monitoring instrumentation as further described in Section 6.2.3 and Section 6.5.

#### 6.2.2.1.5 Location of the Major Components Required for Recirculation

The residual heat removal pumps are located in physically separated compartments at a low elevation in the auxiliary building which is below the ground floor of the containment and auxiliary buildings. The residual heat exchangers are also located in these compartments.

The high head safety injection pumps, component cooling heat exchangers and component cooling pumps are located on the ground floor in the auxiliary building.

The cooling water pumps are located in the screenhouse and the main cooling water supply headers run underground to the Class I portion of the Turbine Building.

### 6.2.2.2 Components

All associated components, piping, structures, and power supplies of the Safety Injection System are designed to Class I seismic criteria.

All components inside the containment are capable of withstanding or are protected from differential pressure changes which may occur during the rapid pressure rise to 46 psig in 10 seconds.

Motors which operate only during or after the postulated accident are designed as if used in continuous service. Periodic operation of the motors and the tests of the insulation ensures that the motors remain in reliable condition.

All motors, instruments, transmitters, and their associated cables located inside the containment which are required to operate following the accident are designed to function under the post-accident temperature, pressure, and humidity conditions. Emergency core cooling components are austenitic stainless steel, and are compatible with the containment spray solution over the full range of exposure in the post-accident environment. Refer to WCAP-7744 (Reference 1) for detailed information.

The quality standards of all safety injection system components are tabulated in summary form in Table 6.2-2.

#### 6.2.2.2.1 Accumulators

The accumulators are pressure vessels filled with borated water and pressurized with nitrogen gas. During normal plant operation each accumulator is isolated from the Reactor Coolant System by two check valves in series.

Should the Reactor Coolant System pressure fall below the accumulator pressure, the check valves open and borated water is forced into the Reactor Coolant System. Mechanical operation of the swing-disc check valves is the only action required to open the injection path from the accumulators to the core via the cold leg.

The accumulators are passive engineered safety features because the gas forces injection; no external source of power or signal transmission is needed to obtain fast-action, high-flow capability when the need arises. One accumulator is attached to each of the cold legs of the Reactor Coolant System.

The design capacity of the accumulators is based on the assumption that flow from one of the accumulators spills onto the containment floor through the ruptured loop, and the flow from the remaining accumulator provides sufficient water to fill the volume outside of the core barrel below the nozzles, the bottom plenum, and one-half the core.

The accumulators are carbon steel, clad with stainless steel and designed to ASME Section III, Class C. Connections for remotely draining or filling the fluid space, during normal plant operation, are provided.

The level of borated water in each accumulator tank is adjusted remotely as required during normal plant operations. Refueling water is added using a safety injection pump. Water level is reduced by draining to the reactor coolant drain tank. Local samples of the solution in the tanks can be taken to check boron concentration.

Redundant level and pressure indicators are provided with read outs on the control board. Each indicator is equipped with high and low level alarms.

The accumulator design parameters are given in Table 6.2-3.

#### 6.2.2.2.2 Deleted

#### 6.2.2.2.3 Refueling Water Storage Tank

The refueling water storage tank consists of a stainless steel plate reinforced with stiffener angles, which is encased in reinforced concrete walls. The concrete tank walls are poured monolithically with the surrounding floors and walls of the Auxiliary Building and thus become a part of the overall Auxiliary Building structure. The dynamic analysis of the Auxiliary Building, which includes the refueling water storage tank, was performed by John A. Blume and Associates (JAB-PS-02) [Ref. 25].

The dynamic seismic pressure on the concrete tank walls caused by the water in the tank were calculated, using the method presented in chapter six, Dynamic Pressure on Fluid Containers, Nuclear Reactors and Earthquakes, TID 7024, August 1963 [Ref. 29].

Stresses due to hydrostatic pressures were calculated and superimposed on the stresses due to the seismic loads.

The maximum stresses which occurred at the base of the tank are as follows:

| Loading Condition | Stresses (psi) Calculated | Allowable |
|---|---|---|
| Dead Load plus OBE: | | |
| Vertical Reinforcement | 12,200 | 20,000 |
| Hoop Reinforcement | 13,100 | 20,000 |
| Dead Load plus DBE: | | |
| Vertical reinforcement | 29,700 | 30,000 |
| Hoop Reinforcement | 13,100 | 30,000 |
| Dead Load plus | | |
| Live Load plus OBE: | | |
| Concrete | 725 | 1,800 |
| Dead Load plus | | |
| Live Load plus DBE: | | |
| Concrete | 976 | 1,800 |

In addition to its normal duty to supply borated water to the refueling cavity for refueling operations, this stainless steel tank provides borated water to the safety injection pumps, the residual heat removal pumps and the containment spray pumps for either a loss-of-coolant accident or a main steam line break accident. During plant operation it is aligned to the suction of the above pumps.

The capacity of the refueling water storage tank is based on the requirement for filling the refueling cavity for delivery. This quantity is in excess of that required for safeguards. This capacity (up to 275,000 gallons) provides an amount of borated water to assure:

a. A volume sufficient to refill the reactor vessel above the nozzles

b. The volume of borated refueling water needed to increase the concentration of initially spilled water to a point that assures no return to criticality with the reactor at cold shutdown and all full-length control rods, except the most reactive RCC assembly, inserted into the core

c. A sufficient volume of water on the floor to permit the initiation of recirculation. The Technical Specification minimum volume of 265,000 gallons satisfies this requirement.

The water in the tank is borated to a minimum concentration of 2600 ppm to assure the acceptance criteria for the LOCA analysis are met. The maximum boric acid concentration is approximately 2 weight percent boric acid. At 32°F, the solubility limit of boric acid is 2.2%. Therefore the concentration of boric acid in the refueling water storage tank is well below the solubility limit at nominal tank temperatures. The tank is sheathed in a concrete shell and is located in the Auxiliary Building.

The water in the refueling water storage tank is normally sampled once a week. If the water does not meet the required chemistry specifications, appropriate valving will; 1) add boric acid or demineralized water as required; or 2) circulate the tank water using its refueling water purification pump (formerly called the refueling water circulation pump) through the spent fuel purification loop until sampling indicates acceptable water quality. Refer to Figure 6.2-5; or 3) circulate the tank water using its refueling water purification pump through alternative purification systems such as reverse osmosis, which can be used to remove silica from the water as needed.

The spent fuel pit purification loop includes a pump, demineralizer and a filter. The demineralizer is a flushable, mixed bed type which removes ionic impurities. The filter removes insoluble particles. Both are capable of functioning at a flow rate of 60 gpm.

Two level indications with low and low-low level alarms are provided.

A dynamic response analysis has been performed to determine the horizontal loads to be applied to this tank for the hypothetical earthquake. Vertical seismic loads have been applied simultaneously. Wave generation in the tank has been taken into account. A membrane stress analysis of the vertical cylindrical tank was performed considering the discontinuities at the base and top.

The design parameters are given in Table 6.2-4.

### 6.2.2.2.4 Safety Injection and Residual Heat Removal Pumps

The two high-head safety injection pumps for supplying borated water to the Reactor Coolant System are horizontal centrifugal pumps driven by electric motors. Parts of the pump in contact with borated water are stainless steel or equivalent corrosion resistant material. A minimum flow bypass line is provided on each pump discharge to recirculate flow to the refueling water storage tank in the event the pumps are started with the normal flow path blocked. The design parameters are presented in Table 6.2-5. Figure 6.2-6 gives the performance characteristics of these pumps. Operator verification of safety injection pump operation is observed on the pump discharge pressure indicators. The discharge pressure transmitters are located outside of containment.

NRC Bulletin 88-04, “Potential Safety-Related Pump Loss” (Reference 15), requested the evaluation of all safety related pumps for 1) pump-to-pump interaction during miniflow operation that could result in the dead-heading of one or more of the pumps and 2) the adequacy of the minimum flow bypass lines with respect to damage resulting from operation and testing in the minimum flow mode. The initial response to NRC Bulletin 88-04, provided by Reference 16, reported that Prairie Island has no safety related system with a pump and piping system configuration that could lead to pump-to-pump interaction during miniflow operation and could therefore result in dead-heading of one or more of the pumps.

A supplemental response to NRC Bulletin 88-04 (Reference 17) reported on the results of evaluations performed on the adequacy of safety related pump miniflow lines. The results of those evaluations found that, with the exception of the safety injection pumps and the auxiliary feedwater pumps, the existing minimum flow piping configurations and flow rates were adequate for the Prairie Island safety related pumps.

An evaluation of the safety injection pump miniflow lines was performed by the safety injection pump supplier in response to NRC Bulletin 88-04. Because this evaluation included hydraulic instability at low flow conditions, not previously addressed, the safety injection pump supplier increased the recommended miniflow rates. In response to the results of this evaluation, the orifices in the safety injection recirculation lines were modified to provide an increased miniflow rate that provided the proper balance between the required safety injection flow rate and the time the pump is expected to be on miniflow.

The SI and RHR pumps were evaluated for wear due to post-LOCA sump recirculation debris as part of the Generic Letter 2004-02 analysis. This analysis is discussed further in section 6.2.2.1.2.

The auxiliary feedwater pump supplier found the existing minimum flow rate to be adequate to prevent catastrophic failure, but recommended that operation at the minimum flow rate be limited to approximately 10 hours per month to minimize cumulative long term damage and to assure a reasonable trouble free period of operation. Since the discharge pressure of the auxiliary feedwater pumps exceeds the steam generator relief setpoints, this 10 hour limitation would only apply to situations where the pumps are run with the motor operated discharge valves closed. Because the auxiliary feedwater pumps are not normally run with the discharge valves closed, and if they were run in that mode it would only be for a short time, no further action was taken in response to the auxiliary feedwater pump supplier's evaluation.

The NRC acknowledged the Prairie Island response to NRC Bulletin 88-04 in Reference 18. Northern States Power informed the NRC of the completion of all the actions required by Bulletin 88-04 in Reference 19.

The two residual heat removal (low head) pumps of the Auxiliary Coolant System are used to inject borated water at low pressure to the Reactor Coolant System. They are also used to recirculate fluid from the containment floor and send it back to the reactor or to the suction of the high head safety injection pumps. These pumps are of the vertical in-line centrifugal type, driven by electric motors. Parts of the pumps which contact the borated water during recirculation are stainless steel or equivalent corrosion resistant material. A minimum flow bypass line is provided on the discharge of the residual heat exchangers to recirculate cooled fluid to the suction of the residual heat removal pumps should these pumps be started with their normal flow blocked. The design parameters are presented in Table 6.2-5.

The pressure containing parts of the residual heat removal pumps are castings conforming to ASTM A-351 Grade CF8 or CF8M. The safety injection pumps are carbon steel forgings procured per ASTM A266 - Class I/with stainless steel cladding. Stainless plate conforms to ASTM A-240, Type 304 or 316. All bolting material conforms to ASTM A-193. Materials such as weld-deposited Stellite or Colmonoy are used at points of close running clearances in the pumps to prevent galling and to assure continued performance in high velocity areas subject to erosion.

Materials for all pressure containing parts of the pumps are traceable to heat numbers and chemical and physical test reports are checked to ensure conformance with the applicable ASTM specification. All pressure containing parts of the residual heat removal pumps were 100% radiographed and liquid penetrant inspected in accordance with Appendix VIII of Section VIII of the ASME Boiler and Pressure Vessel Code. The safety injection pump forgings receive (a) an ultrasonic and (b) magnetic particle or liquid penetrant inspection per the ASME pump and valve code.

These pumps were reviewed with special attention to the reliability and maintenance aspects of the working components. Specific areas include evaluation of the shaft seal and bearing design to determine that adequate allowances have been made for shaft deflection and clearances between stationary parts.

Where welding of pressure containing parts was necessary, a welding procedure including joint detail was submitted for review and approval by Westinghouse. The procedure included evidence of qualification necessary for compliance with Section IX of the ASME Boiler and Pressure Vessel Code Welding Qualifications.

The pressure-containing parts of the pump were assembled and hydrostatically tested to 1.5 times the design pressure for 30 minutes.

Each pump was given a complete shop performance test in accordance with Hydraulic Institute Standards. The pumps are run at design flow and head, shut-off head and three additional points to verify performance characteristics. Where NPSH is critical, this value is established at design flow by means of approved NPSHR test procedures.

Prior to initial construction, to provide additional steam break protection (i.e. to minimize the time required to borate the RCS following a steam line break) a new high head safety injection pump was selected for Units 1 and 2 with a higher shutoff head (2200psi). A detailed study was made to determine what pump characteristic would be most desirable for both steam break and LOCA. The results of the study indicated that the reduced runout capability of a higher head SI pump was more than compensated for by the pump’s ability to inject at system pressures above 1500 psi.

Details of the component cooling and cooling water pumps which serve the Safety Injection System are presented in Section 10.

### 6.2.2.2.5 Heat Exchangers

The two residual heat exchangers of the Auxiliary Coolant System cool the recirculated sump water. These heat exchangers are sized for the normal cooldown of the Reactor Coolant System. Table 6.2-6 gives the design parameters of the heat exchangers.

The ASME Boiler and Pressure Vessel Code has strict rules regarding the wall thicknesses of all pressure containing parts, material quality assurance provisions, weld joint design, radiographic and liquid penetrant examination of materials and joints, and hydrostatic testing of the unit as well as requiring final inspection and stamping of the vessel by an ASME Code inspector.

The designs of the heat exchangers also conform to the requirements of TEMA (Tubular Exchanger Manufacturers Association) for Class R heat exchangers. Class R is the most rugged class of TEMA heat exchangers and is intended for units where safety and durability are required under severe service conditions. Items such as: tube spacing, flange design, nozzle location, baffle thickness and spacing, and impingement plate requirements are set forth by TEMA Standards.

In addition to the above, additional design and inspection requirements were imposed to ensure rugged, high quality heat exchangers such as: confined-type gaskets, general construction and mounting brackets suitable for the plant seismic design requirements, tubes and tube sheet capable of withstanding full shell side pressure and temperature with atmospheric pressure on the tube side, ultrasonic inspection in accordance with Paragraph N-324.3 of Section III of the ASME Code of all tubes before bending, penetrant inspection in accordance with Paragraph N-627 of Section III of the ASME Code of all welds and all hot or cold formed parts, a hydrostatic test duration of not less than thirty minutes, the witnessing of hydro and penetrant tests by a qualified inspector, a thorough final inspection of the unit for good workmanship and the absence of any gouge marks or other scars that could act as stress concentration points, a review of the radiographs and of the certified chemical and physical test reports for all materials used in the unit.

The Residual Heat Exchangers are conventional vertical shell and U-tube type units. The tubes are seal welded to the tube sheet. The shell connections are flanged to facilitate shell removal for inspection and cleaning of the tube bundle. Each unit has a SA-285 Grade C carbon steel shell, a SA-234 carbon steel shell end cap, SA-213 TP-304 stainless steel tubes, SA-240 Type 304 stainless steel channel, SA-240 Type 304 stainless steel channel cover and SA-240 Type 304 stainless steel tube sheet.

The RHR heat exchangers were evaluated for wear due to post-LOCA sump recirculation debris as part of the Generic Letter 2004-02 analysis. This analysis is discussed further in section 6.2.2.1.2.

#### 6.2.2.2.6 Valves

All parts of valves used in the Safety Injection System in contact with borated water are austenitic stainless steel or equivalent corrosion resistant material. The motor operators on the injection line isolation valves are capable of rapid operation. All valves required for initiation of safety injection or isolation of the system have remote position indication in the control room.

Valving is specified for exceptional tightness and, where possible, such as instrument valves, packless diaphragm valves are used. All valves, except those which perform a control function, are provided with backseats which are capable of limiting leakage to less than 1.0 cc per hour per inch of stem diameter, assuming no credit taken for valve packing. Normally closed globe valves are installed with recirculation flow under the seat to prevent leakage of recirculated water through the valve stem packing. Relief valves are totally enclosed.

The check valves which isolate the Safety Injection System from the Reactor Coolant System are installed near the reactor coolant piping to reduce the probability of an injection line rupture causing a loss-of-coolant accident.

The residual heat removal loop is protected by relief valves in the lines leading to the reactor vessel. The valves are located inside the containment and are relieved to the pressurizer relief tank.

The gas relief valves on the accumulators protect them from pressures in excess of the design value.

Motor Operated Valves

The pressure containing parts (body, bonnet and discs) of the motor-operated valves employed in the Safety Injection System are designed per criteria established by the USAS B16.5 or MSS SP66 specifications. The materials of construction for these parts are procured per ASTM A182, F316, GR CF8M, or CF8. All material in contact with the primary fluid, except the packing, is austenitic stainless steel or equivalent corrosion resisting material. The pressure containing cast components are radiographically inspected as outlined in ASTM E-71 Class 1 or Class 2. The body, bonnet and discs are liquid penetrant inspected in accordance with ASME Boiler and Pressure Vessel Code Section VIII, Appendix VIII. The liquid penetrant acceptable standard is as outlined in USAS B31.1 1955 Case N-10 and ASME Section III.

When a gasket is employed the body-to-bonnet joint is designed per ASME Boiler and Pressure Vessel Code Section VIII or USAS B16.5 with a fully trapped, controlled compression, spiral wound asbestos gasket with provisions for seal welding, or of the pressure seal design with provisions for seal welding. The body-to-bonnet bolting and nut materials are procured per ASTM A193 and A194, respectively or ASTM A-453 bolting.

The entire assembled unit is hydrotested as outlined in MSS SP-61 with the exception that the test is maintained for a minimum period of 30 minutes. Any leakage is cause for rejection. The seating design is of the Darling parallel disc design, the Crane flexible wedge design, or the equivalent. These designs have the feature of releasing the mechanical holding force during the first increment of travel. Thus, the motor operator has to work only against the frictional component of the hydraulic unbalance on the disc and the packing box friction. The discs are guided throughout the full disc travel to prevent chattering and provide ease of gate movement. The seating surfaces are hard faced (Stellite No. 6 or equivalent) to prevent galling and reduce wear.

The stem material is ASTM A276 Type 316 condition B or precipitation hardened 17-4 PH stainless procured and heat treated to Westinghouse Specifications. These materials are selected because of their corrosion resistance, high tensile properties, and their resistance to surface scoring by the packing. 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 motor operator is extremely rugged and is noted throughout the power industry for its reliability. The unit incorporates a “hammer blow” feature that allows the motor to impact the disc away from the fore or backseat upon opening or closing. This “hammer blow” feature not only impacts the disc but allows the motor to attain its operational speed.

Each valve was assembled, hydrostatically tested, seat-leakage tested (fore and back), operationally tested, cleaned and packaged per specifications. All manufacturing procedures employed by the suppliers of original plant valve such as hard facing, welding, repair welding and testing were submitted to Westinghouse for approval.

For those valves which must function on the safety injection signal, 10 second operators are provided. For all other valves in the system, the valve operator completes its cycle from one position to the other within approximately 120 seconds.

Valves which must function against system pressure are designed such that they function with a pressure drop equal to full system pressure across the valve disc.

All valves in the ECCS are designed and tested to function under the most adverse conditions that the valves could be expected to experience during their design lifetime, including the highest expected pressure drop across them following a loss-of-coolant accident.

Valves larger than 2 inches which are designated for radioactive service at a fluid temperature greater than 212°F were provided with stuffing box leakoff connections. However, capping the leakoff lines and repacking valves per current maintenance standards has yielded less leakage. The potential for valve stem leakage is reduced by monitoring valves for leakage, early detection, and repacking without leakoff per current maintenance standards. Leakage detection systems are described in USAR sections 6.2.3.9, 6.5 and 6.6. The effects of leakage are described in USAR section 6.7. Post maintenance testing, inservice system leak tests, periodic walkdown surveillance, radiation monitoring, fluid inventory surveillances, and leakage control programs address early leak detection and repair.

Manual Valves

The stainless steel manual globe, gate and check valves are designed and built in accordance with the requirements outlined in the motor operated valve description above.

The carbon steel valves are built to conform with USAS B16.5. The materials of construction of the body, bonnet and disc conform to the requirements of ASTM A105 Grade II, A181 Grade II or A216 Grade WCB or WCC. The carbon steel valves pass only non-radioactive fluids and are subjected to hydrostatic test as outlined in MSS SP-61 except that the test pressure is maintained for at least 30 minutes. Since the fluid controlled by the carbon steel valves is not radioactive, the double packing and seal weld provisions are not provided.

Accumulator Check Valves

The pressure containing parts of this valve assembly are designed in accordance with MSS SP-66. All parts in contact with the operating fluid are of austenitic stainless steel or of equivalent corrosion resistant materials procured to applicable ASTM or WAPD specifications. The cast pressure-containing parts are radiographed in accordance with ASTM E-94 and the acceptance standard as outlined in ASTM E-71. The cast pressure-containing parts, machined surfaces, finished hard facings, and gasket bearing surfaces are liquid penetrant inspected per ASME B&PV Code, Section VIII and the acceptance standard is as outlined in USAS B31.1 Code Case N-10. The final valve is hydrotested per MSS SP-66 except that the test pressure is maintained for at least 30 minutes. The seat leakage is conducted in accordance with the manner prescribed in MSS SP-61 except that the acceptable leakage is 3cc/hr/in of nominal pipe diameter.

The valve is designed with a low pressure drop configuration with all operating parts contained within the body, which eliminates those problems associated with packing glands exposed to boric acid. The clapper arm shaft is manufactured from 17-4 PH stainless steel heat treated to Westinghouse Specifications. The clapper arm shaft bushings are manufactured from Stellite No. 6 material. The various working parts are selected for their corrosion resistant, tensile, and bearing properties.

The disc and seat rings are manufactured from a forging. The mating surfaces are hard faced with Stellite No. 6 to improve the valve seating life. The disc is permitted to rotate, providing a new seating surface after each valve opening.

The valves are in the closed position with a normal differential pressure across the disc of approximately 1550 psi. The valves remain in this position except for testing and safety injection. Since the valves are normally closed and are therefore not subject to impact loads caused by sudden flow reversal, it is expected that these valves will continue to perform their required functions without difficulty.

When the valve is required to operate, a differential pressure of less than 25 psig shears 01450457
any particles that may otherwise prevent the valve from functioning. Although the working parts are exposed to the boric acid solution contained within the reactor coolant loop, a boric acid ‘‘freeze up” is not expected with the low boric acid concentrations used.

Relief Valves

The accumulator relief valves are sized to pass nitrogen gas at a rate in excess of the accumulator gas fill line delivery rate. The relief valves also pass water in excess of the expected leak rate, but this is not necessary because the time required to fill the gas space gives ample opportunity to correct the situation. For an inleakage rate 15 times the manufacturing test rate, there is in excess of 1000 days before water reaches the relief valves. Prior to this, level and pressure alarms would have been actuated.

Leakage Limitations

The specified leakages across valve discs required to meet the equipment specification and hydrotest requirements are as follows:

a. Conventional globe valves - 3 cc/hr/in. of nominal pipe size

b. Gate valves - 3 cc/hr/in. of nominal pipe size for 300 and 150 pound USA Standard.

c. Motor-operated gate valves - 3 cc/hr/in. of nominal pipe size except 10 cc/hr/in for 300 and 150 pound USA Standard.

d. Check Valves - 3 cc/hr/in. of nominal pipe size; except 10 cc/hr/in for 300 and 150 pound USA Standard

e. Accumulator check valves - 3 cc/hr/in. of nominal pipe size

Relief valves are totally enclosed. Leakage from components of the recirculation loop including valves is tabulated in Table 6.2-7.

Valves in the RHR and SI systems exposed to recirculation fluid were evaluated for wear due to post-LOCA sump recirculation debris as part of the Generic Letter 2004-02 analysis. This analysis is discussed further in section 6.2.2.1.2.

#### 6.2.2.2.7 Piping

##### 6.2.2.2.7.1 General

All Safety Injection System piping in contact with borated water is austenitic stainless steel. Piping joints are welded except for the flanged connections as tabulated in Table 6.2-7.

The piping beyond the accumulator stop valves is designed for Reactor Coolant System conditions (2485 psig, 650°F). All other piping connected to the accumulator tanks is designed for 800 psig and 300°F.

The safety injection pump suction piping (210 psig at 300°F) from the refueling water storage is designed for low pressure losses to meet NPSH (net positive suction head) requirements of the pumps.

The safety injection high pressure branch lines (2485 psig at 300°F) are designed for high pressure losses to limit the flow rate out of a potential rupture of a branch line at the connection to the reactor coolant loop.

The piping is designed to meet the minimum requirements set forth in (1) the USAS B31.1 Code for the Pressure Piping, (2) Nuclear Code Case N-7, (3) ASTM Standards, and (4) supplementary standards plus additional quality control measures.

Minimum wall thicknesses are determined by the USAS Code formula in the power piping Section 1 of the USAS Code for Pressure Piping. This minimum thickness is increased to account for the manufacturer's permissible tolerance of minus 12-1/2 per cent on the nominal wall. Purchased pipe and fittings have a specified nominal wall thickness that is no less than the sum of that required for pressure containment, mechanical strength, and manufacturing tolerance.

Thermal and seismic piping flexibility analyses were performed. Special attention is directed to the piping configuration at the pumps with the objective of minimizing pipe imposed loads at the suction and discharge nozzles. Piping is supported to accommodate expansion due to temperature changes during the accident.

Pipe and fitting materials were procured in conformance with all requirements of the ASTM and USAS specifications. All materials were verified for conformance to the applicable specification and documented by certification of compliance to ASTM material requirements. Specifications impose additional quality control upon the suppliers of pipes and fittings as listed below.

a. Pipe branch lines between the reactor coolant pipes and the isolation stop valves conform to ASTM A376 and meet the supplementary requirement S6 ultrasonic testing.

b. Fittings conform to the requirements of ASTM A403. Fittings between the reactor coolant pipes and the isolation stop valves in sizes 3 inch and above have requirements for UT inspection similar to S6 of A376.

Shop fabrication of piping subassemblies was performed by reputable suppliers in accordance with specifications and procedures which define and govern material procurement, detailed design, shop fabrication, cleaning, inspection, identification, packaging and shipment.

Pipes sized 2-1/2″ and larger are butt welded. Reducing tees are used where the branch size exceeds 1/2 of the header size. Branch connections of sizes that are equal to or less than 1/2 of the header size are of a design that conforms to the USAS rules for reinforcement set forth in the USAS B31.1, Code for Pressure Piping. Bosses for branch connections are attached to the header by means of full penetration welds.

