SECTION 10

PLANT AUXILIARY SYSTEMS

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SECTION 10 PLANT AUXILIARY SYSTEMS

10.1 SUMMARY DESCRIPTION

The Plant Auxiliary Systems are the supporting systems required to ensure the safe operation or servicing of the Reactor Coolant System (described in Section 4.1). Various components in some of these systems are shared by Units 1 and 2.

In some cases, the dependable operation of several systems is required to protect the Reactor Coolant System by controlling system conditions within specified operating limits. Certain systems are required to operate under emergency conditions. Tables 1.3-1 and 1.3-2 list those systems which have shared components and list the safety relationships.

The following systems are considered in this section:

a. Fuel Storage and Handling System - This system provides for handling fuel assemblies, control rod assemblies, and material irradiation specimens.

b. Reactor Auxiliary Systems

   1. The Residual Heat Removal system removes the residual heat from the core and reduces the temperature of the Reactor Coolant System during the second phase of plant cooldown.

   2. The Spent Fuel Pool Cooling system removes the heat generated by spent fuel elements stored in the spent fuel pool.

   3. Chemical and Volume Control System - This system provides for boric acid injection, chemical additions for corrosion control, reactor coolant cleanup and degasification, reactor coolant makeup, reprocessing of water letdown from the Reactor Coolant System, and reactor coolant pump seal water injection.

   4. Reactor makeup water deoxygenation system maintains the dissolved oxygen content of the reactor makeup water within the limits identified in 01193402
      PINGP Chemistry procedures.

c. Plant Service Systems - These systems include fire protection, ventilation, air conditioning, emergency lighting, sampling system, and compressed air.

d. Plant Cooling Systems - The Cooling Water System provides cooling water supplies to the auxiliary feedwater pumps, Unit 1 diesel generators, air compressors, component cooling water heat exchangers, containment fan coil units, and the Auxiliary Building unit coolers. The Component Cooling System removes heat from the Reactor Coolant System, via the Residual Heat Removal System, during plant shutdown, cools the letdown flow to the Chemical and Volume Control System during power operation and provides cooling to dissipate waste heat from various primary plant components and the boric acid and waste evaporators.

e. Equipment and System Decontamination

10.2 REACTOR AUXILIARY SYSTEMS

10.2.1 Fuel Storage And Fuel Handling Systems

The Fuel Handling System provides a safe and effective means of transporting and handling fuel from the time it reaches the plant in an unirradiated condition until it leaves the plant after post-irradiation cooling. The system is designed to minimize the possibility of mishandling or maloperations that could cause fuel damage and potential fission product release.

The Fuel Handling System consists basically of:

a. The reactor refueling cavity, which is flooded only during plant shutdown for refueling.

b. The spent fuel pool, shared by the two units, which is kept full of borated water during and after the first refueling and is always accessible to operating personnel.

c. The Fuel Transfer System, consisting of an underwater conveyor that transports fuel assemblies between the reactor refueling cavity and the spent fuel pool.

d. The Spent Fuel Cask Transfer System.

10.2.1.1 Design Basis

Prevention of Fuel Storage Criticality

Criterion: Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls. (GDC 66)

The new and spent fuel storage racks which have accommodations as defined in Table 10.2-1 are designed so that it is impossible to insert assemblies between rack modules or between rack modules and the spent fuel pool wall. In addition, the spent fuel pool has an area set aside for accepting the spent fuel shipping casks. Cask loading is also done under water. Borated water is used to fill the spent fuel storage pool to maintain $k_{eff} \leq 0.95$. During refueling the boron concentration of the reactor coolant system and the refueling cavity shall be sufficient to ensure that the more restrictive of the following conditions is met:

a. $K_{eff} \leq 0.95$, or
b. Boron concentration ≥ 2000 ppm
c. Shutdown Margin as specified in the Core Operating Limits Report

The boron concentration of the spent fuel pool during refueling is maintained near the concentration of the refueling cavity to prevent dilution of the refueling cavity.

In both the new and spent fuel storage racks, criticality of fuel assemblies is prevented by the design of the rack which limits fuel assembly interaction. In the spent fuel pool, this is also done by specifying allowable storage arrays and maintaining soluble neutron poison in the spent fuel pool water.

The regulatory basis for preventing criticality in the Spent Fuel Pool and New Fuel Pit is 10 CFR 50.68(b), which explicitly provides subcriticality criteria, a uranium enrichment maximum limit, and other conditions. Adopting the requirements of 50.68(b) is an alternative to the criticality accident monitoring requirements of 10 CFR 70.24. The applicable criticality criteria of 50.68(b) are summarized below:

(1) The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level.

(2) If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level.

(3) As Prairie Island has taken credit for soluble boron in the spent fuel pool, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

The criticality analysis “allowance for uncertainties” is described in Reference 39 for the New Fuel Pit, and in Reference 45 for the Spent Fuel Pool.

The Prairie Island spent fuel storage racks were analyzed utilizing the Westinghouse Spent Fuel Rack Criticality Analysis Methodology described in WCAP-17400-NP (Reference 45).

As described in the Prairie Island criticality analysis (Reference 45), spent fuel storage configurations are designed to ensure that the spent fuel rack K_{eff} will be less than 1.0 (including allowance for uncertainties), without the presence of any soluble boron in the storage pool. Soluble boron credit is used to demonstrate a subcriticality margin for a K_{eff} less than or equal to 0.95 (including allowance for uncertainties) for normal conditions and accident conditions as well. In the spent fuel criticality analysis, consideration is given to fuel assembly design, initial enrichment, burn-up, decay time, operating conditions, rack configuration, pool boron concentration, pool temperature and the presence of burnable poisons. The Prairie Island racks have been analyzed to allow storage of all fuel assembly types described in USAR Section 3.3.1. The analysis was completed for a maximum spent fuel pool water temperature of 150 ºF for normal operations (212 ºF for loss of cooling accident conditions). The analysis does not take credit for the presence of the spent fuel rack Boraflex neutron absorber panels which are believed to be in a degraded condition.

In each array shown in Technical Specification Figure 4.3.1-1 the fuel assemblies are first categorized by their relative reactivity by using Technical Specifications Tables 4.3.1-1 through 4.3.1-3. The concept of an acceptable array is to combine the use of low-reactivity fuel and/or empty cells to offset higher-reactivity fuel. Parameters that define a fuel category include initial enrichment, burn-up, decay time, and if it was operated in Cycles 1 through 4 (all fuel operated in cycles 1-4 was conservatively assumed to contain Burnable Poison Rod Assembly inserts (BPRA's) during operation). Each of the analyzed arrays and the array interface requirements are described in the Technical Specification Bases.

Special Fuel Considerations: In addition to the conventional fuel assemblies and Consolidated Rod Storage Canisters that were specifically analyzed and defined in Technical Specifications, specific analyses or evaluations have been performed to ensure that variants from the conventional fuel are stored safely in the spent fuel pool and continue to satisfy criticality criteria. Significant variants are discussed below:

With the exception of the array associated with Consolidated Rod Storage Canisters, all array cells designated as "empty" cannot contain any material that displaces water along the entire active fuel elevation.

The failed fuel baskets in use (i.e, Fuel Rod Storage Canister (FRSC) with 10 or less damaged fuel rods and the Failed Fuel Pin Basket (FFPB)) were modeled to show that they are significantly less reactive than the least reactive fuel. Therefore, these failed fuel baskets may be located in any cell where a fuel assembly qualifies for placement.

The following additional non-fuel components were addressed and may be stored in a cell where any fuel assembly qualifies for placement.

- Miniature Incore detectors and thimble tubes.
- Fuel Inserts stored in the assembly guide tubes (such as RCCA's neutron sources, Thimble Plugs, and BPRA's).
- Any amount of non-fuel material (including depleted filters and dummy fuel assemblies)

Any fuel assembly with an instrument tube removed may be stored as if the instrument tube were present (i.e., no impact on storage limitations).

Any assembly discharged in or after 2R28 and experiencing more than 100 MWd/MTU of lifetime core average exposure at full power with rods inserted may not credit any of that exposure for use in determining the coefficients used to categorize fuel assemblies as defined in Technical Specification. For the purposes of this condition, rodded operation is considered only if rods are inserted more than 7.8 inches into the active fuel length at full power. Additionally, all fuel discharged prior to 2R28 has been evaluated for up to 1000 MWD/MTU of rodded operation without impacting use of the Technical Specification coefficients.

Administrative controls implemented by Technical Specifications 3.7.17 and 4.3 ensure that fuel will be stored in the spent fuel pool in accordance with the above restrictions.

The Prairie Island spent fuel rack criticality analysis also addressed postulated accidents in the spent fuel pool.

The criticality analysis confirmed that most spent fuel pool accident conditions will not result in an increase in K_{eff} of the spent fuel racks. Examples of such accidents are the drop of a fuel assembly on top of a rack, between rack modules, between rack modules and the pool wall, and the drop or placement of a fuel assembly into the cask loading area. At Prairie Island, the spent fuel assembly rack configuration is such that it precludes the insertion of a fuel assembly between rack modules or between rack modules and the pool wall. A dropped fuel assembly can only land on the top of the racks or in the cask loading area.

From a criticality standpoint, the dropped fuel assembly accident assumes a fuel assembly in its most reactive condition is dropped onto the spent fuel racks. The rack structure pertinent for criticality is not excessively deformed. Previous accident analysis with unborated water showed that a dropped fuel assembly which comes to rest horizontally on top of the spent fuel rack has sufficient water separating it from the active fuel height of stored fuel assemblies to preclude neutronic interaction. For the borated water condition, the interaction is even less since the water contains boron, an additional thermal neutron absorber. The analysis in Reference 45 concluded that a dropped fuel assembly laying horizontally on the racks is bounded by the fuel assembly misload accident discussed below. This analysis also concluded the reactivity increase resulting from dropping or placing a fuel assembly into the cask loading area is also bounded by the fuel assembly misload accident discussed below.

However, four accidents that could result in an increase in reactivity have been addressed:

- Loss of Spent Fuel Pool Cooling
- Assembly misload into the storage racks
- Inadvertent removal of an RCCA from Array G
- Boron dilution of the pool

The loss of normal cooling to the spent fuel pool causes an increase in the temperature of the water passing through the stored fuel assemblies. This causes a decrease in water density which reduces boron density and causes a positive reactivity addition. The analysis was completed for a maximum pool temperature of 212 ºF. The evaluation established the amount of soluble boron necessary to ensure that the spent fuel rack K_{eff} will be maintained less than or equal to 0.95. The result was <887 ppm of boron which is lower than the spent fuel pool boron concentration limit contained in Technical Specification 3.7.16 (1800 ppm). Based on the double contingency principle, the margin for accident conditions included in the Technical Specification 3.7.16 boron concentration limit does not have to account for multiple accidents, such as a boron dilution event, at the same time.

A fuel assembly misload accident relates to unintentional placement of fuel in storage locations. Special administrative controls are placed on the patterning of assemblies into spent fuel rack cells. The misloading of a fuel assembly potentially constitutes not meeting the enrichment, burnup or decay time requirements of that cell location. The result of the misloading is to potentially add positive reactivity. An evaluation was completed which established the limiting misloading configuration and the resultant boron concentration to keep $K_{eff} < 0.95$ (887 ppm boron). This is less than the spent fuel pool boron concentration limit contained in Technical Specification 3.7.16 (1800 ppm boron). Based on the double contingency principle, the margin for accident conditions included in the Technical Specification 3.7.16 boron concentration limit does not have to account for multiple accidents, such as a boron dilution event, at the same time.

An inadvertent removal of an RCCA from Technical Specification Figure 4.3.1-1 Array G results in a positive reactivity addition to Array G. The amount of soluble boron required to offset this accident was determined to be bounded by the fuel assembly misload accident.

To address an uncontrolled boron dilution event, calculations were performed in order to define the dilution time and volumes for the spent fuel pool (Reference 48). The dilution sources available at Prairie Island were compiled and evaluated against the calculated dilution volume, to determine the potential of a spent fuel pool dilution event. The evaluations show that a large volume of water (approximately 345,000 gallons) is necessary to dilute the spent fuel pool to a soluble boron concentration well above where criticality would be approached in the spent fuel pool.

Technical Specification 4.3.1.1.c requires that the spent fuel rack K_{eff} be less than or equal to 0.95 when flooded with water borated to 400 ppm. The dilution analysis concluded that large volumes of water are necessary to dilute the spent fuel pool water from the 1800 ppm Technical Specification limit to a boron concentration of 750 ppm. 01500515
The availability of such large water supplies on site is limited. In addition, the transferability of the available water supplies to the pool is very low due to the small number of possible flow paths and in many cases impossible due to the physical arrangement of the spent fuel pool relative to the supplies.

A boron dilution event large enough to result in a significant reduction in the spent fuel pool boron concentration will involve the transfer of a large quantity of water from a dilution source and a significant increase in spent fuel pool level which would ultimately result in pool overflow. Such a large water volume turnover, and the likely overflow of the spent fuel pool, would be readily detected and terminated by plant personnel.

In addition, because of the low dilution flow rates available at Prairie Island, and the large quantities of water required, any significant dilution of the spent fuel pool would only occur over a long period of time (hours to days). Detection of a spent fuel pool dilution via level alarms and/or visual inspections would be expected long before a significant dilution would occur.

Therefore, it is highly unlikely that any dilution event in the spent fuel pool could result in the reduction of the spent fuel pool boron concentration from 1800 ppm to less than the 750 ppm analytical endpoint, which leaves ample margin to the Technical Specification value of 400 ppm and the analyzed value of 359 ppm that are both associated with the limiting normal condition. 01500515

The spent fuel pool dilution analysis assumes thorough mixing of all the non-borated water added to the spent fuel pool. It is unlikely, with cooling flow and convection from the spent fuel decay heat, that thorough mixing would not occur. However, if mixing was not adequate, it would be conceivable that a localized pocket of non-borated water could form somewhere in the spent fuel pool. This possibility is addressed by the criticality analysis (Reference 45) which shows that the spent fuel rack K_{eff} will be less than 1.0 on a 95/95 basis with the spent fuel pool filled with non-borated water. Thus, even if a pocket of non-borated water formed in the spent fuel pool, K_{eff} would be expected to be less than 1.0 anywhere in the pool.

The criticality analysis performed on the Prairie Island new fuel racks (Reference 39) found K_{eff} to be less than 0.95 including uncertainties at a 95/95 probability/confidence level for fuel enriched to a batch average 5.0 wt% U-235 and assuming full density moderation. However, the analysis of the new fuel racks under the low density optimum moderation conditions found that the maximum rack reactivity exceeded the design limit of 0.98 if the new fuel rack 8 by 11 storage cell array was assumed to be completely filled. Further evaluation found that if the center 2x7 array of storage cells were removed, thereby reducing the fuel rack storage capacity to 74 assemblies, (Figure 2, Reference 39), the 0.98 K_{eff} limit would be met under low density optimum moderation conditions. Therefore, the new fuel racks have been modified to preclude the storage of fuel assemblies in the center 2x7 array of storage cells assumed to be empty by the criticality analysis.

During cask handling in the spent fuel pool a minimum of 2450 ppm of boron is maintained. The 2450 ppm boron will ensure that k_{eff} for the spent fuel cask, including statistical uncertainties, will be less than or equal to 0.95 for all postulated arrangements of fuel within the cask. This boron concentration was derived from the criticality analyses for the TN-40HT cask design and conservatively bounds the boron requirements for a TN-40 cask. Detailed instructions are available for use by refueling personnel. These instructions, the minimum operating conditions, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety.

The reactor refueling cavity, fuel transfer canal and spent fuel storage pool are reinforced concrete structures with a seam-welded stainless steel plate liner. These structures are designed as Class I structures to withstand the anticipated earthquake loadings so that the liner prevents leakage even in the event the reinforced concrete develops cracks. In addition, a Class I Fuel Pool Enclosure with a special ventilation system is provided to further reduce the consequences of an accidental release of radioactivity.

10.2.1.2 Description

Various sections of the Fuel Handling System are shared by Units 1 and 2. These include a common spent fuel storage pool and a common new fuel handling and storage area.

The reactor is refueled with equipment designed to handle the spent fuel under water from the time it leaves the reactor vessel until it is placed in a cask for removal from the spent fuel pool. Boric acid is added to the spent fuel pools to ensure subcritical conditions at all times. During refueling the concentration of the spent fuel pools is maintained near the concentration of the refueling cavity to prevent dilution of the refueling cavity. During non-refueling times a minimum of 1800 ppm of boron is required to ensure that spent fuel pool k_{eff} including statistical uncertainties, will be less than or equal to 0.95 with the required storage configurations in place.

The Fuel Handling System, shown in Figure 10.2-1, may be generally divided into two areas: the reactor refueling cavity which is flooded during plant shutdown for refueling and the spent fuel pool which is kept full of water during and after the first refueling and is always accessible to operating personnel. These two areas are connected by the Fuel Transfer System consisting of an underwater conveyor that carries the fuel through the transfer tube.

The reactor refueling cavity is flooded with borated water from the refueling water storage tank. In the reactor refueling cavity, fuel is removed from the reactor vessel, transferred through the water and placed in the fuel transfer system by a manipulator crane. Upon arrival in the spent fuel pool, spent fuel will be removed from the transfer system and placed, one assembly at a time, in storage racks using a long-handled manual tool suspended from the spent fuel pool bridge crane. After sufficient decay, the fuel will be loaded into storage casks for storage in the Independent Spent Fuel Storage Installation or into shipping casks for removal from the site. Both the manipulator crane and the long handled tool can handle only one assembly at a time.

New fuel assemblies are received and normally stored in racks in the new fuel storage pit or in the spent fuel pool. New fuel is delivered to the reactor by lowering it into the spent fuel pool and taking it through the transfer system. Seventy-four fuel assemblies can be stored in the new fuel storage pit.

Fuel handling data is given in Table 10.2-1.

10.2.1.2.1 Major Structures Required for Fuel Handling

a. Reactor Refueling Cavity

The reactor refueling cavity is a reinforced concrete structure that forms a pool above the reactor when it is filled with borated water for refueling. The cavity is filled to a depth that limits the radiation at the surface of the water to 50 milliroentgens per hour during those brief periods when a fuel assembly is transferred over the reactor vessel flange.

The reactor vessel flange is sealed between the bottom of the reactor refueling cavity by an elastomeric seal ring which prevents leakage of refueling water from the refueling cavity. This seal is installed after reactor cooldown but prior to flooding the cavity for refueling operations.

The refueling cavity is large enough to provide storage space for the reactor upper internals, the control cluster drive shafts, and miscellaneous refueling tools. Space is also allowed for the storage of the lower internals if required.

A portion of the cavity is at a lower elevation than the reactor vessel flange to provide the greater depth required for the fuel transfer system tipping device and the rod control cluster changing fixture located in the cavity. The fuel transfer tube penetrates the reactor containment and connects the refueling cavity with the fuel transfer canal.

The floor and sides of the refueling cavity are lined with stainless steel or an equivalent corrosion resistant material.

b. Refueling Water Storage Tank

Refueling water used to fill the reactor refueling cavity is stored in the refueling water storage tank which is described under Section 6.

c. Spent Fuel Storage Pool

The spent fuel storage pool is designed for the underwater storage of spent fuel assemblies and control rods after their removal from the reactor. The spent fuel storage facility is a two-compartment pool that, if completely filled with fuel storage racks, would provide up to 1582 storage locations. With the four storage racks in the southeast corner of pool 1 removed a total of 1386 storage locations would be provided. The storage compartments and the fuel transfer canal are connected by fuel transfer slots that can be closed off with pneumatically sealed gates. The elevations of the slot bottoms are above the tops of the active fuel in the fuel assemblies when stored in the racks.

As shown in Figure 10.2-2 spent fuel assemblies are handled by a long handled tool suspended from an overhead monorail electric hoist and manipulated by an operator standing on a moveable bridge over the pool.

The spent fuel pool structure with its 3 to 6 foot thick walls is designed as Class I structure that fully meets the seismic and tornado design criteria given in Section 12. The fuel pool structure is also designed to withstand the hydraulic pressure of the contained water, as well as other credible static and dynamic loadings.

The fuel pool concrete, reinforcing steel and liner were analyzed to account for the additional loadings imposed by the new racks. The structural adequacy was verified using conventional concrete building codes (ACI 318). Results of the analysis for the most severe loading conditions indicate that the maximum stresses are within the allowables, and that the structural members of the fuel pool are adequate to withstand the additional loads imposed by the new 4/20/81 racks and additional fuel. (Reference 59)

The spent fuel storage structure has been designed to minimize the loss of water due to a dropped cask accident. The bottom of the pool is a reinforced concrete slab 5′-11″ thick that conforms to the ACI Standard Code Requirements for Reinforced Concrete (ACI-318-63) and attains a minimum compressive strength of 4000 psi. All reinforcing bars are made from intermediate grade, new billet steel and conform to A615 Grade 40 specifications. In critically loaded areas, of the slab, the reinforcement is arranged with 3 layers of #11 bar (1 3/8″ dia) at the top and bottom in the N-S direction and 1 layer of #11 bars at top and bottom in E-W direction. In addition, the auxiliary building crane has been upgraded to provide a handling system for handling heavy loads in the spent fuel pool area that satisfies the single-failure-proof guidelines of Section 5.1.6 of NUREG-0612, and thus eliminates the need to analyze the effects of drops of heavy loads per the criteria of Section 5.1 of NUREG-0612.

All inside surfaces of the spent fuel pool are lined with Type 304 stainless steel plate with a minimum thickness of 3/16″. A leakage detection and collection system is also provided.

In the unlikely event of dropping a fuel assembly directly on the pool floor, analysis shows that the structural damage to the floor will be insignificant; leakage through the concrete will not occur. (See Reference 1).

For the purpose of moving objects in the spent fuel pool, a heavy load is defined as any object that weighs more than a fuel assembly and its associated handling tool. Anything that weighs more than this must conform to special heavy load procedures. Any object that weighs less than this may be moved about the spent fuel pool with no special restrictions. (Reference 37).

d. New Fuel Storage

New fuel assemblies and rod control clusters are stored in a separate new fuel storage pit, adjacent to the spent fuel storage pool, as shown in Figure 10.2-2. The new fuel storage pit is designed as a Class I structure that fully meets the seismic design criteria given in Section 12. New fuel though may also be stored in the spent fuel pool. This separate storage pit was originally designed to hold 88 new fuel assemblies in specially constructed racks with parallel rows having a center-to-center distance of 21 inches.

However, a criticality analysis was performed in 1993 to increase the maximum fuel enrichment that could be stored in the new fuel storage racks (Reference 39). That analysis found that under the low density optimum moderation conditions the maximum rack reactivity exceeded the design limit of 0.98 if the entire 88 storage cell array was utilized to store the higher enrichment fuel. Further evaluation found that if the center 2x7 array of storage cells were removed, thereby reducing the fuel rack storage capacity to 74 assemblies, the 0.98 K_{eff} limit would be met under low density optimum moderation conditions. Therefore, the new fuel racks have been modified to preclude the storage of fuel assemblies in the center 2x7 array of storage cells assumed to be empty by the criticality analysis.

e. Fuel Pool Enclosure

The fuel pool enclosure is a Class I reinforced concrete building with 12 to 18 inch thick walls and roof, which is integrally connected to the fuel pool structure. The fuel pool enclosure covers both new and spent fuel storage facilities and is completely contained in the auxiliary building. The design of the fuel pool enclosure is shown in Figure 10.2-3. Normal and special ventilation systems are provided for the fuel pool enclosure as described in Section 10.3.7.

As shown in Figure 10.2-3, the fuel pool enclosure is provided with crane access slots and equipment handling doors which physically limits the area of spent fuel pool over which spent fuel casks or heavy objects can be moved. Administrative procedures prohibit movement of heavy objects when fuel is stored in this area unless one of the following measures is used: 1) a single-failure-proof crane with rigging and procedures which implement Prairie Island commitments to NUREG-0612; or 2) spent fuel pool covers with their implementing plant procedures for installation and use.

10.2.1.2.2 Major Equipment Required for Fuel Handling

a. Reactor Vessel Stud Tensioner

The stud tensioner is a hydraulically operated (oil as the working fluid) device provided to permit preloading and unloading of the reactor vessel closure studs at cold shutdown conditions. Stud tensioners were chosen in order to minimize the time required for the tensioning or unloading operations. Three tensioners are provided and they are applied simultaneously to three studs 120° apart. One hydraulic pumping unit operates the tensioners which are hydraulically connected in parallel. The studs are tensioned to their operational load in two steps to prevent high stresses in the flange region and unequal loadings in the studs. Relief valves are provided on each tensioner to prevent over tensioning of the studs due to excessive pressure.

b. Reactor Vessel Head Lifting Device

The reactor vessel head lifting device consists of a welded and bolted structural steel frame with suitable rigging to enable the crane operator to lift the head and store it during refueling operations. The lifting device is permanently attached to the reactor vessel head.

c. Reactor Internals Lifting Device

The reactor internals lifting device is a structural frame used in moving the reactor internals. The device is lowered onto the guide tube support plate of the internals and is manually bolted to the support plate by three bolts. Bushings on the fixture engage guide studs mounted on the vessel flange to provide close guidance during removal and replacement of the internals package.

d. Manipulator Crane

The manipulator crane is a rectilinear bridge and trolley crane with a vertical mast extending down into the refueling water. The bridge spans the refueling cavity and runs on rails set into the floor along the edge of the refueling cavity. The bridge and trolley motions are used to position the vertical mast over a fuel assembly in the core. A long tube with a pneumatic gripper on the end is lowered out of the mast to grip the fuel assembly. The gripper tube is long enough so the upper end is still contained in the mast when the gripper end contacts the fuel. A winch mounted on the trolley raises the gripper tube and fuel assembly up into the mast tube. The fuel is transported while inside the mast tube to its new position. The manipulator can lift only one fuel assembly at a time.

All controls for the manipulator crane are mounted on a console on the trolley. The bridge and trolley are positioned by marks on a mechanical pointer and viewed by the operator on a monitor at the console. The drives for the bridge, trolley, and winch are variable speed and include a separate inching control. Electrical interlocks and limit switches on the bridge and trolley drives protect the equipment. In an emergency, the bridge, trolley, and winch can be operated manually using a handwheel on the motor shaft.

Safety features are incorporated in the system as follows:

1. Travel limit switches on the bridge and trolley drives.

2. Bridge, trolley, and winch drives which are mutually interlocked to prevent simultaneous operation of any two drives.

3. A position safety switch, the GRIPPER TUBE UP position switch, which prevents bridge and trolley main motor drive operation except when it is actuated.

4. An interlock which prevents the opening of a solenoid valve in the air line to the gripper except when zero suspended weight is indicated by a force gauge. As back-up protection for this interlock, the mechanical weight actuated lock in the gripper prevents operation of the gripper under load even if air pressure is applied to the operating cylinder.

5. An EXCESSIVE SUSPENDED WEIGHT trip, which opens the hoist drive circuit in the UP direction when the loading is in excess of either preset limit. The appropriate preset limit, LIGHT or HEAVY as dictated by type of fuel movement, is selected with a 2-position switch.

6. A reduced intermediate hoist speed to reduce fuel assembly grid interaction consequences while inserting a fuel assembly into a core location.

7. A redundant full UP interlock to prevent crane movement when an assembly is not fully inside the protective mast.

8. A hoist underload trip to prevent grid damage on DOWN travel when excessive resistance is encountered. The appropriate preset underload limit, LIGHT or HEAVY as dictated by type of fuel movement, is selected with a 2-position switch.

9. An interlock on the hoist drive circuit in the up and down directions, which permits the hoist to be operated only when either the OPEN or CLOSED indicating switch on the gripper is actuated.

10. An interlock of the bridge and trolley drives, which prevents the bridge drive from traveling beyond the edge of the core unless the trolley is aligned with the refueling canal centerline. The trolley drive is locked out when the bridge is beyond the edge of the core.

    Suitable restraints are provided between the bridge and trolley structures and their respective rails to prevent derailing and the manipulator crane is designed to prevent disengagement of a fuel assembly from the gripper in the event of a Design Basis earthquake.

e. Spent Fuel Pool Bridge Crane

   The spent fuel pool bridge crane is a wheel-mounted walkway, spanning the spent fuel pool which carries electric monorail hoists on an overhead structure. The fuel assemblies are moved within the spent fuel pool by means of a long handled tool suspended from the hoist. The hoist travel and tool length are designed to limit the maximum lift of a fuel assembly to a safe shielding depth.

The spent fuel pool bridge crane has 2 hoists, designated the East Hoist and the West Hoist. The East Hoist is a 2-ton capacity hoist used for general fuel handling and Unit 1 refuelings. The West Hoist is a redundant, 3-ton design rated load (DRL), 3700 pound maximum critical load (MCL) hoist used for general fuel handling, Unit 2 refuelings, and heavy load lifting. The hoist is specifically designed for moving the heavy loads contained in the spent fuel pool enclosure, e.g., the pool divider gates (2600 lbs). In addition, the West Hoist has the capability to transfer out to the New Fuel Crane located outside the pool for movement of new fuel assemblies from the new fuel containers to the New Fuel Pit.

f. The Auxiliary Building Crane

The Prairie Island Auxiliary Building Crane, which is used for handling spent fuel casks, has been upgraded to be in compliance with Section 5.1.6 and Appendix C of NUREG-0612, Control of Heavy Loads at Nuclear Power Plants. This upgrade to the auxiliary building crane has been made to provide a handling system for handling heavy loads in the spent fuel pool area that satisfies the single-failure-proof guidelines of Section 5.1.6 of NUREG-0612, and thus eliminates the need to analyze the effects of drops of heavy loads per the evaluation criteria of Section 5.1 of NUREG-0612. The original crane was designed to handle a load of 125 Tons. The main hoist capacity has not changed with the upgrade, but the auxiliary hoist capacity has been changed from 25 Tons to 15 Tons. The single-failure-proof features have not been incorporated into the design of the auxiliary hoist. All crane structural members have been designed to withstand impact loads per applicable specifications. A seismic evaluation has been performed for the loaded condition. Numerous safety features have been incorporated into the new design of the crane. Among these are the following:

1. Design of the hoist cables are in accordance with that required by Ederer Generic Licensing Topical Report EDR-1(P)-A (Reference 38).

2. All parts subjected to dynamic strains such as gears, shafts, drums, blocks, and other integral parts have a safety factor of five.