All welding was performed by welders and welding procedures qualified in accordance with the ASME Boiler and Pressure Vessel Code Section IX, Welding Qualifications. The Shop Fabricator was required to submit all welding procedures and evidence of qualifications for review and approval prior to release for fabrication.

All high pressure piping butt welds containing radioactive fluid, at greater than 600°F temperature and 600 psig pressure or equivalent, were radiographed.

The remaining piping butt welds are randomly radiographed. The technique and acceptance standards are those outlined in UW-51 of the ASME B&PV Code Section VIII. In addition, butt welds were liquid penetrant examined in accordance with the procedure of ASME B&PV Code, Section VIII, Appendix VIII and the acceptance standard as defined in the USAS Nuclear Code Case N-10. Finished branch welds were liquid penetrant examined on the outside and where size permits, on the inside root surfaces.

A post-bending solution anneal heat treatment was performed on hot-formed stainless steel pipe bends. Completed bends are then completely cleaned of oxidation from all affected surfaces. The shop fabricator was required to submit the bending, heat treatment and clean-up procedures for review and approval prior to release for fabrication.

General cleaning of completed piping subassemblies (inside and outside surfaces) is governed by basic ground rules set forth in the specifications. For example, these specifications prohibit the use of hydrochloric acid and limit the chloride content of cooling water and demineralized water.

**6.2.2.2.7.2** Deleted

#### 6.2.2.2.8 Pump and Valve Motors

Motors Outside The Containment

Motor electrical insulation systems were supplied in accordance with USAS, IEEE and NEMA standards and were tested as required by such standards. Temperature rise design selection is such that normal long life is achieved even under accident loading conditions.

Although the motors which were provided only to drive engineered safety features equipment were normally run only for test, the design loading and temperature rise limits were based on accident conditions. Normal design margins were specified for these motors to make sure the expected lifetime includes allowance for the occurrence of accident conditions.

Criteria for motors of the Safety Injection System require that under any anticipated mode of operation, the motor nameplate rating is not exceeded. The motors have a 1.15 service factor for normal operation. Design and test criteria ensure that motor loading does not exceed the application criteria.

Valve Motor Operators Inside Containment

Valve operator suppliers have conducted loss-of-coolant environmental tests on a Class H unit similar to those used in this plant. WCAP 7410-L (Reference 2) indicates that the unit operated satisfactorily at test conditions more severe than the post-LOCA environment.

In addition, Westinghouse has performed environmental tests on a Class B unit similar to that being used in this plant. The results of the Westinghouse tests indicate that the equipment will perform its required function in the post-LOCA environment.

Tests performed on valve operators, both Class H and Class B, include:

a. Preliminary heat tests (dry heat 16 hrs @ 375°F) on limit and torque switches. All parts operated freely.

b. Preliminary heat tests on actuator. A complete operator was assembled and baked at 325°F for 12 hours. Unit operated every half-hour for 2 minutes full open to full close. All operations satisfactory.

c. Preliminary live steam test. Live steam injected into switch compartment. Unit operated every half-hour for 2 minutes over a period of nine hours. All operations satisfactory.

d. Heat aging of motor. Heat aging at 180°C for 100 hours (equivalent to 40-year life) was performed. Comparison of insulation resistance between new and aged motor indicated no significant insulation degradation.

e. Life cycle test. 150 life cycle test under loaded conditions (valve operator produced approximately 16,500 pounds of thrust). No noticeable change in operator following test.

f. Environmental test. Valve operators subjected to simulated post-LOCA accident environmental conditions and sprayed continuously for a period of 3 hours with a solution of boric acid and sodium hydroxide.

   The Class H operator (actual peak test conditions 320°F at 90 psig), survived the first day of exposure during which 12 complete reversing cycles were accomplished. Following one week's exposure to 247°F and 14.7 psia the unit was operated for two complete reversing cycles. The unit operated satisfactorily.

   The Class B operator survived the first day of exposure with 12 complete reversing cycles. However, after 5 days of exposure, the operator failed (failure found to be a short in the motor winding).

Class H operators are required on all motor operated safeguard valves inside containment that must operate in the post LOCA environment. Class H operators are supplied on all Safety Injection System motor operated valves in containment, including those not required for post-LOCA operation. Class B operators are supplied outside containment.

A production valve motor with Class B insulation was irradiated to a level of 2 x $10^8$ rads using a cobalt-60 irradiation source. The irradiated motor and an identical unirradiated motor were put through a series of reversing tests at room temperature, followed by a series of reversing tests at 275°F. The room temperature test was repeated while both motors were vibrated at a frequency of 30 cycles per second. Both motors operated satisfactorily during all of the tests. No significant difference was evident in the comparison of the data for the two units throughout the test period.

Subsequent testing performed to qualify the units in the post LOCA pressure, temperature, and chemical environment have shown Class H insulation to be the better choice, due to its better thermal characteristics, for in containment applications. The change in insulation material was not connected to the results of the irradiation testing. Furthermore the motor suppliers have assured that the motors with Class H insulation will withstand the effects of high radiation better than those with Class B insulation.

In response to IE Bulletin 79-01B, an evaluation of the environmental qualification of safety related electrical equipment at the Prairie Island facility was conducted. Results of the evaluation were included in a report submitted per Reference 3. This report also addressed the radiation qualification requirement per NUREG-0737, II.B.2. The post accident radiation environment was re-evaluated for impact due to implementation of heavy bundle fuel and incorporation of power measurement uncertainty (Ref. 53). Results of this analysis are incorporated into the Equipment Qualification program.

See Section 8.9 for further discussion of environmental qualification testing.

#### 6.2.2.2.9 Electrical Supply

Details of the normal and emergency power sources for the Safety Injection System are presented in Section 8.

## 6.2.3 Design Evaluation

### 6.2.3.1 Protection Against Dynamic Effects

The two high head injection headers penetrate the containment adjacent to the auxiliary building, and then each split into two branches.

The portion of the high head injection system within containment is connected to each loop's cold leg and the piping leading to the reactor vessel injection nozzles. The portion of the low head injection system within the containment is connected directly to the injection nozzles on the vessel.

For most of the routing, these lines are outside the reactor and steam generator shielding, and hence they are protected from missiles originating within these areas.

The coolant loop supports are designed to restrict dynamic motion well below the allowable displacements of the attached safety injection piping.

All hangers, stops, restraints and anchors are designed in accordance with the applicable codes and design criteria as set forth in Section 12 which provide minimum requirements on materials, design and fabrication with ample safety margins for both dead and operational dynamic loads over the life of the equipment. In addition to the normal load conditions all other combinations shown in Table 12.2-14 and the design requirements of Table 12.2-13 are used in the design of the supports. Specifically, the piping was designed to Power Piping Code, USAS B31.1.0-1967. Paragraph 120.1 of the B31.1.0-1967 Code is a generic opening paragraph addressed to the subject of "Loads on Pipe Supporting Elements." Section (c) of this paragraph states, "Where resonance with imposed vibration and/or shock occurs during operation, suitable dampeners, restraints, anchors, etc., shall be added to remove these effects." Under this paragraph, our design bases has not specifically considered water hammer effects.

The piping systems have been designed to incorporate supporting by means of variable spring hangers, rigid supports, constant support hangers, pipe anchors, guides, and snubbers for considerations of dead load, thermal expansion, seismic occurrences, pipe rupture loads, jet forces in those combinations as listed in Section 12, such that an additional occurrence of water hammer would not cause overstressing of the piping. Additionally, pipes and piping supports were analyzed for waterhammer loads per Generic Letter (GL) 2008-01 as described in Section 6.1.2.9.

All such piping systems are checked for excessive vibration caused by dynamic effects created by transient operations of combinations of valve and pump operation.

### 6.2.3.2 Injection Connections and Flow to the Core

The injection lines from the accumulators, low-head pumps, and high head pumps are connected to the Reactor Coolant System to provide the maximum performance flexibility for a loss-of-coolant accident of any size or location. This performance flexibility is available not only during the injection phase, but also during the long term recirculation.

Each accumulator is attached to a Reactor Coolant System cold leg. The core is therefore rapidly flooded from the bottom to provide the earliest possible cooling of the entire core and the attendant arresting of the clad temperature transient. When the accumulators reflood the bottom regions of the core, rapid steam generation causes a mixture of steam and entrained water droplets to flow through and cool the upper regions of the core.

The residual heat removal pumps (low-head) deliver borated water to the core outlet plenum through nozzles connected to the reactor vessel (See Figure 3.6-5). The low head system thereby serves a basic injection function in the event of large breaks in the Reactor Coolant System. This function is to provide continued make-up following the cooling of the core by the accumulators. A second function of these pumps is to provide continued cooling during the recirculation phase.

The high head system connects to both the cold legs and the core outlet plenum to provide a balanced performance between the steam break and small loss-of-coolant accidents. The headers from each pump are cross connected to provide core outlet plenum and two cold leg connections from each pump. Conventional pressure drop calculations with resistance coefficients and friction factors for commercial steel were used which results in a conservative estimate of the system resistance. However these calculations are used only to verify that the total piping resistance is greater than that required to prevent excessive pump run-out. The required flows down each line were established during the preoperational tests by adjusting live resistance via throttling valves. The selected flows were based on maximum pump runout capability which was determined by the vendor’s shop test. Therefore, final system flow distribution is obtained by adjustment of live resistances during the operational test and pressure drop calculations were used only to verify that the as built system resistance is less than that required to obtain the maximum system performance. This method of establishing system performance insures balanced flows or equal flows down parallel injection paths so that flow lost in the event of a ruptured injection line following a LOCA is clearly defined.

Also given consideration in the design is the special case where a broken safety injection line is the initiating loss-of-coolant accident.

Special attention is given to the factors that could adversely affect the accumulator and safety injection flow to the core. These factors are:

a. Accumulator water carried out of the break, or to other parts of the system during blowdown.

b. Steam bubble formation when accumulator water refloods the core.

c. The effect of the nitrogen gas entering the vessel.

#### 6.2.3.3 Range of Core Protection

The measure of effectiveness of the Safety Injection System is the ability of the pumps and accumulators to keep the core flooded or to reflood the core rapidly where the core has been uncovered by (postulated) large area ruptures. The result of this performance is to sufficiently limit any increase in clad temperature below a value where emergency core cooling criteria are met (Section 6.2.1).

Simulations of a sufficient number of break sizes were performed to demonstrate that the Safety Injection components meet the emergency core cooling requirements.

#### 6.2.3.4 System Response

To provide protection for large area ruptures in the Reactor Coolant System the Safety Injection System must respond to rapidly reflood the core following the depressurization and core voiding that is characteristic of large area ruptures. The accumulators act to perform the rapid reflooding function with no dependence on the normal or emergency power sources, and also with no dependence on the receipt of an actuation signal.

Operation of the Emergency Core Cooling System with one of the two available accumulators delivering their contents to the reactor vessel (one accumulator spilling through the break) meets the principal design criteria for the system.

The function of the safety injection (or residual heat removal) pumps is to complete the refill of the vessel and maintain long term cooling. As discussed earlier, the flow from one safety injection pump or one residual heat removal pump is sufficient to complete the refill with no subsequent loss of level in the core sufficient to cause excessive fuel heat-up and subsequent fuel failure. Moreover, there is sufficient excess water delivered by the accumulators to tolerate a delay in starting the pumps.

Initial response of the injection systems is automatic, with appropriate allowances for delays in actuation of circuitry and active components. The active portions of the injection systems are automatically actuated by the safety injection signal (Section 7). In addition, manual actuation of the entire injection system and individual components can be accomplished from the control room. In analysis of system performance, delays in reaching the programmed trip points and in actuation of components are conservatively established on the basis that only emergency onsite power is available.

The starting sequence of the safety injection pumps and related emergency power equipment is designed so that delivery of full rated flow is reached within 20 seconds after the process parameters reach the set points for the injection signal (Section 8).

Assumptions for the time of SI flow initiation for accident analyses vary. Sections 14.6 and 14.7 (with associated tables) include SI initiation time data.

The starting sequence of engineered safety features is presented in Section 8.

### 6.2.3.5 Single Failure Analysis

A single active failure analysis is presented in Table 6.2-8a. All credible active system failures are considered. The analysis of the loss-of-coolant accident presented in Section 14 is consistent with the single failure analysis.

It is based on the worst single failure (generally a pump failure) in both the safety injection and residual heat removal pumping systems. The analysis shows that the failure of any single active component will not prevent fulfilling the design function.

A redundant train of the safety injection and residual heat removal system can be utilized as an alternative flow path to maintain core cooling if any part of the recirculation flow path becomes unavailable. The actions required to place a redundant train in service can be accomplished from the control room.

A ruptured flow path would be detected and identified by flow indication. Rupture of a residual heat removal loop in a RHR Pit would be detected and identified (and therefore adequately located) by sump water level and by airborne activity indications associated specifically with that loop.

The sump pumps for the RHR pits are each powered by one of the two emergency diesel generators for the unit.

Failure analyses of the component cooling and cooling water system under loss-of-coolant accident conditions are described in Section 10.4.2 and 10.4.1 respectively.

A single failure analysis of the Emergency Core Cooling System (ECCS) and its supporting subsystems is included in a report titled ‘‘ECCS Actuation System” submitted to the NRC dated December 22, 1976 [Ref. 26].

The analysis was reviewed and accepted by the NRC per Reference 4. The analysis demonstrates that the ECCS and supporting subsystems meet the single failure criterion as defined in IEEE STD 279-1971. The NRC requested a Technical Specification change to cover the control of two valves SI-14-1(2SI-14-1)/8814A and SI-14-2(2SI-14-2)/8814B, [Ref. Figure 6.2-1B(Figure 6.2-2B)]. Either of these valves being closed would prevent the ‘‘B” SI pump from injecting into the cold legs. Following plant review, the NRC was informed that these valves are already covered in the Technical Specifications under a general specification dealing with valves which could affect an injection path.

In December 1981, Westinghouse informed the NRC of a potential problem related to the single failure assumptions used in large break LOCA analysis. They stated that the loss of one low head safety injection pump is not the most conservative single failure assumption for some Westinghouse plants. For some plants maximum low head safety injection was more limiting (i.e. no single failure). The additional cold water spilling out the break with maximum low head safety injection during reflood reduces containment pressure. This drop in containment pressure causes a drop in steam density, which decreases the flow of steam out of the vessel and leads to higher temperatures in the core. Prairie Island water inventory, accumulator water, available at the end-of-bypass is not sufficient to fill the reactor vessel and downcomer. Other plants with more accumulators fill the downcomer and pump low pressure injection water spills on the floor. Since the accumulator water does not fill the downcomer, the additional pump running will tend to increase the downcomer level increasing the driving head which increases core cooling. This positive effect on core cooling of running two low pressure safety injection pumps offsets the later negative effect in reflood of spilling more water in containment and dropping containment pressure. Therefore, one low pressure safety injection pump operating during a LOCA is more limiting than two operating. LOCA analysis will continue to assume one low pressure safety injection pump operating.

### 6.2.3.6 Reliance on Interconnected Systems

During the injection phase, the safety injection pumps do not depend on any portion of other systems with the exception of the component cooling water system. During the recirculation phase of the accident for small breaks, suction to a safety injection pump is provided by the associated residual heat removal pump.

The residual heat removal (low head) pumps are normally used in Mode 4, Hot Shutdown, Mode 5, Cold Shutdown, and Mode 6, Refueling. Whenever the reactor is in Mode 1, Power Operation, Mode 2, Startup, or Mode 3, Hot Standby, the pumps are aligned for emergency duty.

### 6.2.3.7 Shared Function Evaluation

Table 6.2-9 is an evaluation of the main components, which have been previously discussed, and a brief description of how each component functions during normal operation and during the accident. There are no components of the Safety Injection System which are shared by the two units.

#### 6.2.3.8 Passive Systems

The accumulators are a passive safety feature in that they perform their design function in the total absence of an actuation signal or power source.

The working parts of the check valves are exposed to fluid of relatively low boric acid concentration contained within the reactor coolant loop. Even if some unforeseen deposition accumulated, calculations have shown that a differential pressure of about 25 psi will shear any particles in the bearing that may otherwise prevent the valve from functioning.

It is our position that administrative control, i.e. operating procedures, is the preferred means for accomplishing those changes in system configurations necessary to change the basic operating mode or status of the plant; e.g. from Mode 1, Power Operation, to Mode 3, Hot Standby, cooldown, heat-up, etc. These changes in basic operating modes, or plant status, result from deliberate intent on the part of the plant operator and involves an operating procedure requiring a multitude of specific operator actions to achieve. Administrative controls are therefore a basic principle in plant operation.

For critical functions, our designs include one, or a combination, of the following indications to show the operator the status of plant systems and to highlight the existence of an incorrect configuration:

a. Indication lights at the control switch for each remote controlled device indicating the operating state or position of the device.

b. A separate monitor light indication grouped with lights for other devices having a similar function such that the lights in the group are all on or all are off to provide for quick operator evaluation of systems status during any mode of plant operation.

c. Alarms redundant to the above indications are to alert the operator to the improper state of a device relative to the plant condition.

The high head SI pump suction isolation valves MV32162(MV32190)/8807A, MV32163(MV32191)/8807B are normally maintained in the open position (which is their safeguards position) and do not receive a signal to open.

These valves (MV32162(MV32190)/8807A, MV32163(MV32191)/8807B) <u>must</u> be closed by the operator in order to transfer from the injection phase to the recirculation phase following a LOCA. Therefore, it would be undesirable to lock these valves in their open position during normal plant operation. However, to ensure that these valves are maintained in their proper position, safeguard monitor lights are provided on the MCB to give the operator rapid intelligence as to the status of these valves during normal plant operation.

The valves in the lines from the refueling water storage tank to the containment spray pumps are motor operated, normally open valves that are left open with the breakers off during reactor operation.

The application of the above indications to supervise the administrative operation of specific valves identified as being of concern, together with other means provided to guard against improper operation, are as follows:

Redundant indications supervising the administrative positioning of the accumulator discharge valves are provided as a check to the operator. These are:

a. Red (open) and Green (closed) position indicating lights at the control switch for each valve. These lights are powered by valve control power and actuated by valve motor operator limit switches. During power operation the valve breaker is off and Red/Green light indication is not available with no control power.

b. A monitor light that is on when the valve is not fully open in an array of monitor lights that are all off when their respective valves are in proper position enabling safeguards operation. This grouping highlights a valve not properly lined up. This light is energized from a separate monitor light supply and actuated by a valve motor operator limit switch.

c. Each valve has an alarm annunciator point which is activated by both a stem mounted limit switch and by a gear operated limit switch whenever an accumulator valve is not fully open for any reason. With the RC system at pressure (the pressure at which the safety injection is unblocked), both circuits will annunciate. The stem mounted limit switch circuit contains a cycle timer for reflash capabilities. This circuit operates only if SI is unblocked. The annunciator will alarm and reflash once every hour, during normal plant operating conditions, unless SI is blocked. When an accumulator is required to be operable by the Technical Specifications and SI is blocked, the gear operated limit switch will provide alarm annunciation if the valve is not fully open. The combination of the two limit switch circuits ensures that true valve position is indicated at all times.

d. In the event a valve is closed for accumulator or valve testing at the time injection is required, a safety injection signal is applied to open the valve, over-riding the test closure.

The addition of a pressure initiated automatic opening feature to the control of these valves is considered undesirable or ineffective for several reasons. First, these valves are required to be closed periodically for accumulator system testing which would require blocking of the automatic opening signal. It is not clear that there is an appropriate system parameter to provide the automatic unblocking as required by IEEE 279-1968. Second, when the plant is shutdown and cooled down for maintenance or refueling maintenance procedures for personnel protection will require these valves to be tagged out and their power supplies locked off to preclude erroneous opening. The automatic opening signal would be of no avail if the operator neglects to have power restored prior to returning to operation. The tag out procedure is administrative.

The isolation valve at each accumulator is closed only when the reactor is intentionally depressurized or momentarily for testing. The isolation valve is normally open and an alarm in the control room sounds if the valve is inadvertently closed.

The check valves are normally closed, with a nominal differential pressure across the disc of approximately 1550 psi. They remain in this position except for testing or when called upon to function. Since the valves are normally closed and are therefore not subject to the abuse of flowing fluids or impact loads caused by sudden flow reversal and seating, they do not experience any wear of the moving parts, and function as required.

As the Reactor Coolant System is pressurized during normal plant heatup, the check valves are checked for back leakage. This check confirms disc seating and whether leakage is within design limits.

The accumulators can accept leakage back from the Reactor Coolant System without effect on their availability. Table 6.2-10 indicates that back leakage rates, over a given time period, require readjusting the level at the end of the time period. In addition, these rates are compared to the maximum allowed leak rates for manufacturing acceptance tests (24 cc/hr, i.e. 2 cc/hr/in).

Back leakage at a rate of 5 cc/hr/inch, 2-1/2 times test, do not require that the accumulator water volume be adjusted more often than once per year. This would indicate that level adjustments can be scheduled for normal refueling shutdowns and that this work can be done at the operator’s convenience.

The accumulators are located inside the reactor containment and protected from the reactor coolant system piping and components by a missile barrier. Accidental release of the gas charge in the two accumulators causes an increase in the containment pressure of approximately 0.1 psi.

During normal operation, the flow rate through the reactor coolant piping is approximately five times the maximum flow rate from the accumulator during injection. Therefore fluid impingement on reactor vessel components during operation of the accumulator is not restricting.

### 6.2.3.9 Recirculating Loop Leakage

Table 6.2-7 summarizes the maximum potential leakage from the recirculation loop leak sources. In the analysis, a maximum leakage is assumed from each leak source. For conservatism, a leakage of 10 drops per minute was assumed from each flange although each flange would be adjusted to essentially zero leakage.

During recirculation following an accident, significant margin exists between the design and operating conditions of the residual heat removal system components, as shown in Table 6.2-11. In addition, during normal plant cooldown operation, the residual heat removal system is initiated when the primary system pressure and temperature have been reduced to 425 psig and 350°F respectively. Since the maximum operating conditions during recirculation following accident are 200 psig and approximately 250°F, significant margin also exists between normal operating and maximum accident conditions. In view of the above margins, it is considered that the leakage rates tabulated in Table 6.2-7 are conservative.

Long term operation of the system occurs only at low pressures and temperatures; therefore, (a) there is not significant internal energy which could cause massive rupture of a weakened pipe, and (b) thermal shock cannot occur during post-accident operation. Moreover, the integrity of the Class I piping system is proven, as stated above, each time that it is used during the normal cooldown and cold shutdown operating modes. During this normal usage the system operates at pressures on the order of ten times higher than those experienced during the long term design basis accident operation; hence, the integrity of the system will be continuously proven. It is concluded that a massive failure of this piping is not credible. The maximum leakage rate possible is 50 gpm which is based on the anticipated flow rate of water through the pump seal if the entire pump seal were wiped out and the area between the shaft and housing were completely open.

A combination of drainage sump monitoring and RHR pit ventilation air radiation monitoring has been provided to detect leakage in the RHR Pits.

The primary method of leakage detection during normal operation is achieved through use of sump level instrumentation. Each RHR pit sump is provided with a 60 gpm pump to transfer leakage to the radwaste system. A manually operated, three-way valve on the discharge of the sump pump allows the operator to send the flow to the containment sump if required during certain accident responses. Pumps are actuated by level control instrumentation with alarms in the control room. Leakage rate monitoring is provided by observation of sump pump operation. Valving and instrumentation is provided to permit manual isolation of individual residual heat removal lines from the control room.

Airborne activity monitors are provided for leakage detection during post-accident conditions. These monitors are discussed in Section 6.5. Airborne activity associated with post-accident conditions will be entrained in the airflow which is maintained through the RHR pits. This air flow is conducted to the Auxiliary Building Special Ventilation System by segregated ventilation ducting which are equipped with radiation monitors. These monitors are designed to detect airborne activity due to leakage associated with post-accident conditions and alarm in the Control Room.

The radiological consequences resulting from leakage of containment sump water being circulated through the residual heat removal systems external to the containment has been evaluated and is discussed in Section 6.7.

#### 6.2.3.10 Service Life Under Accident Conditions

The design service lifetime for the components and associated pipe in the ECCS is based on the original design life of the plant (40 years).

All components in the ECCS are designed to operate without benefit of maintenance and without loss of functional performance for the duration of time the component is required following the accident.

However, the redundancy built into the system is based on maintaining adequate core cooling following an accident in the event of a single active failure during the injection phase or a single passive failure during the recirculation phase of a LOCA.

#### 6.2.3.11 Pump NPSH Requirements

Residual Heat Removal Pumps

The NPSH of the residual heat removal pumps is evaluated for normal plant shutdown operation, and both the injection and recirculation phase operation of the design basis accident. Recirculation operation gives the limiting NPSH requirement. The available NPSH is determined from the containment water level, the pressure drop in the suction piping from sump B to the pumps, and the pressure drop across the debris loaded strainer assemblies.

The potential for air ingestion into the RHR pumps due to vortex formation in the containment sump during long term recirculation operation is evaluated in Reference 42. No vortexing and air ingestion is expected.

During the recirculation phase the residual heat removal pumps can be used to supply recirculated water directly to the reactor vessel via the upper plenum injection lines, to the high head safety injection pumps suction, and to the suction of the containment spray pumps. An attempt to supply water through all of these flow paths simultaneously, would result in residual heat removal pump run out. An analysis was performed (Reference 13) to assess the required flow to the reactor core and the need for containment spray during the recirculation mode. This analysis showed that the required flow following the switch over to the recirculation mode can be met by one high head safety injection pump. It further showed that containment spray flow should not be required following the switch over to recirculation. These minimum flow analyses have been more recently supplemented as described in Section 14.10 of the USAR. In addition, the containment integrity analyses described in Section 14.5.5 and Appendix K of the USAR provide updated confirmation that Containment Spray is not required during recirculation. To avoid residual heat removal pump run out, procedures covering the switch over to recirculation operation now call for securing Containment Spray during the switchover to recirculation.

Safety Injection Pumps

The NPSH for the safety injection pumps is evaluated for both the injection and recirculation phase operation of the design basis accident. The end of injection phase operation gives the limiting NPSH requirement and the NPSH available is determined from the elevation head and vapor pressure of the water in the refueling water storage tank, and the pressure drop in the suction piping from the tank to the pumps.