3. Two separate magnetic brakes are provided as well as an emergency drum band brake. Each magnetic brake provides a braking force of at least 150% of rated load. The emergency drum brake assures that the load can be safely lowered even if power is lost to the crane.

4. The crane is capable of raising, lowering and transporting occasional loads, for testing purposes, of 125 percent of rated load without damage or distortion to any crane part.

5. The crane is provided with a balanced dual reeving system with each wire rope capable of supporting the maximum critical load. The hydraulic load equalizing system allows transfer of the load to the remaining rope, without overstressing it, in the event of a failure of one rope.

6. The main hook has a design safety factor of 10 and was subjected to a 200% overload test followed by magnetic particle inspection.

7. The five step motor control was retained for the bridge and trolley drives. D.E. stepless drive systems have been provided for the main and auxiliary hoists.

8. Both the main and the auxiliary hoists are provided with field weakening for increased hoisting and lowering speeds at reduced loads. Unloaded speed is 200% greater for an unloaded hoist than a fully loaded hoist.

9. The main hoist has been provided with a Hoist Integrated Protection System (HIPS) which includes:

   A. Energy Absorbing Torque Limiter (EATL) - The EATL limits the load imposed on the wire rope in case of load hangup or two-blocking. The kinetic energy on the rotating equipment is dissipated in the form of heat while protecting the reeving system.

   B. Emergency Drum Brake System - The Emergency Drum Brake System is activated by the Failure Detection System or by manual operation of the emergency stop pushbutton.

   C. Failure Detection System - The Failure Detection System is actuated by a loss of mechanical continuity, detection of actuation of the EATL (two-blocking or load hangup), improper rope spooling, broken wire rope, reeving continuity and by drum overspeed.

10. Each hoist has a load cell to protect against load hangup and overload. This is part of the conventional hoist safety system and is not safety related.

11. End of travel limits for bridge and trolley motion have been added.

12. Limit switches have been added to the area of the slot in the roof of the fuel pool enclosure. These limits protect the wire rope by restricting movement of the trolley while operating the main hoist in the slot area.

13. A fail-safe remote radio control system is provided for the crane. The radio controls parallel the master switch connections and all of the safety features built into the control system also apply when the radio transmitter is used. The radio has a very complex system for transmitting and receiving signals so it is impossible to duplicate this signal by any other means. Also, the signal differs for every crane so the transmitter for some other crane would not actuate the receiver panel for this crane.

g. Fuel Transfer System

The fuel transfer system, shown in Figure 10.2-1 is an underwater conveyor car that runs on tracks extending from the refueling canal through the transfer tube and into the fuel transfer canal. The conveyor car receives a fuel assembly in the vertical position from the manipulator crane. The fuel assembly is lowered to a horizontal position for passage through the tube, and then is raised to a vertical position in the fuel transfer canal.

During plant operation, the conveyor car is normally stored in the fuel transfer canal. A blind flange is bolted on the refueling canal end of transfer tube to seal the reactor containment. The terminus of the tube outside the containment is closed by a gate valve.

h. Spent Fuel Storage Racks

Two sizes of spent fuel storage racks are provided; a 7x7 space rack and a 7x8 space rack. Each storage rack consists of storage tubes interconnected with each other through their upper and lower grids. These grids also ensure proper location of the storage tubes on 9.5 inch pitch in both directions. The upper and lower grids are tied together by vertical and diagonal members. Reactivity control is provided by the 9.5 inch storage tube pitch and by fuel loading restrictions described in Technical Specifications. Each storage tube consists of three components: an inner type 304 stainless steel tube, a layer of neutron absorbing material, and an outer skin of type 304 stainless steel. The neutron absorber material is believed to be degraded and is therefore not credited in the spent fuel pool criticality analysis. The rack base is composed of heavy box beams connected at the four corners to box section legs with adjustable feet. The box beams of the base are elevated above the pool floor to allow flow of cooling water below the rack and up into the storage tubes. All material used for rack construction is made of Type 304 stainless steel with the exception of the neutron absorber which is composed of a silicon polymer base material with sufficient boron in the form of boron carbide (Boraflex). This polymer material is degraded and no credit is presently taken for this neutron absorber in criticality analyses.

i. Rod Cluster Control Changing Fixture

A fixture is mounted on the refueling cavity wall for removing rod cluster control (RCC) assemblies from spent fuel assemblies and inserting them into new fuel assemblies. The fixture consists of two main components; a guide tube mounted to the wall for containing and guiding the RCC assemblies, and a wheel-mounted carriage for holding the fuel assemblies and positioning fuel assemblies under the guide tube. The guide tube contains a pneumatic gripper on a winch that grips the RCC assemblies and lifts it out of the fuel assembly. By repositioning the carriage, a new fuel assembly is brought under the guide tube and the gripper lowers the RCC element and releases it. The manipulator crane loads and removes the fuel assemblies into and out of the carriage.

j. Pool No. 1 Protective Cover

In order to protect fuel assemblies in the small spent fuel pool against the accidental drop of a heavy load, a protective cover over the pool is provided. The pool cover is made of 3/16 inch stainless steel plate welded to a grid of structural tees and built-up wide-flange beams which are made of structural steel ASTM A588 Grade A. Underneath each end of the beams, one pad made of one-inch thick compressible material is used between the cover and the concrete floor.

An evaluation was performed on the protective cover when subjected to a postulated drop of 24,800 pounds at a height of 6 inches above the cover.

The results of the evaluation show that although local plastic deformation may occur, the overall structural integrity of the cover will be maintained. Thus, the effect of the postulated drop of this heavy load is considered to be within the acceptable limit.

The spent fuel pool covers are installed and removed in accordance with plant procedures which assure that the covers can not be dropped into the spent fuel pool.

k. Spent Fuel Cask

Prairie Island has an Independent Spent Fuel Storage Installation (ISFSI) licensed under 10 CFR Part 72, issued under materials license SNM-2506. License SNM-2506 authorizes storage of 715.29 TeU of spent fuel assemblies using the TN-40 and TN-40HT storage cask design. Prairie Island also has a General License to use Certified storage cask designs under the provisions of 10 CFR 72.210.

Prairie Island has a General License to use Certified spent fuel transportation casks under the provisions of 10 CFR71.12.

10.2.1.2.3 Refueling Sequence of Operation

In preparation for refueling, the following typical sequence of events occurs:

a. The reactor is shut down and cooled to ≤200°F (Mode 5).

b. A radiation survey is made and the containment vessel is entered.

c. CRDM cables are disconnected.

d. Reactor vessel head insulation and instrument leads are removed.

e. Checkout of the fuel transfer device and manipulator crane is started.

f. Reactor vessel water level is lowered to a foot below the reactor vessel flange.

g. The reactor vessel head nuts are loosened with the hydraulic tensioners.

h. The reactor vessel head studs are removed to storage.

i. Guide studs are installed in three holes and the remainder of the stud holes are plugged.

j. The reactor vessel to cavity seal ring is installed.

k. Final preparation of underwater lights and tools is made. Checkout of manipulator crane and fuel transfer system is completed.

l. The fuel transfer tube flange is removed.

m. The reactor vessel head is unseated and raised a sufficient amount to ensure that rod drive shafts are disengaged, then the reactor vessel head is taken to the storage pedestal.

n. The refueling cavity is filled with water from the refueling water storage tank. The normal Residual Heat Removal System inlet valves from the Reactor Coolant System are closed.

o. When the cavity is filled, the residual heat removal system is restored to normal operation.

p. The full length control rod drive shafts are unlatched.

q. The reactor vessel internals lifting rig is lowered into position and latched to the support plate.

r. The reactor vessel flange protector ring is disengaged from the internals lifting rig.

s. The reactor vessel internals are lifted out of the vessel and placed in the underwater storage rack.

t. The core is now ready for refueling.

The refueling sequence is started utilizing the manipulator crane. The general sequence for fuel assemblies in noncontrol positions is as follows:

a. Spent fuel is removed from the core and placed into the fuel transfer system for removal to the spent fuel pool.

b. Partially spent fuel is rearranged in the core.

c. Replacement fuel assemblies are brought in from the new fuel storage pit through the transfer system and loaded into the core.

d. Whenever new fuel is added to the reactor core, a reciprocal curve of source range neutron count rate is plotted to verify the sub-criticality of the core.

The refueling sequence is modified for fuel assemblies containing Rod Cluster Control (RCC) assemblies, as required. If a transfer of the RCC assembly between fuel assemblies is required, the RCC assembly from one spent-fuel assembly is transferred to another fuel assembly. Such an exchange is required whenever a spent fuel assembly containing RCC assemblies is removed from the core and whenever a fuel assembly is placed in or taken out of a control position during refueling rearrangement.

After refueling is completed, the reactor is prepared for operation by essentially reversing the above sequence (items a. through t.) which was followed in preparing for the refueling.

10.2.1.3 Performance Analysis

Underwater transfer of spent fuel provides essential ease and safety in handling operations. Water is an effective, economic, and transparent radiation shield and a reliable cooling medium for removal of decay heat.

Basic provisions to ensure the safety of refueling operations are:

a. Gamma radiation levels in the containment refueling area and fuel storage areas are continuously monitored (see Section 7.5.2). These monitors provide an audible alarm at the initiating detector indicating an unsafe condition. Continuous monitoring of reactor neutron flux provides immediate indication and alarm in the control room of an abnormal core flux level.

b. Violation of refueling integrity per IT.S.3.9 is not permitted when the reactor vessel head is removed.

c. Whenever new fuel is added to the reactor core, a reciprocal curve of source range neutron count rate is plotted to verify the sub-criticality of the core.

d. The design of the spent fuel storage facility incorporates a Class I fuel pool enclosure for additional protection against the effects of a tornado.

   In all, four barriers protect stored spent fuel against the effects of a tornado. They are:

   1. The Class I fuel pool structure
   2. The twenty-five feet of water that normally covers the spent fuel
   3. The fuel storage racks
   4. The Class I fuel pool enclosure

   The potential effects of a tornado striking the fuel storage pool of existing light-water reactors, i.e., with no Class I fuel pool enclosure, has been thoroughly investigated (References 2, 3, 4). Two key areas were examined:

   a. whether sufficient water could be removed from the pool to prevent cooling of the fuel and

   b. whether missiles could potentially enter the pool and damage the stored fuel.

As concluded by investigation (References 2, 3, 4), existing fuel pool designs are inherently capable of withstanding the effects of a tornado. The design of the fuel pool makes the removal of more than five feet of water due to tornado action highly improbable. With 25 feet of water covering the fuel, the removal of five feet of water is of no concern. Protection is provided against a wide spectrum of tornado-generated missiles by the water which covers the fuel racks.

Results of an evaluation of a spectrum of tornado-driven missiles indicates that a four inch or smaller pipe, wooden debris from the cooling towers or metal panel siding ripped off the building are the most probable sources of missiles.

Such debris would not damage the fuel because of the buoyant force exerted by the twenty-five feet of water covering the fuel. Only by arbitrarily assuming long cylindrical objects to be hurled to the fuel pool by winds acting on their maximum cross-sectional area and then impacting the pool with minimum cross-sectional area could a potential for damage be shown (Reference 3). The probability of such an event has been shown to be less than 10^{-7}. Even in this highly unlikely case, a wide spectrum of these missiles can hit the pool without resulting in fuel damage or liner penetration.

In addition to the inherent protection provided by the fuel pool structure, the 25 feet of water covering the spent fuel, and the fuel storage racks, a Class I fuel pool enclosure has been provided for additional protection. It is, therefore, concluded that adequate protection for tornado-generated missiles and potential pool dewatering has been provided.

e. The auxiliary building crane has been upgraded to provide a handling system for handling heavy loads in the spent fuel pool area that satisfies the single-failure-proof guidelines of Section 5.1.6 of NUREG-0612, and thus eliminates the need to analyze the effects of drops of heavy loads per the criteria of Section 5.1 of NUREG-0612.

The maximum load motion following a drive train failure on the single-failure-proof main hoist of the auxiliary building crane is less than 1.5 foot and the maximum kinetic energy of the load is less than that resulting from 1 inch of free fall of the maximum critical load. The spent fuel pool floor has been analyzed and has been shown to be able to withstand the impact of 1 inch of free fall of the maximum critical load.

Incident Protection

Direct communication between the control room and the refueling cavity manipulator crane operator is available whenever changes in core geometry are taking place. This provision allows the control room operator to inform the manipulator crane operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

Malfunction Analysis

An analysis is presented in Section 14 concerning damage to one complete assembly, assumed as a conservative limit for evaluating environmental consequences of a fuel handling accident.

Effect of Fuel Failure on the Spent Fuel Pool

Experience indicates that there is little radionuclide leakage from spent fuel stored in pools after the fuel has cooled for several months. The predominance of radionuclides in the spent fuel pool water appears to be radionuclides that were present in the reactor coolant system prior to refueling (which becomes mixed with water in the spent fuel pool during refueling operations) or crud dislodged from the surface of the spent fuel during transfer from the reactor core to the SFP. During and after refueling, the spent fuel pool cleanup system reduces the radioactivity concentrations considerably. It is theorized that most failed fuel contains small, pinhole-like perforations in the fuel cladding at the reactor operating condition of approximately 800°F. A few weeks after refueling, the spent fuel cools in the spent fuel pool so that fuel clad temperature is relatively cool, approximately 180°F. This substantial temperature reduction should reduce the rate of release of fission products from the fuel pellets and decrease the gas pressure in the gap between pellets and clad, thereby tending to retain the fission products within the gap. In addition, most of the gaseous fission products have short half-lives and decay to insignificant levels within a few months.

Based on the operational experience there has not been any significant leakage of fission products from spent light water reactor fuel stored in the Morris Operation (MO) (formerly Midwest Recovery Plant) at Morris, Illinois, or at the Nuclear Fuel Services' (NFS) storage pool at West Valley, New York. Spent fuel has been stored in these two pools which, while it was in a reactor, was determined to have significant leakage and was therefore removed from the core. After storage in the onsite SFP, this fuel was later shipped to either MO or NFS for extended storage. Although the fuel exhibited significant leakage at reactor operating conditions, there was no significant leakage from this fuel in the offsite storage facility.

Experience indicates that there is little radionuclide leakage from Zircaloy-clad spent fuel stored in pools for decades. Operators at several reactors have discharged, stored, and/or shipped relatively large numbers of Zircaloy-clad fuel elements which developed defects during reactor exposure. Several hundred Zircaloy-clad assemblies which developed one or more defects in-reactor are stored in the MO pool without need for isolation in special cans. Detailed analysis of the radioactivity in the pool water indicates that the defects are not continuing to release significant quantities of radioactivity.

A Battelle Northwest Laboratory (BNL) report states that radioactivity concentrations may approach a value up to 0.5 μCi/ml during fuel discharge in the SFP. After the refueling, the SFP ion exchange and filtration units will reduce and maintain the pool water in the range of 10^{-3} to 10^{-4} μCi/ml.

In handling defective fuel, the BNL study found that the vast majority of failed fuel does not require special handling and is stored in the same manner as intact fuel. Two aspects of the defective fuel account for its favorable storage characteristics. First, when a fuel rod perforates in-reactor, the radioactive gas inventory is released to the reactor primary coolant. Therefore, upon discharge, little additional gas release occurs. Only if the failure occurs by mechanical damage in the pool are radioactive gases released in detectable amounts, and this type of damage is extremely rare. In addition, most of the gaseous fission products have short half-lives and decay to insignificant levels. The second favorable aspect is the inert character of the uranium oxide pellets in contact with water. This has been determined in laboratory studies and also by casual observations of pellet behavior when broken rods are stored in pools.

10.2.1.4 Inspection and Testing

Upon completion of core loading and installation of the reactor vessel head, certain mechanical and electrical tests are performed prior to criticality. The electrical wiring for the rod drive circuits, the rod position indicators, the reactor trip circuits, and the incore thermocouples are tested before plant operation.

Following the failure of a fuel assembly’s stainless steel sleeve (which connects the guide tubes to upper nozzle) and the resulting dropping of a fuel assembly, the NRC recommended in a letter dated July 11, 1983 [Ref. 61] that a sulfate analysis be included in the routine primary and spent fuel pool water chemistry monitoring and analysis program. Fuel assemblies should also be periodically surveyed for evidence of intergranular stress corrosion cracking.

10.2.1.5 Spent Fuel Consolidation Demonstration Project

10.2.1.5.1 Introduction

NSP and Westinghouse conducted a fuel consolidation demonstration at Prairie Island in the fall of 1987. The goal of fuel consolidation was to store more fuel in a given pool space, thereby increasing the overall spent fuel pool storage capacity. The consolidation process entailed removal of all the fuel rods from two assemblies, reconfiguring them into a close-packed triangular array, and then placing them into a canister of about the same outside dimensions as an assembly. The canister was then stored in a rack cell formerly occupied by a single spent fuel assembly.

10.2.1.5.2 Safety Evaluation

The demonstration project was limited to a maximum of 50 assemblies. The Prairie Island Operations Committee reviewed the project safety evaluation and determined that it could be performed under the provisions of 10CFR50.59, i.e., it did not involve an unreviewed safety question, and no Technical Specification changes were required. Reference 34 transmitted to the NRC the project description, safety evaluation and project topical report. The demonstration project would not increase the amount of spent fuel stored in the pool, it would alter the configuration of some fuel already being stored in the pool. Following is a summary of the analyses performed as part of the safety evaluation; refer to Reference 34 for further details.

10.2.1.5.2.1 Criticality

During the consolidation process, the fuel configuration was significantly changed. A criticality analysis was performed to verify that criticality would not occur during the process in both normal and accident conditions. A more recent criticality analysis addresses storage of the Consolidated Rod Storage Canisters (CRSCs) in the array shown in Technical Specifications Figure 4.3.1-1. This analysis also addressed storage 01450457
of partially loaded CRSCs containing up to 10 fuel rods

10.2.1.5.2.2 Thermal Hydraulic

A thermal hydraulic analysis of the consolidated fuel storage canister was performed to determine the maximum allowable spent fuel decay heat rate which would satisfy the design criteria. The design criteria was to avoid boiling of the pool water under normal conditions, at any time during the process. The analysis determined the required cooling time, as a function of burnup, for any assembly to be consolidated. This also became a limiting condition for consolidation.

10.2.1.5.2.3 Radiological Consequences of Handling Accidents

This analysis evaluated the radiological consequences of a handling accident during consolidation. The analysis assumed total release from two assemblies worth of spent fuel rods, and that the fuel had been out of the reactor for two years. The results of the analysis showed that the radiological consequences of such an accident are much less than the design basis accident, which involves total release from a freshly discharged assembly. A minimum cooling time of two years became a limiting condition for consolidation.

10.2.1.5.2.4 Rack Structural Analysis

The structural capacity of a rack cell to contain a full consolidated fuel storage canister during a seismic event was evaluated and found to be adequate. Placement of the canisters into rack cells for storage was controlled so that no rack contained more mass than what was assumed in the existing licensing basis analysis.

10.2.1.5.3 Demonstration Results

Prior to beginning consolidation, the fuel assemblies were sipped to ensure there were no leaking fuel rods. The candidate assemblies were all of the Westinghouse standard design and included assemblies from regions A, B and C of both units, plus one region D and one region E assembly. The region D assembly was one which exhibited severe rod bow. Assembly average exposures ranged from 28 to 40 GWD/MTU. The minimum cooling time was approximately three years, and most assemblies had been out of core for about ten years.

The Westinghouse consolidation equipment and process is explained in the project topical report (Reference 34). Fuel consolidation began at Prairie Island on October 9, 1987 and continued through November 13, 1987. During this period, 36 spent fuel assemblies were consolidated into 18 canisters. The 18 canisters were placed into a 7 x 8 storage rack in a checkerboard pattern. The assembly cage components were placed into rack storage cells to await later processing.

The cage processing effort included waste classification of the cage components, volume reduction, and appropriate segregation and packaging of the waste material for either in pool storage or disposal. Volume reduction of 6 assembly cages was performed. These assembly cages were reduced to 2 cans which are now stored in the spent fuel pool along with other cage components.

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10.2.2 Spent Fuel Pool Cooling System

10.2.2.1 Design Basis

The spent fuel pool cooling system is designed to remove, from the shared spent fuel pool, the heat generated by stored spent fuel elements.

System design does not incorporate redundant components except for the spent fuel pool pump and the heat exchanger. Alternate cooling capability can be made available under anticipated malfunctions or failures. System piping is so arranged that failure of any pipeline does not drain the spent fuel pool below the top of the stored spent fuel elements.

The system is capable of handling a maximum heat load corresponding to both pools being filled with a combined total of 1362 normally discharged fuel assemblies plus a freshly off loaded core consisting of 121 fuel assemblies.

All piping and components of the Spent Fuel Cooling System are designed to the applicable codes and standards listed in Table 10.2-2. Design data for all major components is listed in Table 10.2-3.

10.2.2.2 Description

Spent fuel is placed in the spent fuel pool during refueling and is stored until it is placed in dry cask storage or shipped to a government repository. Residual heat from the stored fuel is removed by the spent fuel pool cooling system.

The spent fuel pool cooling system consists of two pumps, two heat exchangers, three filters, demineralizer, piping and associated valves and instrumentation. The spent fuel pool pumps draw water from the pool, circulate it through a heat exchanger and return it to the pool. Component cooling water cools the heat exchanger.

The system design incorporates redundant spent fuel pool pumps and heat exchangers. One of the heat exchangers normally serves as an alternate heat sink in the highly unlikely loss of the operating heat exchanger. System piping is arranged to permit manual cut-in of the backup heat exchanger.

The clarity and purity of the spent fuel pool water is maintained by passing approximately 60 gpm of the system flow through a purification loop. The purification loop consists of the two demineralizer inlet filters, the demineralizer, and a demineralizer outlet filter. The two demineralizer inlet filters can also be used independently of the demineralizer by placing the filters in a bypass configuration. Additionally the purification loop can be used to maintain the purity in both units' RWSTs. A separate RWST purification pump for each unit circulates the refueling water storage tanks through the purification loop.

The spent fuel pool pump suction line is located above the fuel assemblies which prevents uncovering fuel assemblies during loss of water as a result of a possible suction line rupture.

The spent fuel pool cooling system is shown in Figure 10.2-4. Description of the major components is discussed below.

a. Heat Exchangers

The spent fuel pool heat exchangers are of the shell and U-tube type with the tubes welded to the tube sheet. Component cooling water circulates through the shell, and spent fuel pool water circulates through the tubes. The tubes are austenitic stainless steel and the shell is carbon steel.

b. Pumps

One or both of the two spent fuel pool pumps circulate water in the spent fuel pool cooling system. All wetted surfaces of the pumps are austenitic stainless steel, or equivalent corrosion resistant material. Pump operation is manually controlled from a local station. The flow in the spent fuel pool cooling line is not continuously monitored. This monitoring is not required for safety or operation. The local instrument is sufficient. The high spent fuel pool temperature alarm is set low enough to give the operator ample time (hours) to make corrections.

c. Strainer

A stainless steel strainer is located at the inlet of each spent fuel pool system suction line for removal of relatively large particles which might otherwise clog the spent fuel pool demineralizer.

d. Filter

Two spent fuel pool filters are located upstream of the spent fuel pool demineralizer to remove particulates in the spent fuel pool water before it goes into the demineralizer. Another spent fuel pool filter is provided downstream of the spent fuel pool demineralizer to remove any particulate matter from the spent fuel pool water coming from the demineralizer.

e. Demineralizer

The demineralizer is sized to pass approximately 60 gpm of the system circulation flow, to provide adequate purification of the spent fuel pool water for unrestricted access to the working area, and to maintain optical clarity.

f. Skimmer

A skimmer pump and filter are provided for surface skimming of the spent fuel pool water. Flow from this pump is returned to the spent fuel pool.

g. Valves

Manual stop valves are used to isolate equipment and lines, and manual throttle valves provide flow control. Valves in contact with spent fuel pool water are austenitic stainless steel or equivalent corrosion resistant material.

h. Piping

All piping in contact with spent fuel pool water is austenitic stainless steel. The piping is welded except where flanged connections are used to facilitate maintenance.

10.2.2.3 Performance Analysis

Whenever a leaking fuel assembly is transferred from the fuel transfer canal to the spent fuel storage pool, a small quantity of fission products may enter the spent fuel pool water. A purification process consisting of filtration and ion-exchange is provided for removing these fission products and other contaminants from the water.

The most serious failure of this system is complete loss of water in the storage pool. To protect against this possibility, piping connections enter the top of the spent fuel pool as stated above except for the drain connection from the transfer canal to the holdup tank recirculation pump. Even if the water in the transfer canal were completely drained, the active portion of the spent fuel would not be uncovered due to the elevation of the bottom of the gate connection in the wall separating the transfer canal from the spent fuel pool. Complete siphon draining of the pool is impossible. The spent fuel pool pump suction connection only goes approximately 4 feet below the normal water level. The cooling water return line which extends 10 feet below the normal water level is prevented from siphon draining the pool by a 0.5 inch hole in the pipe located 4 feet below the normal water level.

The spent fuel pool cooling system consists of two pumps and two heat exchangers. These are cross-connected such that the loss of any one pump or heat exchanger does not prevent the operation of the remaining components. The decay heat is removed from the spent fuel pool heat exchangers by either Unit 1 or Unit 2's Component Cooling Water System. The Unit 1 Component Cooling system supplies the 122 Spent Fuel Pool Heat Exchanger and the Unit 2 Component Cooling System supplies the 121 Spent Fuel Pool Heat Exchanger. Cross-connect valves exist to allow either Spent Fuel Pool to be supplied by either Unit of Component Cooling. Each unit's Component Cooling Water System consists of two, 100% capacity, parallel loops each comprised of one pump and one heat exchanger having a rating of $29x10^6$ BTU/HR. In the unlikely event that a LOCA should occur in the unit whose Component Cooling Water System is connected to the Spent Fuel Pool Cooling System, the operator would conservatively have more than one hour to transfer the pool cooling system to the unaffected unit's Component Cooling Water System. 01422041

The free volume of pool #2 is slightly less than 2.5 times the free volume of pool #1. The volume of water, storage racks and spent fuel in the respective pools are essentially in the same ratio. The water volumes in pool #1 and pool #2 are 12,125 ft^3 and 28,477 ft^3 respectively. Each pool has a high and low water level sensor which provides input to each control room's control board. The spent fuel pool liner seam leakage is directed to a common open sight drain trough for monitoring and then to the waste disposal system.

Temperature detectors are installed at both pools. A high temperature alarm for each pool, nominally set at 120°F, is located on each control room alarm panel. The auxiliary building operator, as a routine shift responsibility, monitors the spent fuel pool water level, temperature, radiation and the leak detection system. The control room operators, as part of their routine shift responsibility, monitor the spent fuel pool radiation levels.

In 1980 the pumps, piping and valving were modified to increase the system heat removal capacity. The heat removal capability was re-analyzed based on a pool heat load determined using the methods in NRC Branch Technical Position ASB 9-2. These were reviewed and approved by the NRC in Reference 59. Subsequently, 122 heat exchanger was replaced with a heat exchanger with the same capability as 121 heat exchanger. New analyses were performed using the same method for determining pool heat load (References 74 and 75). The assumed conditions and resultant maximum pool water temperatures are as follows (the peak heat load in the pool is specified in Table 10.2-3):

a. NORMAL - 1362 normally discharged fuel assemblies are stored in pools 1 and 2. The pools are cooled by a heat exchanger with one pump. The maximum pool water temperature will not exceed 140°F.

b. ABNORMAL - 1362 normally discharged fuel assemblies plus a freshly off loaded core consisting of 121 fuel assemblies are stored in pools 1 and 2. The pools are cooled by both heat exchangers and both pumps. The maximum pool water temperature will not exceed 200°F.

c. FAULTED - 1362 normally discharged fuel assemblies plus a freshly off loaded core consisting of 121 fuel assemblies are stored in pools 1 and 2. The analyses assume the failure of one pump. The maximum pool water temperature will not exceed 200°F.

The elapsed time before pool boiling following the loss of all external pool cooling was investigated (Reference 76).

The two postulated conditions and results are as follows:

a. Pools 1 and 2 contain 1362 normally discharged fuel assemblies plus a full core discharge, and it was assumed that complete mixing of the water in pools 1 and 2 would occur. Calculations indicate that boiling would occur in greater than 8 hours. Further, since this is the maximum heat load the maximum boil off rate would be 65.6 gpm at 212°F or requiring a minimum make-up rate of 66 gpm to maintain pool inventory.

b. Pool #1 contains one full core discharge plus 266 recently discharged fuel assemblies. It was further assumed that there would be no mixing of pool 1 water with pool 2 water. Calculations indicate that boiling would occur in 3.8 hours.

Eight hours is an adequate amount of time to perform minor maintenance on the cooling system and restore it to an operable condition. In order to avoid the short boil off time in the second postulated case, in the event that all of the fuel assemblies (121) or a majority are discharged from the reactor into the spent fuel pool, no more than 45 of the assemblies will be placed in pool #1 and the remaining fuel assemblies be placed in pool #2.

In the unlikely event that all spent fuel pool cooling is lost and boiling occurs, at least six sources of makeup water are available.

One hour or less is required to line up the valves or to carry out the steps necessary in order to make the water available. Six sources of makeup water and their makeup pump or hose station ratings are as follows: (a) Chemical and Volume Control System - 300 gpm, (b) Chemical and Volume Control System Blender - 100 gpm, (c) Refueling Water Storage Tank - 80 gpm, (d) Reactor Makeup Storage Tanks - 80 pm, (e) five demineralized water hose stations, each station rated at 20 gpm, and (f) the fire protection system - there are two fire hose stations near the spent fuel pool each rated at 95 gpm.

A failure analysis of components is given in Table 10.2-4.

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10.2.3 Chemical and Volume Control System

The Chemical and Volume Control System a) adjusts the concentration of chemical neutron absorber for chemical reactivity control, b) maintains the proper water inventory in the Reactor Coolant System, c) provides the required seal water flow for the reactor coolant pump shaft seals, d) processes reactor coolant letdown through filtration and ion exchange, e) maintains the proper concentration of corrosion inhibiting chemicals in the reactor coolant and f) keeps the reactor coolant activity to within design levels. The system is also used to fill and hydrostatically test the Reactor Coolant System. During normal operation, therefore, this system has provisions for supplying:

a. Hydrogen to the volume control tank

b. Nitrogen as required for purging the volume control tank

c. Hydrazine or pH control chemical, as required, via the chemical mixing tank to the charging pumps suction.

d. Hydrogen Peroxide injection, as required, to the RCS via the charging line to the regenerative heat exchanger. 01468418

10.2.3.1 Design Basis

a. Redundancy of Reactivity Control

Criterion: At least two independent reactivity control systems, preferably of different principles, shall be provided. (GDC 27)

In addition to the reactivity control achieved by the Rod Cluster Control (RCC) described in Section 7, reactivity control is provided by the Chemical and Volume Control System which regulates the concentration of boric acid solution in the Reactor Coolant system. The system is designed to prevent uncontrolled or inadvertent reactivity changes which might cause system parameters to exceed design limits.

In the event that injection using the Charging Pumps is not available, the Safety Injection system can perform this function using borated water from the refueling water storage tank. If necessary, the RCS can be sufficiently depressurized to allow injection with the SI Pumps.

b. Reactivity Hot Shutdown (IT.S. Mode 3, Hot Standby) Capability

Criterion: At least two of the reactivity control systems provided shall independently be capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes, sufficiently fast to prevent exceeding acceptable fuel damage limits. (GDC 28)

The reactivity control systems provided are capable of making and holding the core subcritical from any hot standby (Mode 2, Startup) or hot operating (Mode 1, Power Operation) condition, including conditions resulting from power changes. The maximum excess reactivity expected for the core occurs for the cold, clean condition at the beginning of life of the core. The full length RCC assemblies are divided into two categories comprising control and shutdown groups.

The control group, used in combination with chemical shim, provides control of the reactivity changes of the core throughout the life of the core at power conditions. This group of RCC assemblies is used to compensate for short term reactivity changes at power such as those produced due to variations in reactor power requirements or in coolant temperature. The chemical shim control is used to compensate for the more slowly occurring changes in reactivity throughout core life such as those due to fuel depletion, fission product buildup and decay, and load follow.

c. Reactivity Shutdown Capability

Criterion: At least one of the reactivity control systems provided shall be capable of making the core subcritical under any condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margins greater than the maximum worth of the most effective control rod when fully withdrawn shall be provided. (GDC 29)

The reactor core, together with the reactor control and protection system is designed so that the minimum allowable DNBR is no less than the applicable limit and there is no fuel melting during normal operation including anticipated transients.

The shutdown groups are provided to supplement the control group of RCC assemblies to make the reactor at least one per cent subcritical (k_{eff} = 0.99) following trip from any credible operating condition to the hot, zero power condition assuming the most reactive RCC assembly remains in the fully withdrawn position.

Sufficient shutdown capability is also provided to maintain the core subcritical, with the most reactive rod assumed to be fully withdrawn, for the most severe anticipated cooldown transient associated with a single active failure, e.g., accidental opening of a steam bypass or relief valve. This is achieved with a combination of control rods and automatic boron addition via the Safety Injection System.

d. Reactivity Hold-Down Capability

Criterion: At least one of the reactivity control systems provided shall be capable of making and holding core subcritical under any conditions with appropriate margins for contingencies. (GDC 30)

Normal reactivity shutdown capability is provided by control rods, with boric acid injection used to compensate for the xenon transients, and for plant cooldown. When the plant is at power, the quantity of boric acid retained in the boric acid tanks and ready for injection always exceeds that quantity required for normal cold shutdown. This quantity always exceeds the quantity of boric acid required to bring the reactor to Mode 3, Hot Standby and to compensate for subsequent xenon decay.

The boric acid solution is transferred from the boric acid tanks by boric acid transfer pumps to the suction of the charging pumps which inject the solution into the reactor coolant system. Power for the charging pumps and the boric acid transfer pumps is automatically transferred to diesel generator power on loss of AC power. The charging pumps trip off on a loss of offsite power and must be manually restarted while the boric acid transfer pumps will automatically restart. Boric acid can be injected by one charging pump and one boric acid transfer pump at a rate which shuts the reactor down hot with no rods inserted in less than eighty (80) minutes. In eighty (80) additional minutes, enough boric acid can be injected to compensate for xenon decay although xenon decay below the equilibrium operating level does not begin immediately, but could occur up to 26 hours after shutdown, depending upon power history. If two boric acid pumps and two charging pumps are available, these time periods are halved. Additional boric acid is employed if it is desired to bring the reactor to Mode 5, Cold Shutdown conditions.

On the basis of the above, the injection of boric acid is shown to afford backup reactivity shutdown capability, independent of control rod clusters which normally serve this function in the short term situation. Shutdown for long term and reduced temperature conditions can be accomplished with boric acid injection using redundant components.

e. Codes and Classifications

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

All pressure retaining components (or compartments of components) which are exposed to reactor coolant, comply with the following codes:

1. System pressure vessels - ASME Boiler and Pressure Vessel Code, Section III, Class C.

2. System valves, fittings and piping - USAS B31.1, including nuclear code cases.

System component code requirements are tabulated in Table 10.2-5.

The tube and shell sides on the regenerative heat exchanger and the tube side of the excess letdown heat exchanger are designed to ASME III, Class C. This designation is based on the following considerations: (a) each exchanger is connected to the Reactor Coolant System by lines equal to or less than 2″, and (b) each is located inside the reactor containment. Analyses show that the accident associated with a 2″ line break does not result in clad damage or failure. Additionally, previously contaminated reactor coolant, escaping from the Reactor Coolant System during such accident is confined to the reactor containment building and no public hazard results.

10.2.3.1.1 Generic Letter 87-02

By letter dated November 20th, 1995 in response to Generic Letter 87-02 Prairie Island formally transmitted the selected systems and system functions which were reviewed in accordance with the Seismic Qualification Utility Group (SQUG) developed Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment (Reference 95). The non-safety related Chemical and Volume Control System (CVCS) was credited for performing functions for reactor shutdown and to maintain the plant in a hot shutdown condition. CVCS was selected to perform these functions based on the original seismic design requirements of the system and ability to exclude other multiple safety systems from review because of the diverse nature of CVCS capability. Other systems are not similarly described in the USAR for response to Generic Letter 87-02 based on their previously described functions which remained unchanged as a result of Generic Letter 87-02. It is not the intent of the GIP that operators are directed to use the alternate shutdown path as a first priority or change any symptom-based emergency operating procedure.

By submittal of the site response to Generic Letter 87-02, CVCS is described as being capable of providing injection of soluble poison to assure the continuation of hot shutdown conditions following a seismic event, through boron injection with one charging pump at minimum speed, drawing borated water from the Refueling Water Storage Tank (RWST).

By submittal of the site response to Generic Letter 87-02, CVCS is described as being capable of maintaining reactor coolant makeup following a seismic event, by injection through the Reactor Coolant Pump (RCP) seals during the early portion of the event with one pump running at minimum speed. The reactor coolant cold leg injection path will also be utilized as the event progresses but charging flow through the RCP seals will be maintained should thermal barrier cooling to the RCP seals be affected. To prevent the possibility of a solid pressurizer, letdown through the reactor head vent to the Pressurizer Relief Tank (PRT) is utilized as normal letdown would be isolated. If instrument air is lost the normal source for charging pump suction from the Volume Control Tank (VCT) will be manually transferred to the RWST.

10.2.3.2 Description

Various components of the Chemical and Volume Control system are shared by the two units. These components are shown in Table 10.2-6. The following discussion is for the Chemical and Volume Control System for one unit but applies equally to either unit.

The Chemical and Volume Control System, shown in Figures 10.2-5 through 10.2-10B, provides a means for injection of the neutron control chemical in the form of boric acid solution, chemical additions for corrosion control, and reactor coolant cleanup and degasification. This system also adds makeup water to the Reactor Coolant System, reprocesses water letdown from the Reactor Coolant System, and provides seal water injection to the reactor coolant pump seals. Materials in contact with the reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials.

System components whose design pressure and temperature are less than the Reactor Coolant System design limits are provided with overpressure protective devices.

System discharges from overpressure protective devices (safety valves) and system leakages are directed to closed systems. Effluents removed from such closed systems are monitored and discharged under controlled conditions.

During plant operation, reactor coolant flows through the letdown line from a loop intermediate leg on the suction side of the reactor coolant pump and, after processing is returned to the cold leg of the loop on the discharge side of the pump via a charging line and through the inleakage in the reactor coolant pump seals. An excess letdown line is also provided for removing coolant from the Reactor Coolant System.

The letdown line, charging line and the seal water injection lines which are maintained open during normal plant operation have the following redundant valves between the main coolant loop and the CVCS:

a. The letdown line is provided with two fail-closed, air operated valves near the reactor coolant loop. The RCS boundary extends to the second isolation valve. In the event of low pressurizer level, these valves are automatically closed.

b. The charging line is provided with one check valve and one fail-closed, air-operated isolation valve near the reactor coolant loop. The Reactor Coolant System boundary extends to the air operated valve which can be closed from the main control board.

c. The seal injection lines are each provided with two check valves in series. The RCS boundary extends to the second check valve.

The excess letdown line is maintained in the closed position during normal plant operation and is provided with one isolation valve which forms the RCS pressure boundary. The barrier is a normally closed, air-operated isolation valve which can be operated from the main control board.

Reactor coolant entering the Chemical and Volume Control System flows through the shell side of the regenerative heat exchanger where its temperature is reduced. The coolant then flows through letdown orifices which reduce the coolant pressure. The cooled, low pressure water leaves the reactor containment and enters the auxiliary building where it undergoes a second temperature reduction in the tube side of the letdown heat exchanger followed by a second pressure reduction by the low pressure letdown valve. After passing through a letdown filter (when in use) and one of the mixed bed demineralizers, where ionic impurities are removed, coolant flows through the reactor coolant filters and enters the volume control tank through a spray nozzle.

The vapor space in the volume control tank, which is predominantly hydrogen and water vapor pressure is normally maintained by adding hydrogen using the batch method. The hydrogen within this tank is supplied to the reactor coolant for maintaining a low oxygen concentration. Fission gases are periodically removed from the system by venting the volume control tank to the Waste Disposal System.

From the volume control tank the coolant flows to the charging pumps which raise the pressure above that in the Reactor Coolant System. The coolant then enters the containment, passes through the tube side of the regenerative heat exchanger, and is returned to the Reactor Coolant System.

The cation bed demineralizer, located downstream of the mixed bed demineralizers, is provided to control cesium activity in the coolant and also to remove excess lithium which is formed from B^{10} (n, α) Li^{7} reaction.

Boric acid is dissolved in hot water in the batching tank to a concentration of approximately 12 percent by weight. The lower portion of the batching tank is jacketed to permit heating of the batching tank solution with low pressure steam. A transfer pump is used to transfer the batch to the boric acid tanks. Small quantities of boric acid solution are metered from the discharge of an operating transfer pump for blending with reactor makeup water as makeup for normal leakage or for increasing the reactor coolant boron concentration during normal operation. Electric heaters maintain the temperature of the boric acid tanks solution high enough to prevent precipitation. The boric acid piping is heat traced to prevent precipitation.

Excess liquid effluents from the Reactor Coolant System are collected in holdup tanks. As liquid enters the CVCS holdup tanks, the nitrogen cover gas is displaced to the gas decay tanks in the Waste Disposal System through the waste gas vent header. The concentration of boric acid in the holdup tanks varies throughout core life from the refueling concentration to essentially zero at the end of the core cycle.

Liquid effluent in the holdup tanks is processed as a batch operation. This liquid is pumped through ion exchangers and filters which remove corrosion products and fission-products such as long-lived cesium. (Note: The gas stripper/boric acid evaporator packages for Unit 1 and 2, though piped into the system as shown on Figure 10.2-10A, are generally not utilized to process the liquid effluent from the holdup tanks.)

Following effluent processing, the effluent is directed into one of the three CVCS monitor tanks. Subsequent handling of the monitor tank contents is dependent on the results of sample analysis. Discharge from the monitor tanks may be used elsewhere in the plant, reprocessed through the evaporator condensate demineralizers, returned to the holdup tanks for reprocessing or discharged to the environment with the condenser circulating water, when within the allowable activity concentration. If the sample analysis of the monitor tank contents indicates that it may be discharged safely to the environment, at least two valves must be opened to provide a discharge path. As the effluent if being discharged, it is continuously monitored by the Waste Disposal System liquid effluent monitor. If an unexplained increase in radioactivity is sensed, one of the valves in the discharge line to the circulating water discharge header closes automatically and an alarm sounds in the control room.

The deborating demineralizers can be used intermittently to remove boron from the reactor coolant near the end of the core life. When the deborating demineralizers are in operation, the letdown stream passes from the mixed bed demineralizers, through the deborating demineralizers and into the volume control tank after passing through the reactor coolant filter.

During plant cooldown when the residual heat removal system is operating and the letdown orifices are not in service, a flow path is provided to remove corrosion impurities and fission products. A portion of the flow leaving the residual heat exchangers passes through the letdown heat exchanger, letdown filter (when in use), mixed bed demineralizers, reactor coolant filter, and volume control tank. The fluid is then pumped, via the charging pump, through the tube side of the regenerative heat exchanger into the Reactor Coolant System.

10.2.3.2.1 Expected Operating Conditions

Tables 10.2-6 and 10.2-7 list the data for individual system components and the system performance requirements respectively. Reactor coolant equilibrium activities are given in Appendix D.

Reactor Coolant Activity Concentration

The parameters used in the calculation of the reactor coolant fission product inventory, including pertinent information concerning the expected coolant cleanup flow rate and demineralizer effectiveness, are discussed in Appendix D.

10.2.3.2.2 Reactor Makeup Control

The reactor makeup control consists of a group of instruments arranged to provide a manually pre-selected makeup water composition to the charging pump suction header or the volume control tank. The makeup control functions are to maintain desired operating fluid inventory in the volume control tank and to adjust reactor coolant boron concentration for reactivity and shim control.

Makeup for normal plant leakage is regulated by the reactor makeup control which is set by the operator to blend water from the reactor makeup water tank with concentrated boric acid to match the reactor coolant boron concentration. The reactor makeup water pumps have been sized and rated to service the dilution water requirements of the boric acid blender connection.

The makeup system also provides concentrated boric acid or reactor makeup water to either increase or decrease the boric acid concentration in the Reactor Coolant System. To maintain the reactor coolant volume constant, an equal amount of reactor coolant at existing reactor coolant boric acid concentration is letdown to the holdup tanks. Should the letdown line be out of service during operation, sufficient volume exists in the pressurizer to accept the amount of boric acid necessary for Mode 5, Cold Shutdown.

Makeup water to the Reactor Coolant System is provided by the Chemical and Volume Control System from the following sources:

a. The reactor makeup water tanks, which provide water for dilution when the reactor coolant boron concentration is to be reduced

b. The boric acid tanks, which supply concentrated boric acid solution when reactor coolant boron concentration is to be increased

c. The refueling water storage tank, which supplies borated water for emergency makeup

d. The chemical mixing tank, which is used to inject small quantities of solution when additions of hydrazine or pH control chemical are necessary.

e. A temporary chemical injection skid, which may be used in lieu of the chemical mixing tank to manually add hydrogen peroxide as necessary to initiate a crud burst. 01468418

The reactor makeup control is operated from the control room by manually pre-selecting makeup composition to the charging pump suction header or the volume control tank in order to adjust the reactor coolant boron concentration for reactivity control. Makeup is provided to maintain the desired operating fluid inventory in the Reactor Coolant System. The operator can stop the makeup operation at any time in any operating mode. One boric acid transfer pump is normally lined up for automatic operation as required by the makeup controller. One reactor makeup pump is normally running.

A portion of the high pressure charging flow is injected into the reactor coolant pumps between the pump impeller and the shaft seal so that the seals are not exposed to high temperature reactor coolant. Part of the flow is the shaft seal leakage flow and the remainder enters the Reactor Coolant System through a labyrinth seal on the pump shaft. The shaft seal leakage flow cools the lower radial bearing, passes through the seals, is cooled in the seal water heat exchanger, filtered, and returned to the volume control tank.

Seal water inleakage to the Reactor Coolant System requires a continuous letdown of reactor coolant to maintain the desired inventory. In addition, bleed and feed of reactor coolant are required for removal of impurities and adjustment of boric acid in the reactor coolant.

Automatic Makeup

The “automatic makeup” mode of operation of the reactor makeup control provides boric acid solution preset by the operator to match the boron concentration in the Reactor Coolant System. The automatic makeup compensates for minor leakage of reactor coolant without causing significant changes in the coolant boron concentration.

Under steady state plant operating conditions, the mode selector switch is set in the "Automatic Makeup" position. A preset low level signal from the volume control tank level controller causes the automatic makeup control action to open the makeup stop valve to the charging pump suction, open the concentrated boric acid control valve and the reactor makeup water control valve. The makeup control also starts and/or switches to high speed any associated BATPs not in "pull-to-lock." The flow controllers then blend the makeup stream according to the preset concentration. Makeup addition to the charging pump suction header causes the water level in the volume control tank to rise. At a preset high level set-point, the makeup is stopped; the reactor makeup water control valve closes, the concentrated boric acid control valve closes, the makeup stop valve to charging pump suction closes, and the boric acid transfer pumps return to the pre-selected operating condition.

Dilution

The "dilute" mode of operation permits the addition of a pre-selected quantity of reactor makeup water at pre-selected flow rate to the Reactor Coolant System. The operator sets the mode selector switch to "dilute", the reactor makeup water flow controller set point to the desired flow rate, and the reactor makeup water batch integrator to the desired quantity. Upon manual start of the system the stop valve to the volume control tank opens. Makeup water is added to the volume control tank and then goes to the charging pump suction header. Excessive rise of the volume control tank water level is prevented by automatic actuation (by the tank level controller) of a three-way diversion valve, which routes the reactor coolant letdown flow to the holdup tanks. When the preset quantity of reactor makeup water has been added, the batch integrator causes the makeup water stop valve in the charging pump suction to close and the stop valve to the volume control tank to close.

Alternate Dilute

The “alternate dilute” mode is similar to the dilute mode except the dilution water, after passing through the blender, splits and a portion flows directly to the charging pump suction and a portion flows into the volume control tank via the spray nozzle and then flows to the charging pump suction. The operator sets the mode selector switch to “alternate dilute”, the reactor makeup water flow controller set point to the desired flow rate, the reactor makeup water batch integrator to the desired quantity and actuates the makeup start. The start signal causes the makeup control action to open the makeup stop valve to the volume control tank and the makeup stop valve to the charging pump suction header and the reactor makeup control valve. Reactor makeup water is simultaneously added to the volume control tank and to the charging pump suction header. This mode is used for load follow and permits the dilution water to follow the initial xenon transient and simultaneously dilute the volume control tank. Excessive water level in the volume control tank is prevented by automatic actuation of the volume control tank level controller which routes the reactor coolant letdown flow to the holdup tanks. When the preset quantity of reactor makeup water has been added, the batch integrator causes the reactor makeup water control valve and the reactor makeup stop valves to close. This operation may be stopped manually by actuating the makeup stop.

Boration

The “borate” mode of operation permits the addition of a pre-selected quantity of concentrated boric acid solution at a pre-selected flow rate to the Reactor Coolant System. The operator sets the mode selector switch to “borate”, the concentrated boric acid flow controller set point to the desired flow rate, and the concentrated boric acid batch integrator to the desired quantity. Upon manual start of the system the stop valve to the charging pumps opens, and the concentrated boric acid control valve opens. The makeup control also starts and/or switches to high speed any associated BATPs not in “pull-to-lock” and the concentrated boric acid is added to the charging pump suction header. The total quantity added in most cases is so small that it has only a minor effect on the volume control tank level. When the preset quantity of concentrated boric acid solution has been added the associated boric acid transfer pump(s) return to their pre-borate condition, the batch integrator causes the boric acid control valve to close and the makeup stop valve to the charging pump suction to close.

The capability to add boron to the reactor coolant is such that it imposes no limitation on the rate of cooldown of the reactor upon shutdown. The maximum rates of boration and the equivalent coolant cooldown rates are given in Table 10.2-7. One set of values is given for the addition of boric acid from a boric acid tank with one transfer and one charging pump operating. The other set assumed the use of refueling water but with two of the three charging pumps operating. The rates are based on full operating temperature and on the end of the core life when the moderator temperature coefficient is most negative.

Alarm Functions

The reactor makeup control is provided with alarm functions for the following conditions:

a. Deviation of reactor makeup water flow rate from the control setpoint

b. Deviation of concentrated boric acid flow rate from the control setpoint

c. Low level (makeup initiation point) in the volume control tank if the level decreases below the low level makeup initiation setpoint.

Boron Analyzer

A boron analyzer is located on the letdown line. The analyzer provides a continuous indication of the boron concentration in the letdown flow. The boron analyzer is designed to provide a semi-continuous indication of boron concentration in the letdown line during normal operation. The boron concentration indicator is supplied as an option and is not required to operate the plant. Normal sampling procedures, combined with control rod position indication, is the normal method of monitoring boron concentration. The boron concentration indicator is a redundant means of determining the boron concentration. As a general operating aid it should provide information as to when additional check-analysis is warranted rather than a basis for fundamental operating decisions.

The Westinghouse Boron Concentration Measurement System (BCMS) is an electronuclear system which employs neutron-counting principles and electronic circuitry to measure the neutron absorption of boric acid in water, and digitally display the result in terms of ppm boron (parts total boron per million parts of water).

The system employs a sample measurement unit which contains a neutron source and a neutron detector located in a shield tank. Piping within the shield tank is arranged to maintain coolant sample flow between the neutron source and the neutron detector. Neutron absorption by the boron in the coolant sample flow reduces the number of neutrons which contact the detector per unit time. Therefore, the time required to count a fixed number of neutron contacts is a variable which is dependent upon the concentration of boron solution. Electronic circuitry in the console portion of the Boron Concentration Measurement System accepts an amplified signal from the sample measurement unit and converts the signal to a digital display of ppm boron. The digital display is housed in the console, which may be located up to 500 feet from the sample measurement unit. Thus, output signals from the sample measurement unit may be displayed at almost any convenient place within the plant.

Boron Concentration Measurement System Performance Data:

°Range of System Operation	0 to 5000 ppm boron
°System Accuracy	10 ppm for boron concentration between 2800 ppm and 5000 ppm boron
°Measurement Display Frequency	Continuous display of latest concentration measurement
Sample Requirements	
°Pressure	12-225 psig
°Flow	0.2 gpm nominal
°Temperature	70-140°F

The BCMS is calibrated using a known concentration of boric acid.

The BCMS and the CVCS piping system is designed to allow isolation of the BCMS and flushing.

The BCMS is utilized as an advisory system only. It is not connected in any manner to plant control or protective systems. Erroneous readings caused by maloperation of the system would be detected by the operator as boron concentration changes would affect reactivity control and changes in reactivity would be indicated on other instruments.

All piping penetrations are located on the top of the sample tank negating an accidental leakage of the tank water.

A maloperation of the sample tank heater causing it to stay energized would cause the tank water to evaporate resulting in a loss of sample tank water. A local low water level alarm is provided to indicate this condition. Local temperature and level instruments are also provided. Continued energizing of the sample tank heater with the low level would cause the heater to burn out. The abnormal reactor coolant sample temperature would cause erroneous boron concentration readings.

10.2.3.2.3 Charging Pump Control

Three positive displacement variable speed drive charging pumps are used to supply charging flow to the Reactor Coolant System.

The speed of each pump can be controlled manually or automatically. During normal operation, one of the three pumps is operated with its speed control in automatic. A second pump may be operated with its speed control in manual, maintaining a constant speed. The speed of the pump in automatic modulates according to deviation between pressurizer level and level control program. During load changes the pressurizer level set point is varied automatically to compensate partially for the expansion or contraction of the reactor coolant associated with the Tavg changes. The pressurizer level control circuitry provides a demand signal to the charging pump VFD drive (Auto mode) to vary pump speed and to maintain a constant pressurizer level.

If the pressurizer level increases above program, the speed of the pump decreases; likewise, if the level decreases below program, the speed increases. If the charging pump on automatic control reaches the low or high speed limit, an alarm is actuated. Additional charging pumps may be started or stopped as necessary to control pressurizer level. These additional pumps are operated in manual speed control in response to pressurizer level changes.

To ensure that the charging pump flow is always sufficient to meet both the seal water and minimum charging flow requirements, the pump has a variable control stop which does not permit pump flow lower than the specified minimum. This control stop is adjustable to permit higher minimum flow limits to be set if mechanical seal leakage increases during plant life.

10.2.3.2.4 <u>Components</u>

A summary of principal component data is given in Table 10.2-6.

a. Regenerative Heat Exchanger

   The regenerative heat exchanger is designed to recover the heat from the letdown stream by reheating the charging stream during normal operation. This exchanger also limits the temperature rise which occurs at the letdown orifices during periods when letdown flow exceeds charging flow by a greater margin than at normal letdown conditions.

   The letdown stream flows through the shell of the regenerative heat exchanger and the charging stream flows through the tubes. The unit is made of austenitic stainless steel, and is of all-welded construction.

b. Letdown Orifices

One of the three letdown orifices controls flow of the letdown stream during normal operation and reduces the pressure to a value compatible with the letdown heat exchanger design. Either of two letdown orifices, each 40 gpm, is used to pass normal letdown flow. The third orifice, 80 gpm, is designed to be used for high flow purification flow at normal Reactor Coolant System operating pressure and can pass twice the normal letdown flow. The orifices are placed in and taken out of service by remote manual operation of their respective isolation valves. One or both of the standby orifices may be used in parallel with the normally operating orifice in order to bring the letdown flow up to normal when the Reactor Coolant System pressure driving force is below normal. This arrangement provides a full standby capacity for control of letdown flow. Each orifice consists of bored pipe made of austenitic stainless steel.

c. Letdown Heat Exchanger

The letdown heat exchanger cools the letdown stream to the operating temperature of the mixed bed demineralizers. Reactor coolant flows through the tube side of the exchanger while component cooling water flows through the shell side. The letdown stream outlet temperature is automatically controlled by a temperature control valve in the component cooling water outlet stream. The unit is a multiple-tube, multiple-pass shell heat exchanger. All surfaces in contact with the reactor coolant are austenitic stainless steel, and the shell is carbon steel. Since the maximum operating temperature for the components in question is considerably less than 200°F (maximum operating temperatures are 120°F for the letdown heat exchanger shell, 130°F for the charging pump, and 110°F for the monitor tank pump), a design temperature of 200°F provides adequate margin. An increase in design temperature to 250°F would not provide any significant change in equipment design and would not affect reactor safety.

If a significant leak develops in the letdown heat exchanger such that the charging system is unable to maintain pressurizer water level, the letdown line isolation valves near the reactor coolant loop trip closed. The excess letdown path can then be placed in service while maintenance is performed on the letdown heat exchanger.

d. Mixed Bed Demineralizers

Two flushable mixed bed demineralizers maintain reactor coolant purity. A Lithium-7 (or H^+ form) cation resin and a hydroxyl form anion resin are initially charged into the demineralizers. Both forms of resin remove fission and corrosion products, and in addition, the reactor coolant causes the anion resin to be converted to the borate form. The resin bed is designed to reduce the concentration of ionic isotopes in the purification stream, except for cesium, yttrium, and molybdenum, by a minimum factor of 10.

Each demineralizer is sized to accommodate the normal letdown flow. One demineralizer serves as a standby unit for use should the operating demineralizer become exhausted during operation.

The demineralizer vessels are made of austenitic stainless steel, and are provided with suitable connections to facilitate resin replacement when required. The vessels are equipped with a resin retention screen. Each demineralizer has sufficient capacity, after operation for one core cycle with one per cent defective fuel rods, to reduce the activity of the primary coolant to refueling concentration.

e. Cation Bed Demineralizer

A flushable cation resin bed in the hydrogen form is located downstream of the mixed bed demineralizers and is used intermittently to control the concentration of lithium which builds up in the coolant from the B^{10} (n, α) Li^7 reaction. The demineralizer also has sufficient capacity to maintain the Cesium-137 concentration in the coolant below 1.0 µCi/cc with one percent defective fuel. The demineralizer would be used intermittently to control cesium.

The demineralizer is made of austenitic stainless steel and is provided with suitable connections to facilitate resin replacement when required. The vessel is equipped with a resin retention screen.

f. Deborating Demineralizers

When required, two anion demineralizers remove boric acid from the Reactor Coolant System fluid. The demineralizers are provided for use near the end of a core cycle, but can be used at any time.

Hydroxyl-form ion-exchange resin is used to reduce Reactor Coolant System boron concentration. Facilities are provided for regeneration. When regeneration is no longer feasible, the resin is flushed to the spent resin shipping cask.

Each demineralizer can remove the quantity of boric acid that must be removed from the Reactor Coolant System to maintain full power operation near the end of core life without the use of the holdup tanks or evaporators.

g. Reactor Coolant Filter

The filter collects resin fines and particulates larger than 25 microns from the letdown stream. The filter vessel is made of austenitic stainless steel, and is provided with connections for draining and venting. Design flow capacity of the filter is equal to the maximum purification flow rate. Disposable filter elements are used.

h. Letdown Filter

The letdown filter collects particulates larger than 5 microns from the letdown stream before it enters the demineralizers. The vessel is made of austenitic stainless steel, and is provided with the connections for draining and venting. Design flow capacity of the filter is equal to the maximum purification flow rate. Disposable filter elements are used.

i. Volume Control Tank

The volume control tank collects the excess water released from zero power to full power, that is not accommodated by the pressurizer. It also receives the excess coolant release caused by the deadband in the reactor control temperature instrumentation. Overpressure of hydrogen gas is maintained in the volume control tank to control the hydrogen concentration in the reactor coolant at 25 to 50 cc per Kg of water (standard conditions).

A spray nozzle is located inside the tank on the inlet line from the reactor coolant filter. This spray nozzle provides intimate contact to equilibrate the gas and liquid phases. A remotely operated vent valve discharging to the Waste Disposal System permits removal of gaseous fission products which are stripped from the reactor coolant and collected in this tank. The volume control tank also acts as a head tank for the charging pumps and a reservoir for the leakage from the reactor coolant pump controlled leakage seal. The tank is constructed of austenitic stainless steel.

j. Charging Pumps

Three charging pumps are provided for injecting coolant into the Reactor Coolant System. The pumps are the variable speed positive displacement type, and all parts in contact with the reactor coolant are fabricated of austenitic stainless steel and other material of adequate corrosion resistance. Special low-chloride packing is used in the pump glands. These pumps have mechanical packing followed by a leakoff to collect reactor coolant before it can leak to the outside atmosphere. Pump leakage is piped to a local drain for disposal to the Waste Disposal System. The pump design precludes the possibility of lubricating oil contaminating the charging flow, and the integral discharge valves act as check valves.