### 6.2.3.12 Boron Build-up Analysis

A post-LOCA Hot Leg switchover (HLSO) time is calculated to support emergency operating procedures that require a realignment of the SI flow path from the cold legs to the vessel. This realignment of the SI to the vessel precludes boron precipitation in the reactor vessel following certain LOCA accidents. At issue are postulated cold leg breaks where injected SI water boils off due to decay heat, leaving behind boric acid. The concern is that eventually the boric acid solution in the vessel may reach the boron precipitation point. The preclusion of boron precipitation is relied upon as a criterion for assuring core coolable geometry.

For cold leg SI injection plants, the criteria to prevent the possibility of boron precipitation is re-alignment of the SI flow to the hot legs prior to the boron concentration reaching the saturation point. For upper plenum injection (UPI) plants, this hot leg switchover is functionally accomplished by UPI injection, so the SI injection realignment is defined as providing capability of the ECCS system to deliver low head injection (RHR) into the upper plenum of the reactor vessel.

For large breaks, the RCS depressurizes rapidly to a point where the RHR system can begin injecting. For smaller breaks, the pressure may reach equilibrium above the shutoff head of the RHR system. In this case, the RCS must be cooled down sufficiently prior to depressurization. This presents the limiting time frame before which RHR could be injected into the vessel (thus preventing boron precipitation).

A calculation determined that vessel injection must occur before 7.5 hours following a LOCA to prevent boron precipitation. (Reference 52) This then becomes the acceptance criteria for the ability of PI to begin vessel injection. The Prairie Island Emergency Operating Procedures are designed to ensure that vessel injection through the RHR system occurs prior to the 7.5 hours criteria in all cases.

The effect of post-LOCA sump recirculation debris in the reactor vessel has been evaluated for impact on the boric acid precipitation analysis as part of the Generic Letter 2004-02 analysis. It was concluded that the volume of settled debris is small and blockage of coolant passages would be minimal. This will not affect the existing boric acid precipitation analysis. This analysis is discussed further in section 6.2.2.1.2.

## 6.2.4 Tests and Inspections

### 6.2.4.1 Inspection Capability

All components of the Safety Injection System can be inspected periodically to demonstrate system readiness.

The pressure containing systems can be inspected for leaks from pump seals, valve packing, flanged joints and safety valves during system testing.

In addition, to the extent practical, the critical parts of the reactor vessel internals, injection nozzles, pipes, valves and safety injection pumps can be inspected visually or by borescopic examination for erosion, corrosion, and vibration wear evidence, and by non-destructive test inspection where such techniques are desirable and appropriate.

### 6.2.4.2 System Testing

Various preoperational tests were performed to verify the ECCS met the appropriate design criteria. A summary of these tests is contained in Appendix J.

Tests are conducted during each refueling shutdown to demonstrate the flow rates delivered through each injection flow path. This testing ensures that the piping from the RWST through the various valves and pumps into the RCS is filled with borated water.

#### 6.2.4.3 Components Testing

Each active component of the Safety Injection System can be individually actuated on the normal power source at any time during plant operation to demonstrate operability. The test of the safety injection pumps employs the minimum flow recirculation test line which connects back to the refueling water storage tank. Remote operated valves can be exercised and the actuation circuits tested. The automatic actuation circuitry, valves and pump breakers also may be checked during integrated system tests performed during Mode 5, Cold Shutdown of the Reactor Coolant System consistent with Technical Specification surveillance requirements.

The operation of the remote stop valve in the accumulator discharge line is observed by the control board light indication for the valves. A second check is performed by a divergent indication of the accumulator valve status on the “SI NOT READY” panel and the clearing of the valve closed alarm. Test circuits are provided to periodically examine the leakage back through the check valves and to ascertain that these valves seat whenever the reactor system pressure is raised.

Test circuits are provided to periodically examine the leakage back through the accumulator check valves. Tests are conducted whenever the differential pressure has been changed to unseat these check valves. Leakage tests are performed to ascertain that these valves reseat.

If leakage through a check valve should become excessive, the isolation valve may be closed and an orderly shutdown initiated to repair the check valve. The performance of the check valves in this application has been carefully studied and it is concluded that it is highly unlikely that the accumulator lines would have to be closed because of leakage.

The isolation valves are closed and their power supplies opened when the accumulators are not required to be operable. The valves are opened and their power supply opened during plant heatup and pressurization. During normal operation the accumulator outlet MOVs are open and the valve motor breakers are off.

The recirculation piping was initially hydrostatically tested at 150 percent of design pressure of each portion of the loop. The entire loop is also pressurized during periodic testing of the engineered safety features components. Since the recirculation flow path is operated at a pressure in excess of the containment pressure, it is hydrostatically tested during periodic retests at the recirculation operating pressures.

This can be accomplished by running each pump utilized during recirculation (safety injection and residual heat removal pumps) in turn at near shut off head conditions and checking the discharge and recirculation test lines. The suction lines can be tested by running the residual heat removal pumps and opening the flow path to containment spray and safety injection pumps in the same manner as described above.

During the above tests, all system joints, valve packings, pump seals, leakoff connections, or other potential points of leakage can be visually examined and corrections made if required.

### 6.2.4.4 Operational Sequence Testing

The operational sequence of the Safety Injection System was tested to the extent practical prior to initial plant operation. These tests demonstrated the state of readiness and capacity of this system. However a comparison of these test conditions with expected conditions following a DBA is meaningless since very little correlation between the two sets of conditions exists. For example, the preoperational tests were conducted with zero pressure in the reactor coolant system and during the accident the reactor coolant system pressure is variable and decreases to a low but non-zero pressure at the end of the injection phase.

The PINGP Technical Specifications require periodic testing of the operational sequence of the safety injection system. This testing consists of simulating an accident signal coincident with a loss of offsite power. This testing verifies equipment responds as required to the accident signal(s), load rejection/restoration and voltage restoration circuits of the emergency power system.

The safety injection pumps are allowed to automatically start but are prevented from injecting into the Reactor Coolant System. The Residual Heat Removal pumps, which are initially running, are automatically stopped during the load rejection and automatically restart following reenergization of the safeguards power system by the emergency diesel generators. Most ESF equipment is allowed to operate during this integrated test which confirms the automatic start and sequencing logic operation.

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# 6.3 CONTAINMENT AIR COOLING SYSTEM

## 6.3.1 Design Bases

### 6.3.1.1 Containment Heat Removal Systems

Criterion: Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, shall be provided. (GDC 52)

Adequate heat removal capability for the Containment is provided by two engineered safety features systems. These are the Containment Spray System, whose components are described in Section 6.4 and the Containment Air Cooling System whose components operate as described in Section 6.3.2.

The Containment Air Cooling System is designed to recirculate and cool the containment atmosphere in the event of a loss-of-coolant or main steam line break accident and thereby ensure that the containment pressure cannot exceed its design value of 46 psig at 268°F (100% relative humidity). Although the water in the core after a loss-of-coolant accident is quickly subcooled by the Safety Injection System, the Containment Air Cooling System is designed on the conservative assumption that the core residual heat is released to the containment as steam.

Two of the four containment cooling units and one containment spray pump provide sufficient heat removal capability to maintain the post-accident containment pressure and temperature below the design value, assuming that the core residual heat is released to the containment as steam. Analysis has shown that the operation of one containment spray pump during the injection phase and the heat removal capability equivalent to a single fan coil unit at maximum fouling conditions is sufficient to maintain containment pressure less than design. (See Appendix K for more detail.)

Portions of other systems which become part of this containment cooling system are designed to meet the criteria of this section. Neither a single active component failure in such systems during the injection phase nor an active or passive failure during the recirculation phase will degrade the heat removal capability of containment cooling.

### 6.3.1.2 Inspection of Containment Air Cooling Systems

Criterion: Design provisions shall be made to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as, pumps, valves, spray nozzles, torus, and sumps. (GDC 58)

Design provisions are made to the extent practical to facilitate access for periodic visual inspection of all important components of the Containment Air Cooling System. Flow and pressure measuring devices are provided to augment this and obviate entering containment.

#### 6.3.1.3 Testing of Containment Air-Cooling Systems Components

Criterion: The containment pressure-reducing systems shall be designed so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance. (GDC 59)

The Containment Air Cooling System is designed to the extent practical so that the components can be tested periodically, and after any component maintenance, for operability and functional performance.

#### 6.3.1.4 Testing of Operational Sequence of Containment Air-Cooling Systems

Criterion: A capability shall be provided to test under conditions as close to the design as practical the full operational sequence that would bring the containment pressure-reducing systems into action, including the transfer to alternate power sources. (GDC 61)

Means were provided to test initially to the extent practical the full operational sequence of the Containment Air Cooling System including transfer to the emergency power supply.

#### 6.3.1.5 Additional Design Bases

The Containment Ventilation Systems, Section 5.2, including the Containment Air Cooling System, are designed to remove the normal heat loss from equipment and piping in the reactor containment during normal plant operation. The Containment Air Cooling System, working in parallel with the Containment Spray System, is designed to remove sufficient heat from the reactor containment, following the initial loss-of-coolant accident containment pressure transient, to keep the containment from exceeding the design pressure and temperature.

This equipment is designed to operate at the post-accident conditions in excess of those calculated in USAR Appendix K which addresses containment pressure response to LOCA. The actual test conditions for the fan cooler motor units are given in Reference 5. See Section 8.9 for further details of environmental qualification testing.

All components are capable of withstanding or are protected from differential pressures which may occur during the rapid pressure rise to 46 psig in ten (10) seconds.

Where other systems are required to function as part of the Containment Air Cooling System, such systems are designed to meet the performance objective of this system.

Where portions of these systems are located outside of containment, the following features are incorporated in the design for operation under post-accident conditions:

a. Means for isolation of any section.

b. Means to detect and control radioactivity leakage into the environs, to the limits consistent with guidelines set forth in 10CFR100.

### 6.3.2 System Design and Operation

A schematic arrangement of a Containment Air Cooling and Filtration System is shown in Figure 6.3-1A (Figure 6.3-1B).

Individual system components and their supports meet the requirement for Class I structures and for normal vibration which they may experience.

#### 6.3.2.1 Containment Cooling System Characteristics

The Containment Air System consists of four fan coil units, a duct distribution system, and the associated instrumentation and controls.

The fan coil units are located in a missile protected area near the containment wall.

Each fan coil unit consists of a four-sided arrangement of horizontal tube, vertical fin cooling coils, two-speed motor and a fan. During normal operation the fans may be run at high or low speed and during post accident conditions the fans run at low speed. The fan coil units are designed to operate in the post accident environment.

Ducts distribute the cooled air to the various containment compartments and areas. The normal flow sequence through the system is from containment environment, through the cooling coils, axial flow fan, duct system, and into the containment environment. During post-accident conditions, the flow from the axial flow fans is directed to the upper containment environment by actuation of butterfly valves.

Each fan coil unit is designed to remove a minimum of $50 \times 10^6$ Btu/hr from a saturated air-stream mixture at 268°F, with a duct flow rate of 30,000 CFM. Normal operation may see duct flow rates of 62,000 CFM. Refer to USAR Table 5.2-5.

During non-emergency high heat load conditions the preferred heat sink for the fan coils is the containment and auxiliary building chilled water system. In this configuration, it is normal to expect a flow rate of 450 gpm to each fan coil.

During emergencies, the heat sink for the fan coils is provided by the cooling water system. With only one of five cooling water pumps in operation, the design cooling water flow rate through the cooling coils is 900 gpm to each fan coil unit.

In removing heat at the design basis rate, the discharge is predicted to be two phase flow. Calculations, as discussed in 50.59 Evaluation 1019 and Modification 05ZC02, demonstrate that one train of fan coil units continue to remove heat in excess of the minimum rate assumed in safety analyses. Since condensation of water from the air-steam mixture is the principal mechanism for removal of heat from the post-accident containment atmosphere by the cooling coils, the coil fans will operate as wetted surfaces under these conditions. Entrained water droplets added to the air-steam mixture, such as by operation of the containment spray system, will therefore have essentially no effect on the heat removal capability of the coils.

### 6.3.2.2 Actuation Provisions

During accident conditions, all four fan coil units start during the Safety Injection sequence. The cooling media switches from the non-safety related chilled water system to the safety related cooling water system during this sequence also. Capability also exists for manual actuation from the control room. The air flow path through the fan is the same for normal and post-accident operation. The Safety Injection sequence is described in Section 8.4.

The motor in each fan coil unit is connected to the emergency power bus. Depending on the availability of emergency power either all four, or at least two of the fans will be started after an accident with loss of offsite power. Refer to Section 8.4.1.

Overload protection for all fan motors is provided at the switchgear by overcurrent trip devices. If the fans in each fan coil unit are operating in fast speed, a trip of the overload devices may occur during accident conditions; however, a trip of the fast speed overload devices will not prevent slow speed operation.

All fan motors can be operated manually from the control room.

Temperature elements and motor lights for each fan coil unit indicate air is circulating in accordance with the design arrangement. Temperature indicators and motor lights are provided in the control room.

Since the containment fan coil units are used for normal operation, their performance with respect to temperature reduction, water flow, etc. is continuously monitored and logged.

None of the measurement instrumentation associated with the containment air cooling system is required for accident operation. The instrumentation furnished includes air flow measurement test sections; water flow measurement detectors and panel mounted indicators; and air and water temperature sensing and indication equipment. Low cooling water flow through the fan coil units is alarmed in the control room.

Full opening of the water throttling valve from the main control panel and the orifice bypass valve at the outlet of each train of the fan coil units from it's control panel in the Aux Building permits periodic checks of the water flow capacity for each fan coil unit; operation of the fan coil switches on the main control panel demonstrates capability of the fan coil fans to go to "slow speed" (post accident mode) operation; while observation of the "in" and "out" temperatures for each fan coil permits evaluation of the effectiveness of the fan coil units.

Periodic recalibration of the flow sensing differential pressure transmitters and the temperature sensors assures the accuracy of the monitoring output data.

### 6.3.2.3 Flow Distribution and Flow Characteristics

The duct distribution system is designed to promote good mixing of the containment air and ensures that the recirculated cooled air reaches all areas requiring ventilation. The distribution system is represented schematically by the Containment Air Handling Systems Diagram, Figure 6.3-1A (Figure 6.3-1B).

The system includes branch ducts to the primary compartments for distribution of cooled air from the fan coil discharge. The cooled air is circulated upward from the lower primary compartments, through the steam generator compartments to the operating floor level, and to the containment above the operating floor level. Air that has risen to the containment dome is drawn by the containment dome air recirculation fans through four branch ducts which follow the contour of the containment dome upward on opposite sides of the containment. These ducts take suction at the highest point in the center of the containment.

The temperature of the air returning to the air handling units is essentially the ambient existing in the containment vessel.

During an accident the steam-air mixture entering the fan coil units is a function of the energy release and removal rates inside of containment. At peak conditions, as the mixture passes through the cooling coils, part of the water vapor condenses on the cooling coils, and the mixture leaving the coils is saturated at a reduced temperature. The mixture remains in this condition as it flows into the fan, but picks up some sensible heat from the fan before flowing into the ducting. This sensible heat slightly increases the dry-bulb temperature above that exiting the cooling coils and reduces the relative humidity slightly below 100%.

#### 6.3.2.4 Cooling Water for the Fan Coil Units

The cooling water requirements for all four fan coil units during a loss of primary coolant accident and recovery are supplied by two of the five cooling water pumps. However, only one cooling water pump and two associated fan coil units are required for post accident operation. The Cooling Water System is described in Section 10.4.1.

Each fan coil unit is supplied by a line from the containment cooling water header located outside the containment. (See Section 10.4). Each fan coil unit is provided with isolation valves which allows each fan coil unit to be isolated individually for maintenance.

Following a Safety Injection signal the cooling water outlets from the fan coil units are monitored for radioactivity. The solenoid valves located in the sampling lines to monitors R-16 and R-38 are normally closed. The solenoid valves for the affected unit open automatically on a Safety Injection signal. In accordance with Emergency procedures, the operator manually opens the radiation monitor inlet valves of the affected unit which permit sample streams from the common cooling water return flows of the fan coil units to pass through their respective radiation monitors.

Upon indication of radioactivity in the common effluent from the paired fan coils, the operator through selective valve manipulation, isolates one fan coil at a time to determine which fan coil is experiencing leakage from the containment atmosphere into the cooling water header.

Leakage from the containment atmosphere to the cooling water system is prevented by maintaining the cooling water system at a pressure higher than the pressure inside containment during normal and accident conditions. With only one cooling water pump available to serve both units 1 and 2, the containment pressure exceeds the cooling water pressure by less than 20 psi for the first 30 minutes following a DBA. However, since the cooling coils and cooling water lines are a closed system inside the containment, no contaminated leakage into these systems is expected. Flow, temperature, pressure, and radiation are monitored from the control room.

Flow, pressure, and temperature indications are provided outside containment for cooling water from each cooling unit.

The cooling water relief valves are 3/4″ x 1″, set at 150 psig with an orifice area of .06 square inches. For a pressure difference at 150 psi and 10% over pressure, the flow will be 12.0 gpm. The function of the relief valves is to protect the fan coils from overpressure in the event this section of piping is isolated by upstream and downstream valve closure. The possibility of the cooling water line pressure reaching 150 psig is remote since the system normally operates at a much lower pressure.

### 6.3.2.5 Environmental Protection

All system control and instrumentation devices required for containment accident conditions are located to minimize the danger of control loss due to missile damage.

All fan parts, dampers, cooling coils and fins and ducts in contact with the containment fluid are protected against corrosion. The fan motor enclosures, electrical insulation and bearings are designed for operation during accident conditions.

The distribution header and cooling water cooling piping are also located outside the shield. This arrangement provides missile protection for all components.

The damper closed limit switch for each Dome Damper (CD 34072, CD 34074, CD 34076, CD 34078) [CD 34080, CD 34082, CD 34084, CD 34086] is Environmentally Qualified. This switch maintains fan coil unit discharge to the containment after SI Reset.

### 6.3.2.6 General Description of the Fan Cooler System

The fan units are rated 75 HP high speed and 25 HP slow speed, respectively. During a DBA mode of operation, the fan operates at low speed and uses 22 HP with an ambient density of 0.1712 lb/ft$^3$. A sketch of the fan coil unit is shown in Figure 6.3-2.

Short circuit and overload protection is included for each motor. In the event that a fan coil motor experiences an overload condition during a DBA the worst additional load imposed on the diesel generator would be approximately 115 KVA and would not defeat the operation of the diesel generator even during load sequencing. Further information regarding emergency diesel generator loading and sequencing is addressed in USAR Section 8.4 and its tables.

The fan coil units are designed to remove heat from the containment building during normal operation and in the event of a loss-of-coolant accident. The fan coil unit is an engineered safeguard system. It must function during a loss of coolant accident.

During normal operation, air from the containment enters the unit through the cooling coils and is discharged by a fan into the ventilation ductwork.

During the accident mode operation, the air operated dampers on the normal cooling ducting are automatically closed and the air operated dampers on the ducting discharging to the containment dome atmosphere are automatically opened. Air from the containment still enters the unit through the cooling coils and is discharged by a fan into the ventilation ductwork.

The cooling coils remove heat from the air with the fans providing the required air flow rates. Cooling water is supplied by the Cooling Water System during DBA mode of operation. Drains are provided to remove condensate from the cooling coils, and from the motor enclosure.

Each fan coil unit is composed of the following subassemblies:

a. Fan Assembly
b. Motor
c. Motor Enclosure and Support Assembly
d. Cooling Coil Assembly
e. Enclosure

The above assemblies are combined to form the fan coil unit.

### 6.3.2.6.1 Description of Assemblies

#### 6.3.2.6.1.1 Fan Assembly

The fan coil fan has a fabricated steel wheel, fabricated steel fixed-pitch blades, seismic and strike protection, streamlined inlet bell, outlet cone with access door, and vortex breaker. The fan assembly consists of the axial fan, a coupling with stainless steel discs and thermocouples for each bearing. The fan bearings are greased with a special lubricant to assure proper operation and protection during a possible post-accident condition. The thrust loads are carried by the drive end bearing, with a stepped end shaft, with the bearing being specially designed to carry the thrust loads. The second fan bearing is used to position the shaft. Each bearing has a copper constantin thermocouple with the leads brought to the outside of the fan housing and terminated in a pot-head connection.

The fan outlet cones are provided with access doors for servicing the out-board bearing. The inboard bearing is serviced from inside the enclosure.

#### 6.3.2.6.1.2 Motor Assembly

The fan coil unit motor assembly must function satisfactorily during both normal and post accident operation.

Motor

The four containment fan coil units are of the vaneaxial type.

Each fan is designed for a minimum flow rate of 30,000 cfm in the post accident environment.

The fan motor is a Westinghouse 75/25 HP, 1800/900 RPM, 60 cycle, 3 phase, 460 volt, single winding machine.

The motors are designed so that the motor housing serves as an enclosure which isolates the major functional elements of the motor from the containment environment. The objective is to partially exclude the environment which would exist in the containment building under post-accident conditions.

The vital functional elements of the motor are designed specifically for additional assurance of reliable post-accident performance. Special features are incorporated in the motor bearing system to assure that proper lubrication and freedom from contaminants, such as moisture and caustics, are achieved. The motor insulation is form-wound Thermalastic Epoxy Insulation, which is a sealed, vacuum-pressure impregnated system having superior moisture resistance.

Insulation

The insulation is Class F (NEMA rated total temperature 155°C) Thermalastic Epoxy. The basic MICA structure has high-voltage (2300 volt) insulation. It is impregnated and coated to give a homogeneous insulation system which is highly impervious to moisture. Internal leads and the terminal box-motor interconnection are given special design consideration to assure that the level of insulation matches or exceeds that of the motor.

Bearings

The motors are equipped with high temperature grease lubricated ball bearings as would be required if the bearings were subjected to the DBA ambient temperatures.

Motor Enclosure

The motor enclosure is a unit that previously contained the motor heat exchanger and is provided to control the motor environment.

The motor enclosure assembly consists of two main components: the (1) the duct work, and (2) relief valves in the duct work.

Two relief valves per unit permit accident ambient atmosphere (increasing containment pressure) to enter the motor air system so the bearings are not subjected to differential pressure. The drain is piped to the containment fan coil drain system.

The motor heat exchanger cooling coil was mounted within the motor heat exchanger housing (ductwork). The plenum after the coil has a drain connection for collected condensate. The duct system is connected to the motor through flexible connections and is designed for a 2 psig pressure differential. The duct and housing is painted with a special finish and has two relief valves to relieve pressure differentials in case of a possible post-accident pressure transient.

Each relief check valve body and flapper plate are carbon steel, electrolysis nickel coated. The valve bracket, link, shaft, springs, and fasteners are Type 316 stainless steel. The seat seal is a silicone rubber O-ring. Each valve is subjected to a certification test, before shipment, to assure opening time and leak tightness. When the pressure on the outside of the valve increases to the opening pressure (5 inches of water), the valve begins to open to relieve the motor heat exchanger housing and motor internal volume. The maximum pressure differential across the valve cannot exceed 30 inches of water according to the design and tests.

#### 6.3.2.6.1.3 Cooling Coil Assembly

Each fan coil unit contains four banks of continuous water tube cooling coils for normal and post-accident cooling. Each bank consists of two (2) standard velocity Westinghouse-Sturtevant coils, Aerofin coils, Super Radiator Coils (SRC) coils or equivalent. The coil banks are arranged so that there is one bank on each of the four sides of the enclosure to give the effect of 8 coils in the face. Air and water flow paths are arranged for counter flow. Refer to USAR Table 5.2-5 for cooling system design performance data. 01495214

The Westinghouse coils are fabricated of vertical copper fins on horizontal copper tubes. Coil tubes are mechanically and uniformly expanded into the plate fin collars. The resulting intimate bond between the tubes and fins assures rigid durable construction. Positive mechanical expansion effectively cold works the tubes, increases the strength of copper, and assures uniform expansion over the entire length of all tubes. Tubes pass through heavy end-sheets and are supported to prevent tube sagging.

Aerofin coils are constructed with a removable header and tube sheet. The tubes are rolled at the tube sheet. The fins on the coils are helical, “L” footed type that are tension-wound onto the tubes. Tubes pass through heavy end-sheets and are provided with intermediate supports to prevent tube sagging. The coil tubes are constructed from copper-nickel tube material to increase resistance to erosion-corrosion.

SRC coils are constructed with a waterbox head that may be removed from the tube sheet. The tubes are rolled at the tube sheet. The fins are flat plate corrugated. The tubes pass through an end sheet support and intermediate plate supports to prevent tube sagging. The coil tubes are constructed from copper-nickel tube material to increase resistance to erosion-corrosion. 01495214

#### 6.3.2.6.1.4 Enclosure and Motor-Motor Heat Exchanger Support Assembly

Enclosure

The fan coil unit enclosure is a structural steel assembly used to support the axial fan and connect the cooling coil assemblies. The unit is fabricated from ASTM A-36 steel with the major columns being 6″ 25 lb/ft M sections. The top frame beams are 6′ 15.5 lb/ft wide flanges. The top closure plate is 0.25″ A-36 steel plate stiffened with channels and angles. The enclosure is designed for proper seismic forces.

The coils are provided with drain pans and drain piping to prevent flooding during accident conditions. This condensate is drained to the containment sump.

#### 6.3.2.6.1.5 Fan Cooling System Ducting

Each of the four containment fan coil units discharges to either the containment atmosphere (Dome Damper) for accident recirculation, or to a ducted line for equipment (Gap Damper) cooling. Each of the two discharge paths for a single fan coil unit has an air operated damper which controls the selection of the discharge path. The Dome Damper fails open on a loss of power or air and Gap Damper fails closed. A single failure of loss of power or air causes all of the dampers to go the accident recirculation mode. The control circuit wiring for the dampers is routed as non-safety related cabling. This has been evaluated and it has been concluded that no credible failure can cause all of the solenoid valves for the fan coil unit dampers to remain energized when required during post accident mitigation. No single failure can cause all the dampers for all fan coil units to remain energized when required during an accident.

Dampers located close to the fan coil units close off the Gap distribution ductwork in the post accident condition, and the Dome Dampers open to direct an essentially free flow to the upper containment atmosphere.

The ducts are not needed for post accident operations, but are protected to the extent that they will not become missiles within containment. Canvas and rubber-impregnated cloth are utilized in the ductwork to act as pressure-relieving devices. Where flanged joints are used, joints are provided with gaskets suitable for temperatures to 300°F.

Inadvertent damper closure could result in increased component temperatures (Example: the neutron detectors; in this case, temperature sensors on the neutron detectors sense the temperature rise and if the rise is significant, would be alarmed).