Each pump is designed to provide the full charging flow and the reactor coolant pump seal water supply during normal seal leakage. Each pump is designed to provide rated flow against a pressure equal to the sum of the Reactor Coolant System maximum pressure (existing when the pressurizer power operated relief valve is operating) and the piping, valve and equipment pressure losses of the charging system at the design charging flows. Pulsation dampers are provided at the pump suction and discharge.

One of the three charging pumps can be used to hydrotest the Reactor Coolant System.

k. Chemical Mixing Tank

The primary use of the chemical mixing tank is in the preparation of pH control chemical solutions and hydrazine for oxygen scavenging.

The capacity of the chemical mixing tank is determined by the quantity of 35 percent hydrazine solution necessary to increase the concentration in the reactor coolant by 10 ppm. This capacity is more than sufficient to prepare the solution of pH control chemical for the Reactor Coolant System. The chemical mixing tank is made of austenitic stainless steel.

l. Excess Letdown Heat Exchanger

The excess letdown heat exchanger cools reactor coolant letdown flow at a rate equal to the nominal injection rate through the reactor coolant pump labyrinth seal, if letdown through the normal letdown path is blocked. The unit is designed to reduce the letdown stream temperature from the cold leg temperature to 195°F. The letdown stream flows through the tube side and component cooling water is circulated through the shell side. All surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel. All tube joints are welded.

m. Seal Water Heat Exchanger

The seal water heat exchanger removes heat from the reactor coolant pump seal water returning to the volume control tank and reactor coolant discharge from the excess letdown heat exchanger. Reactor coolant flows through the tubes and component cooling water is circulated through the shell side. The tubes are welded to the tube sheet to prevent leakage in either direction, which would result in undesirable contamination of the reactor coolant or component cooling water. All surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel.

The unit is designed to cool the excess letdown flow and the seal water flow to the temperature normally maintained in the volume control tank if all the reactor coolant pump seals are leaking at the maximum design leakage rate.

n. Seal Water Filter

The filters in parallel collect particulates from the reactor coolant pump seal water return and from the excess letdown heat exchanger flow. Each filter is designed to pass the sum of the excess letdown flow and the maximum design leakage from the reactor coolant pump controlled-leakage seals. The filter vessels are constructed of austenitic stainless steel and are provided with connections for draining and venting. Disposable synthetic filter cartridges are used.

o. Seal Water Injection Filters

Two filters are provided in parallel, each sized for the injection flow. They collect particulates from the water supplied to the reactor coolant pump seals.

p. Boric Acid Filter

The boric acid filter collects particulates from the boric acid solution being pumped to the charging pump suction line. The filter is designed to pass the design flow of two boric acid transfer pumps operating simultaneously. The vessel is constructed of austenitic stainless steel and the filter elements are disposable synthetic cartridges. Provisions are available for venting and draining the filter.

q. Boric Acid Tanks

The boric acid tank capacities can be shared by Unit 1 and Unit 2 and are sized to store sufficient boric acid solution for refueling plus enough boric acid solution for a cold shutdown shortly after full power operation is achieved. In addition, each unit-designated tank is sized to provide sufficient boric acid solution to achieve Mode 5, Cold Shutdown, if the most reactive RCCA is not inserted. However, to meet surveillance procedures for readily-available quantities of boric acid to meet Cold Shutdown requirements, credit may be taken for the portion of the available standby tank not reserved for the opposite unit. One tank is normally used with each unit and a third tank acts as a standby.

The concentration of boric acid solution in storage is maintained between 11.5 and 13% by weight. Periodic manual sampling is performed and corrective action is taken, if necessary, to ensure that these limits are maintained. As a consequence, measured amounts of boric acid solution can be delivered to the reactor coolant to control the chemical poison concentration. The combination overflow and breather vent connection has a water loop seal to minimize vapor discharge during storage of the solution. The tanks are constructed of austenitic stainless steel.

The volume of boric acid solution required for Mode 5, Cold Shutdown is 3100 gallons. Refilling the boric acid tanks is a batch process.

r. Boric Acid Tank Heaters

Two 100% capacity electric heaters located near the bottom of each boric acid tank are designed to maintain the temperature of the boric acid solution at 165°F with an ambient air temperature of 40°F thus ensuring a temperature in excess of the solubility limit (for 12 percent boric acid this is 133°F). The temperature is monitored and low temperature is alarmed in the control room. The heaters are sheathed in incoloy. For 11 and 121 Boric Acid Storage Tanks, one heater for each tank is fed from a bus supplied by Diesel Generator 1 and the second heater is fed from a bus supplied by Diesel Generator 2. For 21 Boric Acid Storage Tank, one heater is fed from a bus supplied by Diesel Generator 5 and the second heater is fed from a bus supplied by Diesel Generator 6. Redundant heat tracing circuits are also provided for Boric Tanks and associated lines. Train A heat trace is powered from transferable MCC 1T1 which can be supplied from 480V Diesel Generator 1 (via Bus 112) or Diesel Generator 5 (via Bus 212). Train B heat trace is powered from transferable MCC 1T2 which can be supplied from Diesel Generator 2 (via Bus 122) or Diesel Generator 6 (via Bus 222). Separation criteria established for Train A and Train B safeguards in this plant are applied to power feeds to these redundant systems. Redundant step down transformers, distribution panels, thermostat controllers and failure alarm relays for the heat tracing systems are mounted on the same panel. Separation between redundant heater wires wrapped around the pipe to be traced is maintained by the construction of the heater cables. Each heater cable consists of a heater wire surrounded by magnesium-oxide insulation in a stainless steel sheath, thus two sheaths are placed between redundant heater-circuit wires.

s. Batching Tank

The batching tank, shared by Units 1 and 2 is sized to hold one week's makeup supply of boric acid solution for the boric acid tank. The basis for makeup is reactor coolant leakage of 1/2 gpm at beginning of core life. The tank may also be used for solution storage. A local sampling point is provided for verifying the solution concentration prior to transferring it to the boric acid tank.

The tank manway is provided with a removable screen to prevent entry of foreign material. In addition, the tank is provided with an agitator to improve mixing during batching operation. The tank is constructed of austenitic stainless steel, and is not used to handle radioactive substances. The tank is provided with a steam jacket for heating the boric acid solution to 165°F.

The source of heat for the steam-jacketed boric acid batching tank is process steam which is piped to and from the steam jacket.

The boric acid batching tank is not required under post-accident conditions.

Boric acid can be added to the recirculated emergency coolant using the normal boron makeup system. There is sufficient capacity in the remaining boric acid tanks to provide any necessary makeup.

t. Boric Acid Transfer Pumps

Each unit has two centrifugal pumps with two speed motors (one on each unit may be cross-connected to the other unit) that are used to circulate or transfer boric acid. The pumps circulate boric acid solution through the boric acid tanks and transfer boric acid into the charging pump suction header.

Although one pump is normally used for boric acid batching and transfer and the other for boric acid injection, either pump may function as standby for the other. The design capacity of each pump is equal to the normal letdown flow rate. The design head at high speed is sufficient, considering line and valve losses, to deliver rated flow to the charging pump suction header when volume control tank pressure is at the maximum operating value (relief valve setting). All parts in contact with the solutions are austenitic stainless steel and other adequately corrosion-resistant material.

The transfer pumps are operated either automatically or manually from the main control room. The pumps may also be operated in manual from a local control point. Any pump not in pull-to-lock will be started or transfer to high speed when boric acid solution is required for makeup or boration.

u. Boric Acid Blender

The boric acid blender enhances thorough mixing of boric acid solution and reactor makeup water from the reactor makeup supply circuit. The blender consists of a conventional pipe elbow fitted with a perforated tube insert. All material is austenitic stainless steel.

v. Holdup Tanks

Three holdup tanks which can be shared by Units 1 and 2, are provided to collect water resulting from various Reactor Coolant System evolutions, such as: startup, shutdown, load changes, and from boron dilutions to compensate for burnup. The contents of the non-aerated sump tank are usually directed to the CVCS holdup tanks also. The contents of the holdup tanks are processed as needed using ion exchangers and filtration to achieve desired effluent activity limits and maintain adequate holdup capacity.

The design pressure for these large volume, low pressure storage tanks was selected to allow the use of the API-620 code. This code provides more definitive guidelines for the design of this class of vessel.

w. Holdup Tank Recirculation Pump

The holdup tank recirculation pump is used to mix the contents of a holdup tank, transfer the contents of one holdup tank to another, or transfer holdup tank water to the spent fuel pit.

The wetted surface of this pump is constructed of austenitic stainless steel.

x. Gas Stripper Feed Pumps

The three gas stripper feed pumps, which can be shared by Units 1 and 2, provide feed from a holdup tank to the evaporator feed ion exchanger/filtration process. Water is generally processed in a batch mode using one pump. The non-operating pumps are available for operation in the event the operating pump malfunctions. These centrifugal pumps are constructed of austenitic stainless steel.

y. Evaporator Feed Ion Exchangers

Four flushable ion exchangers can be shared by Units 1 and 2 to remove impurities from the holdup tank effluent. The resin is selected as needed to achieve desired decontamination factors. (See Appendix D for typical cesium decontamination factors.) These ion exchangers may be operated in parallel or series. Each vessel is constructed of austenitic stainless steel and contains a resin retention screen.

z. Ion Exchanger Filters

These filters collect resin fines and particulates from the evaporator feed ion exchangers. The vessel is made of austenitic stainless steel and is provided with connections for draining and venting. Disposable filter cartridges are used.

aa. Gas Stripper/Boric Acid Evaporator Equipment

Although normally not used, the gas stripper/boric acid evaporators can be shared by the two units to remove N_2, H_2, and fission gasses and concentrate boric acid for reuse in the Reactor Coolant System. The boric acid evaporators currently are retired in place and are not used for processing CVCS water.

bb. Evaporator Condensate Demineralizers

Two condensate demineralizers, which can be shared by Unit 1 and 2, are available to provide cleanup of water processed from the CVCS Holdup Tanks. The demineralizers are constructed of stainless steel.

cc. Condensate Filter

This filter removes resin fines and particulate from water processed from the CVCS Holdup Tanks. The vessel is made of stainless steel, is provided with connections for venting and draining, and uses disposable filter elements.

dd. Monitor Tanks

The monitor tanks, which can be shared by Unit 1 and Unit 2, permit continuous processing of CVCS Holdup Tank water. When one tank is filled, the contents are analyzed and either reprocessed, discharged via the Waste Disposal System or used elsewhere in the plant as needed. These tanks are stainless steel.

ee. Monitor Tank Pumps

Two monitor tank pumps, shared by Units 1 and 2, discharge water from the monitor tanks. The wetted surfaces of these pumps are constructed of austenitic stainless steel.

ff. Reactor Makeup Water Tanks

Two reactor makeup water tanks per unit are provided and furnish the suction supply to the two reactor makeup pumps of each unit. The tanks receive this makeup water from the monitor tanks via the monitor tank pumps, or the condensate makeup storage tank via the condensate recycle and transfer pumps or from the demineralized water system. These tanks are constructed of coated carbon steel.

gg. Reactor Makeup Water Pumps

Two reactor makeup water pumps per unit are provided and furnish demineralized water to the pressurizer relief tank, to the boric acid blender and chemical mixing tank, to the spent fuel pool, and makeup to the component cooling surge tank. These pumps take suction supply from the reactor makeup water tanks. The wetted surfaces of these pumps are constructed of austenitic stainless steel. Each pump is sized to match the high flow letdown purification flow of 80 gpm.

hh. Concentrates Filter

A disposable synthetic cartridge type filter removes particulates from the evaporator concentrates. Design flow capacity of the filter is equal to the boric acid evaporator concentrates transfer pump capacity. The vessel is made of austenitic stainless steel.

ii. Concentrates Holding Tank

The concentrates holding tank, shared by Units 1 and 2, is sized to hold the production of concentrates from one batch of evaporator operation. The tank is supplied with an electrical heater which prevents boric acid precipitation and is constructed of austenitic stainless steel.

jj. Concentrates Holding Tank Transfer Pumps

Two holding tank transfer pumps, shared by Units 1 and 2, discharge boric acid to the boric acid tanks. The wetted surfaces of these pumps are constructed of austenitic stainless steel.

kk. Electrical Heat Tracing

Electrical heat tracing is installed under the insulation on all piping, valves, line-mounted instrumentation, and components normally containing concentrated boric acid solution. The heat tracing compensates for heat loss due to cooling and prevents boric acid precipitation. The lines and components of the Chemical and Volume Control System (CVCS) which are provided with heat tracing are shown in Figures 10.2-6, 10.2-8 and 10.2-10b.

The boric acid tanks, boric acid batching tank, and the Concentrates holding tank are provided with individual means of heating and need not be heat traced.

Redundant electrical heat tracing is installed on all sections of the CVCS normally containing boric acid solution, to provide standby capacity if the operating section malfunctions. The power supply for the redundant lines of heat tracing is connected to the diesel-powered buses to ensure continuous operation during a prolonged outage of normal power supplies.

The combination of electrical heat tracing and insulation maintains the temperature of the piping and contents at 160° to 180°F with an ambient air temperature of 40°F. Separate thermostatic controls are provided for each of the duplicate sets of heat tracing to maintain the temperature within the specified control band. High and low alarms are provided in the control room for selected heat trace circuits to warn of failure to maintain the normal temperature control band for the piping and equipment containing concentrated boric acid solution. Transfer of control between the redundant heat tracing is an automatic operation.

II. Valves

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 surveillance, and leakage control programs address early leak detection and repair. Globe valves are installed with flow over the seats when such an arrangement reduces the possibility of leakage. Basic material of construction is stainless steel for all valves except the batching tank steam jacket valves which are carbon steel.

Isolation valves are provided at all connections to the Reactor Coolant System. Lines with flow into the reactor containment also have check valves inside the containment to prevent reverse flow from the containment. Further discussion of isolation criteria is presented in Section 5.2.1.2.1.

Relief valves are provided for lines and components that might be pressurized above design pressure by improper operation or component malfunction. Pressure relief for the tube side of the regenerative heat exchanger is provided by the check valve which bypasses the charging line isolation valve.

mm. Piping

All Chemical and Volume Control System piping handling radioactive liquid is austenitic stainless steel. All piping joints and connections are welded, except where flanged connections are required to facilitate equipment removal for maintenance and hydrostatic testing. Piping, valves, equipment and line-mounted instrumentation, which normally contain concentrated boric solution, are heated by redundant electrical tracing to ensure solubility of the boric acid.

nn. Temporary chemical injection skid

A temporary chemical injection skid may be connected to the charging line to the regenerative heat exchanger. Use of the skid is controlled as a recurring temporary modification.

The primary use of the temporary chemical injection skid is to bypass the seal water injection flowpath for the injection of hydrogen peroxide in order to initiate a crud burst. 01468418

The skid is designed to interface with the permanently installed equipment through a vent/drain valve to limit the possible leakage from the RCS should any failure occur. The skid is only to be used in mode 5 in order to limit the impact on containment operability. The skid is designed to maintain a CVCS pressure of up to 2,735 PSIG. The injection pressure of the skid is designed only to provide flow at the expected CVCS pressure during Mode 5.

10.2.3.3 Performance Analysis

10.2.3.3.1 Availability and Reliability

A high degree of functional reliability is assured in this system by providing standby components where performance is vital to safety and by assuring fail-safe response to the most probable mode of failure. Special provisions include duplicate heat tracing with alarm protection of lines, valves, and components normally containing concentrated boric acid.

Each unit has three high pressure charging pumps, each capable of supplying the normal reactor coolant pump seal and makeup flow.

The electrical equipment of the Chemical and Volume Control System is arranged so that multiple items receive their power from various 480 volt buses (see Section 8).

The two boric acid transfer pumps are powered from separate 480 volt buses. One charging pump and one boric acid transfer pump are capable of meeting Mode 5, Cold Shutdown requirements shortly after full power operation. In cases of loss of AC power, the charging pumps and the boric acid transfer pumps can be manually transferred to diesel generator power.

10.2.3.3.2 Control of Tritium

The Chemical and Volume Control System is used to control the concentration of tritium in the Reactor Coolant System. Essentially all of the tritium is in chemical combination with oxygen as a form of water. Therefore, any leakage of coolant to the containment atmosphere carries tritium in the same proportion as it exists in the coolant. Thus, the level of tritium in the containment atmosphere, when it is sealed from outside air ventilation, is a function of tritium level in the reactor coolant, the cooling water temperature at the cooling coils, which determines the dew point temperature of the air, and the presence of leakage other than reactor coolant as a source of moisture in the containment air.

There are two major considerations with regard to the presence of tritium:

a. Exposure of plant personnel to tritium during access to the containment. Leakage of reactor coolant during operation with a closed containment causes an accumulation of tritium in the containment atmosphere. It is desirable to limit the accumulation to allow containment access for short times for in-core instrumentation maintenance and equipment inspection.

b. Exposure to the public due to release of tritium.

Neither of these considerations is limiting in this plant.

The uncertainties associated with estimating the amounts of tritium generated are discussed in Appendix D.

Periodic determinations of tritium concentrations is made by liquid scintillation counting of condensed water vapor from the containment.

Tritium releases to the atmosphere via the containment purge system are made in accordance with limits given in the Offsite Dose Calculation Manual (ODCM). Normally, tritium releases are expected to be much lower than allowed by the referenced limits.

10.2.3.3.3 Leakage Prevention

Quality control of the material and the installation of the Chemical and Volume Control System valves and piping, which are designated for radioactive service, is provided in order to essentially eliminate leakage to the atmosphere. The components designated for radioactive service are provided with welded connections to prevent leakage to the atmosphere. However, flanged connections are provided in each charging pump suction and discharge, on each boric acid pump suction and discharge, on the relief valves inlet and outlet, on three-way valves, on the flow meters and elsewhere where necessary for maintenance.

The positive displacement charging pumps stuffing boxes are provided with leakoffs to collect reactor coolant before it can leak to the atmosphere. 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.

Diaphragm valves are provided where the operating pressure and the operating temperature permit the use of these valves. Leakage to the atmosphere is essentially zero for these valves. In some cases, due to heat-traced piping and other environmental conditions, diaphragm valves have a tendency to fail. In these cases, other types of valves such as globe valves, gate valves, etc. are used.

10.2.3.3.4 Incident Control

The letdown line and the reactor coolant pumps seal water return lines penetrate the reactor containment. The letdown line contains two air-operated valves inside the reactor containment upstream of the regenerative heat exchanger. These two valves form the RCS pressure boundary and close on low pressurizer level. In addition, three parallel air-operated orifice block valves inside the reactor containment are backed up by an air-operated valve outside the reactor containment. These four valves are automatically closed by the containment isolation signal.

If the charging system is unable to maintain pressurizer water level, the letdown line isolation valves located close to the reactor coolant loop are tripped closed by a low level signal from the pressurizer level instrumentation. Redundant valves in series and redundant level channels powered by separate and independent buses are provided to ensure letdown isolation even in the event of a single active failure. The isolation valves provided are air-operated, fail-closed valves (i.e., in the event of a loss of air or an electrical failure to the solenoid valves in the air supply line, the valves fail to the closed position). Isolation of the letdown line following a LOCA is accomplished at the containment penetration which has fail-closed, air-operated valves inside containment downstream of the letdown orifices and one fail-closed, air-operated valve outside the containment near the penetration.

If a valve failure causes the letdown line to be isolated, the charging line must also be removed from service. However, the charging pumps can continue to supply seal injection flow to the reactor coolant pumps. The excess letdown line can be placed in service, as required, to letdown the labyrinth seal inleakage to the Reactor Coolant System. Therefore, the failure of these isolation valves has no effect on the safety of the plant and operation of the plant can continue until corrective action is taken and normal letdown is restored.

The reactor coolant pumps seal water return line contains one motor-operated isolation valve outside the reactor containment which is automatically closed by the containment isolation signal.

The two seal water injection lines to the reactor coolant pumps and the charging line are inflow lines penetrating the reactor containment. Each line contains two check valves inside the reactor containment to provide isolation of the reactor containment if a break occurs in these lines outside the reactor containment.

The check valves inside the containment on the seal water injection lines are capable of withstanding the full pressure differential resulting from a pipe break in the seal injection line outside the reactor containment.

The first check valve in the seal water supply lines prevents loss of reactor coolant due to a line break inside the loop compartment walls. The second check valve located as close as practicable to the containment wall provides the first containment isolation barrier. The seal water injection lines and header are connected to a closed system outside containment. This closed system serves as the secondary containment isolation boundary. Check valves, and one locked closed manual valve provide the closed system boundary at the suction side of the charging pumps. As this is considered a leakage collection system, the boundary valves are leak tested per ASME, Section XI requirements. However, there are manually operated needle valves in the seal water injection lines which may be closed to provide a second isolation barrier outside containment.

In the event that the manual closure of these valves is required, an operator would be exposed to the post LOCA radiation field from the shielded containment vessel. Conservatively assuming that 15 minutes is required to operate these valves at one hour post LOCA, an operator would receive about 10 rem whole body exposure.

10.2.3.3.5 Malfunction Analysis

To evaluate system safety, failures or malfunctions were assumed concurrent with a loss-of-coolant accident and the consequences analyzed and presented in Table 10.2-8. As a result of this evaluation, it is concluded that proper consideration has been given to plant safety in the design of the system.

If a rupture were to take place between the reactor coolant loop and the first isolation valve or check valve, this incident would lead to an uncontrolled loss of reactor coolant. The analysis of loss-of-coolant accidents is discussed in Section 14.6 and 14.7.

Should a rupture occur in the Chemical and Volume Control System outside the containment, or at any point beyond the first check valve or remotely operated isolation valve, actuation of the valve would limit the release of coolant and assure continued functioning of the normal means of heat dissipation from the core. For the general case of rupture outside the containment, the largest source of radioactive fluid subject to release is the contents of the volume control tank. The consequences of such a release are considered in Section 14.

When the reactor is subcritical, the relative reactivity status (neutron source multiplication) is continuously monitored and indicated by BF_3 counters and count rate indicators. Any appreciable increase in the neutron source multiplication, including that caused by the maximum physical boron dilution rate, is slow enough to give ample time to start a corrective action (boron dilution stop and/or emergency boron injection) to prevent the core from becoming critical. The maximum dilution rate is based on the abnormal condition of three charging pumps operating at full speed delivering unborated primary water to the Reactor Coolant System at a particular time during refueling when the boron concentration is at the maximum value and the water volume in the system is at a minimum. (See Section 14 for results of analysis).

At least two separate and independent flowpaths are available for reactor coolant boration; i.e., the charging line or through the two reactor coolant pump labyrinths. The malfunction or failure of one component does not result in the inability to borate the Reactor Coolant System. An alternate flow path is always available for emergency boration of the reactor coolant. As backup to the boration system, the refueling water storage tank outlet can be aligned to the suction of the charging pumps.

Concentrated boric acid can be injected into the reactor coolant system by means of the charging pumps through two flow paths; (a) Normal charging line (30 gpm) and (b) Seal water supply lines to the two reactor coolant pumps while bypassing seal injection filters (8 gpm per pump - 2.5 gpm of which leaks back into the CVCS and 5.5 gpm of which 01450457
passes into the RCS). Each flow path is provided with a flow indicator. Normally, both charging paths are used for a combined charging rate of 40 gpm. However, if the charging line is isolated, the same charging can be achieved by increasing the leakage through the reactor coolant pump labyrinth seals to the reactor coolant system.

Suction to the charging pumps can be delivered through three flow paths; (a) the blender and flow control valve, (b) a local manual valve path, or (c) an emergency boration path through the motor operated valve. Each flow path is provided with a flow meter.

A letdown or charging line break would be indicated by excessive auto makeup to the volume control tank. A break in the charging line or upstream of the letdown orifices in the letdown line would also result in an increase in charging pump speed and a possible hi-speed alarm, depending on the break size. If the break size is such that the charging/letdown system is unable to maintain pressurizer water level, the letdown line isolation valves near the reactor coolant loop would be tripped closed by a low level signal from the pressurizer level instrumentation (redundant valves and level channels are provided to ensure letdown isolation in the event of a single active failure).

A single charging pump has sufficient capacity (60 gpm) to compensate for normal letdown purification flow (40 gpm) and seal return (normally 2.5 gpm/pump) and thus 01450457
maintain the proper reactor coolant total inventory, temperature and pressure.

The refueling water storage tank becomes the source of makeup in the event that leakage makeup is required when the reactor makeup water storage tank is empty.

In the event that the letdown line must be isolated, the charging line is also isolated to avoid thermal shocking of the charging line penetrations into the Reactor Coolant System. If the charging line must be isolated, the letdown line is also isolated to avoid flashing of the letdown stream as the pressure of the high temperature flow is reduced.

With the letdown and charging lines isolated, the Reactor Coolant System can be borated via the reactor coolant pump labyrinth seals by allowing the pressurizer water level to increase, i.e., the pressurizer has sufficient volume to accept the amount of 12 wt% boric acid solution required to borate the system for a cold shutdown. As the system is cooled down, makeup for contraction would be provided through the labyrinth seals at the Mode 5, Cold Shutdown boron concentration. Therefore, the letdown and normal charging path are not required to go to Mode 5, Cold Shutdown condition. If letdown of reactor coolant is necessary, the excess letdown line is capable of letting down a flow equivalent to the total labyrinth seal inleakage from both reactor coolant pumps.

Design seal injection to each reactor coolant pump is 8 gpm with the majority of the flow passing through labyrinth seals into the reactor coolant system and the remainder passing through the shaft seals. The shaft seals are designed to divert nominally 2.5 gpm back to the chemical and volume control system, with a small fraction flowing into the waste liquid system via the reactor coolant drain tank (RCDT). (References 98 and 99) 01450457

With the charging line isolated, the rate of injection of 12 wt.% boric acid solution into the reactor coolant system could be increased to 12 gpm if required (6 gpm from the labyrinth seal leakage through each reactor coolant pump). This charging rate can borate the reactor coolant system to the concentration necessary for Mode 5, Cold Shutdown in less than three hours and is equivalent to a cooldown rate of 2.0°F/min as given in Table 10.2-7.

If the normal letdown line is not in service, the charging line must also be isolated to avoid thermal shocking of the charging line penetration into the Reactor Coolant System. As stated above, with the charging and letdown lines out of service, the length of time necessary to bring the reactor to Mode 5, Cold Shutdown, is not affected because of the capability of increasing leakage through the reactor coolant pump seals to the reactor coolant system. If the charging flow through the reactor coolant pump labyrinth seals is not increased following isolation of the charging line, the time necessary to borate the reactor coolant sufficiently for a cold shutdown is approximately 3 hours.

In the event that injection using the Charging Pumps is not available, the Safety Injection system can perform this function using borated water from the refueling water storage tank. If necessary, the RCS can be sufficiently depressurized to allow injection with the SI Pumps.

On loss of seal injection water to the reactor coolant pump seals, seal water flow may be re-established by manually starting a standby charging pump. Even if the seal water injection flow is not immediately re-established, the plant can continue to operate temporarily since the thermal barrier cooler has sufficient capacity to cool the reactor coolant flow which would pass through the thermal barrier cooler and seal leakoff from the pump volute.

Short term operation is possible but, in order to prevent the reactor coolant pump seals from being exposed to prolonged flow of unfiltered reactor coolant and potentially exceeding their operating temperature limit, it is recommended that this condition be minimized. The effect of continuous reactor coolant pump operation without injection water would be to possibly cause clogging or damage of the seals with the introduction of unfiltered water. 01450457

The thermal barrier cooling coil is a backup to seal injection for cooling the reactor coolant pump bearings and seals and no overheating would result from continued operation without seal injection water for a period of at least one hour following the onset of a loss of seal injection event.

10.2.3.3.6 Deleted

10.2.3.3.7 Fuel Element Failure Detection

The fuel element failure detection system monitors letdown flow using one channel (R-9) of the Area Radiation Monitoring System (see Section 7.5.3.2 and Table 7.5-1).

Delay time for the monitor in detecting fuel element failure ranges from approximately one minute to approximately three minutes, depending on the letdown flow rate. At the high flow letdown flow rate (80 gpm), approximately 1-1/2 minutes pass before the reactor coolant, contaminated by fission product release from the failed fuel element, reaches the monitor. At the normal letdown rate (40 gpm), approximately three minutes pass before the contaminated flow reaches the detection area.

The monitor detects fuel element failure release of fission products against a background of:

a.	N-16 source (assuming a 60 second decay)	4.5×10^1 mR/hr
b.	Nominal corrosion product sources	1.1×10^1 mR/hr
c.	Previous fuel element defects	Determined during operation

Fuel failure severity can be determined by relating any increase in radiation detected to a corresponding increase in either rod gap release or general fuel element defects:

a.	.01% fuel element defects	7.2×10^1 mR/hr
b.	1 rod gap release	3.2×10^2 mR/hr

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10.2.4 Residual Heat Removal System

10.2.4.1 Design Basis

The residual heat removal system is shown in Figures 10.2-11 and 10.2-12. The residual heat removal system is designed to remove residual and sensible heat from the core and reduce the temperature of the Reactor Coolant System during the second phase of plant cooldown. During the first phase of cooldown, the temperature of the Reactor Coolant System is reduced by transferring heat from the Reactor Coolant System to the Steam and Power Conversion System.

Separate and independent residual heat removal systems are supplied for the two units. The description contained herein is equally applicable to either unit.

All active system components which are relied upon to perform their function during an accident are redundant.

The system design precludes any significant reduction in the overall reactor shutdown margin when the system is brought into operation for residual heat removal or for emergency core cooling by recirculation.

The system design includes provisions to enable hydrostatic testing to applicable code test pressures during shutdown.

The system design and operation has been evaluated against Generic Letter 87- 12, “Potential for Loss of Residual Heat Removal (RHR) While the Reactor Coolant System (RCS) is Partially Filled”. It was concluded that the procedures currently in place, in conjunction with the additional procedures, training and modifications described in NSP’s response to that Generic Letter, (Reference 33) provide assurance that this problem will not occur at Prairie Island.

System components, whose design pressure and temperature are less than the Reactor Coolant System design limits, are provided with overpressure protective devices and redundant isolation means.