Electrical Supply

Each set of dampers for a single fan coil is controlled from a separate switch on the local control panel to preclude interaction between the four fan coil units.

Details of the normal and emergency power sources are presented in Section 8.

Further information on the components of the Containment Air Cooling System is given in Section 5.

## 6.3.3 Design Evaluation

### 6.3.3.1 Range of Containment Protection

The Containment Air Cooling System provides the design heat removal capacity for the containment following a loss-of-coolant accident assuming that the core residual heat is released to the containment as steam. The system accomplishes this by continuously recirculating the air-steam mixture through cooling coils to transfer heat from containment to cooling water.

Each of the containment fan coil units is capable of transferring heat at the rate of 13,900 Btu/sec (50 x $10^6$ Btu/hr) from the containment atmosphere during post-accident conditions from a saturated air-steam mixture at 268°F with a flow rate of 30,000 CFM. Refer to USAR Table 5.2-5.

A failure analysis has been made on all active components of the system to show that the failure of any single active component will not prevent fulfilling the design function. This analysis has been summarized in Table 6.3-1.

On a safety injection signal, the air operated dampers on the normal cooling ducting are automatically closed and the air operated dampers on the ducting discharging to the containment dome atmosphere are automatically opened.

On switching from the normal to the accident mode, the following are isolated from fan coil air flow:

11 (21) Steam Generator
12 (22) Steam Generator
Excess Letdown Heat Exchanger
Regenerative Heat Exchanger
Reactor Vessel Support
Neutron Detector Well
11 (21) Pressurizer

The Reactor Vessel Support Structure does not require cooling ventilation during post-accident conditions. None of the remaining items listed above are required during post-accident conditions and hence cooling air flow is also not required.

The containment fan coil units and the associated discharge dampers to the containment dome are safeguards Class I and as such meet the requirements of IEEE 279-1968.

The cooling water piping for the fan coil units is designated design Class I and as such has been fully analyzed for maximum thermal, deadweight and OBE and DBE conditions. In addition this piping was taken into consideration in the pipe whip analysis of the high pressure piping in containment and has been duly protected from jet impingement which could occur as a result of a double ended rupture of a high pressure line. Therefore cooling water flow is assured to the fan coils units for OBE and DBE conditions, satisfying the single failure criteria. Furthermore each of the two sets of two fan coil units is supplied from different trains of the cooling water system, thus assuring cooling water flow to at least two fan coil units with any single failure postulated.

#### 6.3.3.2 System Response

The starting sequence of the containment cooling fans and the related emergency power equipment is designed so that delivery of the minimum required air and cooling water flow is reached in a time consistent with post accident requirements. In the analysis of the containment pressure transient, USAR Appendix K, a delay time of 60 seconds was assumed for the initiation of containment cooling.

#### 6.3.3.3 Single Failure Analysis

A failure analysis has been made on all active components of the system to show that the failure of any single active component will not prevent fulfilling the design function. This analysis is summarized in Table 6.3-1.

The analysis of the loss-of-coolant accident presented in Section 14 is consistent with the single failure analysis.

#### 6.3.3.4 Reliance on Interconnected Systems

The Containment Air Cooling System is dependent on the operation of the electrical and cooling water systems. Cooling water to the coils is supplied from the cooling water system. Five cooling water pumps are provided, only one of which is required to operate during the post-accident period. Each emergency diesel generator is capable of supplying the required emergency power to one train of the fan coil unit fans.

#### 6.3.3.5 Shared Function Evaluation

Table 6.3-2 is an evaluation of the main components which have been discussed previously and a brief description of how each component functions during normal operation and during the accident.

### 6.3.3.6 Reliability Evaluation of the Fan Coil Motor

The basic design of the motor as described herein is such that the accident environment is prevented, in any major sense, from entering the motor winding.

It should be noted that the motor insulation hot spot is not expected to exceed normal temperature even under accident conditions.

During the lifetime of the plant, these motors perform part of the normal heat removal service and as such are loaded to approximately 56 HP.

The bearings are designed to perform in the accident ambient temperature conditions. It is expected that bearing temperatures would not exceed 140°C by any significant amount even under accident conditions.

The insulation has high resistance to moisture, and tests performed indicate the insulation system would survive the accident ambient moisture condition without failure. Additionally, the motors are furnished with insulation margin beyond the operating voltage of 460 V.

Following the accident there is only an insignificant mixing of the motor (closed system) environment and the containment ambient environment.

Environmental tests of the motor unit has been described in WCAP-7829, (Reference 5), ‘‘Fan Cooler Motor Unit Test”. Proof testing went beyond any simulation need to meet plant requirements, actually including nine separate accident cycles. To further demonstrate the ruggedness of the motor, windings were directly exposed to containment conditions in three of these cycles.

Absence of damage from these rigorous tests confirms that the motor unit is more than adequate for the intended service.

In the event of a DBE, the immediate transient environment enters the motor enclosure through two small relief valves. As the enclosed volume is relatively small, the valves are only partially open for a period of a few seconds. Once the pressure equalizes, the valves close.

The motor unit has been shown to not be adversely affected by continued operation in the peak DBE external environment including pressure above 75 psig for 24 hours. Maintenance of high pressure implies peak fan loads, therefore sustains high motor temperature and high humidity, a combination which could promote rapid hydrolytic decomposition of insulation. Worst case predicted insulation hot spot temperatures are below the rated temperature for the insulation.

Component and motor unit testing (with and without the heat exchanger), have substantiated the conclusion that the environment assumed in Safety Guide No. 4 will not adversely affect the motor units. The equipment will operate successfully in this environment.

#### 6.3.3.6.1 Fan Coil Insulation Irradiation Testing

The testing program has been completed on the effects of radiation on the WF- 8AC "Thermalastic"[1] Epoxy insulation system used in the reactor containment fan coil motor. Tests description and results are presented in the Westinghouse Proprietary report, WCAP 7343-L, "Topical Report - Reactor Containment Fan Cooler Motor Insulation Irradiation Testing", July, 1969 (Reference 6).

Irradiation of form wound motor coil sections was accomplished up to exposure levels exceeding that calculated for the design basis loss-of-coolant accident. Three coil samples received the following treatment sequence: Irradiation, high-potential test and breakdown voltage test. Nine coil samples received an alternate treatment sequence: Thermal aging, high-potential test, irradiation, high-potential test and breakdown voltage test.

All coil samples passed the high potential tests. The breakdown voltage levels of all coils were well in excess of those required by the design, and clearly indicate that the reactor containment fan cooler motor insulation system will perform satisfactorily following exposure to the radiation levels calculated for the design basis accident.

In response to IE Bulletin 79-01B, an evaluation of the environmental qualification of safety related electrical equipment at the Prairie Island facility was conducted. Results of the evaluation were included in a report submitted per Reference 3. This report also addressed the radiation qualification requirement per NUREG-0737, II.B.2. The post accident radiation environment was re-evaluated for impact due to implementation of heavy bundle fuel and incorporation of power measurement uncertainty (Ref. 53). Results of this analysis are incorporated into the PINGP Equipment Qualification program.

See Section 8.9 for further discussion of environmental qualification testing.

#### 6.3.3.6.2 Fan Coil Unit Motor Lubricant Irradiation Testing

The lubricant originally used in the fan coil unit fan and motor bearings was Westinghouse type No. 773A773G05. Section 6.3.3 of the FSAR summarized the results of tests performed on unirradiated and irradiated samples of this lubricant which showed that there was no significant change in lubricating properties as a result of irradiation at levels anticipated in the containment following a DBA. Chevron SRI lubricant is the qualified equivalent of W style No. 773A773G05 and is currently in use in the fan coil unit fan and motor bearings (Reference 7).

### 6.3.4 Inspection and Testing

#### 6.3.4.1 Inspection Capability

Access is available for visual inspection of the containment air cooling system components including fans, cooling coils, motor and ductwork.

[1] Westinghouse Electric Corporation Trademark

### 6.3.4.2 Testing

Component Testing

One of the containment cooling fans is shop tested for conformance to the AMCA (Air Moving and Conditioning Association) ratings performance criteria using air at standard conditions. Application of conventional fan laws verifies their ability to perform as designed under post-accident conditions.

The Prairie Island fan coil units motors use NEMA #449 frames as compared to the 588.5 frame motor used in reported tests. IEEE guide #334 requirements are met in that the same "essential components (insulation system, winding type, bearings, seals and essential accessory devices and equipment)" are used in test and installed motors. Actual winding horse power is not necessarily a significant criterion.

The features proven by test are common to both machines. These include:

a. The insulation system is a 2400 volt Thermalastic Epoxy system. The same low voltage stress is applicable; and all these stations are electrically tested while submerged in water to detect imperfections (pin-holes). This is the equivalent of a brief steam exposure on every production motor. No other practical way is known to insure that pin holes will not become common mode failures when such motors are exposed to chemically ionized moisture in a design basis event.

b. The Thermalastic Epoxy system is a form winding. There is no significant difference in windings except that of physical strength. The subject Prairie Island winding cross section and extension beyond station support are within Westinghouse experience and qualification tests.

   The two speeds were obtained by reconnecting a single winding. This did entail making more connections and bringing out more leads for external re-connection. However, attachments, leads and entrance seals are all the same as those qualification tested and therefore require no extrapolation.

   The Prairie Island motors are of the same type Westinghouse manufacture as was the test motor described in WCAP 9003 [Ref. 27]. That is, heat exchanger - motor unit which has the same qualified features such as lubricant, seals, check valves, materials, stresses, etc. The Prairie Island motor heat exchangers have been removed. WCAP 7829 [Ref. 5] also tested motors without heat exchangers.

The motor bearing design temperature is 355°F (180°C) with a short time maximum of 400°F as set by the bearing manufacture. The insulation is rated class "F", i.e., 155°C. Designs are based on class "A" average operating temperatures in normal, pre-DBE, service. The bearing lubricant is capable of withstanding post-accident containment environmental conditions without inhibiting motor operation.

The internal vacuum volume relief valves used on the Prairie Island motor units are 4 inches in diameter, two valves per motor unit. The tests performed by Westinghouse employed duplicate type valves as were used on the Prairie Island motor units. No blockage was observed at anytime during those tests. Post inspection of the motor unit internal parts showed no indication of blockage of the air cooling passages.

All motor units tested by Westinghouse have used cooling coils with fin spacing of 8.5 fins per inch of tube length, fin thickness being 0.008 inches. Prairie Island motor units have the cooling coils removed.

The motor unit tested during 1971 used a 588.5 frame having an internal volume of about 12 $ft^3$. The Prairie Island motors are 449 frames with internal volumes of about 4 $ft^3$. As previously mentioned, both motor units used 2 - 4″ diameter valves of duplicate design. Thus the Prairie Island motors would not be subjected to as high pressure differentials as the test motor; and are more conservative in design in this regard. Greater internal volumes would require larger quantities of steam-air flow for pressure equalization. Fewer boron crystal would enter the Prairie Island motors as compared to the test motor as this is a function of volume flow rate.

The environmental testing described in Section 6.3.1.5 is sufficiently conservative for the pressure-temperature calculated for the accident and post accident conditions.

Nine tests were conducted. Seven tests were conducted for the purpose of collecting performance and development data. Two test runs were conducted to:

a. Simulate operation during an accident cycle and observe the performance

b. Expose the motor unit to prolonged post-accident environment and observe its condition

c. “Qualify” the design for operation in a nuclear power station.

Bearing temperatures first rose to the 140°C region toward the end of test number 4 and continued into subsequent tests which showed that the 100° readings were unusual, not the 130-140°C ones.

A review of the tests shows that glass wool was placed around the motor assembly to improve the heat balance calculations relative to B.T.U. removed by the heat exchanger water. This insulation, which was never again used, is assumed to have become water-logged, allowing the steam to dictate bracket temperatures in combination with the cooling by heat-exchanger air. This data and subsequent experience with a similar motor, having no heat exchanger, shows the following conditions to be typical:

| | With Heat Exchanger | Without Heat Exchanger |
|---|---|---|
| Steam Temperature (Sat. at 80 psig) | 162°C | 162°C |
| Air Inside Motor | 62°C | 180°C |
| Bracket | 135°C | 173°C |
| Bearings | 135 ± 5°C | 173 + 2°C |

As one would expect, the brackets of the motor heat-exchanger "see" the cooled air inside and steam on the outside of the motor, thus arriving at a compromise level between these extremes. Bearings typically run within a few degrees of bracket temperatures.

When no heat exchanger is used, windings must rise above ambient steam to expel losses. Thus, the 80 psig test internal air temperature may be 180°C when the steam is 162°C. Brackets, now cooled by the steam, arrive at a temperature of about 173°C and bearings about 175°C. It is noted that post-accident (LOCA and MSLB) containment conditions are well below the 80 psig/162°C test conditions, providing additional margin.

In response to IE Bulletin 79-01B, an evaluation of the environmental qualification of safety related electrical equipment at the Prairie Island facility was conducted. Results of the evaluation were included in a report submitted per Reference 3. This report also addressed the radiation qualification requirement per NUREG-0737, II.B.2. The post accident radiation environment was re-evaluated for impact due to implementation of heavy bundle fuel and incorporation of power measurement uncertainty (Ref. 53). This evaluation is maintained as part of the PINGP Equipment Environmental Qualification Program. This program includes the evaluation of motor operation without the heat exchanger.

See Section 8.9 for further discussion of environmental qualification testing.

The original Westinghouse coil heat transfer performance was calculated by computer using the HECO code described in WCAP-7336-L [Ref. 28]. This WCAP is a proprietary topical report that was submitted to the AEC in early 1969. A prototype test was conducted to confirm the HECO code. The prototype test was also described in WCAP-7336-L. Replacement heat exchanger coils were supplied by Aerofin. The Aerofin heat exchanger performance computer programs were similarly confirmed with a test program in 1991. A prototype unit was tested under simulated accident

conditions. The construction and design of the test unit compares favorably to the supplied unit. The essential components of the cooling coils are the number of rows, passes, tube size, thickness, fin size, thickness, spacing and materials. The differences in the number of passes and tube thickness are not critical values. These parameters are accounted for in standard calculations for film thickness given the number of tube rows or tube size. The minor difference in fin spacing is within American Refrigeration Institute test criteria and computer program limits. Alternate tube material, such as copper-nickel tubes in lieu of copper tubes, is modeled by the use of an outside fouling factor to account for the tube material thermal conductivity. The extension of the test to full size units is similar to the Westinghouse report, and merely an increase in component size and total flow quantities, but not a change in controlling parameters. A procurement audit verified software control. The Unit 2 Aerofin cooling coils were replaced by Super Radiator Coils (SRC) cooling coils. The SRC cooling coil performance during accident conditions was confirmed by testing at NTS Huntsville and 01495214
by SRC computer codes to demonstrate the SRC cooling coils are equivalent to the Aerofin coils as documented in Equivalency Evaluation EC-25792 [Ref. 75].

System Testing

Each fan cooling unit was tested after installation for proper flow and distribution through the duct distribution system. All four of the fan cooling units are used during normal operation.

Operational Sequence Testing

Periodic tests can be conducted to demonstrate proper sequencing of the accident fan motor supplies to the diesel generators in the event of loss of offsite power. These tests can be conducted at the time the diesel generators are tested.

### 6.3.5 System Evaluation

The containment pressure suppression systems (i.e. containment fan coils and containment spray) are designed to provide sufficient heat removal capability to maintain the post accident containment pressure and temperature below the design limit with two of the four containment cooling fan coils and one of the two containment spray pumps running. Since two of the four motor operated throttle valves on the fan coil units are powered from Train A 480 volt safeguards power and two from Train B no single failure can prevent all four throttle valves from opening. Also, since the throttle valves are normally partially open, a failure to open still provides partial cooling on the failed fan coil unit. The effect of a single throttle valve failure to open is significantly less than a loss of a train of safeguards electric power which is an acceptable failure. Analysis has shown that the operation of one containment spray pump during the injection phase and the heat removal capability equivalent to a single fan coil unit at maximum fouling conditions is sufficient to maintain containment pressure less than design. (See Appendix K for more detail.)

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# 6.4 CONTAINMENT VESSEL INTERNAL SPRAY SYSTEM

## 6.4.1 Design Bases

The basis for containment vessel internal spray system design is discussed in Section 6.3.

The primary purpose of the Containment Vessel Internal Spray System is to spray cool water into the containment atmosphere, when appropriate, in the event of a loss-of-coolant accident and thereby ensure that containment pressure does not exceed its design value of 46 psig at 268 degrees F. (100% R.H.). This protection is afforded for all pipe break sizes up to and including the hypothetical instantaneous circumferential rupture of a reactor coolant pipe. Pressure transients for loss-of-coolant accidents are presented in USAR Appendix K.

One containment spray pump and two of the four containment fan coil units provide sufficient heat removal capability to maintain the post-accident containment pressure below the design value, assuming that the core residual heat is released to the containment as steam. Analysis has shown that the operation of one containment spray pump during the injection phase and the heat removal capability equivalent to a single fan coil unit at maximum fouling conditions is sufficient to maintain containment pressure less than design (See Appendix K for more detail).

### 6.4.1.1 Inspection of Containment Vessel Internal Spray Systems

Criterion: Design provisions shall be made to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as, pumps, valves, spray nozzles, torus, and sumps. (GDC 58)

Where practicable, all active components and passive components of the Containment Vessel Internal Spray Systems are inspected periodically to assure system readiness. The pressure containing systems are inspected for leaks from pump seals, valve packing, flanged joints and safety valves. During operational testing of the containment spray pumps, the portions of the systems subjected to pump pressure are inspected for leaks. Design provisions for inspection of the Safety Injection System, which also functions as part of the Containment Vessel Internal Spray System, are described in Section 6.2.4.3.

### 6.4.1.2 Testing of Containment Vessel Internal Spray Systems Components

Criterion: The containment pressure-reducing systems shall be designed so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance. (GDC 59)

All active components in the Containment Vessel Internal Spray systems were adequately tested both in pre-operational performance tests in the manufacturer's shop and in-place testing after installation. Thereafter, periodic tests are performed after any component maintenance. Testing of the components of the Safety Injection System used for containment spray purposes is described in Section 6.2.4.

The component cooling water pumps and the cooling water pumps which supply the cooling water to the residual heat exchangers are in operation on a relatively continuous schedule during plant operation. All pumps may be tested during normal operation.

#### 6.4.1.3 Testing of Containment Vessel Internal Spray Systems

Criterion: A capability shall be provided to test periodically the delivery capability of the containment spray system at a position as close to the spray nozzles as is practical. (GDC 60)

Permanent test lines for all the containment spray loops are located so that all components up to the spray header manual isolation valves are checked.

Containment spray nozzles are verified to be free of obstructions following any maintenance activity that could result in nozzle blockage (by introducing foreign material into the system). Spray nozzle testing is not required following maintenance activities that successfully comply with the foreign material exclusion program.

#### 6.4.1.4 Testing of Operational Sequence of Containment Vessel Internal Spray Systems

Criterion: A capability shall be provided to test under conditions as close to the design as practical the full operational sequence that would bring the containment pressure-reducing systems into action, including the transfer to alternate power sources. (GDC 61)

Capability is provided to test the operational startup sequence beginning with transfer to alternate power sources and ending with near design conditions for the Containment Vessel Internal Spray System, including the transfer to the alternate diesel-generator power source.

#### 6.4.1.5 Performance Objectives

The Containment Vessel Internal Spray System is designed to spray 2,400 gpm of borated water into the containment building whenever the coincidence of three sets of one out of two (Hi Hi) containment pressure signals occurs, or a manual signal is given. Either of two subsystems containing a pump and associated valving and spray headers are independently capable of delivering one-half of this flow, or 1200 gpm.

The design basis is to provide sufficient heat removal capability to maintain the post-accident containment design pressure. This requires a heat removal capacity of the subsystem, with either pump operating, at least equivalent to two fan-coil units heat removal capability at the containment design conditions.

The spray system is designed to operate during the injection phase, following a primary coolant system failure as required to restore and maintain containment conditions at near atmospheric pressure. It has the capability of reducing the containment post-accident pressure and consequent containment leakage taking into account any reduction due to single failures of active components.

Portions of other systems which share functions and become part of the containment cooling system when required are designed to meet the criteria of this section. Any single failure of active components in such systems does not degrade the heat removal capability of containment cooling.

Those portions of the spray systems located outside of the containment which had been designed to circulate radioactively contaminated water collected in the containment, under post-accident conditions, meet the following requirements:

a. Shielding to maintain radiation levels as stated in Section 7.5.

b. Collection of discharges from pressure relieving devices into closed systems.

c. Means to limit radioactivity leakage to the environs, consistent with guidelines set forth in 10CFR100.

System active components are redundant. System piping located within the containment is redundant and separable in arrangement unless fully protected from damage which may follow any primary coolant system failure.

System isolation valves relied upon to operate for containment cooling are redundant, with automatic actuation or manual actuation. After the recirculation phase has been initiated as described in Section 6.2.2 the Containment Spray system is shutdown.

### 6.4.1.6 Service Life

All portions of the system located within containment are designed to withstand, without loss of functional performance, the post-accident containment environment and operate without benefit of maintenance for the duration of time needed to reduce containment pressure during the injection phase.

### 6.4.1.7 Codes and Classifications

Table 6.4-1 tabulates the codes and standards to which the containment vessel internal spray system components are designed.

## 6.4.2 System Design and Operation

### 6.4.2.1 System Description

Adequate containment cooling is provided by the Containment Vessel Internal Spray System shown in Figure 6.2-5 whose components operate to spray a portion of the contents of the refueling water storage tank into the entire containment atmosphere using the containment spray pumps.

The principal components of the Containment Vessel Internal Spray System which provide containment cooling following a loss of coolant accident consist of two pumps, spray ring headers and nozzles, and the necessary piping and valves. The containment spray pumps are located in the auxiliary building and take suction directly from the refueling water storage tank.

The 360° ring headers are located as follows:

| | |
|---|---|
| Ring No. 1 | EL. 867′-5 11/16″ |
| Ring No. 2 | EL. 868′-4 5/16″ |
| Ring No. 3 | EL. 882′-8 1/16″ |
| Ring No. 4 | EL. 883′-6 11/16″ |

The spray is actuated by the coincidence of three sets of one out of two high- high containment pressure signals. This starting signal starts pump, opens discharge motor-operated valves and caustic addition-containment spray pump control valves. The pump start opens the RWST-containment spray pump motor-operated valves and blocks the discharge motor-operated valves open. The pump breaker is inhibited from closing until the safeguards actuation sequencer permits closing. This assures proper loading of the diesel generators. It should be noted that there is an inherent time delay after the actuation signal which is due to the loading sequence.

After being started, a similar inhibit can occur if offsite power is lost, however, the pump can be reloaded on its diesel generator when the loading sequence permits.

Once in operation a containment spray pump may be stopped (if the actuation signal is not present) by resetting at the main control board. Further, at any time, the operator may stop the containment spray pump at the control panel switch furnished, by moving the switch handle to the stop position. If the operator does not ‘‘pull-out” the manual switch in this position, the pump will restart automatically (if the switch spring returns) unless the operator has previously pressed the reset function.

If required, the operator can manually actuate the system from the control room, and periodically, the operator can actuate system components to demonstrate operability.

The system design conditions were selected to be compatible with the design conditions for the low pressure injection system since both of these systems share the same suction source.

After the injection operation spray flow is discontinued while maintaining containment pressure reduction with the containment fan coil units, and returning all of the recirculated water to the core. In this mode, the bulk of the core residual heat is transferred directly to the sump by the spilled coolant to be eventually dissipated through the residual heat exchanger once the sump water becomes heated. The heat removal capability equivalent to a single fan coil unit is sufficient to remove the corresponding energy addition to the vapor space resulting from steam boil off from the core assuming flow into the core from one residual heat removal pump at the completion of injection without exceeding containment design pressure, hence it is not expected that continued spray operation would be required for containment cooling.

Remote operated valves of the Containment Spray System, which are under manual control (that is, valves that normally are in their ready position and do not receive a containment spray signal) have position indicators on a common portion of the control board. The position indicator lights function only when the respective motor valve's breaker is energized. Normally during operation, the RWST supply motor valve to Containment Spray is open with its power turned off, and the RHR supply motor valve to Containment Spray is closed with its power turned off.

#### 6.4.2.1.1 Chemical Addition

As discussed later in this section under pH control, direct additive (NaOH) to the spray system will be done automatically whenever spray is actuated. Means are provided to remotely sample and monitor the recirculated containment sump liquid in the chem lab during the post-accident period, and additives could be supplied to the sump water if necessary by manually initiating the containment spray system.

### 6.4.2.2 Components

All associated components, piping, structures, and power supplies of the Containment Vessel Internal Spray System are designed to Class I criteria.

The Containment Vessel Internal Spray System shares the refueling water storage tank liquid capacity with the Safety Injection System. Refer to Section 6.2.2 for a detailed description of this tank.

#### 6.4.2.2.1 Pumps

The two containment spray pumps are of the horizontal centrifugal type driven by electric motors. These motors can be powered from both normal and emergency power sources. The containment spray flow rate is expected to be 1300 gpm per pump as shown in Figure 6.4-1.

The design head of the pumps is sufficient to continue at rated capacity with minimum level in the refueling water storage tank against a head equivalent to the sum of the design pressure of the containment, the head to the upper-most nozzles, and the line and the nozzle pressure losses. The materials of construction are stainless steel or equivalent corrosion resistant material. Design parameters are presented in Table 6.4-2.

The pump motors are direct-coupled and non-overloading to the end of the pump curve.

Details of the component cooling pumps and cooling water pumps, which serve the Safety Injection System, are presented in Section 10.

#### 6.4.2.2.2 Spray Nozzles

The spray nozzles, of the ramp bottom design, are not subject to clogging by particles less than 1/4 inch in maximum dimension, and are capable of producing a mean drop size of approximately 700 microns in diameter with the spray pump operating at design conditions and the containment at design pressure.

The spray nozzles are stainless steel and have a 3/8 inch diameter orifice. The nozzles are connected to four 360 degrees ring headers with two ring headers per train. There are 84 Spraco Model 1713 nozzles per train.

The nozzles and headers are so oriented as to ensure adequate coverage of the containment volume.

When a spray drop enters the hot saturated steam-air environment, the vapor pressure of the water at its surface is much less than the partial pressure of the steam in the atmosphere. Hence, there will be diffusion of steam to the drop surface and condensation on the drop. This mass flow will carry energy to the drop. Simultaneously the temperature difference between the atmosphere and the drop will cause a heat flow to the drop. Both of these mechanisms will cause the drop temperature and vapor pressure to rise. The vapor pressure of the drop will eventually become equal to the partial pressure of the steam and the condensation will cease. The temperature of the drop will be essentially equal to the temperature of the steam-air mixture.