The equipment utilized for residual heat removal is also used for emergency core cooling during loss-of-coolant accident conditions. This is described in Section 6.

It is the intent of the design of the safety related actions of the Residual Heat Removal System to meet the requirements of IEEE 279 “Standard, Nuclear Power Plant Protection Systems”, August 1968.

10.2.4.2 Description

Two pumps and two residual heat exchangers perform the residual heat removal functions for the reactor. After the Reactor Coolant System temperature and pressure have been reduced to 350°F and 425 psig respectively, residual heat removal is initiated by aligning the pumps to take suction from the hot legs of both reactor coolant loops and discharge through the heat exchangers into the cold leg of one reactor coolant loop. If only one pump and one heat exchanger are available, reduction of reactor coolant temperature is accomplished at a lower rate. For a typical case of 85°F cooling water, such availability would extend the total time of cooldown to refueling shutdown from approximately 27 hours to 159 hours. 01183316

During plant shutdown reactor coolant flows from the Reactor Coolant System to the residual heat removal pumps, through the tube side of the residual heat exchangers and back to the Reactor Coolant System. The heat loads are transferred by the residual heat exchangers to the component cooling water. The cooldown rate of the reactor is controlled by regulating the reactor coolant flow through the tube side of the residual heat exchangers. A by-pass line and an automatic flow control valve around the residual heat exchangers are used to maintain a constant flow through the residual heat removal system.

Redundant, remotely operated valving in the residual heat removal system inlet lines is provided to isolate the system from the Reactor Coolant System. One set of isolation valves, those adjoining the reactor coolant system, are interlocked with a pressure signal to prevent their being opened whenever the system pressure is greater than 425 psig. The valves will also be automatically closed whenever the system pressure increases above 600 psig. This interlock and automatic closing action is derived from one process control channel.

The other set of valves, those adjoining the residual heat removal system, are similarly interlocked and automatically closed with the actions being derived from a second process control channel. A remotely operated valve and a check valve isolate the line to the Reactor Coolant System cold leg. Overpressure in the system is prevented by a relief valve in the inlet piping which discharges to the pressurizer relief tank and by a relief line from the outlet piping to the CVCS letdown line.

Design parameters for the residual heat removal system components are presented in Table 10.2-9.

10.2.4.2.1 Components

a. Residual Heat Exchangers

The two residual heat exchangers are of the shell and U-tube type with the tubes welded to the tube sheet. Reactor coolant circulates through the tubes, while component cooling water circulates through the shell side. The tubes and other surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel.

b. Residual Heat Removal Pumps

The two residual heat removal pumps are vertical centrifugal units with special seals to prevent reactor coolant leakage to the atmosphere. All pump parts in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant material.

c. Residual Heat Removal Valves

The valves used in the residual heat removal system are constructed of austenitic stainless steel or equivalent corrosion resistant material.

Manual stop valves are provided to isolate equipment for maintenance. Throttle valves are provided for remote and manual control of the residual heat exchanger tube side flow. Check valves prevent reverse flow through the residual heat removal pumps.

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.

Manually operated valves have backseats to facilitate repacking and to limit the stem leakage when the valves are open.

d. Residual Heat Removal System Piping

All residual heat removal system piping is austenitic stainless steel. The piping is welded with flanged connections at some components for ease of maintenance.

10.2.4.3 Performance Analysis

The residual heat removal system is connected to the hot legs of both reactor coolant loops on the suction side and to the cold leg of one reactor coolant loop on the discharge side. On the suction side the connections are through two electric motor operated gate valves in series with both valves automatically closed when the Reactor Coolant System pressure exceeds a selected fraction of the design pressure of the Residual Heat Removal System. Once closed, each valve is interlocked to prevent opening unless the Reactor Coolant System pressure is below the Residual Heat Removal System design pressure. On the discharge side the connection is through a check valve in series with an electric motor-operated gate valve. All of these motor-operated valves are closed during normal operation.

Two pumps and two heat exchangers are utilized to remove residual and sensible heat during plant cooldown. If one of the pumps and/or one of the heat exchangers is not operative, safe operation of the plant is not affected; however, the time for cooldown is extended. The function of this equipment following a loss of coolant accident is discussed in Section 6. The entire system is seismic Class I design and the Components are designed to the codes given in Table 10.2-2.

Welded construction is used where possible throughout the residual heat removal system piping, valves, and equipment to minimize the possibility of leakage. Each of the two residual heat removal pumps and associated piping is located in a shielded compartment as described in Section 6. An analysis of potential leakage and the resulting offsite radiological effect has been evaluated and is discussed in Sections 6 and 14. A failure analysis of components is given in Table 10.2-4.

10.2.5 Reactor Makeup Water Deoxygenation System

The Reactor Makeup Water Deoxygenating System is a part of the Reactor Makeup (RM) System. It is common to both units 1 and 2 and is designed to remove oxygen from water stored in any of the four Reactor Makeup Water Tanks. The System can be aligned to degas the water in any one of the tanks at a time.

10.2.5.1 Design Basis

The Reactor Makeup Water Deoxygenation System is designed to: a) maintain the dissolved oxygen content of the Reactor Makeup Water within the limits identified in PINGP Chemistry procedures; and b) deoxygenate all aerated water prior to storage in the Reactor Makeup Tanks, or deoxygenate in a batch process once in the storage tank.

10.2.5.1.1 Codes and Classifications

All components and piping are designed in accordance with the ASME Boiler and Pressure Vessel Code Section VIII and ANSI B31.1 respectively. The system is designated QA Type III. Seller's inspection/test program for the components in this system are in accordance with the requirements of FPI TS QA III, Rev. 2, 1976.

10.2.5.2 Description

The Reactor Makeup Water Deoxygenation System is shown in Figure 10.2-9, and periodically is used to deoxygenate the contents of each Reactor Makeup Water Tank.

The existing Reactor Makeup Water Pump piping is designed so that the pumps can supply water either to the Reactor Makeup services or to the degasifier.

The deoxygenating equipment consists of a packed column degasifier. The water is introduced near the top of the degasifier and is broken up into drops as it trickles down over the plastic packing and collects in the storage section at the bottom of the degasifier.

A full capacity vacuum pump maintains the necessary vacuum in the column. Oxygen and other gases given up by the water are discharged outside the Turbine Building into the atmosphere. The vacuum pump has a pump of equal capacity installed as a backup.

The deoxygenated water, collected in the storage section of the column, is transferred into a predetermined Reactor Makeup water tank by means of a transfer pump. A second transfer pump is installed as a backup.

The degasification equipment is shared between Units 1 and 2.

All components in the Reactor Makeup water deoxygenation system are located in the Unit 1 turbine building. The degasifier, the transfer pumps and the vacuum pumps are located on the ground floor, 695′ - 0″ elevation.

10.2.5.2.1 Components

A summary of principal component data is given in Table 10.2-10.

Degasifier

The Reactor Makeup Water Degasifier is designed for continuous processing of water at a rate of 80 gpm. It is designed to provide a water effluent of 50 ppb (max) dissolved oxygen content based on 2 ppm oxygen content and ammonia free demineralized water influent at 60°F.

Transfer Pumps

Two magnetic drive positive displacement vane type pumps are provided. All wetted parts of the pumps are constructed from stainless steel. The pumps are provided with drains and shut-off valves in both suction and discharge lines for ease of maintenance. 01400506

Vacuum Pumps

Two Rotary Piston, oil lubricated, belt driven vacuum pumps are provided.

10.2.5.3 Performance Analysis

The Reactor Makeup Water Deoxygenation System is not an engineered safety features system since it is not required for safe shutdown.

10.2.5.3.1 Instrumentation and Controls

The Reactor Makeup Water Pumps are manually aligned with the degasifier and manually started. The level in the degasifier is controlled by a control valve in the inlet line. This control valve closes upon detection of low inlet pressure.

One of the two vacuum degasifier transfer pumps may be manually started by means of local switches. Low level in the degasifier automatically trips the operating vacuum degasifier transfer pump in the RM degasifier area. To resume operation, the operator should manually restart the pump. The second pump is held in standby, and if needed, may be manually put in operation.

The discharge flow is controlled by throttling the manual valve in the vacuum degasifier Transfer Pumps discharge.

The vacuum degasifier vacuum pumps are manually started by means of local switches. High levels in the degasifier automatically trip the operating vacuum pump. At the same time an alarm sounds in the control room. To resume operation, the operator should manually restart the pump. The second pump is held in standby, and if needed, may be manually put in operation.

A pressure gage is provided in the discharge of the vacuum degasifier transfer pumps. A vacuum gage and a level gage is also provided on the degasifier. A pressure switch is located upstream of the inlet control valve. Upon low pressure detection by this switch, the inlet control valve closes and an alarm sounds in the control room. The level in the degasifier will drop. The degasifier transfer pump stops on low level and an alarm is sounded in the control room.

An alarm is sounded in the control room when a vacuum loss is sensed in the degasifier with either transfer pump running.

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10.3 PLANT SERVICE SYSTEMS

10.3.1 Plant Fire Protection Program

10.3.1.1 Design Basis

Criterion: The reactor facility shall be designed (1) to minimize the probability of events such as fires and explosions and (2) to minimize the potential effects of such events to safety. Noncombustible and fire resistant materials shall be used whenever practical throughout the facility, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features. (GDC 3)

The Fire Protection System is designed to provide a defense-in-depth principle by achieving an adequate balance in:

a. reducing the likelihood of occurrence of fires;

b. promptly detecting and extinguishing fires if they occur;

c. maintaining the capability to safely shutdown the plant if fires occur; and

d. preventing the release of a significant amount of radioactive materials if fires occur.

NSP is committed to meet General Design Criterion (GDC) 3 as proposed by the Atomic Energy Commission (AEC) as published in the Federal Register on July 11, 1967. The fire protection system was originally designed to AEC GDC 3. When 10CFR50.48 became effective, the NRC's basic criterion for fire protection as set forth in GDC 3, Appendix A to 10CFR50 became applicable to PINGP on October 29, 1980, which states:

> "Structures, systems and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.
>
> "Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room.
>
> "Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems and components important to safety.
>
> "Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems and components."

Following a fire at the Brown's Ferry Nuclear Station in March 1975, the Nuclear Regulatory Commission initiated an evaluation of the need for improving the fire protection programs at all licensed nuclear power plants. As part of this continuing evaluation, the NRC, in February 1976, published the report by a special review group entitled "Recommendations Related to Brown's Ferry Fire," NUREG-0050. This report recommended that improvements in the areas of fire prevention and fire control be made in most existing facilities and that consideration be given to design features that would increase the ability of nuclear facilities to withstand fires without the loss of important functions. To implement the report's recommendations, the NRC initiated a program for reevaluation of the fire protection programs at all licensed nuclear power stations and for a comprehensive review of all new licensee applications.

The NRC issued new guidelines for fire protection programs in nuclear power plants which reflect the recommendations in NUREG-0050. These guidelines are contained in the following documents:

"Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-75/087, Section 9.5.1, "Fire Protection," May 1976, which includes "Guidelines for Fire Protection for Nuclear Power Plants" (BTP APCSB 9.5-1), May 1, 1976.

"Guidelines for Fire Protection for Nuclear Power Plants" (Appendix A to BTP APCSB 9.5-1), August 23, 1976.

"Supplementary Guidance on Information Needed for Fire Protection Evaluation," September 30, 1976.

"Sample Technical Specifications," May 12, 1977.

"Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance," June 14, 1977.

"Manpower Requirements for Operating Reactors," June 1978.

"Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," Appendix R to 10CFR50, November 19, 1980.

"Fire Protection Rule (45 FR 76602, November 19, 1980) Generic Letter 81-12", February 20, 1981.

'NRC Positions on Certain Requirements of Appendix R to 10 CFR 50 (Generic Letter 83-33)", October 19, 1983.

In order to conform with new NRC guidelines issued prior to November, 1980, fire hazard analysis and modifications were submitted to the NRC (see References 7-11, 13, 14). These modifications were based upon fire hazard analysis and NRC's onsite evaluation of the fire protection program. Summary of the modifications and result of NRC's evaluations are discussed in References 12, 15 and 16. Completed modifications of the fire protection system are included in this Section. In November, 1980 Appendix R to 10 CFR 50 was issued by the Staff in an effort to close out those remaining open items from the review initiated after the Browns Ferry Fire and to insure consistency on certain generic topics. Only those generic topics of Sections III.G, III.J and III.O of Appendix R remained open and were required to be reviewed for compliance for the Prairie Island Plant. References 17 thru 19, 21, 23, 24, 26 and 27 contain the safe shutdown analysis, the proposed modifications and the needed exemption requests to show compliance with those applicable requirements of the Appendix R ruling. References 20, 22, 25, 28 and 29 summarize the modifications and result of NRC's evaluation for the exemption requests.

The results of the fire hazards analysis were subsequently incorporated into Operations Manual F5, Appendix F, "Fire Hazards Analysis" (Ref. 50). Similarly, the compliance descriptions for 10CFR50, Appendix R Sections III.G and III.O (including exemptions to 10CFR50, Appendix R requirements) were subsequently incorporated into Operations Manual F5, Appendix E, "Fire Protection Safe Shutdown Analysis Summary" (Ref. 51). These plant documents are maintained current with the plant conditions. Subsequent to completion of the fire hazards analysis and safe shutdown analysis, additional guidance was provided by the NRC which are used when performing changes that may affect the fire hazards analysis and safe shutdown analysis.

Generic Letter 86-10, "Implementation of Fire Protection Requirements"

Generic Letter 86-10, Supplement 1, "Fire Endurance Test Acceptance Criteria for Fire Barrier Systems Used to Separate Redundant Safe Shutdown Trains within the Same Fire Area"

Generic Letter 88-12, "Removal of Fire Protection Requirements From Technical Specifications"

In accordance with the guidance provided in Generic Letters 86-10 and 88-12, the operating license conditions were revised and fire protection systems were removed from the Technical Specifications. Amendment Nos. 120 and 113 revised the Operating License condition 2.C.(4) for fire protection for Units 1 and 2, respectively. The technical specifications for fire protection systems were removed and incorporated into Operations Manual F5, Appendix K, "Fire Detection and Protection Systems" (Ref. 52). In addition, the implementing procedures of the fire protection program were added into Technical Specifications 5.4.1.D, "Plant Operating Procedures."

The following documents include a description of the Fire Protection Program and are therefore, part of the USAR by reference:

Operations Manual, Section F5, Appendix E
Operations Manual, Section F5, Appendix F
Operations Manual, Section F5, Appendix K

See Section 7.8.4 for discussion related to control room design for the prevention of fires. 01484954

10.3.1.2 Description

10.3.1.2.1 Fire Protection System Interface with Other Plant Systems

In addition to the two dedicated fire pumps, a third pump, electric motor-driven and having a capacity of 2000 gpm at a pressure of 125 psi, normally assigned to the screen wash function, can be aligned to pump into the fire water system.

The cooling water system provides a backup source of water for the fire protection headers. Via normally closed valves, the cooling water system can provide a backup water supply for the following safeguards equipment fire protection spray and sprinkler systems: auxiliary feedwater pumps, diesel generators, containment cable penetrations, and safeguards cooling water pumps. The fire protection system flow diagrams are shown in Figures 10.3-1 through 10.3-5.

10.3.1.2.2 Deleted

10.3.1.2.3 Emergency Communication and Lighting Systems

The plant is equipped with emergency communication and lighting systems that are effective and useful for the fire fighting operation (see Sections 10.3.6 and 10.3.8).

10.3.1.2.4 Administrative Controls

The administrative controls for the plant's fire protection consist of the fire protection organization and its qualifications, fire brigade training, the controls over combustibles and ignition sources, methods for assuring the availability of the fire protection systems and equipment, procedures for fighting fires, fire watch, and quality assurance provisions for the fire protection program.

A description of the administrative controls for fire protection was submitted to the NRC and was found to conform to the provisions of Appendix A to BTP 9.5-1 (see References 9, 12, 13 and 14). The fire protection program implements selected elements of the Quality Assurance Topical Report, NSPM-1.

Implementing procedures of the fire protection program are governed by Technical Specification 6.4.D [ITS 5.4.1].

10.3.1.3 Performance Analysis

10.3.1.3.1 General

The performance description and operability requirements of the fire detection and fire protection systems are described and governed by Operations Manual F5, Appendix K.

10.3.1.3.2 deleted per 98133

10.3.1.3.3 deleted per 98133

10.3.1.4 Inspection and Testing

The fire protection systems and components were tested prior to placement in service. Surveillance and inspection requirements are identified in Operations Manual F5, Appendix K, “Fire Detection and Protection Systems.”

10.3.1.5 Safe Shutdown Systems Analysis

10.3.1.5.1 General

To comply with 10CFR50, Appendix R, Section III.G, a safe shutdown analysis was performed (Ref. 17) which was subsequently included in Operations Manual F5, Appendix E. This procedure describes the safe shutdown methodology, including the identification of safe shutdown systems, safe shutdown components, associated circuits concerns (common power supply, common enclosure, and spurious operation concerns), and a summary of approved exemptions from Appendix R requirements. This procedure also summarizes the analysis of safe shutdown equipment and circuits, and is maintained current with plant design.

For a fire in the control room and relay room, which requires control room evacuation, alternative shutdown capability is implemented from the hot shutdown panel location. The equipment protected from a fire in the control room and relay room was analyzed in the safe shutdown analysis. The alternate shutdown methodology is described in Operations Manual F5, Appendix E.

Operations Manual F5, Appendix E also includes a summary of compliance with 10CFR50, Appendix R, Sections III.J (Emergency Lighting) and III.O (Reactor Coolant Pump Lube Oil Collection Systems).

10.3.1.5.2 deleted per 98090

10.3.2 Auxiliary Building Normal Ventilation Systems

10.3.2.1 Design Basis

The Auxiliary Building Normal Ventilation System is designed to provide maximum safety and convenience for operating personnel, with equipment arranged so that potentially contaminated areas are separated from clean areas.

The ventilation equipment complies with accepted industry standards for power plant equipment and with all applicable state and local codes and regulations. Redundant equipment is provided for those systems where, in case of malfunctions, public health and safety may be endangered or where safeguard equipment operation may be impaired. Particulate filters, where used, are of the high-efficiency type capable of removing at least 99.97% of 0.3 micron diameter DOP smoke particles. These filters meet all requirements of AEC Health and Safety Bulletin 212, dated June 25, 1965.

The design of the Auxiliary Building Normal Ventilation (ZD) system is to maintain temperatures of 10°F or less above the outdoor ambient in critical area; i.e., motors, instrumentation, etc. To account for time periods when the ZD system is not fully operational or extremely hot outside air temperatures, the following temperature limits have been established in the Auxiliary Building (these limits do not apply during post accident operation):

a. On the 695′ floor, the established limit is 104°F.

b. On floors above 695′, the limiting temperature sensitive equipment is the Motor Control Centers (breakers, overloads, etc.). Thus, two different temperature limits are established.

    1. In the vicinity of the MCCs, the established limit is 122°F.

    2. Other areas the established limit is set by the assumptions in the EQ program.

    (Reference 89)

The ventilating equipment is accessible for periodic testing and inspection during normal operation.

10.3.2.2 Description

The Auxiliary Building has separate normal ventilation systems to serve the auxiliary equipment areas, the Spent Fuel Pool Area, and the nonradioactive area, as shown in Figure 10.3-6. The path of ventilating air is from clean, or low activity areas toward areas of progressively higher activity. Ventilation air is drawn from outside, through two makeup air units (two each per unit). The system is balanced to maintain the Auxiliary Building at a pressure slightly negative with respect to atmospheric and adjacent Turbine Building pressures.

Special provisions are also included to exhaust air through activated charcoal beds and high-efficiency filters from areas subject to possible radioactive contamination. This is accomplished through local ventilation systems and the use of the Auxiliary Building Special Ventilation System. The Auxiliary Building Special Ventilation System is actuated by either a Safety Injection Signal, or a high radiation signal from the Auxiliary Building vent monitor. This system is described in Section 10.3.4.

The Spent Fuel Pool has been provided with normal and special ventilation systems. The completely enclosed pool area is normally ventilated and exhausted through roughing and HEPA filters. In the event of high radiation in the pool area, the normal system is isolated and the special exhaust system is actuated, directing air flow through redundant, roughing, HEPA and charcoal filters as described in Section 10.3.7.

10.3.3 Control Room Ventilation System

10.3.3.1 Design Basis

The control room ventilation system shall isolate the outside atmosphere and recirculate a portion of the control room atmosphere through PAC filters to maintain the dose to the control room operator less than the limits specified in 10 CFR 50, Appendix A, GDC 19, for a design basis accident. Isolation and recirculation is accomplished through an automatic start signal generated from either a SI signal or a high radiation signal. Dose to control room operators are maintained less than the limits through minimizing inleakage to the control room envelope and clean up of the activity that may enter the control room.

The control room shall be maintained at suitable temperature conditions for personnel habitability and equipment operability. This is accomplished by providing safeguards chilled water (Section 10.4.3) to the control room ventilation air handling units.

The control room ventilation system is capable of performing the design functions in the event of a single active failure. Redundant trains of control room ventilation are provided for this purpose.

The control room ventilation system is capable of performing the required functions in the event of a loss of off-site power. In the event of a loss of off-site power, the ventilation system is automatically powered from the Emergency Diesel Generators (Section 8.4.3).

The control room envelope, the control room ventilation system and the structure housing it is designed to withstand a seismic event (Section 12).

The control room ventilation system components are capable of performing their safety functions under the worst case design basis environmental conditions. The control room ventilation system and boundary shall be capable of providing adequate protection in a high energy line break event (Appendix I). The control room ventilation system is located in a mild environment.

The control room ventilation system and boundary shall be capable of providing adequate protection to the operators in a toxic gas release event (Section 2.9.4).

The control room ventilation system and boundary shall be capable of providing adequate protection in a smoke event such that the operators can achieve and maintain safe shutdown from either the control room or the hot shutdown panels.

10.3.3.2 System Description

10.3.3.2.1 Control Room Envelope

The control room envelope consists of the control room and the two chiller rooms. The control room is a common structure that contains the controls for both Unit 1 and Unit 2. The control room is located at elevation 735′ within the Auxiliary Building approximately equidistance between Unit 1 and Unit 2. The chiller rooms are located directly above the control room at elevation 755′. The cable spreading room on the 715′ elevation (directly below the control room) is not part of the control room envelope. HVAC ducting between the cable spreading room and the control room has been isolated by closing the dampers and installing blanks. The control room ventilation system is entirely located within the two chiller rooms (one train of ventilation system in each room), with the exception of the outside air supply. The outside air supply ducting is routed through the Auxiliary Building. The outside air supply dampers are located at the envelope boundary. There are no other ventilation systems that penetrate the control room envelope.

The Auxiliary Building is a Class I structure, designed to withstand a seismic event; described in USAR, Section 12. The Control Room and Chiller Rooms are protected from adverse environmental conditions by the ceilings, walls and floors. In areas where the ventilation system provides possible communication paths, dampers in the ventilation system automatically close to isolate the envelope from a steam environment (referred to as Steam Exclusion). Protection from a High Energy Line Break is described in more detail in USAR, Appendix I. The concrete walls and roof provide shielding for the Control Room (Section 12.3.2.2.2).

The walls, floor and ceiling of the Control Room are of fire resistant construction and have a three-hour fire rating. Ratings for doors, penetrations and ventilation dampers are discussed in the site Fire Hazards Analysis (Reference 88).

10.3.3.2.2 Control Room Ventilation System

The Control Room ventilation system is designed to provide a reliable means of cooling and filtering air supplied to the Control Room under both normal and post-accident conditions. Figure 10.3-7 shows a simplified drawing of the Control Room Ventilation system. The Control Room ventilation system has two modes; a normal mode and an emergency mode.

<u>Normal Mode</u>

During normal operation one train is running and the other train is in standby. For the operating train, the air handler would be operating and the clean-up fan would be in standby with no air flow through the PAC filter. The operating train recirculates the control room envelope air and draws in fresh air. This recirculation flow rate is not filtered; i.e., the Air Handler Unit roll filter is not credited. Air is exhausted from the Control Room Envelope at a rate equivalent to the quantity of fresh air brought in. The design flow rates are 10,000 cfm recirculation flow rate and 2000 cfm fresh air for a total air handler flow rate of 12,000 cfm. For the dose analysis, the normal recirculation flow rate is not credited and, thus, is not an input. The input for normal operation in the dose analyses and toxic gas analyses is the fresh air supply flow rate. Humidifiers are provided as part of the ventilation system for maintaining the humidity in the control room. Functioning of the humidifiers is not required for habitability or equipment operability.

<u>Emergency Mode</u>

In response to a Safety Injection or high radiation signal, both trains start. Starting both trains automatically isolates the fresh air supply and exhaust and starts the Clean-Up fan. The portion of the air that is drawn by the clean-up fan passes through a PAC filter that is credited in the dose analysis. In this alignment, the system is recirculating and filtering the control room atmosphere. To account for a single active failure, only one train of control room ventilation system is credited in the dose analysis. In the emergency mode, the clean-up fan is designed to provide 4000 cfm $\pm$10%.

The safety injection signal is generated from low Pressurizer Pressure, low Main Steam Line Pressure or high Containment Pressure. The high radiation signal is generated from radiation monitors (R-23 and R-24) located in the control room. The radiation monitors sensing lines penetrate the control room supply ductwork downstream of the control room HVAC filter unit inside the Control Room in the supply distribution ductwork. A “high” signal from either detector will automatically switch the control room ventilation system from the normal mode of operation to the emergency mode.

On a Steam Exclusion signal (high temperature in the outside air supply or exhaust ductwork) redundant outside air dampers automatically close to protect the control room from a potential steam environment. The RTDs that sense the high temperature conditions are set to actuate the dampers at a temperature less than 120°F. It has been demonstrated (Reference 84) that the outside supply air ducting that routes through the Auxiliary Building is capable of withstanding maximum external pressures predicted for the high energy line break events.

10.3.3.3 Performance Analysis

The discussion of performance analysis is separated into overall design evaluations and equipment performance requirements. Overall design evaluations (Section 10.3.3.3.1) are the radiological, toxic chemical and smoke analysis and assessments. Equipment performance requirements (Section 10.3.3.3.2) provide the discussion of the performance capabilities of various components.

10.3.3.3.1 Design Evaluations

Control room habitability is evaluated for radiological, toxic chemical and smoke considerations.

10.3.3.3.1.1 Radiological Analysis

Radiological analyses are performed to demonstrate that the dose to the control room operators is less than the limitations in 10CFR 50.67. Specifically, the acceptance criteria in the dose analyses for the control room operators is 5 Rem TEDE.

The dose analyses are discussed in more detail in Section 14.

10.3.3.3.1.2 Toxic Chemical Analysis

As discussed in more detail in Section 2.9.4, protection from toxic chemicals is provided by detection (smell) and the ability to don SCBAs prior to the chemical concentration in the control room reaching debilitating levels.

10.3.3.3.1.3 Smoke

Using the guidance in NEI 99-03, Rev 1 (Reference 85), Appendix A, it has been shown that a single smoke event originating from inside or outside the Control Room would not affect both the Control Room and Hot Shutdown Panel areas. Operators would be able to achieve and maintain safe shutdown (reactor control capability) from either the Control Room or the Hot Shutdown Panels if needed. In the event that control room evacuation were necessary, the plant can be shutdown from outside the control room as described in USAR, Section 7.8.5. This is for control room evacuation for a non-Appendix R fire related event. For shutdown outside the control room for a fire event, the methods are described in the Prairie Island (Appendix R) safe shutdown analysis and associated procedures. This is discussed in more detail in the Prairie Island response to Generic Letter 2003-01, Control Room Habitability (Reference 86).

10.3.3.3.2 Equipment Performance Capabilities

The control room ventilation system design basis functions are met by performance of the following functions.

10.3.3.3.2.1 Automatic Actuation

The control room ventilation system is aligned to the emergency mode by either a SI or High Radiation signal. In most instances, the high radiation signal and the SI signal are redundant. The exceptions are that, for the following scenarios, the high radiation signal is relied on to back-up the SI signal.

- If 121 CR Vent is initially running and Train A SI signal fails to actuate, the High Radiation signal (R-23 and/or R-24) is relied on to close the 121 CR Vent outside air dampers.

- If 122 CR Vent is initially running and Train B SI signal fails to actuate, the High Radiation signal (R-23 and/or R-24) is relied on to close the 122 CR Vent outside air dampers.

The timing of the actuation signals is a consideration in the dose analyses. Until the outside air dampers are closed, the potential exists for activity to enter the control room envelope through the normal outside air supply ducting. In order to actuate the high radiation signal, the activity level needs to build-up to reach the radiation monitor setpoint. The time to reach the high radiation setpoint could be longer than the time to actuate SI. The dose analyses accounts for the longer time delay of the SI and high radiation signals.

10.3.3.3.2.2 Air Handling Units

The air handling unit is provided for control room cooling. Safeguards chilled water system provides the heat sink for the air handling units. There are two redundant air handling units; either one is capable of providing the required cooling. An air handling unit is automatically started upon start of the associated Clean-Up fan or the air handling unit may be started manually. Starting the air handling unit opens the associated discharge damper.

10.3.3.3.2.3 Clean-Up Fans

The clean-up fans are provided, in the event of a radiological accident, to recirculate a portion of the control room atmosphere through PAC filters to reduce the dose to the control room operators. Two redundant clean-up fans are provided; either one is capable of providing the credited flow rate. A clean-up fan is automatically started by SI signal from either unit or high radiation signal detected in the control room. Manual initiation of the clean-up fans is also provided. Starting a clean-up fan opens the associated suction and discharge dampers. The radiological analyses credits a clean-up fan air flow rate of 3600 cfm; corresponding to one train at minimum flow. No credit is taken for clean-up fan flow in the toxic chemical or smoke assessments.

10.3.3.3.2.4 PAC Filters

The PAC filters are provided, in the event of a radiological accident, to filter the associated clean-up fan air flow. Two redundant PAC filters are provided; either one is capable of providing the credited filtration. Each PAC filter consists of particulate, absolute and charcoal filters. Each PAC filter is designed for 4000 cfm air flow. Filter efficiency testing is based on maximum air flow (4400 cfm) with a residence time of 0.185 seconds. Per the ventilation filter testing program (T.S.5.5.9), the HEPA filter is tested to 0.05% penetration and system bypass and the charcoal is tested to 2.5% penetration. Charcoal adsorber samples are laboratory tested at a face velocity of 54 fpm, 30°C [86°F], and 95% RH. (Reference 90)

10.3.3.3.2.5 Outside Air Dampers

During normal operation the outside supply air dampers associated with the operating air handling unit and the discharge dampers from 122 Chiller Room to the Auxiliary Building Special Ventilation Zone are typically open. Two redundant outside air dampers are provided in each flow path. The outside air dampers are automatically closed by a SI signal, high radiation signal in the Control Room or a Steam Exclusion signal. Time delays for closing of the outside air dampers are factored into the dose analyses in conjunction with the time delays for the actuation signal. The operations of other dampers in the system are discussed within the discussion of the associated fans.

For post-accident operation, the operator may elect to stay in the 100% recirculation mode, or the operator can add outside air to the Control Room. By proper damper manipulation, the operator may direct outside makeup air through the PAC filter, using the clean-up fan, then to the associated air handling unit. The outside air sources are physically separated to permit taking fresh air from an area of low airborne contamination.

10.3.3.3.2.6 Unfiltered Inleakage

Unfiltered inleakage to the control room envelope was determined using tracer gas testing techniques in 1998 and 2004 (LOOP results are documented in Reference 87). The results for total unfiltered inleakage to the control room envelope from 2004 are as follows:

System Configuration	Inleakage Rate (cfm)
High Radiation	115 ± 36
SI	114 ± 21

The system configuration for the High-Radiation vs. the SI signal affects the ventilation systems in the adjacent spaces (Aux Bldg and Turbine Bldg); but, it does not affect the emergency alignment of the control room ventilation system. The alignment of the ventilation systems in the adjacent spaces affects the differential pressures across the control room envelope boundaries; which can affect the unfiltered inleakage. With the High Radiation signal, the configurations of the ventilation systems in the adjacent areas is more representative of a FHA. With the SI signal, the configuration of the ventilation systems in the adjacent areas is more representative of a MSLB or LOCA event.

10.3.3.4 Inspection and Testing

The control room ventilation system is tested per the associated Technical Specification Surveillance Requirements. These include testing of the PAC filter, flow verification of the clean-up fan and verification that the system actuations are functioning properly.

Door seals for doors that are part of the control room envelope boundary are maintained in good condition. The door seals are inspected and replaced (if necessary) on an annual basis as part of the Preventative Maintenance program.

Periodic inspections are performed of the outside air damper (supply and exhaust) seating surfaces to ensure that the dampers that provide control room envelope integrity are in good condition.

Electrical and Mechanical control room boundary penetrations are inspected and maintained in accordance with the Fire Protection preventative maintenance program.

The Control Room Habitability Program ensures the Control Room Envelope (CRE) is adequately maintained. Unfiltered in-leakage testing of the CRE is performed in accordance with the Control Room Habitability Program.

Each of the Control Room Chiller Rooms has a single floor drain that is part of the Waste Liquid System (WL). The floor drains have loop seals that are designated as Auxiliary Building Cat I Vent Zone boundaries. Control Room in-leakage testing acceptance criteria can be met with the loop seals full or empty but they are kept full to provide additional Control Room Envelope in-leakage margin.

10.3.4 Auxiliary Building Special Ventilation System (Category I Ventilation Zone)

10.3.4.1 Design Basis

The Auxiliary Building Special Ventilation System conforms to the intent of Proposed IEEE Criteria for Nuclear Power Plant Protection Systems IEEE 279-68 and is designed to reliably collect significant portions of any potential containment system leakage that might bypass the Shield Building annulus, as discussed in Section 5.4.3, and to cause it to pass through charcoal filters before reaching the environment. The system is also provided to filter any leakage from systems which could recirculate primary coolant during LOCA mitigation. The filter assemblies of the auxiliary building special ventilation system and the shield building ventilation system are similar in design parameters. Table 5.3-3 and Figure 5.3-3 apply to both systems. These design parameters were originally used for equipment sizing and selection, and do not necessarily reflect operating conditions. To ensure confinement of such by-pass leakage, and its removal by the Auxiliary Building Ventilation System, the areas within the Auxiliary Building where there is the potential for such leakage are designated as a Category I Ventilation Zone. Plan and elevation views of the Auxiliary Building Category I Ventilation Zone are given in Figures 1.1-5, -6, -7, -8, -15 and -16. As shown, the Shield Buildings of Units 1 and 2 comprise partial boundaries of this zone. Some of the major systems within this zone are:

a. Residual Heat Removal System

b. Safety Injection System

c. Containment Vessel Internal Spray System

d. Component Cooling System

e. Chemical and Volume Control System

f. Waste Gas Handling System

The nature of the potential leakage paths from the Containment System to the Auxiliary Building is discussed in detail in Appendix G. From this above discussion it is apparent that the only leakage of any consequence that could by-pass the Shield Building would be from Group VI penetrations and through packings, etc., of some of the other penetration groups. Because these potential sources of leakage are minor (mostly small valves or stem packing) it is expected that leakage during a LOCA into the Auxiliary Building Special Ventilation Zone would be much less than the assumed value in the offsite dose analysis (see Section 14.9).

The entire Category I Ventilation Zone is housed within Class I structures and is designed to function reliably under the environment conditions discussed in Section 12. The system is capable of drawing a measurable vacuum in the zone within 20 minutes after initiation.

10.3.4.2 Description

The entire perimeter of the Auxiliary Building special ventilation zone is defined as a medium-leakage type barrier. Most walls, ceilings and floors are constructed of poured concrete with sealed joints where required. Doors penetrating the perimeter of the Auxiliary Building Category I Ventilation Zone are fitted and weather-stripped to minimize leakage.

The floor drains in the Auxiliary Building are part of the Waste Liquid System (WL). Various loop seals located within the Auxiliary Building are designated as Auxiliary Building Cat I Vent Zone boundaries. Auxiliary Building Special Ventilation testing has proven effective with the loop seals either empty or full.

The Auxiliary Building Special Ventilation System is provided with redundant exhaust ducts that connect redundant filter assemblies to the exhaust ductwork; electric heaters; roughing, absolute, and charcoal filters; and then to redundant exhaust fans, as shown in Figure 10.3-6 and discussed in Section 5.3.2.

The Auxiliary Building Special Ventilation System provides a sub-atmospheric pressure in, ventilation for, and fission-product removal from the Auxiliary Building Category I Ventilation Zone. The single system of trunk and branch ducting has 100% redundant roughing-particulate-charcoal filters, exhaust fans, and discharge ducting to the Shield Building’s vent stacks.

The initiating signal for the Auxiliary Building Special Ventilation System is a Safety Injection Signal, described in Section 7, or a signal from the detection of high radiation in the Auxiliary Building Vent. When the Auxiliary Building Special Ventilation System is actuated, the normal supply and exhaust ducts from the Category I zone are closed automatically, and the normal supply and exhaust fans for the Auxiliary Building are tripped.

To provide continued cooling for safety features components located within the Category I zone under these conditions, small unit coolers are located throughout the Auxiliary Building to recirculate and cool the air as required. These units receive cooling water from the Class 1 Cooling Water System or from the Containment and Auxiliary Building Cooling System shown in Figure 10.3-17. Specific analyses has demonstrated that the cooling units in the SI, CS and CC Pump areas are not required to ensure component operation in an accident scenario.

10.3.4.2.1 Doors

The Category I ventilation zone is bounded by a system of single doors and pairs of doors for low traffic usage. A local audible alarm will sound if a door of the single door configuration is opened. A control room visual alarm will illuminate if a door of the single door configuration is held open for more than 30 seconds. The air lock type passage doors (pair-configuration) may be normally open one at a time. When both doors are open, an alarm sounds locally. A control room visual alarm will illuminate if both doors of the pair-configuration are held open for more than 30 seconds. All Category I zone doors are furnished with heavy duty closers. Air flow into the Category I zone due to the differential pressure assures that these doors remain closed for proper functioning of the Auxiliary Building Special Ventilation System.

10.3.4.2.2 Fans

The Category I ventilation zone exhaust fans are axial flow type, direct-connected fans of standard construction.

10.3.4.2.3 Filter Assemblies

The filter assemblies are composite units consisting of electric heating elements, roughing filters, HEPA filter section, impregnated charcoal bed filter sections. Each section is designed as follows:

a. The heating coil is designed to dry incoming air at 100% saturation by increasing the temperature of the air entering the charcoal bed. The air is then dry enough to support the charcoal adsorber iodine removal efficiency requirements. An interlock with the Category I ventilation fans is provided to assure that the electric heater is functional before the fan starts.

b. The roughing filter is designed as a standard particulate removal air filter.

c. The high-efficiency particulate filters are designed to be capable of removing 99.97 percent minimum of particulate matter, 0.3 micron or larger in size. Filter design is water and fire resistant and meets all requirements of AEC Health and Safety Bulletin 212-1965.

d. The iodine filter is an impregnated activated-charcoal bed, designed to remove 99.9 percent minimum of elemental iodine and 95% minimum of methyl iodine. These filters were sized based upon an atmosphere of 150°F, 70 percent relative humidity, a filter depth of 2 inches and a residence time of 0.25 seconds. The ignition temperature for the charcoal used is greater than 330°C.

Filter design, evaluation, reliability, and testing is the same as the Shield Building Ventilation System described in Sections 5.3.2.2.4, 5.3.2.3, and 5.3.2.4, except as noted above. Filter testing conditions, per the ventilation filter testing program (T.S.5.5.9) envelope expected worst-case operating conditions. (Reference 90)

The charcoal filters were sized based on ventilation flow requirements. This gave filters that are similar (including fission product retention capability) to those being used in the Shield Building Ventilation System described in Section 5.3.2.2.4. Since the iodine handling capability of these charcoal filters is so much greater than any possible iodine loading in this system, no further calculations are required.

Sufficient charcoal is provided to accommodate 100% of the radioiodine postulated to leak from the primary containment. This amount of leakage does not load the Auxiliary Building charcoal in excess of 10 mg/gram of elemental iodine or 3 mg/gram of organic iodine. For purposes of charcoal bed sizing, it was assumed that 10% of the total iodine would occur in organic form.

The exhaust air is discharged through the Shield Building Vent Stack which extends through the roof of the Shield Building. The Auxiliary Building Special Ventilation System flow capacity is sufficient to provide a measurable negative pressure in the Category I Ventilation Zone under the credible environmental and operating conditions. Shield walls are provided to protect personnel from radiation exposure due to contaminated charcoal filters during post accident operation of vital motor control center areas.

10.3.4.3 Performance Analysis

The leakage specification for the Category I Ventilation Zone stipulates there be no exfiltration of air from the Category I boundary with only one of the redundant trains of the Auxiliary Building Special Ventilation System in operation. Initial acceptance tests demonstrated the integrity of the Category I Ventilation Zone Perimeter, as well as the capacity of the Ventilation System. Because of the steep head versus capacity characteristic curve for the system fans, the flow is relatively unaffected by the vacuum attained, therefore vacuum has little significance in demonstrating the functional performance of Category I Ventilation Zone and the associated Auxiliary Building Special Ventilation System.

The steady state leakage with one train operating represents an inleakage of approximately 600%/day.

The significant parameter in demonstrating the performance of the Category I Ventilation Zone is the face velocity across openings in the perimeter. The industrial ventilation practice as given by the American Conference of Government Industrial Hygienists recommends a minimum face velocity of 50 ft/min for exhaust hoods. Margin has been provided for the Prairie Island Design by sizing the Auxiliary Building Special Ventilation System to produce an adequate negative pressure, so that with a door open, air flow through the opening has a minimum face velocity of 80 ft/min.

The capacity of the Auxiliary Building Special Ventilation System was increased by replacing the original 4800 cfm units with 9000 cfm fans. The modification enables the system to attain the design conditions of negative pressure within the Category I Ventilation Zone. Measured train flow was increased to range from approximately 7500 cfm to approximately 9000 cfm. Train flow is subject to the leaktightness of the building, variations in air density, etc. Such variability is expected.

10.3.4.4 Inspection and Testing

The following inspections and tests were performed to provide assurance that the functional intent of the system is achieved during the manufacture of the components and the construction of the system.

a. All ducting and filter assemblies were given a pneumatic test and leak test.

b. Each filter assembly received a filter performance test. Each HEPA and charcoal filter was tested in place to verify performance.

c. Dimensional tolerances on filter assemblies and frame assemblies were checked to assure that suitable gasket compression was uniformly achieved on the filter sealing faces.

d. Charcoal filters used Barnebey Cheney Type 727 or equivalent activated charcoal. This material was qualified by previous NRC testing. Each batch was tested by the manufacturer to assure it has removal capability equivalent to the activated charcoal that was used in the qualification testing.

e. Each charcoal bed filter was assembled at the manufacturer's shop and given a Flow Resistance Test and a Leak Test.

f. High efficiency particulate absolute filters were randomly tested to demonstrate the filter's ability to withstand a pressure differential of 10 inches of water without loss of filtering efficiency.

g. HEPA filter of identical design to those in the filter assemblies was subjected to a rough handling test (3/4-inch amplitude at 200 cycles/min.) following which the filter must demonstrate no loss of filtering efficiency.

A pre-operational acceptance test was performed to demonstrate the capability of the Auxiliary Building Special Ventilation System to accomplish the following functions:

a. To start both trains of redundant equipment upon initiation by a simulated normal actuation signal with coincident isolation of the normal ventilation supply and exhaust ducts for the Special Ventilation Zone.

b. To demonstrate by means of smoke tests, for each redundant train, that there is no exfiltration of air from various size openings in the Category I boundary. Further to establish a limiting size for openings and to serve as an operating limit whenever containment operability is required. NOTE: These tests were performed with a simulated dirty filter condition which would be the normal minimum flow for the system.

The acceptance test also determined the limiting total leakage area in the Category I ventilation zone which, with only one filter train and fan operating, will still obtain sufficient in leakage velocities to prevent air exfiltration from the Category I Ventilation Zone. This limiting area was established as an operating limit whenever containment operability, as defined in Technical Specification is required.

Detailed visual inspection of the filters is done during surveillance procedures for each filter. The procedures require inspecting for broken or damaged filters and improper fitting of gaskets. Dioctyl phthalate (DOP) and iodine removal efficiency testing is performed in accordance with the Technical Specifications. Filter efficiency testing is based on maximum air flow (8800 cfm) with a residence time of 0.139 seconds. Per the ventilation filter testing program (T.S.5.5.9), the HEPA filter is tested to 0.05% penetration and system bypass and the charcoal is tested to 10% penetration. Charcoal adsorber samples are laboratory tested at a face velocity of 72 fpm, 30°C [86°F], and 95% RH. (Reference 90)

10.3.5 Sampling System

10.3.5.1 Design Basis

a. Performance Requirements

This system provides samples for laboratory analysis to evaluate reactor coolant and other reactor auxiliary systems chemistry during normal operation. It has no emergency function. This system is normally isolated at the containment boundary.

Sampling system discharge flows are limited under normal and anticipated fault conditions (malfunctions or failure) to preclude any fission product releases beyond the limits of 10CFR20. Each unit has an identical sampling system and no equipment is shared between units except the drains and vents to the waste disposal system. The description contained herein is equally applicable to either unit.

b. Design Characteristics

The system is capable of obtaining reactor coolant samples during reactor operation and during cooldown when the system pressure is low and the residual heat removal loop is in operation. Access to the containment is not required.

Sampling of other process coolants, such as of tanks in the Waste Disposal System, is accomplished locally. Equipment for sampling secondary and non-radioactive fluids is separated from the equipment provided for reactor coolant samples. Leakage and drainage resulting from the sampling operations are collected and drained to tanks located in the Waste Disposal System, the Volume Control Tank and the Steam Generator Blowdown System.

Two types of samples are obtained by the system: 1) high temperature, high pressure Reactor Coolant System and steam generator blowdown samples which originate inside containment; and 2) low temperature, low pressure samples from the Chemical and Volume Control and Auxiliary Coolant Systems.

1. High Pressure - High Temperature Samples

A sample connection is provided from each of the following:

a. The pressurizer steam space

b. The pressurizer liquid space

c. One hot leg

d. Blowdown from each steam generator

The omission of a sampling line for sampling the discharge from the charging pumps for Prairie Island has no safety significance. This sampling line is not included in Westinghouse standard design and is not considered to be a critical part of the sampling system.

2. Low Pressure - Low Temperature Samples

A sample connection is provided from each of the following:

a. The mixed-bed demineralizer inlet header

b. The mixed-bed demineralizer outlet header

c. The residual heat removal loop, just downstream of the heat exchangers

d. The volume control tank gas space

e. The automatic gas analyzer vent line (common to Units 1 and 2)

c. Expected Operating Temperatures

The high pressure, high temperature samples and the residual heat removal loop samples leaving the sample heat exchangers are held to a temperature of 120°F to minimize the generation of radioactive aerosols.

d. Codes and Standards

System component code requirements are given in Table 10.3-1.

10.3.5.2 Description

The Sampling System, shown in Figure 10.3-10, provides the representative samples for laboratory analysis. Analysis results provide guidance in the operation of the Reactor Coolant, Residual Heat Removal, Steam and Chemical and Volume Control Systems. Analyses show both chemical and radiochemical conditions. Typical information obtained includes reactor coolant boron and chloride concentrations, fission product radioactivity level; hydrogen, oxygen, and fission gas content; corrosion product concentration; and chemical additive concentration.

The information is used in regulating boron concentration adjustments, in evaluating fuel element integrity, mixed-bed demineralizer performance, and in regulating additions of corrosion controlling chemicals to the systems. The Sampling System is designed to be operated manually on an intermittent basis. Samples can be withdrawn under conditions ranging from full power to Mode 5, Cold Shutdown.

Reactor coolant liquid lines, which are normally inaccessible and which require frequent sampling, are sampled by means of permanently installed tubing leading to the sampling room.

Sampling System equipment is located inside the auxiliary building with most of it in the sampling room. The sample lines with remotely operated valves are located inside the reactor containment.

Reactor-coolant hot-leg liquid, pressurizer liquid, and pressurizer steam samples originating inside containment flow through separate sample lines to the sampling room. Each of these connections to the Reactor Coolant System has a remotely operated valve located close to the sample source, plus a remotely operated isolation valve immediately outside the reactor containment. Each sample line isolation valve shown on Figure 10.3-10 receives an isolation trip signal automatically following a loss-of-coolant accident. The samples pass through the containment, to the Auxiliary Building, and into the sampling room, where they are cooled (pressurizer steam samples are condensed and cooled) in the sample coolers. The sample steam pressure is reduced by manual throttling valves located in each sample line train. The sample stream is purged to the Volume Control Tank in the Chemical and Volume Control System until sufficient purge volume has passed to permit collection of a representative sample. After sufficient purging, the sample pressure vessel is isolated and then disconnected for laboratory analysis of the contents.

Alternately, liquid samples may be collected by bypassing the sample pressure vessels. After sufficient purge volume has passed to permit collection of a representative sample, a portion of the sample flow is diverted to the sample sink where the sample is collected.

The reactor coolant sample originating from the Residual Heat Removal loop is connected into the sample line coming from the hot leg at a point upstream of the sample cooler. Samples from this source can be collected either in a sample pressure vessel or at the sample sink as with hot leg samples.

Liquid samples originating at the Chemical and Volume Control System letdown line at the mixed-bed demineralizer inlet and outlet pass directly through the purge line to the volume control tank. Samples are obtained by diverting a portion of the flow to the sample sink. The sample line from the gas space of the volume control tank delivers gas samples to the volume control tank sample pressure vessel in the sampling room. Purge flow for volume control tank vapor space sample is discharged to the vent header to the waste gas compressor.

Samples of the steam generator liquid are obtained from the blowdown lines of each steam generator by separate sample lines. These lines are missile protected within the containment and are equipped with double shut-off valves at the sample lines take-off and remote operated isolation valves on both sides of the containment penetrations. The remotely operated valves are automatically closed upon receipt of a signal from the containment isolation system.

The blowdown sample lines are routed to the sample room where they are split into two branches: the first is cooled, pressure-reduced, and routed to the sample sink to provide periodic samples for chemical and radiochemical analyses; the second branch is routed to an automatic analysis apparatus. This automatic analysis apparatus handles a continuous flow for the continuous automatic determination of pH, conductivity, and sodium.

The sample sink, which is contained in the laboratory bench as a part of the sampling hood, has a drain line to the Waste Disposal System.

Local instrumentation is provided to permit manual control of sampling operations and to ensure that the samples are at suitable temperatures and pressures before diverting flow to the sample sink.

10.3.5.2.1 Components

A summary of principal component data is given in Table 10.3-2.

a. Sample Heat Exchangers

   Five sample heat exchangers reduce the temperature of samples from the pressurizer steam space, the pressurizer liquid space, each steam generator, and the reactor coolant to 120°F before samples reach the sample vessels and sample sink. The tube side of the heat exchangers is austenitic stainless steel.

   The inlet and outlet tube sides have Conax joints for connections to the high pressure sample lines. Connections to the component cooling water lines are screwed joints. The samples flow through the tube side, and cooling water from the component cooling system circulates through the shell side.

b. Sample Pressure Cylinders

The high-pressure sample trains, the residual heat removal loop sample train, and the volume control tank gas space sample train each contain sample pressure cylinder connections which are used to obtain liquid or gas samples. The hot leg and the residual heat removal loop sample lines have a single sample pressure vessel in common. Integral isolation valves are furnished with O-ring seal type disconnects connected to nipples extending from the valves on each end. The vessels, valves, and couplings are austenitic stainless steel.

c. Sample Sinks

The sample sinks are located in the chemical fume hoods which are equipped with exhaust ventilators that discharge through particulate absolute and charcoal (PAC) filters. The work area around the sinks and the enclosures is large enough for sample collection and some analytical work. The sinks have raised edges to contain any spilled liquid.

d. Piping and Fittings

All liquid and gas sample lines are austenitic stainless steel tubing and are designed for high pressure service.

Socket welded joints are used throughout the sampling system within containment. The remaining sampling system lines have compression fittings. Lines are so located as to protect them from accidental damage during routine operation and maintenance.

e. Valves

Remotely operated valves are used to isolate all sample points from inside containment. Remotely operated containment isolation valves are provided in accordance with the requirements of Section 5.2.1.2.1. Manual stop valves are provided for component isolation and flow path control at all normally accessible Sampling System locations. Manual throttle valves are provided to adjust the sample flow rate as indicated on Figure 10.3-10.

Check valves prevent gross reverse flow of gas from the volume control tank into the sample sink.

All valves in the system are constructed of austenitic stainless steel or equivalent corrosion resistant material.

An isolation valve is provided, outside the containment on all sample lines leaving the containment, which trips closed upon actuation of the containment isolation signal.

10.3.5.3 Performance Analysis

Leakage of radioactive reactor coolant from this system within the containment is evaporated to the containment atmosphere and removed by the cooling coils of the Containment Air Cooling System. Leakage of radioactive material from the most likely places outside the containment is collected by placing the entire sampling station under a hood provided with a ventilation connection to the Unit 2 Auxiliary Building normal exhaust via the Sample Room PAC Filters. Liquid leakage from the valves in the hood is normally drained to the waste disposal system.

The Sampling System operates on an intermittent basis, and under administrative manual control.

To evaluate system safety, the failures or malfunctions are assumed concurrent with a loss-of-coolant accident, and the consequences analyzed. The results are presented in Table 10.3-3.

10.3.6 Emergency Lighting 01471873

Normal lighting in the control room is divided approximately equally, with the first group fed from 480V MCC 1T1 and the second group from 480V MCC 1T2. These MCC's can be fed from either Unit No. 1 or Unit No. 2 safeguards buses.

Emergency lighting in the control room is provided by strategically placed automatic standby units, each consisting of sealed-beam lamp heads, rechargeable battery and trickle charger. Lights operate instantly on loss of AC power and are extinguished when AC power is restored.

To further facilitate access to remote areas of the plant by the fire brigade, two battery powered portable handlights are provided in the fire brigade dress-out area.

Lighting units equipped with an 8-hour capacity battery for backup power are located in areas having equipment needed to safely shut down the plant and along access routes to this equipment. A description of the method of compliance with Section III.J of 10CFR50, Appendix R, is included in Operations Manual F5, Appendix E.

10.3.7 Spent Fuel Pool Ventilation Systems

10.3.7.1 Design Basis

The Spent Fuel Pool Special Ventilation System (SFPSVS) is designed to provide specialized ventilation of the Spent Fuel Pool area in the event that high radiation is detected. This is a safeguards system and complies to the requirements of Proposed Criteria for Nuclear Power Plant Protection Systems IEEE 279-68, as well as accepted industry standards for power plant equipment and with all applicable state and local codes and regulations. The minimum flow requirements of this system were sized to maintain a negative pressure in the Spent Fuel Pool enclosure.

The Spent Fuel Special and Containment In-Service Purge exhaust fans, see Figure 6.3-1A (Figure 6.3-1B), are used to exhaust air flow through PAC filters before exhausting out the Shield Building vent stack. This system is redundant having two complete trains, each capable of meeting the design requirements. One train exhausts through the Unit 1 Shield Building exhaust stack while the other train exhausts through the Unit 2 Shield Building Exhaust stack. This air is filtered through high efficiency and activated charcoal filters identical to the filters provided for the Shield Building ventilation (See Section 5.3.2 and Section 10.3.4.) with the exception that the temperature switch in the filter heater control circuit is bypassed.

10.3.7.2 Description

The completely enclosed Spent Fuel Pool area is normally ventilated and exhausted through roughing and HEPA filter. In the event of high radiation in the pool area, signals from radiation monitors in the normal ventilation exhaust duct isolate and shut down the normal ventilation system and initiate spent fuel pool special ventilating system. Ventilation is then accomplished via the Spent Fuel Pool Special Ventilation System which shares the exhaust portion of the Containment In-Service Purge System. The air flow is therefore directed through these redundant roughing, HEPA, and charcoal filters in this system.

Each PAC filter consists of particulate, absolute and charcoal filters. Each PAC filter is designed for 5200 cfm air flow. This system is not included in the ventilation filter testing program as the system is not credited in the radiological consequence analyses.

The Spent Fuel Pool Special Ventilation System (SFPSVS) is actuated automatically by a high radiation signal from one of the radiation monitors (R-25 for train A and R-31 for train B) located in the exhaust ducts of the system. This high radiation signal also automatically shuts down the Spent Fuel Pool Normal Ventilation System. (See Section 7.5.)

Since this system is completely redundant, either of the two trains can be tripped manually from the control room. This system is shown in Figure 6.3-1A (Figure 6.3-1B).

The SFPSVS was designed to maintain negative pressure with all SFP enclosure doors closed. Opening of these doors is acceptable for personnel use providing the doors are not blocked open.

10.3.7.3 Performance Analysis

The consequences of a fuel handling accident are discussed in Section 14.5.1. The analyses do not credit operation of the Spent Fuel Pool Special Ventilation System.

10.3.8 Emergency Communication Systems

A fixed public address system interfaced with a UPS powered Private Branch Exchange (PBX) telephone system provide normal and emergency communications. In the event of a PBX failure, extensions operating on the Xcel Energy Sherco Plant Telephone Switch could be utilized to conduct emergency communications. In addition, a sound 01469902
powered communications system is installed with communications jacks located throughout the plant. The sound powered system requires no external power, and headsets for use with the system are readily available.

Due to high ambient noise in some plant areas, and the potential for fire damage to wired systems, portable wireless communications is sometimes desirable. The site radio system utilizes hand-held portable radios, mobile radios, and stationary radio consoles to facilitate two way communications between out-plant personnel and control points such as the Control Room, Central Alarm Station, or Technical Support Center. The radio transmitters and the radio system controller are UPS powered with backup 01465889
transmitters located outside the plant in the Guardhouse and Microwave building.

10.3.9 Technical Support Center Air Conditioning System

10.3.9.1 Design Basis

The air conditioning systems supply air to the upper and lower level of the Administration Building Annex, which is designated as the Technical Support Center (TSC). The system is designed to provide the TSC with tempered air resulting in a room temperature between 65°F in the winter and 78°F in the summer. The temperature is maintained independent of the outside air temperature variation of -16°F to 92°F. Approximately 1000 cfm of outside air provides a supply of fresh air and pressurization air to the TSC.

10.3.9.2 Description

The TSC Air Conditioning System is shown in Figure 10.3-12.

During normal operation, outside air is mixed with return air through control dampers and is supplied to the TSC through two air handling units. Each unit consists of a filter, a cooling coil (with an air cooled condensing unit), a heating coil, and a fan. The air is returned to the air handling unit through a return fan.

During emergency operation, the normal outside air inlet will be closed and the return fan will be shut off. Fresh air will be supplied to the air handling units at the return air opening through a HVAC filter, a carbon filter, a downstream HEPA, and a fan. The air into the HVAC Cleanup Unit is a combination of outside air and return air such that the TSC will be under a positive pressure.

10.3.9.3 Performance Analysis

This system operates to satisfy the heating, cooling, and pressurization requirements of the TSC. A two-position maintained control switch (EMERGENCY/NORMAL) is provided on the TSC control panel to allow manual changeover from normal to emergency mode of operation or vice versa. The control switch is set to “Emergency” position when the TSC is activated. A portion of TSC return and outside air is diverted through the TSC cleanup unit and a constant air flow of approximately 3000 scfm is provided through the system. The TSC will be pressurized to about 1/8 inch W.C. with respect to atmosphere pressure. All outside air to the TSC passes through the cleanup unit before being conditioned and entering the room.

10.3.10 Compressed Air System

10.3.10.1 Design Basis

The primary design function of the Instrument and Station Air System is to provide both units with a continuous supply of oil-free, dry compressed air to the plant instruments and controls. The system also provides compressed air to hose stations throughout the plant for maintenance operations and oil-free compressed air to the condensate polishing (CP) system.

The Instrument Air system design and operation have been evaluated against the requirements of Generic Letter 88-14, “Instrument Air Supply System Problems Affecting Safety-Related Equipment”. The evaluation included the verification of air quality; a review of the adequacy of air system maintenance, emergency procedures and training; and a review of the system design and valve failure positions. The evaluation concluded that minor procedural and system changes would be required to fully address the Generic Letter recommendations. Those changes are described in NSP’s response to Generic Letter 88-14 (Reference 35).

10.3.10.2 Description

10.3.10.2.1 Instrument and Station Air System

The Instrument and Station Air System consists of two independent subsystems: Station Air (SA) and Instrument Air (IA). The two systems can be cross-connected. (See Figures 10.3-15 and 10.3-16.)