Analysis performed shows that containment spray drop temperature rises to the containment steam-air mixture temperature in less than 0.5 seconds, which will occur before the drop has fallen 5 feet, hence thermal shock as a result of cold containment spray on the hot (268°F) containment wall will not occur.

The spray nozzles are oriented such that there is no direct impingement of spray water on the Reactor Coolant piping.

#### 6.4.2.2.3 Valves

The valves for the Containment Vessel Internal Spray System are designed in accordance with specifications discussed for the valves in the Safety Injection System.

Valving descriptions and valve details are shown in Section 5.2 and 6.2.2.

The pressure containing parts of safety related manual valves installed for modification 03CS02 are designed per criteria established by the USAS B16.34 specifications. These valves were procured to Section 3.2.1.5 of the Engineering Manual, Specification for Valves.

#### 6.4.2.2.4 Piping

The piping for the Containment Vessel Internal Spray System is designed in accordance with specifications discussed for the piping in the Safety Injection system (Section 6.2).

The system is designed for 200 psig and 300 degrees F. on the suction side and for 500 psig and 300 degrees F. on the discharge side up to the nozzles in the containment.

The division between the suction side design and the discharge design on the full flow recirculation loop added per modification 03CS02 is at the isolation valve closest to the pump suction piping.

#### 6.4.2.2.5 Motors for Pumps and Valves

The motors for the Containment Vessel Internal Spray System are designed in accordance with specifications discussed for motors in the Safety Injection System. (Section 6.2)

### 6.4.2.3 Electrical Supply

Details of the normal and emergency power sources are presented in the discussion of the Electrical System, Section 8.

### 6.4.2.4 Environmental Protection

The four spray headers are located at the upper portions of the containment vessel outside and above the reactor and steam generator concrete shield for missile protection. During operation movable shielding plugs provide missile protection for the area immediately above the reactor vessel. The spray headers are therefore also protected from missiles originating within this shield.

All of the active components of the containment vessel internal spray system are located outside the containment, and hence are not required to operate in the steam-air environment produced by the accident. See Section 8.9 for further discussion of environmental qualification testing.

### 6.4.2.5 Material Compatibility

Parts of the system in contact with borated water are stainless steel or an equivalent corrosion resistant material.

## 6.4.3 Design Evaluation

### 6.4.3.1 Range of Containment Protection

During the injection phase following the maximum loss-of-coolant accidents (i.e., during the time that the containment spray pumps take their suction from the refueling water storage tank) this system, in conjunction with the containment fan coil units, provides the design heat removal capacity for the containment. After the injection phase, each train of the recirculation system provides sufficient cooled recirculated water to keep the core flooded. This applies for all reactor coolant pipe ruptures up to and including the hypothetical instantaneous circumferential rupture of a reactor coolant pipe. Only one low head safety injection pumping train and one heat exchanger are required to operate for this capability at the earliest time recirculation is initiated.

One containment spray pump and two of the four fan coil units will provide sufficient heat removal capability to maintain the post-accident containment pressure below the design value as shown in USAR Appendix K, Figures K-2, K-5 and K-9.

During the injection, spray water is raised to the temperature of the containment in falling through the steam-air mixture. The minimum fall path of the droplets is approximately 70 ft. from the lowest spray ring headers to the top of the steam generators and approximately 112 ft. to the operating deck. The actual fall path is longer due to the trajectory of the droplets sprayed out from the ring header. Heat transfer calculations, based upon 1000 micron droplets, show that thermal equilibrium is reached in a distance of approximately five feet. Thus, the spray water reaches essentially the saturation temperature.

At containment design pressure, 46 psig, 1200 gpm of boric acid solution is injected into the containment atmosphere by one spray pump. At containment air temperature of 268 degrees F the total heat absorption capability of one spray pump is about $110 \times 10^6$ Btu/hr based on addition of 100 degrees F refueling water. The heat removal by the spray system credited in the containment analysis is determined in the model as a function of spray flow rate and liquid temperature.

In addition to heat removal, the spray system is also effective in scrubbing fission products from the containment atmosphere.

### 6.4.3.2 pH Control

To prevent the potential possibility of stress corrosion cracking of stainless steel RHR components during the recirculation phase, a caustic solution (NaOH) is added to the containment spray water. Upon automatic actuation of the Containment Spray system, a 9 wt.% to 11 wt.% NaOH solution is supplied to the pump suctions. The added NaOH assures a pH of less than 10.5 in the spray and greater than 7.0 in the recirculated water. The portions of the Caustic Addition system which are required to perform this function are Class 1.

A schematic of the arrangement is shown in Figure 6.4-2, and consists of a standpipe, vacuum breakers, a recirculating pump, a surge tank, a feed tank, and the necessary piping and valves. The 2590 gallons of 9 wt.% to 11 wt.% NaOH solution is contained in a standpipe located alongside the RWST which has the same equivalent height. Upon receipt of a containment spray signal, air operated valves open and allow the NaOH solution to mix with water being taken from the RWST. Redundant valve trains, and level indicators are provided. A level readout is located in the control room so that flow can be verified.

The gravity feed arrangement is used for introduction of the solutions because the caustic tank and RWST liquid levels remain identical, assuring the caustic flow is fed continuously in a fixed proportion to the RWST flow.

The potential for stress corrosion cracking of stainless steel by boric acid solution was evaluated in Westinghouse WCAP-7798-L (Reference 33). That report reviewed experiments in which stressed and unstressed as well as sensitized and unsensitized stainless steel samples were exposed to conditions conservatively simulating those anticipated in the emergency core cooling system following the hypothetical design basis accident. Tests were conducted in boric acid solutions with varying pH levels and chloride concentrations. The results of the tests showed that pH adjustment of the solution was found to protect all stainless steel, both sensitized and non-sensitized, from stress corrosion at chloride concentrations much greater than expected for periods as long as one year. Based on WCAP-7798-L, Branch Technical Position MTEB 6-1 (Reference 34) recommends a minimum of pH level of 7.0 for post accident emergency coolant water to reduce the probability of stress corrosion cracking of austenitic stainless steel.

### 6.4.3.3 System Response

The starting sequence of the containment spray pumps and their related emergency power equipment is designed so that delivery of the minimum required flow is reached within 60 seconds (see Section 8). 01313069

### 6.4.3.4 Single Failure Analysis

A failure analysis has been made on all active components of the system to show that the failure of any single active component will not prevent fulfilling the design function. This analysis is summarized in Table 6.4-3.

### 6.4.3.5 Reliance on Interconnected Systems

The Containment Vessel Internal Spray System initially operates independently of other engineered safety features taking suction directly from the RWST and caustic addition standpipe following a loss-of-coolant accident. Spray pump cooling is supplied from the component cooling loop.

### 6.4.3.6 Shared Function Evaluation

Table 6.4-4 is an evaluation of the main components which have been discussed previously and a brief description of how each component functions during normal operation and during the accident.

### 6.4.3.7 Containment Spray Pump NPSH Requirements

The NPSH for the containment spray pumps is evaluated for injection operation. The end of the injection phase gives the limiting NPSH requirement. The NPSH available is determined from the elevation head and vapor pressure of the water in the RWST and the pressure drop in the piping to the pump.

## 6.4.4 Inspection and Testing

### 6.4.4.1 Inspection Capability

All components of the Containment Vessel Internal Spray System can be inspected periodically to demonstrate system readiness.

The pressure containing systems are inspected for leaks from pumps seals, valves packing, flanged joints and safety valves during system test. During the operational testing of the containment spray pumps, the portions of the system subjected to pump pressure are inspected for leaks.

### 6.4.4.2 Component Testing

All active components in the Containment Spray System were tested both in pre-operational performance tests in the manufacturer's shop and in-place testing after installation.

The containment spray pumps can be tested singularly by opening valves in the full flow test line and opening the pump discharge MOV. Each pump in turn can be started by operator action and checked for flow establishment. The spray injection valves can be tested with the pumps shut down.

Initially the containment spray nozzle availability is tested by blowing smoke or a gas mixture through the nozzles and observing the flow through the various nozzles in the containment visually or by tell tales.

During these tests the equipment is visually inspected for leaks. Leaking seals, packing, or flanges are tightened to eliminate any leak. Valves and pumps are operated and inspected after any maintenance to ensure proper operation.

### 6.4.4.3 System Testing

Permanent test lines for all containment spray loops are located so that the system, up to the manual isolation valves at the spray header, can be tested.

A periodic test is performed to verify that spray nozzles are not obstructed.

### 6.4.4.4 Operational Sequence Testing

The functional test of the Safety Injection System described in Section 6.2.4.4 demonstrates proper transfer to the diesel generator power source in the event of loss of power. A test signal simulating the containment spray signal will be used to demonstrate operation of the spray system up to the isolation valves on the pump discharge.

Actuation signals for the containment spray system are generated by the same pressure transmitters used for safety injection, and the testing of this instrumentation is the same (with the exception of set points and actual trip units) as for the safety injection system.

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# 6.5 LEAKAGE DETECTION SYSTEMS

## 6.5.1 Introduction

General Design Criterion 30 of Appendix A to 10CFR50, "Quality of Reactor Coolant Pressure Boundary", states: "Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage." This detection of abnormal RCS leakage is accomplished by a variety of methods. NRC report entitled "Coolant Leakage Detection System Performance at the Prairie Island Nuclear Generating Plant", dated March 31, 1976 (Reference 22), summarized the effectiveness of these leakage detection methods over the first two years of operation of Unit 1. Section 3.6 of this report described the performance of the Containment Radioactive Particulate and Gas Monitors and the method by which they were evaluated. ENG-ME-792 (Reference 65) presently supersedes the detector response information for 1R-11 (2R-11) contained in that report, which was based on actual 01450457 measured coolant activity levels from December 1975 and which may not be conservative relative to current plant operating coolant levels.

Reactor coolant leakage can be broadly classified as controlled leakage or uncontrolled leakage. Controlled leakage consists of leakoff from No. 3 reactor coolant pump seal. This leakage is normally about 6 gallons/hour per pump and it is piped to the reactor 01450457 coolant drain tank in containment. All other coolant leakage is by definition uncontrolled leakage.

Uncontrolled coolant leakage can be further categorized as identified leakage or as unidentified leakage. Identified leakage consists of:

a. Leakage that is captured and conducted to a sump or collecting tank.

b. Leakage into the containment atmosphere from sources that are specifically located and known not to be from non-isolable perforations in the reactor coolant system boundary.

c. Reactor coolant leakage through the steam generator into the secondary system.

Category (a) includes a number of components designed to operate with some small amount of coolant leakage. All such components located within containment are equipped with leakoff connections that conduct this leakage to the reactor coolant drain tank or the pressurizer relief tank. Components designed to operate with some small amount of coolant leakage that are located outside containment are similarly provided with means for conducting leakage to the drain systems. This leakage is normal and amounts to less than 1 gallon/minute under ordinary circumstances from all sources.

Category (b) includes leakage which is not collected and which can be traced to components not a part of the reactor coolant pressure boundary. This leakage normally approaches zero within containment.

Category (c) leakage results from through-wall cracks in one or more steam generator tubes. Since primary-to-secondary leakage is a sensitive indicator of the general condition of steam generator tubes, special limits apply to this category of identified leakage.

All reactor coolant leakage which is not included in categories (a), (b) or (c) is termed unidentified leakage. Unidentified leakage can originate from components located either inside or outside containment. Unidentified leakage from components located outside containment can always be isolated and is a concern only from the standpoint of control of radioactive materials. Instruments and techniques used to detect unidentified leakage outside containment are designed to detect significant levels of leakage and initiate isolation of the affected components.

Unidentified leakage inside containment may be the result of cracks in the reactor coolant pressure boundary. Relatively sophisticated, sensitive, and rapid leakage detection techniques can permit identification and repair of these cracks before they can propagate and result in component failure. For this reason, most of the emphasis placed on leakage detection is placed on instruments and techniques to detect unidentified leakage inside containment.

A formal program has been established to increase assurances that the Reactor Coolant System integrity is not degraded by boric acid corrosion caused by the leakage of fluids containing boric acid. This program has been described in NSP's response to Generic Letter 88-05 (Reference 14).

### 6.5.2 Design Basis

The Prairie Island Technical Specifications establish the maximum permitted reactor coolant leakage rates for uncontrolled sources. In order to determine the leakage rate from uncontrolled sources, the leakage from the reactor coolant pump No. 3 seal (controlled leakage) may be subtracted from the as-found leak rate. Controlled leakage subtracted in this way is limited to the maximum permitted No. 3 reactor coolant pump seal leakage specified by the pump vendor (References 73 and 74). 01450457 Leakage from No. 3 seal in excess of the maximum specified by the pump vendor must be considered "uncontrolled" leakage. The Technical Specification leakage rate limits established for each unit are:

| | |
|---|---|
| Unidentified Leakage | 1 gallon/minute |
| Identified Leakage | 10 gallons/minute |
| Steam Generator Tube Leakage | 150 gallons/day |

If any leakage limit is reached, the unit is brought to Mode 5, Cold Shutdown for identification and correction of the excessive leakage.

The limit on unidentified leakage of 1 gallon/minute is based on the following considerations:

a. It is well above the minimum detection capability of the reactor coolant leakage detection methods inside containment.

b. It is well below the leakage rates calculated for critical through wall cracks in pipes 3 inches in diameter and larger (Reference 9). A break in lines smaller than 3 inches in diameter does not lead to failure of fuel cladding.

The limit on identified leakage of 10 gallons/minute is based on a rate which will not mask a smaller unidentified reactor coolant leak rate from the coolant leakage detection system.

The total reactor coolant system to secondary coolant system leakage limit through any one steam generator of a unit is 150 gallons per day (GPD) and is based upon voltage-based repair criteria per the guidance of NRC Generic Letter 95-05. (Reference 23). This leakage limit of 150 GPD is more restrictive than the historical leak rate limit of 1 gpm corresponding to a through wall crack less than 0.6 inches long based on test data. (Reference 10). The more restrictive 150 GPD limit is intended to provide an additional margin to accommodate a crack which might grow at a greater than expected rate or unexpectedly extend outside the thickness of the support plates. (Reference 24).

Application of Leak-Before-Break (LBB) methodology to reactor coolant system (RCS) branch piping (Reference 65) required detection of RCS leakage of 0.2 gpm. The RCS leakage detection systems described in Reference 22 include multiple methods for detecting small RCS leakage. There are only two methods that use instrumentation whose reliability is maintained in accordance with the PINGP Technical Specifications and that have sufficient sensitivity to detect 0.2 gpm leaks, notably the Containment Radiological Particulate Monitors (1R-11 and 2R-11), and the Containment Sump A pump run time indicators, as described below (Reference 66).

### 6.5.3 Systems Design and Operation

Various methods are used to detect leakage from either the primary system or the auxiliary system.

If an accident involving gross leakage from the Reactor Coolant System occurred it could be detected by the following methods.

a. Pump Activity

   During normal operation one charging pump is operating with its speed control in automatic. If a gross loss of reactor coolant to another closed system occurred which was not detected by other methods, charging pump speed would indicate the leakage.

The leakage from the reactor coolant system will cause a decrease in the pressurizer liquid level. The decrease would be within the sensitivity range of the pressurizer level indicator. The speed of the charging pump in automatic will increase to try to maintain the equivalence between the letdown flow and the combined charging line flow and flow across the reactor coolant pump seals. If the pump reaches a high speed limit, an alarm is actuated.

A break in the primary system would result in reactor coolant flowing into the containment sump. Gross leakage to this sump would be indicated by a rise in sump level or the frequency of operation of the containment sump pumps. High level in this sump will actuate an alarm.

b. Instrument Thimble Leak Detection

The Leak Detection System consists of a drain header connecting the 10-path transfers, a pressure switch and drainage solenoid valve installed in the drain header, and an alarm and reset push-button mounted on the distribution panel in the Control Console. Liquid collecting in a 10-path transfer due to a leak will cause the water level to rise in the drain header and thus actuate the pressure switch. A contact from the pressure switch will energize the leak alarm and drainage solenoid, dumping the water to the plant drain. This and an additional description of the Incore Instrumentation System is discussed in WCAP-7607 (Reference 11).

c. Liquid Inventory

Gross leaks might be detected by unscheduled increases in the amount of reactor coolant makeup water which is required to maintain the normal level in the pressurizer. This is inherently a low precision measurement, since makeup water is necessary as well for leaks from systems outside the containment.

A high level alarm for the component cooling water surge tank and high radiation and temperature alarms actuated by monitors in the component cooling headers could also indicate a thermal barrier cooling coil rupture in a reactor coolant pump. However, in addition to these alarms, high temperature and high flow on the component cooling outlet line from the reactor coolant pump would activate alarms.

### 6.5.3.1 Reactor Coolant System

During Mode 5, Cold Shutdown or Mode 6, Refueling, personnel can enter the containment and make a visual inspection for leaks. The location of any leak in the Reactor Coolant System would be determined by the presence of boric acid crystals near the leak. The leaking fluid transfers the boric acid outside the Reactor Coolant System and the process of evaporation deposits crystals.

During normal operation and anticipated reactor transients the following methods are employed to detect leakage from the Reactor Coolant System.

#### 6.5.3.1.1 Containment Air Particulate Monitor

Refer to Section 7.5.2.2. This system takes continuous air samples from the containment atmosphere and measures the air particulate beta radioactivity. The samples, drawn from outside the containment, are in a closed, sealed system and are monitored by a scintillation counter - filter paper detector assembly.

The containment air particulate monitor is the most sensitive instrument of those available for detection of reactor coolant leakage into the containment.

The containment air particulate monitor (1R-11 and 2R-11) is relied upon for detection of small reactor coolant system leaks in the Leak-Before-Break analyses (Reference 65). During steady state plant operations, the R-11 monitor can detect leaks as small as 0.2 gpm and is also capable of detecting a 1 gpm leak within one hour, consistent with Regulatory Guide 1.45.

The raw count rate data from the containment air particulate monitor fluctuates over short time intervals so datapoints to determine the rate of change of 10 minute average R-11 outputs have been established in the PINGP Emergency Computer Response System (ERCS). An ERCS datapoint to indicate the R-11 count rate increase corresponding to a potential 1 gpm leak calculates the rate of change over 1 hour. An ERCS datapoint to indicate the response to a potential 0.2 gpm leak calculates the rate of change over 4 hours.

To alert plant operators to indications of a 1 gpm leak within one hour, two alarms are provided on the ERCS. One alarm indicates a rate of change corresponding to a 1 gpm leak. In addition, to alert operators to a slowly growing leak that reaches 1 gpm but where the increase in R-11 count rates was too gradual to trip the rate of change alarm, an alarm is also provided based on a high average count rate setpoint. The high count rate ERCS alarm will not be activated for approximately 4 hours after a 1 gpm leak occurs. This high average count rate ERCS alarm is not associated with any high count rate alarms provided on the containment air particulate monitor itself. Operator response actions were included in plant procedures so that operators will check other indicators of RCS leakage as described later in this section. If a potential leak of 0.2 gpm or greater is confirmed by at least one other indicator then investigative actions will be undertaken, up to and including potential containment entry.

The R-11 monitors are capable of detecting increases in containment radiation levels corresponding to leaks that appear over a relatively short time period. Leaks that start much smaller and grow slowly will be more readily detected by changes in the Containment Sump A pump run time indicators as discussed below.

### 6.5.3.1.2 Containment Radiogas Monitor

Refer to Section 7.5.2.3. This system measures the gaseous radioactivity in the containment by taking continuous air samples from the containment atmosphere, after they pass through the air particulate monitors, and drawing the samples through a closed, sealed system to a gas monitor assembly.

The containment radioactive gas monitor is inherently less sensitive than the containment air particulate monitor, and would function in the event that significant reactor coolant gaseous activity exists from fuel cladding defects.

### 6.5.3.1.3 Humidity Detector

The humidity detection instrumentation offers another means of detection of leakage into the containment. Although this instrumentation has not nearly the sensitivity of the air particulate monitor, it has the characteristics of being sensitive to vapor originating from all sources within the containment, including the reactor coolant and steam and feedwater systems.

The sensitivity of this method depends on cooling water temperature, containment air temperature variation and containment air recirculation rate.

### 6.5.3.1.4 Condensate Measuring System

This leak detection method is based on the principle that the condensate collected by the cooling coils matches, under equilibrium conditions, the leakage of water and steam from systems within the containment. This principle applies because conditions within the containment promote complete evaporation of leaking water from hot systems. The air and internal structure temperatures are normally 80°F to 105°F, the relative humidity of the air is well below the saturation point, and the cooling coils provide the only significant surfaces at or below the dew point temperature.

The containment fan coil units are designed to remove the sensible heat generated within the containment. The resulting large coil surface area has the effect that the exit air from the coils has a dew point temperature which is very nearly equal to the cooling water temperature.

Measurement of the condensate drained from each of the fan coil units is made to determine condensation rate and thus leak rate.

Should a leak occur, the condensation rate will increase above the previous steady state due to the increased vapor content of the fan coil air intake. A new equilibrium rate will be approached within approximately 200 minutes after the start of the leak. Detection of the increasing condensation rate is possible, however, within 5 to 10 minutes for initial condensation rates in the order of .05 gpm and larger (which correspond to leakage rates of .5 gpm and larger). Readout of each condensate flow measuring device is provided in the Auxiliary Building. A high flow alarm is provided in the control room to alert the operator to significant increases in the condensate flow rate.

#### 6.5.3.1.5 Component Cooling Liquid Monitor

Refer to Section 7.5.2.11. A large heat exchanger leak from the Reactor Coolant System side to the Component Cooling side could result in reactor coolant flowing into the component cooling water and raising the liquid level in the component cooling water surge tank. A high level alarm for the CC surge tank or high radiation or temperature alarms actuated by monitors in the component cooling water headers would alert the operator. The Component Cooling Liquid Monitor can identify RCS leakage as described in Section 7.5.2.11.

#### 6.5.3.1.6 Condenser Air Ejector Gas Monitor

Refer to Section 7.5.2.6. This system monitors the discharge from the air ejector exhaust header of the condensers for gaseous radiation which is indicative of a primary to secondary system leak. The gas discharge can be routed to either the Auxiliary Building normal ventilation system or to the Auxiliary Building special ventilation system.

#### 6.5.3.1.7 Steam Generator Liquid Sample Monitor

Refer to Section 7.5.2.13. This system monitors the liquid phase of the secondary side of the steam generator for radiation which would indicate a primary-to-secondary system leak, providing backup information to that of the condenser air ejector gas monitor. Steam generator blowdown (SGBD) from both steam generators is mixed in the flash tank and is continuously monitored during blowdown via an in-line scintillation detector at the SGBD heat exchanger outlet.

#### 6.5.3.1.8 Containment Sump A Pump Run Time Indication

Containment Sump A pump run time indication is relied upon to detect small reactor coolant leakage in the Leak-Before-Break analyses (Reference 65). Reactor coolant system leaks that develop too slowly to be detected by the rate of change ERCS datapoints for the containment air particulate monitor will be detected by monitoring the containment sump A pump run time indication.

Sump pump operation is controlled by high and low level switches, and more frequent pump operation indicates potential RCS leakage into the sump. The volume of Containment Sump A between switch setpoints is such that a 0.2 gpm leak would result in pump operation on an approximately daily basis. Using this method, a 0.2 gpm leak can be detected within approximately 48 hours. Potential reactor coolant leakage indicated by increased sump pump operating frequency must be investigated to determine the leak source because containment sump A collects leakage from all sources within containment. In this case, other indications of leakage as discussed above can be used.

### 6.5.3.2 Residual Heat Removal System

The out of containment portions of the two residual heat removal systems are physically located within separate shielded and isolated RHR pits as described in Section 6.2. Each RHR pit has its own sump, a 60 gpm sump pump and associated level instrumentation. One of the means for leakage detection is through use of this sump level instrumentation.

The radiation monitoring system portion of the leak detection system is discussed in Section 7.5.2.9. The Residual Heat Removal Cubicle Air Monitors (R-26 and R-27) are discussed in Section 7.5.2.9.

Should a large tube side to shell leak develop in a residual heat exchanger, the water level in the component cooling surge tank would rise, and the operator would be alerted by a high water alarm. Radiation and temperature monitors in the component cooling water headers will also signal an alarm. (For additional information, see Sections 7.5.2.11 and 10.4.2). Should the seal of a residual heat removal pump break, the leakage would drain to the sump in the pit and would be detected by high sump water level, sump pump operations, and by airborne activity indications.

### 6.5.3.3 Component Cooling System

Leakage from the component cooling system inside the reactor containment might be detected by the fall in level of the surge tank and/or the humidity detector and/or the condensate measuring system (For additional information see Section 10.4.2 and the coverage on Reactor Coolant System leak detection).

Visual inspection inside the containment is possible in any operational mode.

If the leakage is from a part of the component cooling system outside the containment, it would be directed by floor drains to an auxiliary building sump. The auxiliary building sump pumps then transfer the leakage to the waste holdup tank. (For additional information see Section 9.2).

### 6.5.3.4 Cooling Water System

The containment fan cooler cooling water monitor checks the containment fan for radiation indicative of a leak from the containment atmosphere into the cooling water.

The Containment Fan Coil Cooling Water Monitors (R-16 and R-38) are provided for leak detection as described in Section 7.5.2.10.

Gross leakage of cooling water due to a faulty cooling coil in the Containment Air Cooling System can be detected by stopping the fans and continuing the cooling water flow. Any significant cooling water leakage would be seen as flow into a collecting pan.

## 6.5.4 Leakage Experience From a Similar Plant Design (Historical Reference)

Operating plant experience from Ginna has indicated that average total primary system leakage (Reactor Coolant System and charging and letdown portion of the Chemical and Volume Control System) would be on the order of 0.5 gpm. Major sources of this leakage were the reciprocating charging pump seals, averaging about .2 gpm, and the pressurizer spray and spray bypass valves leakage, which ranged from 0.2 to 0.5 gpm during the period between repacking.

The plant operates with spray valves of the bellows seal type and spray bypass valves of the packless diaphragm type, thereby eliminating this source of leakage entirely. A new type packing and plunger is used in the reciprocating charging pumps which has limited the leakage from this source to less than 0.07 gallons per hour per pump in long term tests. The design leak rate on this packing plunger combination is 0.3 gallons per hour per pump.