Valve and piping configuration is provided in the air supply systems to allow isolation of individual components for maintenance, duty rotation, and repairs.

10.3.10.2.1.1 Instrument Air System

Compressed air is supplied to the IA system by three 50% capacity water cooled, oil lubricated rotary screw compressors, each with its own oil mist eliminator, after-cooler, moisture separator and air receiver. The compressors are water cooled from either unit's cooling water system. 01424465

The IA compressors discharge to an IA header which is common to both units. The common IA header supplies a separate IA header for each unit. Each unit's IA header is provided with an air dryer and supplies instrument air to that unit's turbine building, reactor building, and auxiliary building.

The Unit 1 IA header normally supplies the screenhouse, water treatment system, the control room, and the auxiliary building common areas. A cross-tie is provided between the Unit 1 and 2 IA headers. The cross-tie is normally-closed via a motor-operated valve.

Primary control of each air compressor is from a local controller display module. The controller internal sequencing is designed to operate a group of multiple compressors feeding a common header. As air demands change, the compressors are controlled to deliver the required compressed air in an efficient manner by sequencing the compressors based on run hours and maintaining system pressure from 88 to 103 psig. The controller designates a Lead, 1st Backup and 2nd Backup, or a compressor can be manually set to run independent of the other compressors. 01424465

Operators have the ability to manually inhibit the start of a compressor through the Pull Out switch position from the control room if its safeguards bus is de-energized. This Pull Out position initiates a remote “Lockout” contact that removes control power to each of the compressors. In the event that normal control is lost or the control room is evacuated, remote/local transfer switches are provided at the compressor skids in order to maintain the ability to operate the compressors locally. The compressors are designed to start back up into manual operation after a loss of power. In manual operation the compressor maintains the new set point range of 93 to 103 psig.

10.3.10.2.1.2 Station Air System

Compressed air is supplied to the SA system by two 100% capacity air cooled rotary screw compressors.

The SA compressors discharge to an SA header which is common to both units. The SA system supplies compressed air to the turbine areas, including the louver valve assemblies and the Condensate Polishing (CP) systems for resin backwashing operations, the auxiliary building, the reactor building, the radwaste building, the resin disposal building, the chlorine house, the screenhouse, the hydrogen house, and the neutralizing tank house. Fittings are also provided to supply SA to containment; these are blind-flanged outside containment during normal operation.

The SA compressors have controllers which communicate to determine which air compressor is the lead and which is in standby. The lead compressor starts when the SA header pressure drops to approximately 95 psig and the standby air compressors starts at approximately 90 psig.

10.3.10.2.2 System Operation

When Instrument and Station Air Systems are separated, only two of the three IA compressors are required to supply the IA header requirements for both units, and only one of the two SA air compressors is required to supply the SA header requirements for both units. The other two compressors serve as standbys.

The SA compressors are networked together through a digital control system and are sequenced based on compressor run time. The IA compressors are also networked together and are sequenced based on run hours through a digital controller. 01424465

If required, IA may be aligned to supply SA by opening isolation valve MV-32318. The isolation valve automatically closes if IA header pressure drops below 85 psig.

If desired, SA may be aligned to supply IA by opening manual valve CP-40-7 or opening isolation valve MV-32318.

SA may be aligned to supply IA through either of two flow paths. The CP-40-7 flow path from the SA dryers to a connection on the common header upstream of the IA dryers. Air flow back through CP-40-7 to the SA system is prevented by a check valve upstream of CP-40-7. This prevents a low pressure condition on the SA system from affecting the IA system.

The second flow path through MV-32318 supplies SA to a connection on the Unit 2 IA header downstream of the IA dryers. Two pressure regulating valves (PRVs) are provided in the tie-in between the SA compressors and the SA header. The PRVs were designed to function such that SA compressors supply air to the SA header only if 1) the Condensate Polishing system air requirements are already being fully met, and 2) the three IA compressors are already being used to full capacity or are not cross connected to the SA header.

When the IA system is cross connected to the SA header through MV-32318, the PRVs operate in the following manner: PRV “B” is actuated by SA pressure and begins to close if the upstream pressure (from SA compressors) drops to approximately 80 psig. PRV “A” is actuated by SA header pressure and begins to open if the SA header pressure drops to approximately 88 psig.

If air demand exceeds the capacity of all five compressors, IA header pressure will drop below 85 psig and cause the SA header to isolate from the IA header. In this event IA compressors are dedicated to supplying instrument air, and the SA compressors are the sole source of station air. PRV “A” will remain open and PRV “B” will throttle to maintain Condensate Polishing air pressure at approximately 80 psig.

During heavy use periods, such as outages, it is possible to tie an auxiliary compressor into the system.

10.3.10.3 Performance Evaluation

The compressed air system has sufficient capacity to provide a continuous supply of instrument and station air under the most extreme plant air demands. The failure of one compressor or an air line leak are the events most likely to cause loss of instrument air. Isolation valves and standby compressors are provided to ensure isolation of cause and to maintain air supply to the undamaged portion of the system.

The IA supply to the containment is non safety-related and is isolated via solenoid-operated valves upon receipt of a signal developed by the combination of a main steam isolation signal and a containment isolation signal. In the event that the containment is isolated, air requirements inside containment are supplied by accumulators. The accumulators have sufficient capacity to enable their associated equipment to perform its function while the containment is isolated. The accumulators are kept charged by the IA system during normal operation.

10.3.11 Post Accident Sampling System

The Post Accident Sampling System (PASS) has sampling and analysis capability to promptly obtain and analyze reactor coolant samples and containment atmosphere samples. The PASS provides for a grab sample of reactor coolant for pH, dissolved hydrogen, chloride, boron and radionuclide analysis. Containment atmosphere grab samples provide measurements of containment atmosphere hydrogen and radionuclides. Provisions for ensuring samples are representative have been made. Instrumentation and analytical procedures meet the recommended ranges provided in NSPs response to NUREG-0737 Item II.B.3. Training is conducted on an annual basis.

10.3.12 D5/D6 Building and Room Ventilation Systems

10.3.12.1 Design Basis

The D5/D6 Building HVAC System, Figures 10.3-13 and 10.3-14, is designed to provide for removal of electrical equipment heat loads to maintain equipment within qualification limits and operating limits during all plant operating conditions. It also provides sufficient air flow to prevent buildup of unwanted gases.

The D5/D6 Diesel Generator Room Ventilation System is designed to limit ambient temperature within the Diesel Generator Rooms to maintain equipment ratings.

10.3.12.2 Description

The D5/D6 Building HVAC System includes the following subsystems for each diesel generator train:

a. Lube Oil Storage Tank and Fuel Oil Day Tank Rooms Exhaust System

b. Emergency Switchgear Areas Ventilation system

c. Diesel Generator Control Room and Switchgear Area Auxiliary Cooling System

d. Diesel Generator Engine Room

The HVAC System is designed to limit the D5/D6 Control Room/Excitation Room and 4KV electrical equipment areas to ambient temperatures of 104°F without supplementary cooling. The system mixes outside air with recirculated air to limit the minimum air supply temperature to 50°F. Supply air is filtered by a roughing filter.

The D5/D6 Control Room/Excitation Room and 4KV electrical equipment areas have been evaluated for ambient temperatures up to 120°F (Ref. 97). 01435305 01450457

The Control Rooms and 480V switchgear areas are provided with a direct expansion type mechanical cooling system for non-safety related auxiliary cooling. Each cooling system consists of an outdoor compressor/condenser unit, a room-mounted evaporator/fan unit, interconnecting tubing, control equipment, and power supply.

The safety-related building ventilation system includes two 100% capacity vane axial type supply fans and two 100% capacity exhaust fans per train. The fans have direct-drive totally enclosed, air over (TEAO) motors. One supply and one exhaust fan per train are normally operating, with the other fan in standby. All safety-related equipment is powered from safeguards buses.

The Lube Oil and Fuel Oil Tank Rooms are provided with exhaust fans to prevent the buildup of fumes.

The D5/D6 Diesel Generator Room Ventilation system includes the non-safety related normal mode ventilation fans, provided for human comfort; the safety-related cooling fans (one per train) which operate upon receipt of a diesel generator start signal; and the associated ductwork and controls. The diesel generator room cooling fans will function to limit the maximum ambient temperature to 120°F in conformance with equipment operating temperature ratings, and supply the volumetric flow of air necessary to cool the generator bearings. The outside air is modulated with return air to maintain a minimum temperature of 50°F in the Diesel Generator Engine Room. The diesel generator room cooling fans are vane axial type with direct-drive TEAO motors, and are supplied from safeguards buses. The ventilating equipment is accessible for periodic testing and inspection during normal operation.

10.3.13 Fuel Oil System

10.3.13.1 Design Basis

10.3.13.1.1 Function

The fuel oil system receives and stores diesel fuel oil and delivers it to the fuel oil systems of both safety and non-safety related components consisting of six diesel generators, two diesel driven cooling water pumps, one heating boiler, and a diesel driven fire pump. Since four of the diesel generators, D1/D2 and D5/D6, and the cooling water pumps are safety related components, provisions are made in the design and installation to ensure that these components will operate in the event of an abnormal or emergency condition. The fuel oil system for Unit 1 provides fuel oil for the operation of all but two safety related emergency diesel generators, D5/D6; the fuel oil system for Unit 2 provides fuel oil solely for the operational needs of emergency diesel generators D5/D6.

The fuel oil system also provides a means of transferring fuel oil between fuel oil storage tanks and a means of filtering new and transferred oil.

10.3.13.1.2 Design Minimum Storage Capacity

10.3.13.1.2.1 Unit 1

The Unit 1 design minimum required diesel fuel oil volume for a DBA is based on a 7 day diesel fuel oil volume for each emergency diesel generator (EDG), D1 and D2, and each diesel driven cooling water pump (DDCLP), 12 and 22, in accordance with Technical Specifications. The Unit 1 design minimum required diesel fuel oil volume for the maximum external flooding event, discussed in USAR Section 02, is based on a 14 day diesel fuel oil supply to operate one EDG, D1 or D2, and one DDCLP, 12 or 22, at a bounding diesel loading. (Reference 72). The 7 day and 14 day minimum fuel oil volumes are both licensing requirements; therefore, the most limiting calculated fuel oil volume will always be maintained. The onsite Unit 1 fuel oil supply is sufficient to operate one EDG and one DDCLP for longer than the time to replenish the onsite supply from outside sources.

10.3.13.1.2.2 Unit 2

The Unit 2 design minimum required diesel fuel oil volume for a DBA is based on a 7 day diesel fuel oil volume for each EDG, D5 or D6, in accordance with Technical Specifications. The Unit 2 design minimum required diesel fuel oil volume for the maximum external flooding event, discussed in USAR Section 02, is based on a 14 day diesel fuel oil supply to operate one EDG, D5 or D6, at a bounding diesel loading (Reference 73). The 7 day and 14 day minimum diesel fuel oil volumes are both licensing requirements; therefore, the most limiting calculated fuel oil volume will always be maintained. The onsite Unit 2 fuel oil supply is sufficient to operate one EDG for longer than the time to replenish the onsite supply from outside sources.

10.3.13.2 Description

10.3.13.2.1 Unit 1

Four Design Class I fuel oil storage tanks supply fuel oil to the two emergency diesel generators (EDGs) D1/D2. Each tank is equipped with a transfer pump to pump fuel from the tank to the nominal capacity 500 gallon day tank of either EDG. For the two diesel driven cooling water pumps, fuel oil supply is from two Design Class I fuel oil storage tanks. Similarly each tank is provided with a transfer pump to transfer fuel to either diesel driven cooling water pump nominal capacity 500 gallon day tank. Two non-Design Class I fuel oil storage tanks supply fuel oil to the heating boiler booster pumps and/or the D3/D4 fuel oil day tanks supply pumps. A small 4,000 gallon non-Design Class I fuel oil storage tank supplies the day tank for the diesel fire pump. Interconnecting piping, valves, and instrumentation are shown in Figure 10.3-18.

The fuel oil storage tanks are normally filled from the fuel oil transfer house through a filter. The safety related diesel generator fuel oil storage tanks are filled either from the normal fill connection or from an alternate fill connection. The diesel cooling water and diesel fire pump fuel oil storage tanks are also filled from either the normal fill connection or alternate fill connection. An alternate fill connection is provided for each of the heating boiler oil storage tanks.

Piping and valving is provided to transfer fuel oil from any one fuel oil storage tank to any other fuel oil storage tank by using the proper valve lineup. A submersible pump associated with each fuel oil storage tank pumps the fuel oil to the transfer house where the oil is then routed to the desired fuel oil storage tank. The boundary between safety and non-safety related components is provided by normally closed isolation valves. In addition, any fuel storage tank can be pumped to any diesel's day tank by performing a special valve lineup.

10.3.13.2.2 Unit 2

The Unit 2 fuel oil system consists of a non-Design Class I fuel oil receiving tank, four Design Class I fuel oil storage tanks, four fuel oil transfer pumps, two fuel oil day tanks, one fuel oil recirculating pump, four fuel oil transfer recirculation filters, a receiving tank recirculating filter, and associated piping, valving and instrumentation. The basic drawing of the Unit 2 fuel oil system is shown in Figure 10.3-19.

The fuel oil receiving tank supplies fuel oil to the fuel oil storage tanks and the fuel oil recirculation pump. The four fuel oil storage tanks supply fuel oil to the fuel oil day tanks via the fuel oil transfer pumps. The two fuel oil day tanks each have a nominal capacity of 600 gallons.

The fuel oil receiving tank is filled from a truck fill connection. The fuel oil storage tanks are filled from the fuel oil receiving tank by gravity feed. Also, there is an emergency fuel oil storage tank fill connection located in the D5/D6 building. The emergency fill connection can be utilized in the event of flooding which could render the receiving tank fill connection inaccessible. The fuel oil day tanks can only be filled from the fuel oil storage tanks.

Piping is provided to permit the transfer of fuel oil from one storage tank to another storage tank or to the receiving tank by using the appropriate valve lineup. The transfer can be accomplished by performing a tank recirculation lineup through the fuel oil transfer recirculation filters and pumping the tank contents through the recirculation header to another tank.

The fuel oil storage tanks of each diesel generator building are located in a Seismic Category I reinforced concrete fuel oil storage vault. The fuel oil storage vaults are located below ground level. The vaults provide the required three-hour rated fire protection barrier and are designed to withstand the effects of tornado generated missiles, site flood and buoyancy force considerations. The storage vaults are provided with leak detection sumps. If a sump fills with water and/or fuel oil, an alarm will activate on the main diesel generator local control panel.

The base of the fuel oil receiving tank is located below ground level and the tank is in a concrete lined retention basin. The basin is sized to contain the tank volumetric contents, should a leak occur.

The fuel oil day tank associated with one EDG can be filled directly from the fuel oil storage tanks of the other EDG by performing a valve lineup through the receiving tank return line. However, each fuel oil day tank is dedicated to its respective EDG and cannot supply fuel oil to the other EDG directly.

10.3.13.3 Performance Analysis

10.3.13.3.1 Unit 1

There are four Design Class I fuel oil storage tanks (19,500 gallons nominal each) for the Unit 1 EDGs and two Design Class I fuel oil storage tanks (19,500 nominal gallons each) for the diesel driven cooling water pumps. The six Design Class I tanks are interconnected such that any tank can be manually aligned to supply any diesel storage tank. Therefore, any combination of the tanks will meet the 14 day fuel oil supply requirements. The 7 day fuel oil supply configuration requirements will be maintained in accordance with Technical Specifications. The Unit 1 Design Class I fuel oil storage tanks can also be refilled from either of the two non-Design Class I heating boiler fuel oil storage tanks (35,000 nominal gallons each).

10.3.13.3.2 Unit 2

There are four Unit 2 Design Class I fuel oil storage tanks (32,800 nominal gallons each). The four tanks are interconnected such that any storage tank can be manually aligned to supply either diesel day tank. Therefore, any combination of the tanks will meet the storage capacity requirements for the 14 day fuel oil supply requirements. The 7 day fuel oil supply configuration requirements will be maintained in accordance with Technical Specifications.

10.3.13.4 Testing and Inspection

Fuel oil sampling is done in accordance with the Diesel Fuel Oil Testing Program for each Design Class I fuel oil storage tank and verified by analysis that the samples are within acceptable limits. In addition, the operability of the fuel oil transfer pumps and fuel flow paths from associated storage tanks to associated days tanks are verified in accordance with the Technical Specifications.

The interconnecting capability for the 14 day fuel oil supply requirement will be tested every refueling cycle for the associated Unit. The testing will verify flow path integrity for all the various storage tank to day tank configurations.

The Diesel Fuel Oil Testing Program is credited as part of the License Renewal Fuel Oil Chemistry Aging Management Program as described in Appendix L.

10.3.13.5 Instrumentation and Control

10.3.13.5.1 Unit 1

The fuel oil storage tanks are provided with level instrumentation to indicate tank liquid level at the control room (except for the heating boiler fuel oil storage tank) and filter house. High level will actuate an alarm in the control room except for the heating boiler fuel oil storage tank.

Level switches are provided in the fuel oil day tanks for automatic makeup.

10.3.13.5.2 Unit 2

The fuel oil storage tanks are provided with level instrumentation, mounted on the diesel generator benchboard control panel and monitored by a computer. Tank high and low levels annunciate in the diesel generator control room. A separate leak detection monitoring instrument system is provided for each pair of tanks serving a single diesel generator set.

Level instrumentation is provided with each fuel oil day tank for automatic makeup.

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10.4 PLANT COOLING SYSTEM

10.4.1 Cooling Water System

10.4.1.1 Design Basis

The Cooling Water System has been designed to provide redundant cooling water supplies with isolation valves to auxiliary feedwater pumps, Unit 1 diesel generators, air compressors, component cooling water heat exchangers, containment fan-coil units, and the Auxiliary Building unit coolers.

The Mississippi River is the source of cooling water. The design maximum temperature for the Cooling Water System is 85°F. This value was chosen based on historical data gathered at several sites on the Mississippi River. The value is that temperature that is not expected to be exceeded more than 1% of the time. Safeguards components are operable with cooling water inlet temperature up to 95°F. Thus, it is acceptable to continue plant operation with cooling water inlet temperature up to 95°F, when external conditions (e.g., several hot, humid days) cause cooling water inlet temperature to exceed 85°F.

The system is sized to assure adequate heat removal based on highest expected cooling-water temperatures, maximum loading, and leakage allowances. The system is monitored and operated from the Control Room. Isolation valves are incorporated in all cooling water lines penetrating the containment.

Electrical power requirements for the Cooling Water System are supplied by the power sources described in Section 8.

The preferred source of water to the containment fan coil units and some Auxiliary Building unit coolers is provided by chilled water. The Cooling Water System provides an alternate source of water to these components.

10.4.1.2 Description

The Cooling Water System flow diagrams are shown on Figures 10.4-1a through 10.4-1d and 10.4-2a through 10.4-2c. The diesel cooling water pumps, vertical motor driven pump and their associated equipment are located in the Class I portion of the cooling water screenhouse and separated by a concrete wall. A ring header which is shared by Units 1 and 2 can be isolated automatically to provide two redundant independent sources of cooling water for all essential services. One-half of essential services for each Unit is supplied from each side of the isolable loop. Each side of the loop is designed to supply the needs for all essential services for both Units. Thus, failure of one side of the loop still provides for the operation of all equipment required for the safe shutdown of both Units.

Table 10.4-1 lists the nominal design flow rates of the components serviced by the cooling water system for Mode 1, Power Operation, Mode 2, Startup, Mode 3, Hot Standby, normal cooldown and slow cooldown.

Normal operation utilizes two horizontal pumps with the vertical motor-driven pump as a standby. Two vertical diesel driven pumps are provided for emergency operation. The diesel driven pumps are used whenever an engineered safety features sequence is initiated, when discharge header pressure drops below its setpoint, or on a loss of offsite power (due to loss of motor driven pumps and resultant low discharge header pressure).

The vertical motor driven pump also functions as a replacement when a diesel driven pump is taken out of service. The pump is aligned manually to the appropriate train of Unit 2 safeguards power, depending upon which diesel pump is out of service. The motor operated valves are placed in the desired position and administratively disabled.

The capacity of each motor-driven pump and each diesel driven pump is nominally 17,500 gpm. This exceeds the maximum flow required for a single pump serving one unit in hot shutdown and a second unit in the long-term post-accident condition, which is calculated to be less than 14,500 gpm (Reference 41). 01457655

As stated above, two diesel driven pumps are provided common to both Units 1 and 2. Train “A” Safety Injection signal from either unit will start 12 diesel driven pump. Train “B” Safety Injection signal from either unit will start 22 diesel driven pump. The vertical motor driven pump will also start upon any of these signals, but will trip if both diesel driven pumps operate. If a diesel driven pump fails to start, the vertical motor driven pump will continue to operate. The diesel driven pumps also start automatically on a drop in their respective cooling water discharge header pressure below 75 psig for longer than 15 seconds: low pressure in cooling water discharge header A(B) starts 12(22) diesel driven pump. Each diesel driven pump is provided with both local and control room manual controls.

Electrical control for the diesel driven cooling water pumps is installed in accordance with the separation criteria established for redundant Class IE Circuits as outlined in Section 8.7. The safeguards power supply cable to the vertical motor driven pump (from Bus 27 to the motor) is run separate from both the Unit 2 A and B safeguards trains.

Each diesel engine was factory tested which consisted of the regular run-in plus full load testing on a dynamometer for 50 hours. During the factory testing, engine operating data was recorded at periodic intervals.

A separate test of each diesel engine and its accessories and auxiliaries such as the jacket water heat exchanger, control panel and air starting package was conducted. This test consisted of a minimum of 15 cold starts, coming up to a minimum of 50% load and running for a minimum of 10 minutes after each start. Tests were also run to check the effectiveness of all protective and monitoring devices such as engine overspeed, low lube oil pressure, high jacket water temperature, etc. These tests confirmed each engine's ability to start, attain speed and carry load in 15 seconds or less.

Each diesel engine provides 1000 hp continuous at 1200 rpm and 1200 hp intermittent at 1200 rpm. The engines have the capability of attaining the rated speed and pump design horsepower within 15 seconds or less.

Fuel oil is supplied to the diesel engine from a 500 gallon day tank (one per diesel engine) by an engine driven gear-type fuel oil transfer pump which circulates the fuel oil through a filter to the engine cylinders with excess fuel oil returning to the day tank. The day tank fuel oil is normally supplied from external underground storage tanks (one 19,500 gal. tank per diesel engine) by a motor operated transfer pump. Refer to Figure 10.3-18.

A manual fuel priming pump is provided to fill the engine fuel system after maintenance has been performed on the piping or for initial filling of the system.

Compressed air is used for starting the diesel engine and is stored at approximately 250 psig in two air accumulators. An A.C. motor driven air compressor unit is used to maintain air pressure in the accumulators. A manually operated interconnecting tie is provided between the air compressors of diesel engines No. 12 and No. 22. Two solenoid valves, connected in parallel, admit air to two starter motors for cranking.

During shutdown of the diesel engine, the jacket water is warmed by two electric immersion heaters which maintain the water at 110°F minimum. The heated jacket water flows by natural circulation throughout the engine flow passages. As the heated jacket water flows through the lube oil cooler, the circulating lube oil is warmed by the jacket water.

Control power for starting and stopping the diesel engines is supplied from separate trains of safeguards 125V DC systems. 12[22] DDCLP is fed from 125V DC panel 17[18]. Panel 17[18] is fed from a manual transfer switch which can be aligned to DC panel 11[12] or 21[22].

The cooling water pumps normally are supplied with water from behind the circulating water traveling screens. An alternate source of water (Figures 10.4-3a and 10.4-3b) is provided for the vertical pumps. This alternate source of water is drawn through a Class I intake canal which communicates with the river so that if the normal intake canal became obstructed, a source of water will always be available for the pumps.

All valves in the piping that provides cooling to components required for safe shutdown including engineering safety systems are either open (manual valves), receive an "open valve" signal directly or indirectly from the Safety Injection signal (motor-operated and control valves) or are fail open valves (control valves). Valves in the header which are required to split the system are redundant and receive a "close valve" signal. These valves may also be positioned by the operator from the Control Room after resetting the safety features signal.

There are two cooling water strainers in each cooling water header. Their purpose is to remove particulate from the cooling water before it enters the cooling water header. The backwash valves on each strainer provide for the clearing of debris from the strainers to prevent loss of flow due to debris build up. The motive force for these valves is provided by the instrument air system. Air pressure maintains the valves closed and upon a loss of air pressure, the valves fail open, which then diverts water from the cooling water headers. To prevent this diversion of water from the header, two safety related backup compressed air systems were installed for these control valves on Modification 00CL02. Each compressed air supply serves one strainer backwash valve on each cooling water header. This configuration is similar to the configuration of the power supply for these control valves. Plant procedures ensure minimum air cylinder pressure of 500 psig, which will provide 99 actuations of the strainer backwash control valves, or 4 actuations per hour for a 24 hour period (see Calc. PI-M-014). There are four spare air cylinders located in close proximity to the installed air cylinders for quick cylinder replacement. The type and size of air cylinder used is the same as other air cylinders in the plant so that additional air will be readily available if needed.

The temperature control valves on the outlet of the component cooling heat exchangers are air operated valves that are positioned by a control signal from a temperature controller. Normally, a temperature setpoint is set on the temperature controller, and the control valve is opened or closed based on a deviation from the setpoint. If air pressure or control power is lost, the valve fails to the open position. Since the air system is not safety-related, it may not be available during a design basis accident. The temperature control valve would fail to the open position. Analysis (Reference 41) has shown that this passes a disproportionate amount of cooling water flow through the component cooling heat exchangers. Additional analysis (Reference 42) was done to determine the required cooling water flow rate to support the assumed component cooling water heat exchanger heat transfer rate. From this information, a maximum valve position was determined. The valve actuator is furnished with threaded openings for use of either a minimum or maximum flow stop. The valve stop is mechanical, typically a threaded bolt with some combination of bolts and/or pins. The maximum flow stop was positioned through a Design Change (Reference 43). Analysis was performed of the capability to provide long term cooling following a LOCA with the valve stop installed. This analysis (Reference 77) concluded that under bounding conditions, adequate cooling could be provided as long as the valve stop was removed (or backed out) within the first 24 hours following the accident.

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The ring supply header of the cooling water system shown in Figure 10.4-1a is normally fed by the two horizontal electrical pumps to supply normal cooling loads. A safety injection signal from one unit automatically accomplishes all of the following:

a. Starts both diesel driven pumps, causing them to supply the ring header directly.

b. Starts the vertical motor driven pump, which will trip if both diesel pumps reach operating speed.

c. Isolates the portion of the ring header between the diesel pumps by closing two normally open motor-operated valves and ensures isolation of the far side of the ring by providing closure signal to two motor-operated valves, such that the ring header is split into two independent supply paths, each supplied by a diesel-driven pump and each serving half the essential equipment in each reactor.

d. Causes the three motor-driven valves associated with each fan coil unit and the orifice bypass valves in the fan coil unit return line of each train on the affected unit to go full open. The motor valves are normally open but one can be partly closed for throttling. Other essential equipment in the affected unit is open to the cooling water supply, or is indirectly opened by the safety injection signal.

The return side of the cooling water system disposes of the water to the circulating water conduits. Each of the two discharge headers of the turbine building is provided with a 30-inch standpipe. The emergency dump connection is located physically above the normal circulating water discharge on the standpipe. In the event the normal discharge line is blocked, the cooling water level in the standpipe rises until it reaches the emergency dump connection. From there, the cooling water flows through the 24-inch dump line, outside the building, and is dumped to grade. This dump line is a passive device in that it requires no manual or automatic actuation.

The auxiliary building header is provided with an emergency dump valve to be used in the event that both dump lines in the turbine building fail. This valve is motor-operated and controlled, either locally or from the Control Room.

By proper sloping of the grade outside the turbine building, natural drainage is provided back to the intake and recirculating canals. Concrete splash blocks are provided at each Turbine Building dump line discharge.

A maximum of about 15,000 gpm is discharged from each standpipe or a total of 30,000 gpm. When the auxiliary building emergency dump valve is used, a maximum of 15,000 gpm is discharged.

The fan coil units are supplied by lines which are split outside the containment so that each unit can be individually isolated, controlled, and monitored. With all systems operating normally, the water pressure always exceeds containment pressure. A small by-pass line is provided to pressurize the space between the two fan coil outlet valves so that there is no leakage of the containment atmosphere to the environment. Outlets from the fan coil units are individually monitored for radiation to permit isolation of failed units.

Component Cooling water heat exchangers are temperature controlled. The control valves are air operated valves, which fail to the open position upon loss of air pressure or control power.

Cooling water is supplied to D-1 Diesel Generator from Loop A Cooling Water header, and D-2 Diesel Generator is supplied from Loop B. The Unit 2 diesel generators have dedicated radiator cooling systems.

Cooling water is available to the auxiliary feedwater pumps and provides the secondary source of auxiliary feedwater.

A minimum flow of cooling water is furnished to the turbine generators to prevent wiping of bearings and consequent damage during coast-down of the turbines. This source of water may be manually isolated.

The cooling water system also provides water for safeguards-related equipment such as pump oil coolers, pump gland seals, unit coolers, and the control room air-conditioning chillers.

Because the cooling water pumps are diesel driven, water is available for all services at the time required by the equipment served. If auxiliary power is available, the normal cooling water pumps continue to operate. Only one cooling water pump is required for the safe shutdown of both units (accident in only one unit). Since the cooling water system is in operation at all times it is in a high state of readiness and available for normal or emergency types of operations.

In the event of a design basis accident, adequate flow to essential equipment is maintained. Non-essential equipment in the Turbine Building is isolated by automatic action, when necessary. Actuation circuitry will automatically close the 24″ motor operated valve that supplies the Turbine Building. Low cooling water supply header pressure coincident with an SI signal for a preset time delay will actuate the isolation logic.

In the event of a loss of offsite power, the loss of the motor driven pumps results in lowering discharge header pressure. When the low pressure setpoint is reached, the associated diesel driven pump starts and will provide adequate cooling to the associated Unit 1 diesel generator and other cooling loads. Further, the 121 vertical motor-driven cooling water pump, upgraded to a safeguards status, having an essential safeguards diesel-backed power supply provides a diverse means of providing cooling water independent of diesel-driven pumps. One diesel driven pump is sufficient to meet all the cooling system loads required for the safe shutdown of both units (accident in only one unit).