Based on these improvements in equipment, it is anticipated that with proper maintenance of valves and pumps, the leakage rates from this experience can be reduced to values much lower than those previously experienced.

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# 6.6 LEAKAGE PROVISIONS

Provisions are made for the isolation and containment of any leakage.

## 6.6.1 Design Basis

The provisions made for leakage are designed to prevent uncontrolled leaking of reactor coolant or auxiliary cooling water. This is accomplished by (1) isolation of the leak by valves, (2) designing relief valves to accept the maximum flow of water from the worst possible leak, (3) supplying redundant equipment which allows a standby component to be placed in operation while the leaking component is repaired and (4) routing the leakage to various sumps and holdup tanks.

## 6.6.2 Design and Operation

Provisions for leakage avert unmonitored leakage from the primary and auxiliary coolant systems.

### 6.6.2.1 Reactor Coolant System

Adequate identification of leakage from components within the containment vessel is, under most conditions, possible on the basis of relative indication of particulate activity, gaseous activity, humidity, or the flow of condensate collection from the fan coil units; by monitoring of cooling water and component cooling water; by changes in detector response following temporary shutdown or isolation of redundant system components, and by careful evaluation of these indications.

Entry into the containment might, in some instances, be necessary to confirm the identification, and shutdown would possibly be required in the case of leakage from major primary loop components.

If a given source of leakage has been identified and established as safe and acceptable within the Technical Specification limitations, and if further leakage then occurs during subsequent operation, the source of incremental leakage would be evaluated in the same manner. An identical pattern of indication would suggest increased flow of identified leakage, but, in the case of previously identified primary coolant leakage, this would also suggest a deteriorating condition requiring further evaluation and continued surveillance.

When significant leakage from the Reactor Coolant System is detected, action is taken to prevent the release of radioactivity to the atmosphere outside the plant. Automatic actions are performed to mitigate the consequences of a detected leak. See Sections 7.5.2.2, 7.5.2.3, 7.5.2.11 and 7.5.2.13 for details. 98139

If a leak from the Reactor Coolant System to the component cooling system was a gross leak or if the leak could not be isolated from the component cooling system before the inflow completely filled the surge tank, the excess water will flow through the surge tank overflow line. This overflow line is routed to the waste holdup tank in the auxiliary building. 98139

A large leak in the Reactor Coolant System pressure boundary, which does not flow into another closed system, would result in reactor coolant flowing into the containment sump.

A most important aspect of identification is determination of whether the leakage is indeed primary coolant. This identification, plus location with regard to leakage into the containment or into specific closed systems, is normally accomplished adequately by the systems provided. Location of leakage within the containment is not generally possible without entry, except in the case of a fan coil leak, which will cause continued liquid collection after the cooling water flow is stopped and the condensation source removed. In general, direct means of locating leaks within the containment were not intended in design of these systems.

### 6.6.2.2 Residual Heat Removal System

If leakage from the residual heat removal system into the component cooling system occurs, the component cooling radiation monitor will actuate as described in Section 98139
7.5.2.11, an alarm and the valve in the component cooling surge tank vent line is automatically closed to prevent gaseous radioactivity release. If the leaking component (i.e., a residual heat exchanger) could not be isolated from the component cooling system before the inflow completely filled the surge tank, the excess water will flow through the surge tank overflow line and the effluent would be discharged to the waste hold up tank. 98139

Gross leakage from the section of the residual heat removal system inside the containment, which does not flow into another closed system would result in reactor coolant flowing into the containment sump.

Other leakage provisions for the residual heat removal system are discussed in Section 6.5.3.2. 98139

### 6.6.2.3 Component Cooling System

Gross leakage from the section of the component cooling system inside the containment which does not flow into another closed loop flows into the containment sump. Outside the containment major leakage is drained to an auxiliary building sump. From here it is pumped to the waste holdup tank.

Other provisions made for leakage from the component cooling system are discussed in Sections 6.5.3.3 and 10.4.2. 98139

## 6.7 EFFECTS OF LEAKAGE FROM ENGINEERED SAFEGUARDS SYSTEMS

### 6.7.1 Leakage from Residual Heat Removal Systems

Additional release of fission products might be postulated to occur during the Design Basis accident as the result of leakage of containment sump water being circulated through the Residual Heat Removal Systems external to the containment, and the backleakage of containment sump water into the RWST through leaking valves.

As described in Section 6.2, the two trains of the Residual Heat Removal System (RHR) are completely independent and physically separated from each other. Outside of containment, each train is largely housed in a separate below grade vault with floors sloped to separate sumps with separate sump pumps, and the communicating pipe trenches are separated.

Any significant leakage would be detected by sump level instrumentation and by either of two airborne activity monitors in the segregated ventilation ducts from the vaults. Either type of indication would identify which system is leaking and permit its prompt isolation at the containment.

#### 6.7.1.1 ESF Leakage Method of Analysis

Leakage rates through the ESF are controlled by periodic surveillance procedures that establish maximum leakage limitations. Consistent with Regulatory Guide 1.183, a safety factor of two is applied in the AST LOCA analysis. A ESF leakage rate of 4 gph is used in the dose analysis described in Section 14.9.

For determination of the dose contribution from ESF leakage, all radionuclides assumed to be released from the core (except noble gases) are assumed to be instantaneously and homogeneously mixed in the containment sump. Actual leakage from the RCS sump through ESF equipment would not start until after the recirculation phase of the accident begins. For conservatism, and to decouple the dose analyses from the actual calculated recirculation start time, ESF leakage is assumed to begin at t = 0.

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The fraction of total iodine in the liquid that becomes airborne is assumed to be equal to the fraction of leakage that flashes to vapor. Consistent with Regulatory Guide 1.183, Appendix A, this flash fraction (FF) is determined using a constant enthalpy process based on the maximum time-dependent temperature of the sump liquid circulating outside of containment. Values are conservatively selected.

#### 6.7.1.2 RWST Back-Leakage Method of Analysis

The maximum permitted valve seat leakage for valves that could result in backleakage to the RWST is a total of five gallons per hour (5 gph). Consistent with Regulatory Guide 1.183, a safety factor of two is applied in the AST LOCA analysis. A backleakage rate to RWST of 10 gph is used in the dose analysis described in Section 14.9.

For the backleakage to the RWST, the leakage needs to transit long lengths of liquid filled piping prior to reaching the RWST. Due to the time required for the leakage to transit through the piping prior to reaching the RWST, a conservative transit time of 35 hours is used before the leakage reaches the RWST. 01406407

The fraction of total iodine in the liquid that becomes airborne is assumed to be equal to the fraction of leakage that flashes to vapor. Consistent with Regulatory Guide 1.183, Appendix A, this flash fraction (FF) is determined using a constant enthalpy process based on the RWST liquid temperature. Values are conservatively selected.

Radiological consequences of the LOCA, including this RWST back-leakage, are described in Section 14.9.

### 6.7.2 Postulated Major System Leak

More significant releases would result from a major system leak, as from failures of a pump seal. The maximum loss rate from such occurrence might be that associated with complete wipeout of a pump seal, and this could result in a leakage rate of 50 gpm.

The rate of leakage, however, is relatively unimportant except as it affects the total volume that will leak from the system. This total volume includes the partial system drainage that will necessarily occur after isolation for any detected leak rate, and the maximum possible drainage volume would be the largest component system volume between valves, which is approximately 2000 gallons.

Thus a maximum total leakage volume that might be postulated is 3500 gallons--a 50 gpm loss rate that is assumed to persist for thirty minutes prior to detection and isolation, plus 2000 gallons maximum drainage.

A massive break in either system is not regarded as credible because its long-term operation occurs only at low pressures and temperatures and the systems are protected from environmental conditions by Class I structures. Also, incipient breaks preceded by very low leakage from faults that might propagate further would be anticipated by sump level indication, which would permit prompt isolation and depressurizing of a system. Substantially larger leakage rates, although not regarded as credible, might be encompassed by the volume loss assumption, depending on drainage location.

### 6.7.2.1 Method of Analysis

Fifty percent of the total inventory of iodine, or 3.97 x $10^7$ equivalent curies of $I^{131}$, is assumed to be uniformly mixed in 315,000 gallons of containment sump water (1.2 x $10^9$ cc), giving a specific activity of .0331 equivalent curies/cc or 126 curies/gallon. This conservatively assumes that the AEC Safety Guide 4 content of iodine is entirely contained in the water, and it neglects the substantial amount of plateout that may be expected to occur on walls and equipment within the containment.

Accident-related activity is first introduced to the RHR Systems when their operation is changed over from the injection mode to the recirculation mode and this action is not anticipated until nearly a half-hour after the accident. A passive failure of a RHR Pump Seal is not assumed for the initial twenty-four hours. The sump water temperature is conservatively calculated to be well below 212°F at this point after the accident. Thus flashing of any leakage that might occur just at this time would not occur.

Any leakage liquid is therefore subcooled. The iodine contained in any leakage water will largely be retained in the liquid, which will be removed by the sump.

Various estimates of the fraction of iodine that may be expected to escape from the liquid indicate factors on the order of $10^{-4}$. Indeed the leakage dose from identical equipment and the possible activity release from subcooled leakage has been ignored without question in most recent instances of licensing review. However, it is conservatively assumed that ten percent of the iodine is released.

The iodine released from spilled coolant would largely be plated out within structures before release through a charcoal filter that is expected to remove 99 percent or more iodine. For no assumed plateout and a filter efficiency of only 90 percent for elemental iodine and 70 percent for organic iodine, the inhalation doses are calculated in Reference 21.

Results

The added inhalation dose from 3500 gallons total leakage is indicated in the following table.

| Distance | Thyroid Filtered Dose (Rem) | Unfiltered Thyroid Dose (Rem) |
|---|---|---|
| Exclusion Area Boundary (EAB) | 49.4 | 455.2 |
| 1 1/2 Miles | 7.6 | 69.7 |
| 5 Miles | 1.2 | 10.7 |

These estimated doses, which are regarded as extremely conservative, would not cause the guidelines of 10CFR100 to be exceeded if they were added to the radiation doses calculated for the Design Basis Accident. Per NRC Standard Review Plan 15.6.5 Guidelines, dose contribution due to leakage from an ESF System outside containment is not considered provided a filtration system is available. The RHR System is located within the Auxiliary Building Special Ventilation Zone; thus, the above doses are not included with the LOCA dose analysis in Section 14. If this dose contribution were considered with the LOCA dose analysis, the EAB dose would be ignored since the release is assumed at 24 hours and the EAB dose limit is based on the initial two hours of the event. Consistent with Regulatory Guide 1.183, a similar analysis of dose due to a major system leak is not included as part of the Alternative Source Term dose consequence analysis. 01406407

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61. Letter, A Giambusso to AV Dienhart, “Clarification of Guidelines and Criteria Regarding a Postulated Break in a Pipe Carrying a High-Energy Fluid”, January 11, 1973. (7208/1180)

62. USNRC Generic Letter 87-11, Relaxation In Arbitrary Intermediate Pipe Rupture Requirements, June 19, 1987. (30404/2619)

63. EC 16940, Condenser Pit Fill Time due to a Random Pipe Failure

64. ENG-ME-448, Auxiliary Building Flooding Analysis.

65. ENG-ME-792, “R-11 Response to 1.0 GPM and 0.2 GPM RCS Leaks”, dated December, 2010.

66. Letter, T.J. Wengert (NRC) to M.A. Schimmel (NSPM), “Issuance of Amendments [204/191] Re: Request to Exclude the Dynamic Effects Associated with Certain Postulated Pipe Ruptures From the Licensing Basis Based Upon Application of Leak-Before-Break Methodology,” October 27, 2011.

67. Letter, M.A. Schimmel (NSPM) to Document Control Desk (NRC), “Supplement to License Amendment Request to Exclude the Dynamic Effects Associated with Certain Postulated Pipe Ruptures From the Licensing Basis Based Upon Application of Leak-Before-Break Methodology – Response to Request for Additional Information (TAC Nos. ME2976 and ME2977),” L-PI-11-070, August 9, 2011.

68. GEN-PI-079, POST-LOCA EAB, LPZ & CR Dose - AST.

69. Letter T.J. Wengert (NRC) to J.E. Lynch (NSPM), “Prairie Island Nuclear Generating Plant, Units 1 and 2 - Issuance of Amendments RE: Adoption of Alternative Source Term Methodology (TAC Nos. ME2609 and ME2610)”, January 22, 2013. (Adams Accession Number ML112521289). 01459998

70. Letter (NRC) to Kevin K. Davison (NSPM), “Prairie Island Nuclear Generating Plant Units 1 and 2 – Closeout of Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors,” April 10, 2015. (ADAMS Accession Number ML 15062A301). 01482599

71. WCAP-16793-NP-A, Revision 2, “Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid” (October 2011). 01482599

72. Final Safety Evaluation by the Office of Nuclear Reactor Regulation of WCAP-16793-NP, Revision 2 (April 2013); ADAMS Accession No. ML13084A154.

73. EC 21789 Unit 2 RCP Seal Replacement Modification 01450457

74. EC 21790 Unit 1 RCP Seal Replacement Modification

75. EC 25792 Replace Fan Coil Unit Faces-Unit 2 01495214

**TABLE 6.1-1 SYSTEMS CONSIDERED ENGINEERED SAFETY FEATURES FOR THE PURPOSES OF REPORTING UNDER 10 CFR PART 50.72 (b)(2)(ii)**

Auxiliary Feedwater

Component Cooling

Cooling Water

Containment Spray

4160V Safeguards Distribution (System EA)

Emergency Diesel Generators

Residual Heat Removal

Reactor Protection

Safety Injection 94342

**TABLE 6.2-1 SAFETY INJECTION SYSTEM - CODE REQUREMENTS**

01312188

| COMPONENT | CODE |
|---|---|
| Refueling Water Storage Tank | API 650 |
| Residual Heat Exchanger | |
| Tube Side | ASME Section III Class C |
| Shell Side | ASME Section VIII, TEMA Class R |
| Accumulators | ASME Section III Class C |
| Valves | USAS B16.5 |
| Piping | USAS B31.1 |

## TABLE 6.2-2 QUALITY STANDARDS OF SAFETY INJECTION SYSTEM COMPONENTS

(Page 1 of 4)

01484978

### RESIDUAL HEAT EXCHANGERS

A. Test and Inspection

1. Hydrostatic Test
2. Radiograph of longitudinal and girth welds (tube side only)
3. UT of tubing or eddy current tests
4. Dye penetrant test of welds
5. Dye penetrant test of tube to tube sheet welds
6. Gas leak test of tube to tube sheet welds before hydro and expanding tubes

B. Special Manufacturing Process Control

1. Tube to tube sheet weld qualification procedure
2. Welding and NDT and procedure review
3. Surveillance of supplier quality control and product

### COMPONENT COOLING HEAT EXCHANGER

A. Test and Inspections

1. Hydrostatic Test
2. Dye penetrant test of welds
3. Radiography of shell side nozzle joints
4. Ultrasonic examination of tubing
5. Shell side gas leak tests

B. Special Manufacturing Process Control

1. Welding and NDT and procedure review
2. Surveillance of supplier quality control and product

**TABLE 6.2-2 QUALITY STANDARDS OF SAFETY INJECTION SYSTEM COMPONENTS**
(Page 2 of 4)

01484978

**SAFETY INJECTION, AND RESIDUAL HEAT REMOVAL PUMP**

A. Test and Inspections

1. Performance Test
2. Dye penetrant of pressure retaining parts
3. Hydrostatic test
4. Radiograph and liquid penetrant in accordance with B&PV Code
5. SI pump forgings receive UT and magnetic particle or liquid penetrant of barrel

B. Special Manufacturing Process Control

1. Weld, NDT and inspection procedures for review
2. Surveillance of suppliers quality control system and product

**ACCUMULATORS**

A. Test and Inspections

1. Hydrostatic test
2. Radiography of longitudinal and girth welds
3. Dye penetrant/magnetic particle of weld

B. Special Manufacturing Process Control

1. Weld, fabrication, NDT and inspection procedure review
2. Surveillance of suppliers quality control and product

**TABLE 6.2-2 QUALITY STANDARDS OF SAFETY INJECTION SYSTEM COMPONENTS**
(Page 3 of 4)

01484978

**VALVES**
(Not Including Auxiliary Relief Valves)

A. Test and Inspections

(a) 200 psi and 200°F or below (cast or bar stock)

1. Dye penetrant Test
2. Hydrostatic Test
3. Seat Leakage Test

(b) Above 200 psi and 200°F

(i) Forged Valves

1. UT of billet prior to forging
2. Dye penetrant 100% of accessible areas after forging
3. Hydrostatic Test
4. Seat Leakage Test

(ii.) Cast Valves

1. Radiograph 100%[1]
2. Dye Penetrant all accessible areas[1]
3. Hydrostatic Test
4. Seat Leakage

(c) Functional Tests Required for:

1. Motor Operated Valves
2. Auxiliary Relief Valves
3. Air Operated Valves

[1] For valves in radioactive service only.

**TABLE 6.2-2 QUALITY STANDARDS OF SAFETY INJECTION SYSTEM COMPONENTS**

(Page 4 of 4)

01484978

B. Special Manufacturing Process Control

1. Weld, NDT, performance testing, assembly and inspection procedure review

2. Surveillance of suppliers quality control and product

3. Special Weld process procedure qualification (e.g. hard facing)

**PIPING**

A. Tests and Inspections

Class 1501 and below

Seamless or welded pipe if welded 100% radiography is required.

Shop Fabricated and field fabricated pipe weld joints are inspected as follows:

301R - radiograph inspection

151R - 100% liquid penetrant examination

B. Special Manufacturing Process Control for Reactor Control Piping and Pressurizer Surge Line

Surveillance of suppliers quality control and product.

C. Hydrostatic Testing in accordance with USAS B31.1

**TABLE 6.2-3 ACCUMULATOR DESIGN PARAMETERS**

| | |
|---|---|
| Number (per Unit) | 2 |
| Type | Stainless steel clad/ carbon steel |
| Total volume, $ft^3$ | 2000 |
| Design pressure, psig | 800 |
| Design temperature, °F | 300 |
| Normal pressure, psig | 750 |
| Minimum pressure, psig | 700 |
| Relief valve set point, psig* | 800 |
| Operating temperature, °F | 70 - 120 |
| Minimum water volume at operating conditions, $ft^3$ | 1250 |
| Minimum Boron concentration (as boric acid), ppm | 2300 |

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* The relief valves have soft seats and are designed and tested to ensure zero leakage at normal operating pressure.

**TABLE 6.2-4 REFUELING WATER STORAGE TANK DESIGN PARAMETERS**

| | | |
|---|---|---|
| Number (per Unit) | 1 | |
| Material | Stainless Steel | |
| Total volume, gal. | 275,000 | |
| Minimum volume, (solution) gal. | 265,000 | 01124489 |
| Normal pressure, psig | atmospheric | |
| Minimum Operating temperature, °F | 70°F | |
| Design pressure, psig | atmospheric | |
| Design temperature, °F | 200 | |
| MaximumBoron concentration (as boric acid), ppm | 3500 | |

**TABLE 6.2-5 PUMP PARAMETERS**
(Page 1 of 2)

| Safety Injection Pump Design Parameters | |
|---|---|
| Number | 2 |
| Type | Horizontal centrifugal |
| Design pressure, psig | 2485 |
| Design temperature, °F | 300 |
| Design flow rate, gpm | 700 |
| Design head, ft | 2760 |
| Shutoff head, ft | 5100 maximum |
| Material clad | Carbon Steel Forging-18-8 |
| Motor H. P. | 800 |

| Residual Heat Removal Pump Design Parameters | |
|---|---|
| Number of pumps | 2 |
| Type | Vertical centrifugal in-line |
| Design pressure, discharge, psig | 600 |
| Design temperature, °F | 400 |
| Design flow, gpm | 2000 |
| Design head, ft | 280 |
| Material | Austenitic Stainless steel |
| Shutoff head, ft | 340 |
| Motor H. P. | 200 |

**TABLE 6.2-5 PUMP PARAMETERS**
(Page 2 of 2)

| | **Safety Injection Pumps** | **Residual Heat Removal Pumps** | |
|---|---|---|---|
| Required NPSH @ design flow | 17 FT @ 700 gpm | 8 FT @ 2000 gpm | |
| Required NPSH @ maximum runout flow | 21 FT @ 835 gpm | 14 FT @ 2600 gpm | |
| Minimum available NPSH during the injection phase | 24 FT | 47 FT (Reference 31) | 01473445 |
| Minimum available head loss without strainer | Note 1 | 28.3 FT (Reference 32) Note 2 | |

NOTES:

1. The suction of the safety injection pumps during high head recirculation are boosted by RHR pumps.

2. Head loss across strainer and debris bed is shown to be less than the difference between NPSH available and NPSH required. (Reference 39)

**TABLE 6.2-6 RESIDUAL HEAT EXCHANGERS DESIGN PARAMETERS**

| | | |
|---|---|---|
| Number | | 2 |
| Design heat duty, Btu/hr (Normal) | | $26.0 \times 10^6$ |
| Design UA, Btu/hr/°F | | $0.68 \times 10^6$ |
| Type | | Vertical Shell and U-tube |
| | Tube-Side | Shell-Side |
| Design pressure, psig | 600 | 150 |
| Design Temperature °F | 400 | 350 |
| Design flow, lb/hr | $1.0 \times 10^6$ | $1.25 \times 10^6$ |

## TABLE 6.2-7 ORIGINAL MAXIMUM POTENTIAL RECIRCULATION LOOP LEAKAGE*

| | Items | No. of Items per unit | Type of Leakage Control and Unit Leakage Rate Used in the Analysis | Leakage to Atmosphere cc/hr | Leakage Sump Tank cc/hr |
|---|---|---|---|---|---|
| 1. | Residual Heat Removal Pumps | 2 | Mechanical seal with backup labyrinth. Essentially zero leakage from mechanical seal | 0 | 0 |
| 2. | Spray Pumps | 2 | Same as Residual | 0 | 0 |
| 3. | Safety Injection Pumps | 2 | Same as Residual | 0 | 0 |
| 4. | Flanges: | | Gasket - adjusted to zero leakage following any test - 10 drops/min/flange used in analysis | | |
| | a. Pumps (4) | 8 | | 240 | 0 |
| | b. Valves bonnet to body (larger than 2I) | 37 | Startup strainer flange at pumps | 1120 | 0 |
| | c. Control Valves (butterfly) | 4 | | 120 | 0 |
| | d. Heat exchanger | 2 | | 60 | 0 |
| | e. Startup strainers | 12 | | 360 | - |
| 5. | Valves - Stem Leakoffs | 20 | Packed without backseating or leakoff. | 0 | 20 |
| 6. | Misc. Large Valves (larger than 2I) | 13 | Double packing 1 cc/hr | 13 | 0 |
| 7. | Misc. Small Valves | 100 | Flanged body packed stems - 1 drop/min. used | 300 | 0 |
| | | | | 2213 | 20 |

* Table represents the original maximum design leakage for each recirculation loop. Current dose consequence analysis uses outside of containment leakage rates that follow Reg. Guide 1.183, Guidance for Alternate Source Tem. These leakage rates are: 2 gph (7,560 cc/hr) allowable, 4 gph (15,120 cc/hr) assumed for dose analysis. (Ref. 68, 69)

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## TABLE 6.2-8a SINGLE FAILURE ANALYSIS-SAFETY INJECTION SYSTEM
## Page 1 of 2

| | Component | Malfunction | Comments |
|---|---|---|---|
| A. | Accumulator (injection phase) | Delivery to broken loop | Totally passive system with one accumulator per loop. Evaluation based on one accumulator delivering to the core and one spilling from ruptured loop. |
| B. | Pump: (injection phase) | | |
| | 1) High head safety injection | Fails to start | Two provided. Evaluation based on operation of one. |
| | 2) Residual heat removal | Fails to start | Two provided. Evaluation based on operation of one. |
| | 3) Component cooling[1] | Fails to start | Two provided. One required for recirculation cooling |
| | 4) Cooling Water | Fails to start | Five provided. Evaluation based on operation of one. (See also Section 10.4.1) |
| C. | Automatically Operated Valves: Open on SIS - Injection phase | | |
| | 1) Not used | | |
| | 2) Residual heat removal pump isolation valves at reactor vessel | Fails to open | Two parallel valves. Both of which are already open with breakers off. |
| | 3) Not used | | |

[1] Recirculation phase

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## TABLE 6.2-8a SINGLE FAILURE ANALYSIS-SAFETY INJECTION SYSTEM


| | Component | Malfunction | Comments |
|---|---|---|---|
| D. | Valves Operated From Control Room for Recirculation: (recirculation phase) | | |
| | 1) Containment sump recirculation isolation | Fails to open | Two lines in parallel with two valves in series in each line, one pair of valves in either line is required to open |
| | 2) Safety injection pump suction valve at residual heat exchanger discharge | Fails to open | Two parallel valves. One required to open |
| | 3) Isolation valve on the test line returning to the refueling water storage tank | Fails to close | Two valves in series. One required to close |
| | 4) Isolation valve at suction header from refueling water storage tank | Fails to close | Two parallel valves. One required to close. Interlocks assure the selected valve is closed before the associated supply valve from the residual head removal heat exchanger can be opened. |
| | 5) Isolation valves at residual heat removal pump suction line from refueling water storage tank | Fails to close | Two valves in series (one a check valve). One required to close. See Section 6.2.2.1. |

The status of all active components of the Safety Injection System is indicated on the main control board. Reference is made to Table 6.2-2.

**TABLE 6.2-9 SHARED FUNCTIONS EVALUATION**
**Page 1 of 2**

01-091

| Component | Normal Operating Function | Normal Operating Arrangement | Accident Function | Accident Arrangement |
|---|---|---|---|---|
| Refueling Water Storage Tank (1/unit) | Storage tank for refueling operations | Lined up to suction of safety injection residual heat removal, and spray pumps | Source of borated water for core and spray nozzles | Line up to suction of safety injection residual heat removal, and spray pump |
| Accumulators (2/unit) | None | Lined up to cold legs of reactor coolant piping | Supply borated water to core promptly | Lined up to cold legs of reactor coolant piping |
| Safety Injection (2/Unit) | None | Lined up to reactor vessel plenum nozzles and cold legs of reactor coolant piping (no automatic supply to reactor vessel plenum nozzles) | Supply borated water to core | Lined up and open to cold legs of reactor coolant piping |
| Residual Heat Removal Pumps (2) (2/unit) | Supply water to loop to remove residual heat during shutdowns | Lined up to take suction from refueling water storage tank and deliver to reactor vessel | Supply borated water to core through reactor vessel nozzles | Lined up to take suction from refueling water storage tank and deliver to reactor vessel |
| Cooling Water* Pumps (5) | Supply cooling water to component cooling heat exchangers | Two pumps in service during operation of both units (See Section 10.4) (one per unit) | Supply cooling water to component cooling heat exchangers and containment fan coil units (under accident conditions) | Two pumps in service during operation of both units (See Section 10.4) (one per unit) |

* Shared
** One tank associated with each unit, the third tank is a common standby

## TABLE 6.2-9 SHARED FUNCTIONS EVALUATION


| Component | Normal Operating Function | Normal Operating Arrangement | Accident Function | Accident Arrangement |
|---|---|---|---|---|
| Component Cooling Pumps (2) | Supply cooling water to plant nuclear components | One pump in service | Supply cooling water to residual heat exchangers and residual heat removal pump seals and to S. I. pump seal cooler heat exchanger and lube oil coolers | One pump in service |
| Residual Heat Exchangers (2) | Remove residual heat from core during shutdown | Lined up for recirculation | Cool water in containment sump for core cooling | Lined up for recirculation |
| Component Cooling Heat Exchangers (2) | Remove heat from component cooling water | One heat exchanger in service | Cool water for residual heat exchangers and residual heat removal pump seals and to S. I. pump seal cooler heat exchanger and lube oil coolers | One heat exchanger in service |

**TABLE 6.2-10 ACCUMULATOR INLEAKAGE**

01414695

| Time Period Between Level Adjustments | Assumed Leak Rate cc/hr | (Assumed Leak Rate) ÷ (Max. Allowed Design) |
|---|---|---|
| 1 month | 1377 | 57.4 |
| 3 months | 459 | 19.1 |
| 6 months | 229 | 9.6 |
| 9 months | 153 | 6.4 |
| 1 year | 115 | 4.8 |

## TABLE 6.2-11 RESIDUAL HEAT REMOVAL SYSTEM DESIGN, OPERATION AND TEST CONDITIONS

| | Pumps | Heat Exchangers | Valves | Pipe and Fittings |
|---|---|---|---|---|
| Design Conditions of Systems | | | | |
| Pressure, psig | 600 | 600 | 600 | 600 |
| Temperature, °F | 400 | 400 | 400 | 400 |
| Operating Conditions (Max)* | | | | |
| Pressure, psig | 200 | 200 | 200 | 200 |
| Temperature, °F (approx) | 250 | 250 | 250 | 250 |
| Test Pressure, psig | 1200 | 900 | 1100 | 900 |
| Allowable Pressure at Operating Temp. psig | $\leq$ 600 | $\leq$ 600 | $\leq$ 690 | $\leq$ 850 |

01-072

*During post loss-of-coolant recirculation

TABLE 6.3-1 SINGLE FAILURE ANALYSIS - CONTAINMENT AIR COOLING SYSTEM

| Component | Malfunction | Comments and Consequences |
|---|---|---|
| A. Containment Fan Coil Unit | Fails to Start | Four provided. Evaluation based on two fans and one containment spray pump operating. |
| B. Cooling Water Pumps | Fails to Start | Five provided. Two required for operation of 4 coils, one for two coils. |
| C. Automatically Operated Valves: (Cooling water open on automatic engineered safety features sequence signal and chilled water close) | Cooling Water Valves fail to open and chilled water fail to close | Two redundant cooling water trains provided with 2 fan coils each train. Evaluation based on 2 fans (one train) and one containment spray pump operating |

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## TABLE 6.3-2 SHARED FUNCTION EVALUATION

| Component | Normal Operating Function | Normal Operating Arrangement | Accident Function | Accident Arrangement |
|---|---|---|---|---|
| Containment Fan Cooling Units (4) | Circulate and cool containment atmosphere and equipment | Four fan units in service | Circulate and cool containment atmosphere | Four initially and two fan units in service later during the accident |
| Chilled Water Pumps (Non-Safeguards Equipment) | Supply chilled water to fan units | One pump in service | None | N/A |
| Cooling Water Pumps (5) | | | Supply cooling water to fan units | Two pumps in service immediately, one later. |

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## TABLE 6.4-1 CONTAINMENT VESSEL INTERNAL SPRAY SYSTEM-CODE REQUIREMENTS

| Component | Code |
|---|---|
| Valves | USAS B16.5 or B16.34 |
| Piping (including headers and spray nozzles) | USAS B31.1 |

05-012

**TABLE 6.4-2 CONTAINMENT VESSEL INTERNAL SPRAY PUMP DESIGN PARAMETERS**

| | | |
|---|---|---|
| Quantity | 2 | |
| Design pressure, discharge, psig | 500 | |
| Design temperature, °F | 300 | |
| Design flow rate, gpm | 1300 | |
| Design head, ft. | 500 | |
| Shutoff head, ft. | 550 | 98139 |
| Motor HP | 250 | |
| Type | Horizontal-Centrifugal | |

## TABLE 6.4-3 SINGLE FAILURE ANALYSIS - CONTAINMENT SPRAY SYSTEM

| Component | Malfunction | Comments and Consequences |
|---|---|---|
| A. Spray Nozzles | Clogged | Large number of nozzles (168) renders clogging of a significant number of nozzles as incredible. |
| B. Pumps<br>1) Containment Spray Pump | Fails to start | Two provided. Evaluation based on operation of one pump in addition to two out of four containment cooling fans operating during injection phase. |
| 2) Cooling Water | Fails to start | Five provided. Operation of one pump required. |
| 3) Component Cooling | Fails to start | Two provided. Operation of one pump required. |
| C. Automatically operated valves: coincidence of three - 1/2 Hi Hi containment pressure signals) | | |
| 1) Containment spray pump discharge isolation valve | Fails to open | Two provided. Operation of one required (per header). |

01312188

**TABLE 6.4-4 SHARED FUNCTIONS EVALUATION**

98139

| Component | Normal Operating Function | Normal Operating Arrangement | Accident Function | Accident Arrangement |
|---|---|---|---|---|
| Containment spray pumps (2) | None | Lined up to spray headers | Supply spray water to containment atmosphere | Lined up to spray headers |

Note: Refer to Section 6.2 for a brief description of the refueling water storage tank, residual heat removal pumps, conventional cooling water pumps, component cooling pump, residual heat exchangers and component cooling heat exchangers which are also associated either directly or indirectly with the Containment Vessel Internal Spray System. 98139

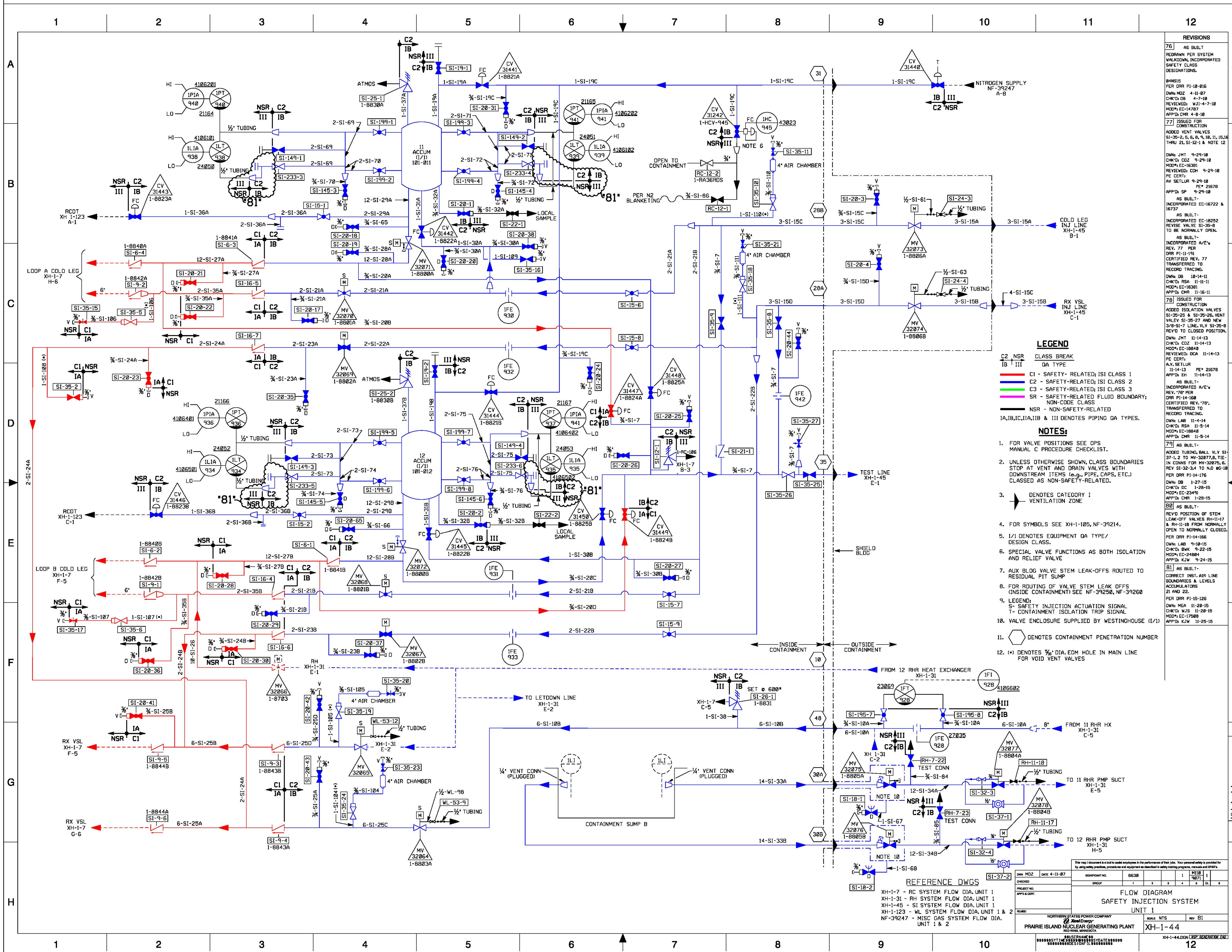


FIGURE 6.2-1A REV. 34

01516979

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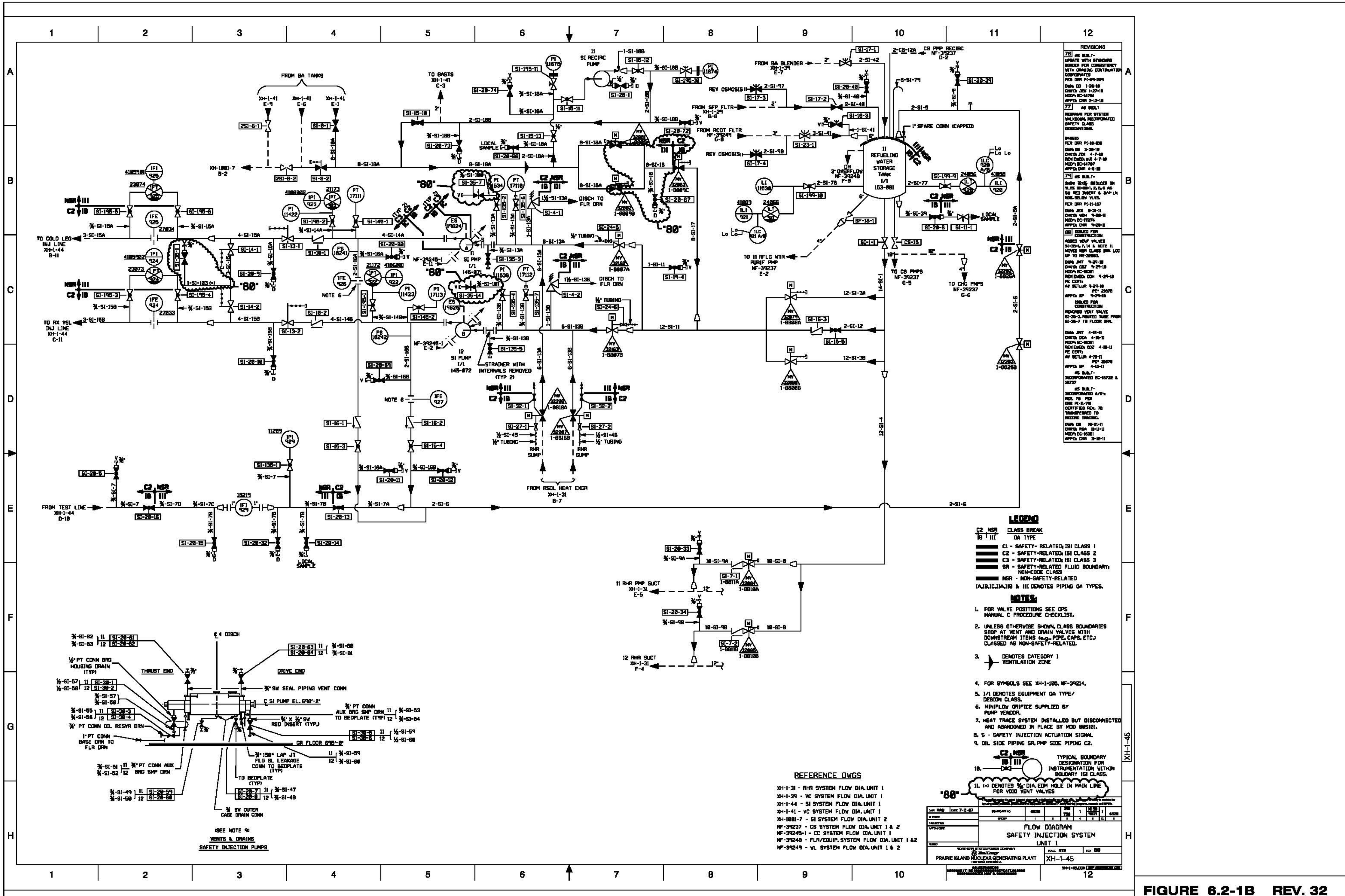


FIGURE 6.2-1B REV. 32

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**FIGURE 6.2-2A REV. 34**

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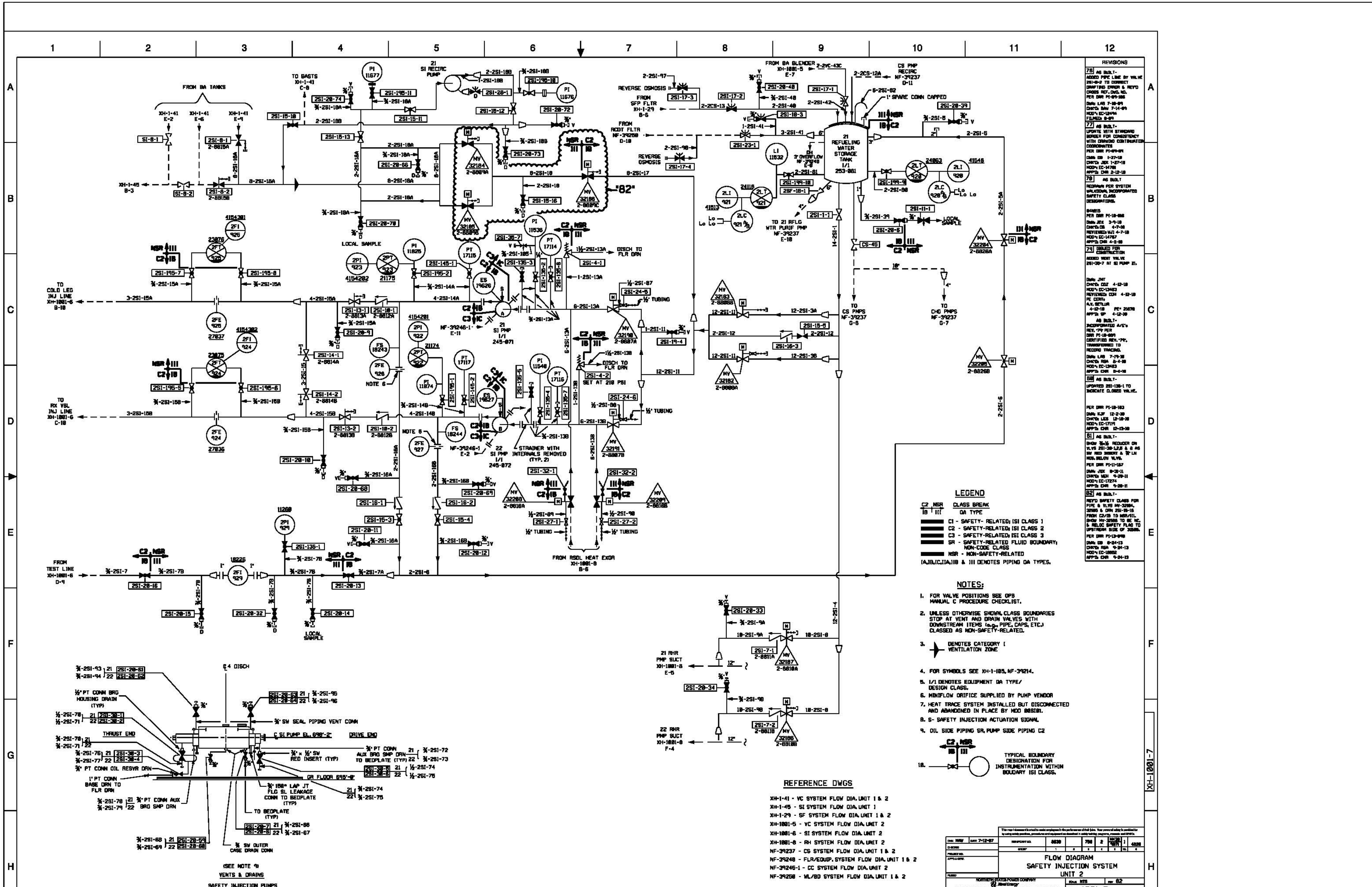


FIGURE 6.2-2B REV. 33

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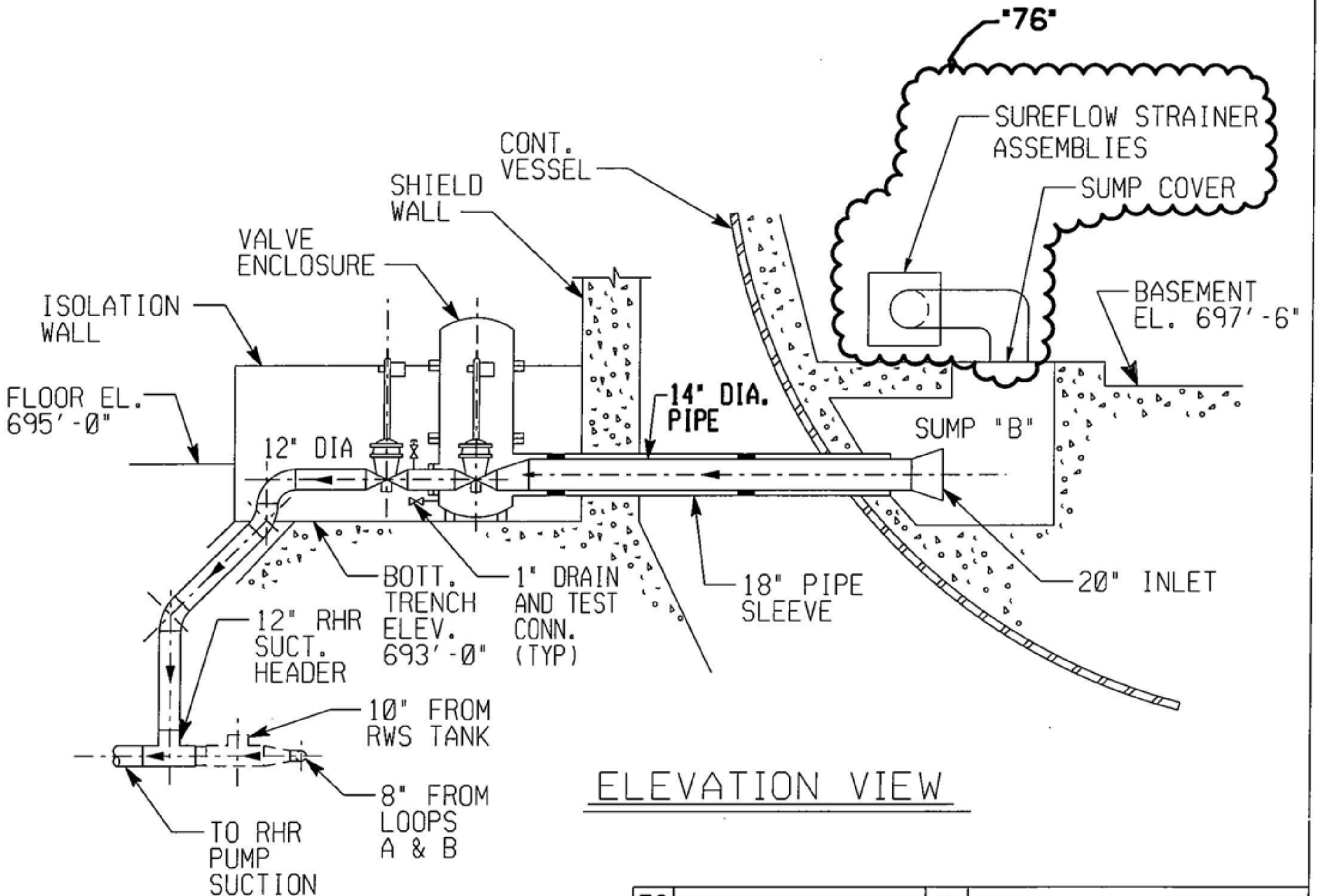
NL-173001-1




ELEVATION VIEW

| 76 AS BUILT- | A AS BUILT- |
|---|---|
| ADDED SUREFLOW STRAINER ASSEMBLIES INSTALLED PER EC-378 | INCORPORATE DRAWING INTO NSP DRAWING SYSTEM PER NDR PI-98-102 |
| PER DRR PI-10-047 | |
| DWN: DMN 5-28-10 | DWN: BMS 12-7-98 |
| CHK'D: JWL 6-2-10 | CHK'D: PAS 12-22-98 |
| MOD#: EC-14086 | MOD: |
| APP'D: CMR 6-2-10 | FILMED 1-99 |

| DWN TAM | DATE 10-25-97 | SIGNIFICANT NO. | 8630 | | 210 | 1&2 | M610 / 6900 | 3 | 3310 |
|---|---|---|---|---|---|---|---|---|---|
| CHECKED | | GROUP | 1 | 2 | 3 | 4 | 5 | CL | 6 |
| PROJECT NO. | | | | | | | | | |
| APP'D & CERT. | | CONTAINMENT SUMP B ELEVATION VIEW | | | | | | | |
| FILMED | | | | | | | | | |

NORTHERN STATES POWER COMPANY
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RED WING, MINNESOTA

SCALE NONE REV 76

NL-173001-1

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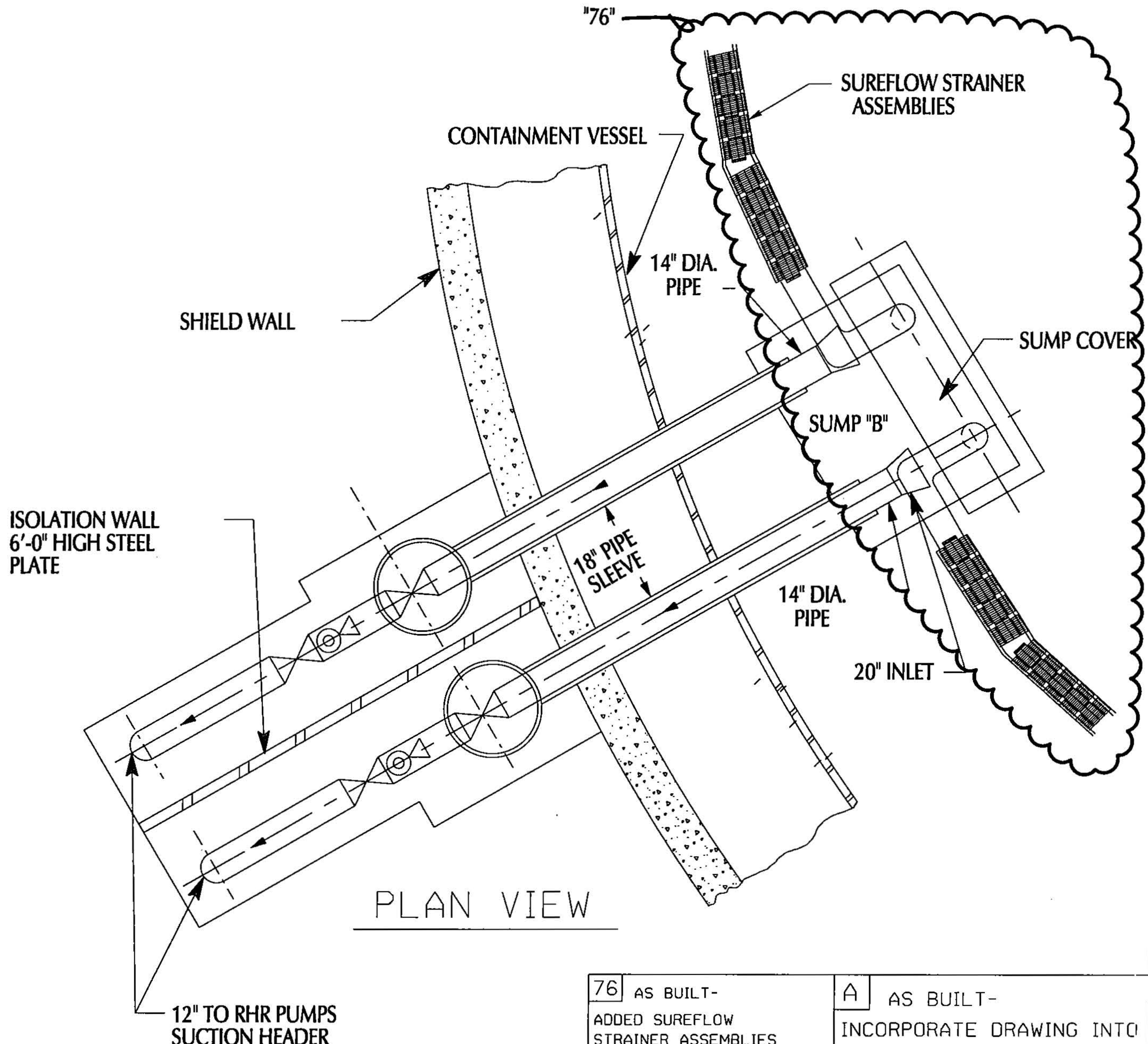
**FIGURE 6.2-3 REV. 31**

NL-173001-2




PLAN VIEW

| 76 AS BUILT- | A AS BUILT- |
|---|---|
| ADDED SUREFLOW STRAINER ASSEMBLIES INSTALLED PER EC-378<br><br>PER DRR PI-10-047<br>DWN: DMN 5-28-10<br>CHK'D: JWL 6-2-10<br>MOD#: EC-14086<br>APP'D: CMR 6-2-10 | INCORPORATE DRAWING INTO NSP DRAWING SYSTEM PER NDR PI-98-102<br>DWN: BMS 12-7-98<br>CHK'D: PAS 12-22-98<br>MOD: —#—<br>FILMED 1-99 |

| DWN TAM | DATE 11-15-97 | SIGNIFICANT NO. | 8630 | | 210 | 1&2 | M610 / 6900 | 3 | 3310 |
|---|---|---|---|---|---|---|---|---|---|
| CHECKED | | GROUP | 1 | 2 | 3 | 4 | 5 | CL | 6 |

PROJECT NO.

APP'D & CERT.

FILMED

# CONTAINMENT SUMP B
# PLAN VIEW

NORTHERN STATES POWER COMPANY
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RED WING, MINNESOTA

SCALE NONE | REV 76

NL-173001-2

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**FIGURE 6.2-4 REV. 31**

01516979


FIGURE 6.2-5 REV. 34

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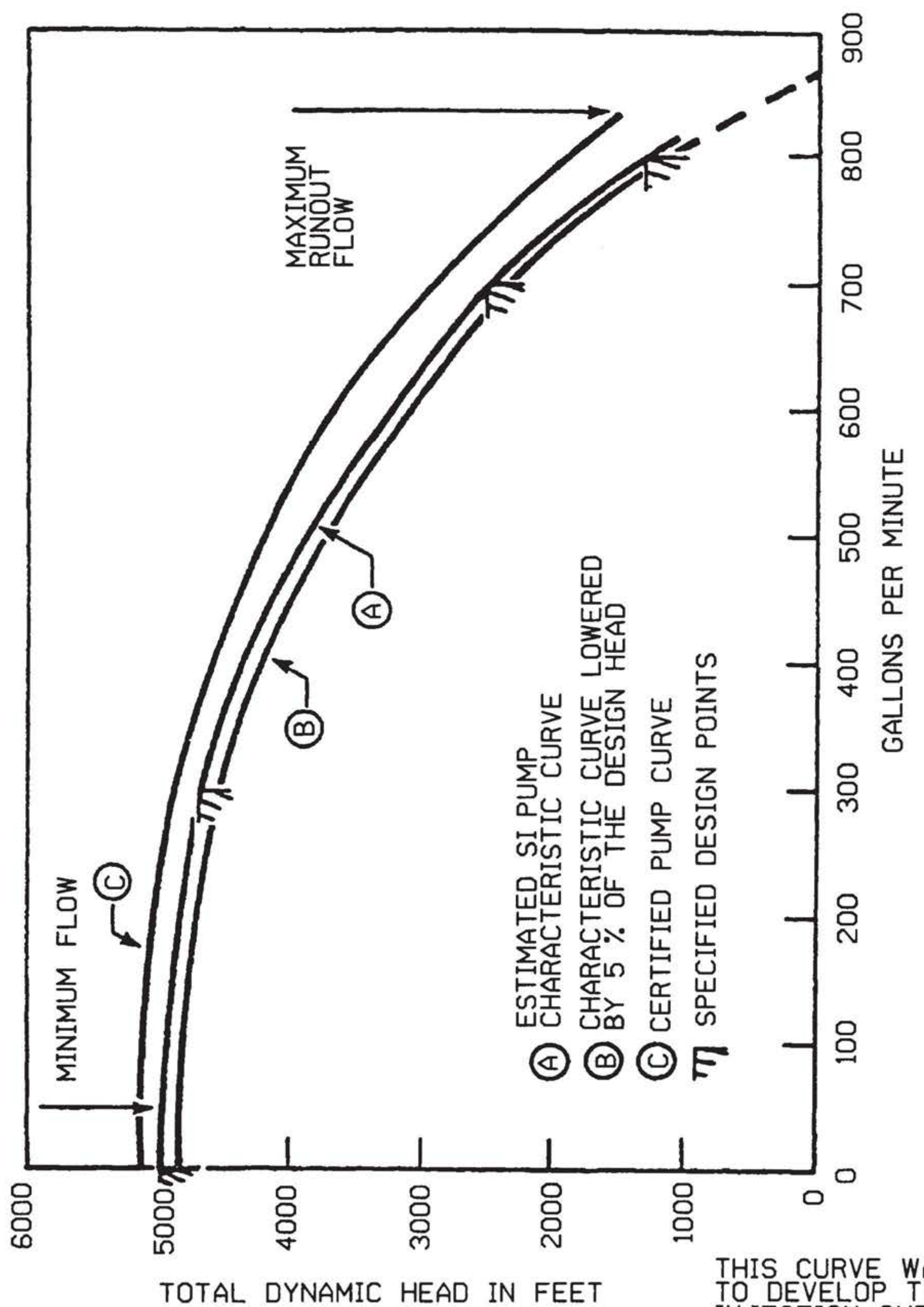



THIS CURVE WAS USED TO DEVELOP THE INJECTION CURVES PRESENTED IN CHAPTER 14

SAFETY INJECTION PUMP CHARACTERISTICS

| | | | |
|---|---|---|---|
| DWN T. MILLER | DATE 6-23-99 | NORTHERN STATES POWER COMPANY | SCALE: NONE |
| CHECKED | CAD FILE U06206.DGN | PRAIRIE ISLAND NUCLEAR GENERATING PLANT<br>RED WING MINNESOTA | **FIGURE 6.2-6 REV. 18** |

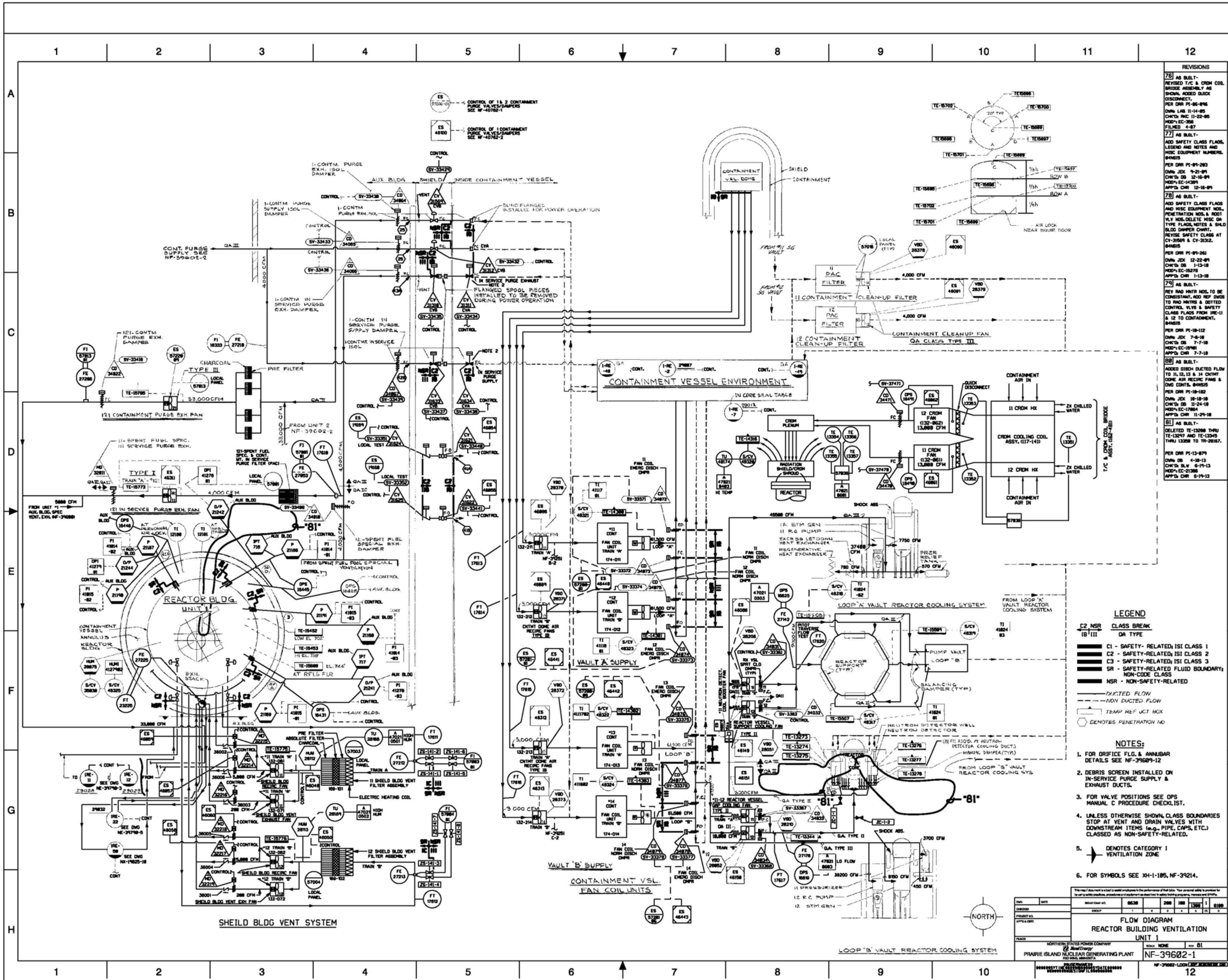



**FIGURE 6.3-1A REV. 33**

01429088

CAD FILE: U06301A.DGN

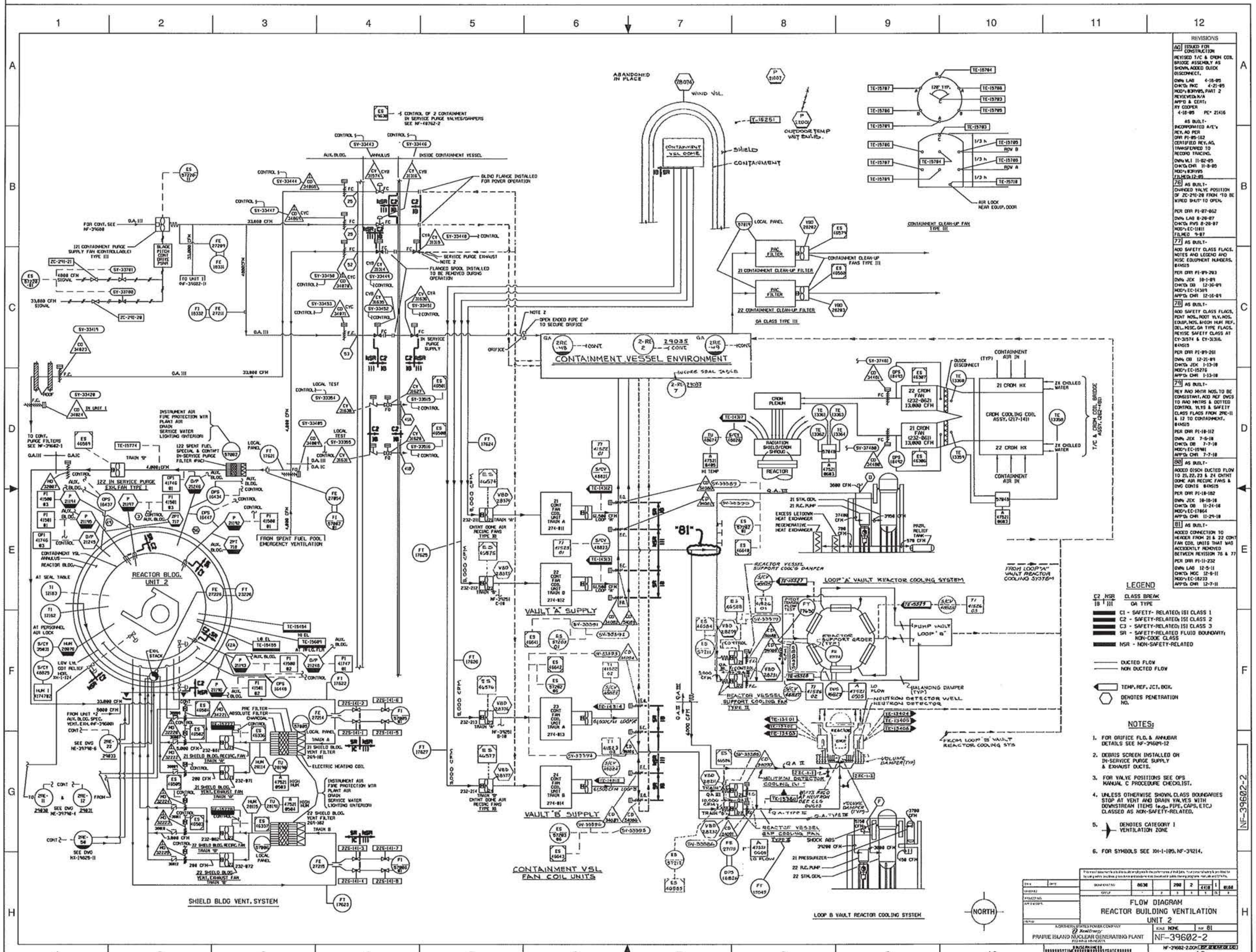



**FIGURE 6.3-1B REV. 32**

01352784

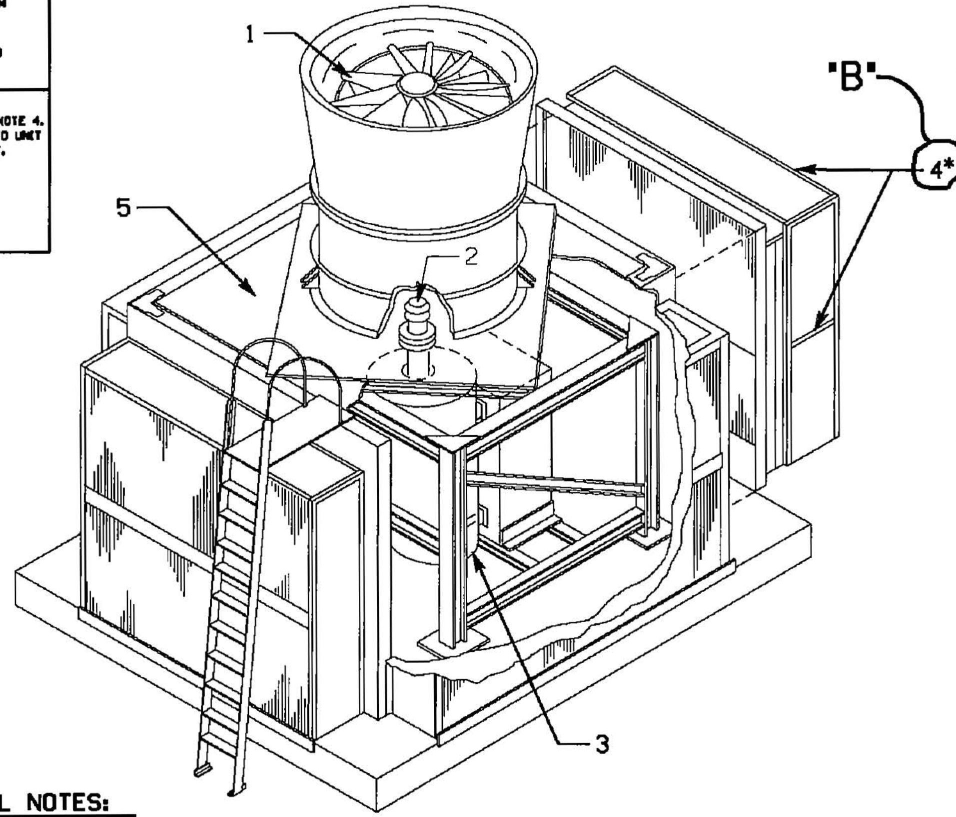
CAD FILE: U06301B.DGN

NL-173001-3

A AS BUILT-
INCORPORATE DRAWING INTO
NSP DRAWING SYSTEM
PER NDR PI-98-102
DWN: BMS 12-7-98
CHK'D: PAS 12-22-98
MOD:
FILMED 1-99

B AS BUILT-
MADE ADDITION TO NOTE 4.
REFLECTS CHANGE TO UNIT
2 FCU REPLACEMENT.

PER DRR PI-05-126
DWN: DMN 7-18-05
CHK'D: CMR 8-2-05
MOD#: 05ZC02
FILMED 9-05




## SPECIAL NOTES:

1. WESTINGHOUSE STURTEVANT SPECIAL MODEL L1054 VANE AXIAL FAN WITH NON-ADJUSTABLE BLADES. NOMINALLY RATED AT 60,000 CFM AT 1760 RPM AND 4.6" W.G. TOTAL STATIC PRESSURE

2. COUPLING

3. WESTINGHOUSE FR 449 TZ FORCE VENTILATED RCFC AC MOTOR. 460 VOLTS, 3 PHASE, 60 HERTZ, 1800 RPM-75 HP/900 RPM-25 HP SINGLE WINDING.

4. COOLING COILS WITH 0.035" COPPER TUBES, COPPER FINS, HEADERS AND FLANGED PIPE CONNECTIONS. TWO (2) COILS PER ENCLOSURE SIDE OR EIGHT (8) PER ENCLOSURE.

"B" * UNIT 2 COOLING COILS WITH 0.049" COPPER-NICKEL TUBES AND .065" U-BENDS.

5. ENCLOSURE TOP PLATE WITH ACCESS HATCH.

## GENERAL NOTES:

CORROSION PROTECTION-FAN WHEEL HUB, FAN HOUSING, MOTOR AND ENCLOSURE ARE COATED.

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| DWN TAM DATE 10-22-97 | SIGNIFICANT NO. | 8630 | | 210 | 1&2 | M300 6900 | 3 | 1180 |
| CHECKED | GROUP | 1 | 2 | 3 | 4 | 5 | CL | 6 |

PROJECT NO.

APP'D & CERT.

FILMED

# CONTAINMENT FAN COIL UNIT

NORTHERN STATES POWER COMPANY
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RED WING, MINNESOTA

SCALE NONE
REV B

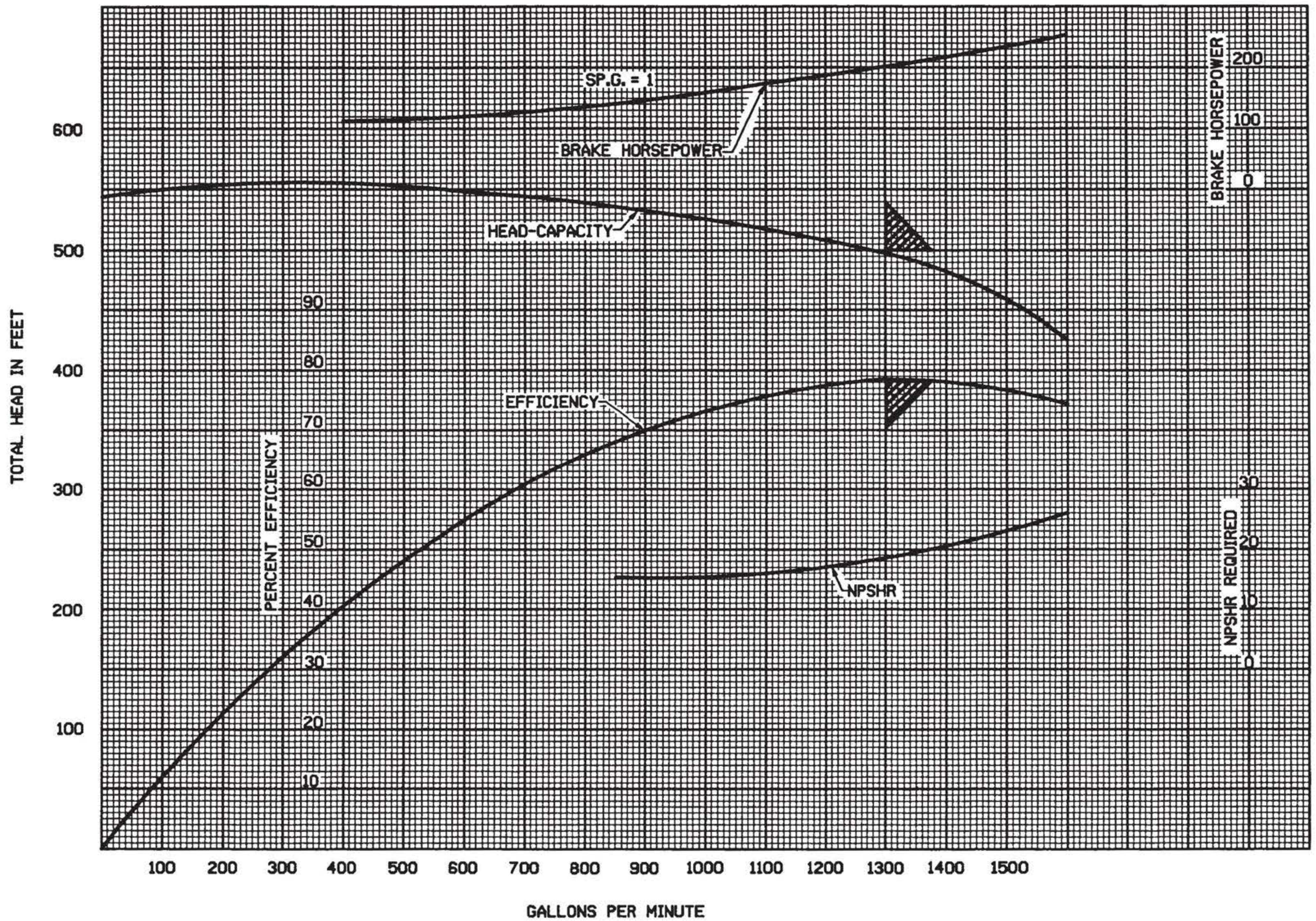
NL-173001-3

173001-3.DGN NSP GENERATION CAD

# FIGURE 6.3-2 REV 30

01169405

FIGURE 6.3-2




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 U06401.DGN | | | | | | | | |

CONTAINMENT SPRAY PUMP
CHARACTERISTIC CURVES

NORTHERN STATES POWER COMPANY
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RED WING MINNESOTA

SCALE NONE | REV

**FIGURE 6.4-1 REV. 18**

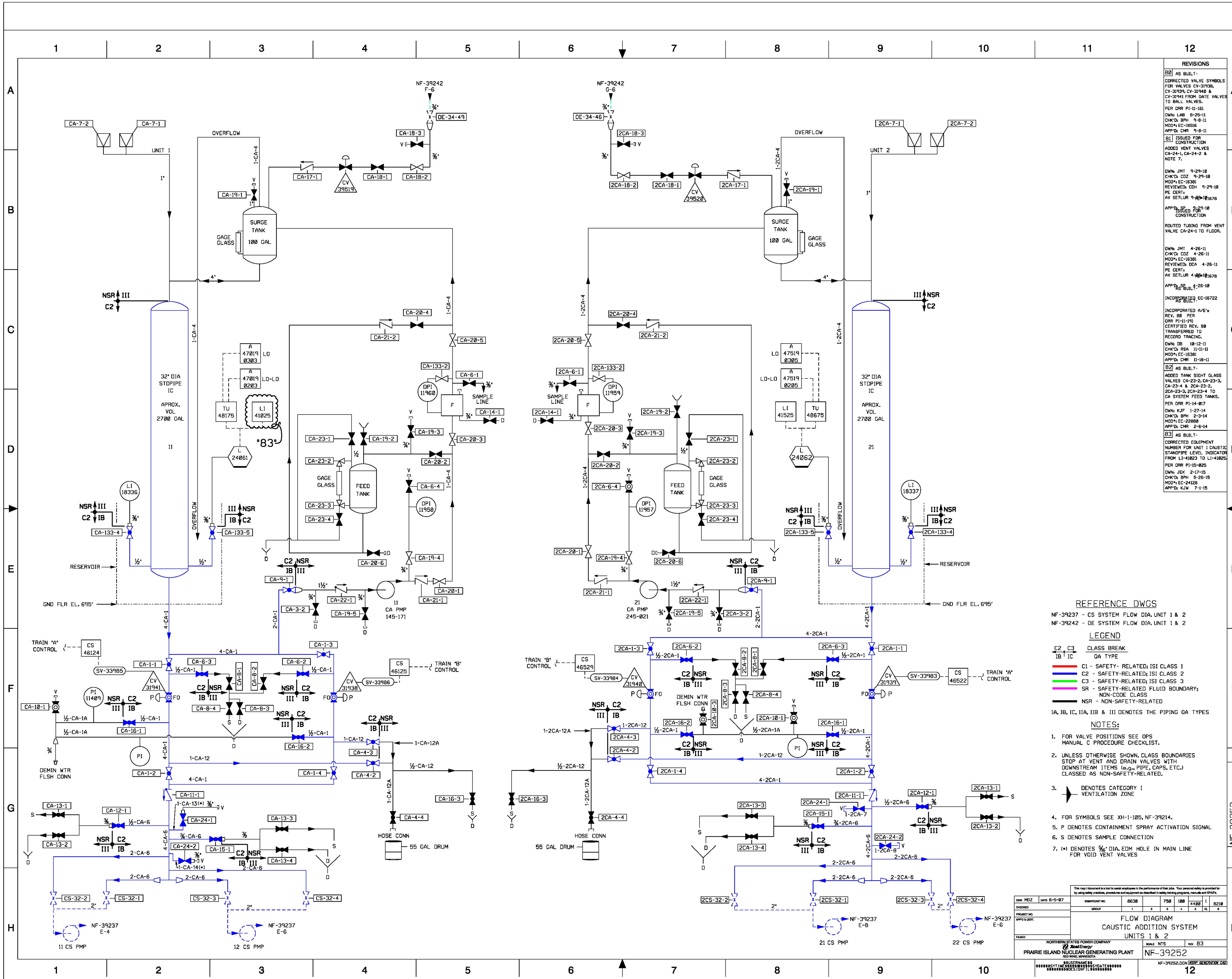



FIGURE 6.4-2 REV. 34

01516979

CAD FILE: U06402.DGN