10.4.1.2.1 Auxiliary Building and Containment Chilled Water System

A common non-safety related chilled water system was added for both units 1 and 2. The system provides chilled water during normal plant operation for the containment air cooling fan coil units and the control rod drive mechanism shroud cooling coils, as well as for the new and existing unit coolers located within the auxiliary building. The chilled water system was integrated with the cooling water system. The combined system provides chilled water during normal plant operation but upon loss of power or safety signal actuation, the system will return to its original design integrity and configuration by isolating the chilled water system from the cooling water equipment. The chilled water system was refurbished and upgraded by modification 01ZX01. Changes to the system include corrosion resistant coating in the piping and a chemical feed tank.

10.4.1.2.2 Emergency Cooling Water Intake

The Emergency Cooling Water Intake provides water to maintain safe shutdown for both units after a Design Basis Earthquake. This intake is a 36-in. pipe buried approximately 40 feet below the Circulating Water Intake Canal water level in nonliquefiable soil, connecting the screenwell to a submerged intake crib in a branch channel of the Mississippi River. This Emergency Cooling Water Intake is a Class I structure as is the Approach canal which supplies its intake crib from the main channel of the Mississippi. The intake crib is designed to exclude trash, and means are provided for back flushing. This emergency intake supplies a bay in the screenwell which has Class I traveling screens and which provides suction for the Class I cooling water pumps.

To maintain both units in safe shutdown, the safeguards cooling water pumps must provide sufficient flow to remove heat from the required components. The supply to the safeguards pumps' suction must be greater than or equal to that demand. In the event of a design basis seismic event, non-seismic qualified components cannot be relied upon to maintain safe shutdown. The effect on the cooling water system is that only the safeguards pumps are available, off-site power is lost, and the instrument air system is not available. Since many control valves fail to the open position upon loss of air or control power, the flow demand in the cooling water system increases. With two safeguards cooling water pumps operating, the cooling water system demand will exceed the supply capacity of the emergency intake line (EIL).

At the onset of the seismic event, the emergency intake line and the intake canal both supply the suction of the safeguards pumps. The design basis seismic event assumes that Lock & Dam No. 3 is destroyed by the seismic event. Failure of Lock & Dam No. 3 causes the upstream and downstream pools to equalize. Over time the upstream pool level is postulated to decrease to 666′ -6″, the normal level of the downstream pool. This provides 4.5 feet of submergence above the emergency intake line. Due to discrepancies between original design calculations and pre-operational testing results regarding intake line capacity vs. head loss, a test was performed on the emergency intake line (Reference 91). The results from this test indicate that there has been some degradation since original construction. Original design calculations predicted a minimum supply capacity of 18000 gpm. However, the new test results, when extrapolated for minimum submergence, demonstrated that only 12,850 gpm is actually available. (Reference 96) 01457655

Upon occurrence of the design basis seismic event, the cooling water system flow demand is approximately 40,000 gpm (Reference 41). This is with the two diesel driven safeguards cooling water pumps operating, since this creates the highest demand on the suction supply. There is an additional 2000 gpm demand from the diesel fire pump. Initially, the supply to the safeguards cooling water pumps is from both the intake canal and the emergency intake line. The stability of the intake canal banks has been evaluated (References 54 & 55). The evaluations demonstrate that the intake canal will support the safeguards function of the cooling water system. The volume in the intake canal provides approximately 3.3 hours for a flow demand of 47,000 gpm (Reference 55). 01457655 01457655

Assuming no makeup from the river to the intake canal, the volume in the intake canal is depleted in approximately 3.3 hours. After this time, the emergency intake line will be the sole supply of water to the cooling water pumps. It is necessary for the operators to reduce the cooling water system flow demand to a value within the capacity of the emergency intake line. Procedural guidance directs the operator which cooling water system loads to secure to reduce demand. Instrumentation provides the operator with cooling water header flow and pressure. The procedure ensures components needed to maintain safe shutdown are available. 01457655

An evaluation was performed (Reference 56) comparing the minimum water volume of the Intake Canal required for operator action to the minimum water volume of the Intake Canal available post-DBE. The minimum water volume available in the Intake Canal after a DBE, for both the design basis and bounding analysis cases, is nearly twice the minimum volume required for operator action. This demonstrates a significant operating margin. 01457655

The capacity of the EIL must support the minimum equipment required for safe shutdown. As stated above, it is assumed the equipment that is not qualified to seismic criteria does not function. Therefore, off-site power is lost and the instrument air system is not available. The following is the minimum equipment for safe shutdown and the design flow rate.

1 - Unit 1 Emergency Diesel Generator	900 gpm
2 - Auxiliary Feedwater Pumps (1 per unit)	440 gpm
2 - Component Cooling Heat Exchangers (1 per unit)	3600 gpm
1 - Control Room Chiller	320 gpm
2 - Containment Fan Coil Units (1 per unit)	900 gpm
Total	6160 gpm

Taken by itself, this would be the minimum required flow capacity of the EIL. However, cooling water system loads that are not isolated from the control room must also be considered as cooling water system demand. Through operator action, this flow rate can be redistributed within the cooling water system based on the specific event.

The basic design intent for the emergency pipe was to provide enough flexibility in the system to withstand earthquakes. This was accomplished by introducing four flexonics expansion joints, two near the screenhouse and the other two in the pipe riser at the intake crib. The articulation provided by the joints is expected to act in a fashion similar to paired flexible joints in steam lines.

In order for the emergency intake pipe to behave elastically, as intended, the portion of the pipe embedded in the screenhouse was wrapped with rodoform to alleviate localized stresses due to the settlement of the soil. Special backfill material was placed around the pipe to prevent liquefaction of the soil which would result in flotation of the pipe. All natural material has been replaced by nonliquifiable backfill materials up to the liquefaction level in accordance with the recommendations of Dames & Moore in Appendix E.

The design of the 36″ emergency intake pipe and the approach canal are based upon recommendations by earthquake consultants J. A. Blume & Associates and Dames & Moore. Professor H. Bolton Seed of the University of California at Berkeley, in his letter dated June 3, 1970 to Mr. Garrison Kost of John A. Blume & Associates in San Francisco, stipulates the following minimum criteria to ensure that the emergency service water intake pipe at Prairie Island would not be disrupted by displacements due to soil liquefaction:

a. “The slope of any liquefiable material should not exceed about 1 degree.

b. The pipe line should be supported or protected against settlement or uplift due to liquefaction of the underlying soils.

c. The pipe should be located at least 25 times the height of any bank beyond the toe of the bank in order to protect it from lateral forces due to movement of liquefied soil.”

Professor Seed then proceeds to make the following specific recommendations: It will be possible to design:

a. “A section near the plant where the pipe would be placed in non-liquefiable soils.

b. A section to be stabilized against liquefaction by densification and in which the pipe would be brought up to a higher elevation, and

c. A section designed in accordance with the criteria listed above so that the pipe would not be disrupted even if the underlying and adjacent soil should liquefy.”

The design of the 36″ emergency intake pipe and the approach canal applies to the following criteria:

Near the screenhouse, where the pipe line is above the liquefiable horizon, we have removed all liquefiable material around the pipe and replaced it by non-liquefiable material.

The east-west run of the emergency intake pipe has been placed below the horizon of the liquefiable soil. Trench backfill materials are non-liquefiable up to the horizon of liquefaction.

At the intake, where the pipe line rises vertically through potentially liquefiable strata, we have provided secure anchorage of the intake crib by piling into the non-liquefiable strata. In order to protect the riser pipe itself, we have designed a considerable degree of flexibility into the riser by installing two Flexonics joints which are capable of swiveling 6° in any direction.

In accordance with the explanation and criteria set forth by Dr. Seed, lateral movements of liquefied soil layers are not expected in the intake area, nor do we expect a covering of the intake itself, because the intake crib is located in a 575 ft. wide intake canal which has been sized by applying the 25 to 1 slough angle cited by Dr. Seed. The bottom of the canal has been kept flat.

The bed of the branch channel of the Mississippi River in which the emergency intake crib is located has been backfilled to Elevation 660 in order to minimize any potential gradients which might cause a flow of liquefied materials. The slough angle of 25 to 1 has again been observed at the underwater bank which rises from Elevation 660 to Elevation 664.5.

The non-compacted, non-liquefiable backfill has been designed and specified according to recommendations by John Blume Associates, according to which the sieve analysis is:

70% passing 0.742 in. screen opening
36% - 50% passing #4 screen,
10% passing #10 screen.

The material used for the non-liquefiable backfill closely approximates that recommendation.

The profile of the Approach Canal to the emergency intake pipe is monitored periodically per surveillance testing to ensure the design function is maintained.

10.4.1.2.3 Waterhammer in FCU Piping (GL 96-06)

NRC Generic Letter 96-06 (Reference 79) identified a concern regarding the potential for waterhammer to occur in the service water (cooling water at Prairie Island) piping associated with the containment fan coil units. The specific concern in Generic Letter 96-06 related to the potential for excessive piping and piping support loads due to the formation of steam in the fan coil units and associated piping during the initial time period following an accident (LOCA, MSLB) with subsequent condensation induced water hammer during refill.

EPRI developed a methodology to evaluate the waterhammer issues identified in the generic letter (EPRI technical reports TR-113594, Volumes 1 and 2; which were subsequently renumbered as EPRI Technical Reports 1006456 and 1003098, respectively). NRC acceptance of TR-113594 along with limitations for use of the methodology are documented in the NRC Safety Evaluation Report (Reference 80).

The methods in TR-113594, Volumes 1 and 2, were applied to determine the potential waterhammer loads in the CL Piping associated with the containment fan coil units (FCUs) (References 81 and 82). This review/evaluation was performed as follows:

- Limiting system configurations were determined, paying particular attention to system alignments, single failures and component operation that could maximize the severity of the postulated waterhammer. Each of the components that could impact the severity of the event was identified with their corresponding operating times (after event initiation) initially assuming the component was operating normally. Through this review, it was confirmed that the limiting waterhammer event would be a column closure waterhammer.

- Following identification of the limiting system alignments, the limiting location of the waterhammer event was identified. It was determined that the potential existed for a column closure event in several of the FCU supply lines. A waterhammer event in the supply line would be more limiting than in a return line because a higher refill velocity would exist in the supply line due to the lower total flow resistance from the supply header to the point of void collapse.

- The closure velocity at the predicted void closure location was determined for each of the limiting configurations based on the hydraulic characteristics of the system and the pump operating curve. To maximize the refill capacity, the maximum safeguards cooling water pump curve was used (maximum is defined by upper capacity limit from the In-Service Testing pump curves).

- Based on the calculated closure velocity, the magnitude, the rise time and time duration of the waterhammer pressure pulse was determined. Due to different system configurations (based on time line, operating configurations and postulated single active failures), several different cases were considered.

- The results from the different cases were then evaluated for the effect on the piping and pipe supports. The cases were analyzed using piping model to determine the system response. In the model, the pressure pulse was characterized as a trapezoid. Piping analyses showed that the piping and supports satisfied the acceptance criteria in USAR, Section 12. 01450457

In Reference 83, the NRC documented that the staff was satisfied with the response to GL 96-06 and considers the waterhammer element of GL 96-06 closed.

10.4.1.2.4 System Functionality Following a Seismic Event

The Cooling Water (CL) System is shared between the two units and provides cooling to both safety related and non-safety related loads, as shown on Figures 10.4-1A through 10.4-2C. The single largest non-safety related supply line (one per unit), which supplies the majority of the non safety loads in the Turbine Building, can be isolated by closing a safety related motor operated valve. The isolation valve in each unit closes automatically on a safety injection signal coincident with a low-pressure condition in the safety related supply header (for post-accident response) or can be remotely closed from the Control Room. The other valves that could be used to isolate the non-safety related lines from the safety related supply headers are manual valves. During a seismic event, without a SI signal, all of the valves (both the motor operated Turbine Building isolation valves and the manual valves) would remain open unless closed by operator action.

The safety related portions of the CL system are designed to withstand a seismic event (Class I piping). Consistent with the plant design requirements (Section 12.2.1.5.2.1), portions of the non-Class I piping may be analyzed with seismic loads to support the Class I piping analysis. The other areas of the non-Class I piping are not typically analyzed for seismic loads. Therefore, following a seismic event, there will be several runs of non-Class I non-analyzed piping that remain connected to the Class I piping.

To account for the non-Class I piping that has not been analyzed for seismic loads, continued functionality of the Cooling Water system following a seismic event is demonstrated using hydraulic analysis techniques. The hydraulic analysis is performed assuming a complete failure (rupture) of a non-seismic pipe at the worst-case location. The intent of assuming a complete failure of one pipe is to bound possible cases where partial failures (cracks) may occur in multiple lines. The size of the non-seismic piping break can be reduced using stress analyses techniques of the non-Class I cooling water piping for seismic loads as discussed in Section 12.2.1.5.2.4.

The worst case may not necessarily be the largest pipe if the pressure at smaller pipe locations is significantly higher. The analysis includes evaluation of other potential break locations to ensure the worst-case location is identified. In response to a seismic event, the function of the cooling water system is to support achieving and maintaining safe shutdown. Thus, an accident is not assumed concurrent with the seismic event.

The results from the hydraulic analysis (Reference 41) show: 01457655

- Sufficient flow is provided to the Unit 1 Emergency Diesel Generators.
- Sufficient flow is provided to Diesel Driven Cooling Water Pump Auxiliaries (e.g., Jacket Water Heat Exchanger).
- Flow demand on the CL Pumps is within the pump capacity.
- Sufficient flow and pressure is available for suction to the AFW Pumps. 01457655

10.4.1.3 Performance Analysis

The cooling water system is designed to prevent a component failure from curtailing normal station operation. The system has been designed and equipment furnished to provide the highest degree of reliability and availability. The combination of redundancy, isolation and system monitoring assures that, in the event of a major failure, safe shutdown of the station is not compromised. Any equipment can be isolated without shutting down the system. Sufficient pumping capacity has been provided for two-unit operation.

The cooling water system can accommodate a passive failure such as rupture of a pipe and still perform its intended function during long-term cooling following a loss-of-coolant accident. Cooling water supply for both units branch out of the main supply header. On a safety injection signal, this single header is automatically split into two headers by closure of redundant isolation valves. Each of these two supply headers then supplies both units, and either is capable of meeting the cooling water requirements for long-term cooling of both units with an accident having occurred in one. Thus, in spite of a passive failure in one loop, the other loop is available to supply sufficient cooling water for long-term cooling.

The following components of the Cooling Water system have safeguards functions:

Containment Fan Coil Units

Component Cooling Heat Exchangers

Safeguard Traveling Water Screens

Auxiliary Feedwater Pumps (alternate suction)

Unit 1 Emergency Diesel Generator Cooling (D1/D2)

Control Room Chillers

Safeguards Cooling Water Pumps

Safeguards Isolation Valves and Dump Valves

The preoperational tests for the cooling water system to assure its operation as required to function, following a loss-of-coolant accident, included the following.

The cooling water system was operated in its various safeguards modes with the supply valves to all components opened. Total system flows and flows to the containment fan coil units and the component cooling heat exchangers were measured, using existing flow meters, and compared to the required maximum flows. System pressures were measured across the remaining safeguards components to assure that flow is proportionally distributed under maximum flow conditions. The temperature controls which modulate the control valves for the coolers were checked when heat loads were available. System tests were performed with supply water being taken through the emergency intake line.

Safeguards modes of operation checked included:

Two pumps operating supplying water to both safeguard trains.

One pump operating supplying water to one safeguard train.

Both above modes of operations using normal return flow path and alternate safeguards return path.

The auxiliary feedwater pumps were operated using cooling water and discharging through the recirculation line. There was sufficient recirculation capacity to obtain the full required flow.

10.4.1.4 Testing and Inspection

The safeguard motor driven and the two diesel driven cooling water pumps were preoperationally tested.

The pre op test functionally tested the system and its components.

For the diesel driven pumps the first portion of the test proved that each control, protective shutdown or alarm device functions properly and properly actuates its end device. Such functions as governor speed changer function, low oil pressure shutdown and/or alarm and engine coolant temperature high alarm were proven for the diesel driven pumps.

The second portion tested the ability of the diesel engines to start, accelerate to operating speed and pump its rated flow of cooling water into the plant system. This part of the test included gathering a complete set of baseline data on the diesel engine and pump after both the engine and pump reached steady state operation.

A third part of the test provided the data proving the diesel engine will:

a. Start reliably.

b. Start and reach operating conditions in the required time.

This portion of the pre-op test provided starting data from 150 starts on each diesel engine. A copy of the test and test data sheets is available to Directorate of Reactor Licensing (DRL) review.

Following a turbine trip from 100% power, the cutback in cooling water flow was measured and the cooling water system capability to supply safeguards flow determined.

10.4.2 Component Cooling System

10.4.2.1 Design Basis

The component cooling system is shown in Figures 10.4-4A through 10.4-4B and 10.4-5A through 10.4-5B. The component cooling system is designed to remove heat from major components in the Nuclear Steam Supply System under normal conditions and from all components associated with removal of reactor core decay heat under accident conditions.

The component cooling system consists of two pumps and two heat exchangers for 100% redundancy within each unit. A single active or passive failure does not impair the function of the system for post-accident operation. The Unit 1 and Unit 2 systems communicate via an expansion line connecting the surge tanks of each component cooling system.

The system design provides for detection of radioactivity entering the system from reactor coolant sources.

It is the intent of the design of the safety related actions of the Component Cooling System to meet the requirements of IEEE 279 “Standard, Nuclear Power Plant Protection System”, August 1968 [Ref. 60].

10.4.2.2 Description

The component cooling system consists of heat exchangers, pumps, surge tank, piping and the necessary valves and instrumentation. During operation component cooling water is circulated through the shell side of the component cooling water heat exchanger and then to the various system components. The temperature of the component cooling water at the outlet of the heat exchanger is normally maintained between 80°F and 105°F. A maximum temperature of 120°F is permissible for a 2 hour period. Cooling water flows through the tubeside of the component cooling water heat exchanger. The returns from the components are pumped through the component cooling water heat exchanger and circulated back to the system components. The surge tank is provided to accommodate the expansion and contraction of the component cooling water, to assure a continuous supply of component cooling water, and adequate NPSH for CC pump.

Component cooling is provided for the following heat sources:

a. Residual Heat Removal Heat Exchangers

b. Residual Heat Removal Pumps

c. Safety Injection Pumps

d. Containment Spray Pumps

e. Reactor Coolant Pumps

f. Spent Fuel Pool Heat Exchangers

g. Letdown Heat Exchanger

h. Excess Letdown Heat Exchanger

i. Seal Water Heat Exchanger

j. Boric Acid Evaporator Package

k. Primary Sample Coolers

l. Steam Generator Blowdown Sample Panel

m. Hot Chemistry Lab Chiller

n. Waste Gas Compressors

o. Waste Evaporator Package

p. Waste Gas Recombiners

Nominal design flow rates for these heat loads for various modes of operation are listed in Tables 10.4-2a and 10.4-2b.

One pump and one component cooling heat exchanger fulfill the heat removal function during normal full load operation for various components located in the auxiliary and containment buildings. During plant cooldown, two pumps and two heat exchangers are utilized to remove the residual heat. If one of the heat exchangers is not operative, only one RHR loop is effective and cooldown then is at a slower rate. Makeup water is taken from the reactor makeup tanks or demineralized water system and delivered to the component cooling surge tank.

A portion of the Unit 1 Component Cooling piping is located outside of a Class 1 structure and is susceptible to tornado generated missiles. The piping at risk includes part of the supply piping and part of the return piping for the 122 Spent Fuel Pool Heat Exchanger. To prevent Component Cooling system inventory loss during a postulated tornado event, two air operated valves (AOVs) on the supply to the heat exchanger will automatically close either on a loss of Component Cooling discharge pressure from the 122 SFP HX or on a low level in the 11 Component Cooling Surge Tank, and two check valves on the return from the heat exchanger will close on low pressure in the upstream piping. 01422041

As indicated in Figures 10.4-4A and 10.4-4B (Figures 10.4-5A and 10.4-5B), the two component cooling loops associated with one unit are interconnected downstream from the heat exchangers so as to effectively form an open loop supply header both for loads which are essential and those that are nonessential. Some of the nonessential loads are isolated by the Safety Injection signal, (e.g., Evaporators, excess letdown etc.).

A safety injection signal causes automatic closure of two valves in this loop, which splits it in three parts, effectively isolating the middle section. The isolated loads normally served by this middle section are those that are nonessential during the post-accident period. The unisolated loads served by each segment immediately downstream from each heat exchanger are the essential loads, consisting primarily of the safety injection pumps, the residual heat removal pumps and heat exchangers, containment spray pumps, reactor coolant pumps, spent fuel pit heat exchangers, and sample coolers. These loads are arranged redundantly so that each heat exchanger supplies a complete set of the engineered safety features cooled by component cooling water.

The essential loads other than the residual heat exchangers are normally valved open to the supply header, and they discharge to the suction of the component cooling pump with which they are normally associated, so that component cooling water is circulated continuously through the essential loads during normal operation. The safety injection signal causes both component cooling pumps to start, although at least one would normally be running. In addition the component cooling pumps are capable of being automatically started by a low discharge pressure signal. If a running component cooling pump fails, the low pressure switch associated with the other pump, if it is in standby, senses the loss of pressure and automatically starts the standby pump.

Each of the component cooling inlet lines to the residual heat exchangers has a normally closed remotely operated valve which is opened automatically by startup of the RHR pump upon a safety injection signal. The Unit 1 CC inlet lines to the RHR HXs will also open when the AOVs on the supply to the 122 Spent Fuel Pool Heat Exchanger auto close and two Unit 1 CC Pumps are running. If the accident sequence continues into the recirculation mode, the RHR pumps discharge to the RHR heat exchangers where the residual heat is delivered to the component cooling water flow. 01422041

To prevent one component cooling pump from overpowering the other pump during two pump operation, flow needs to be aligned to one of the spent fuel pool heat exchangers or to one RHR heat exchanger. If the Unit 1 component cooling flow to the 122 Spent Fuel Pool Heat Exchanger is automatically isolated by one or both of the supply AOVs during two pump operation, the corresponding train RHR inlet MOV will open. 01422041

In anticipation of a potentially large cooling demand during Post-Accident Conditions, the control room operator can isolate the component cooling inlet to the spent fuel pool heat exchangers or shift to the other unit's component cooling system using remotely operated (from control room) and/or manually operated valves. Temperature alarm switches, which sense the spent fuel pool temperature, provide alarm in the main control room for high temperature in the spent fuel pool. The boric acid evaporator supply is isolated by a Safety Injection signal to MV-32120 (MV-32122) and MV-32121 (MV-32123).

The operation of the system is monitored with the following instrumentation:

a. A temperature detector in the outlet lines of the component cooling heat exchangers;

b. A pressure detector on the line between the component cooling pumps and the component cooling heat exchanger.

c. A flow indicator in the outlet lines from the component cooling heat exchangers.

d. A radiation monitor for the outlet flow from either component cooling heat exchanger and/or the system flow to the component cooling pumps suctions.

e. A level detector on the surge tank.

Design parameters for the component cooling system components are presented in Table 10.4-2.

10.4.2.2.1 Components

a. Component Cooling Heat Exchangers

Two component cooling water heat exchangers per unit are provided to cool the discharge of the component cooling water pumps before circulating the water through the system. Component cooling water flows through the shell side and cooling water flows through the tube side of the heat exchanger. The tube-tube sheet joint is rolled and the tubes are stainless steel. The shell of the heat exchanger is carbon steel.

b. Component Cooling Pumps

Two component cooling water pumps per unit are provided to circulate component cooling water. Makeup water to the system is from the component cooling surge tank. The pump casing and impeller are constructed of carbon steel and bronze respectively.

c. Component Cooling Surge Tank

One surge tank is provided per unit to provide makeup water to the component cooling water system. The makeup water is provided from either the demineralized water system or the reactor makeup water pumps. Each unit's surge tank communicates with the other via an open expansion line that serves to equalize volume and pressure in the two tanks. The tank is fabricated of pressure vessel quality steel.

d. Component Cooling Valves

The valves used in the component cooling system are constructed of carbon steel with stainless steel trim. Since the component cooling water is not normally radioactive, special valve features (such as special leakoff lines to the Waste Disposal System) to prevent leakage to the atmosphere are not provided.

Self-actuated spring loaded relief valves are provided for lines and components that could be pressurized to their design pressure by improper operation.

e. Component Cooling Piping

All component cooling system piping is carbon steel with welded joints and connections except at certain components where flanged connections are used to facilitate maintenance.

10.4.2.3 Performance Analysis

Each unit has its own component cooling system, consisting of two parallel loops that are normally interconnected. The two loops associated with each unit are isolated into independent loops upon a safety injection signal for that unit, with each loop serving a redundant set of essential equipment and each alone capable of providing the long term cooling required by this equipment following a loss-of-coolant accident.

Thus, a single active or passive failure in any component or part of one loop, including either of the valves that isolate the loops from each other, can result at most in loss of capability of that loop and the equipment which it serves. The remaining loop would still be available to provide adequate long-term cooling. This capability is demonstrated in Reference 77.

This capability to tolerate any single active or passive failure extends also to one condition of sustained component outage, the case where two units are operating with one of the four component cooling pumps unavailable. The pumps are interconnected so that either pump from either unit can be manually aligned to serve any heat exchanger, and the capability of manually switching the pumps enables the two unit system to tolerate loss of an additional pump with one pump already out of service.

The heat exchangers are not similarly interconnected, so loss or outage of a heat exchanger disables one set of essential equipment in one unit, and loss of a second heat exchanger in that unit then cannot be accommodated.

In summary, outage of a single heat exchanger can exhaust redundancy with regard to ability to meet a single failure, but outage of one pump can be tolerated in the case of two unit operation. The conditions of permissible sustained outage are accordingly limited by the Technical Specifications.

A test (References 92 & 93) was performed that demonstrated even with low water level in the surge tank there was sufficient volume of water in each of the loops for adequate component cooling until such time as the operator takes appropriate action.

Both of the redundant systems are started following a loss of coolant incident. All electrical equipment is supplied with power from engineered safeguard buses. During normal power operation, this system is in service and is in a state of readiness at all times.

Following a loss of offsite power event, the load restoration provides a permissive start signal to the component cooling pump; the low pressure switch provides the actual signal to automatically start the component cooling pump. The resulting component cooling flow to the reactor coolant pump (RCP) prevents the loss of offsite power event from progressing to a LOCA due to loss of RCP seals (see Section 4.3.3.2.3).

For continued cooling of the reactor coolant pumps and the excess letdown heat exchanger inside the containment, most of the piping, valves, and instrumentation are located outside the primary system concrete shield at an elevation well above the anticipated post-accident water level in the bottom of the containment. Outside the containment, the residual heat removal pumps, the residual heat exchangers, the spent fuel pool heat exchanger, the component cooling pumps and heat exchangers, and associated valves, piping and instrumentation can be maintained and inspected during power operation. System design provides for the replacement of one component cooling pump while the other three pumps are in service.

The system materials are, in general, carbon steel. The component cooling water contains a corrosion inhibitor to protect the carbon steel. Welded joints and connections are used except where flanged closures are employed to facilitate maintenance. The entire system is seismic Class I design. The components are designed to the codes given in Table 10.2-2. In addition the components are not subjected to any high pressures (see Table 10.4-2) or stresses; hence a rupture or failure of the system is very unlikely.

As stated in the definition of Single Failure (USAR Section 1.5), examples of passive failures in piping systems include the failure of a check valve to move to its correct position when required or leakage from failed components (such as a pump seal or valve packing). Such leakage failures are limited to 50 gpm, maximum. That is, random piping breaks are not considered credible in addition to the initiating event.

The CC system is not high energy and Appendix I type pipe breaks are not required to be considered. Current regulatory guidance (NUREG 0800, Section 3.6.2, BTP MEB 3-1, Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment, dated July 1981) would consider the system to be of moderate energy and only a limited crack would be considered for this piping. The PINGP is not committed to NUREG 0800.

With the above discussion in mind, the following describes the effects of line breaks within the CC system. Line breaks can disable one entire loop which includes one heat exchanger and the pump that is normally associated with it and which remains open to the return flow from the heat exchanger regardless of how the pumps are aligned. Thus, accommodation of a break in one loop requires that the heat exchanger of the redundant loop in that unit be available and not out of service. If the pump of the redundant loop of that unit were out of service, however, its function could be performed by manually switching over a pump from the second unit.

Should a rupture of the reactor coolant pump thermal barrier occur, high flow would result and an air operated, remotely controlled gate valve on the component cooling water outlet of the thermal barrier heat exchanger would close, isolating all thermal barrier cooling flow. Therefore, all piping from the check valve on the pump inlet to the air-operated control valve on the pump outlet would be subjected to reactor coolant system pressure and temperature. A safety valve discharging to the reactor containment sump is provided on each pump outlet line to relieve excess pressure.

Should the component cooling water become contaminated with radioactive water due to a leak in any heat exchanger tube in the Chemical and Volume Control, the Sampling, Residual Heat Removal Systems, or a leak in the cooling coil for the reactor coolant pump thermal barrier, the leak would be detected by a combination of surge tank level variations and radiation monitoring.

Should a large tube side to shell side leak develop in a residual heat exchanger, the water level in the component cooling surge tanks would rise, and the operator would be alerted by a high water alarm. The atmospheric vents on the tanks are automatically closed in the event of high radiation level at the component cooling heat exchanger outlet header. If the leaking residual heat exchanger is not isolated from the component cooling system before the inflow completely fills the surge tanks, the excess water will flow through the surge tank overflow line. This overflow line is routed to the auxiliary building waste holdup tank.

Leakage from the component cooling system can be detected by a falling level in the component cooling surge tanks. Redundant air operated valves on the Unit 1 component cooling supply to the 122 Spent Fuel Pool heat exchanger will automatically close either on a loss of Component Cooling discharge pressure from the 122 SFP HX 01422041
or on low surge tank level. The rate of water level decrease and the area of the water surface in the tank permit determination of the leakage rate. The component which is leaking can be located by sequential isolation or inspection of equipment in the system. During normal operation the leaking exchanger could be left in service with leakage out of the component cooling system up to the capacity of the makeup line to the system from the Demineralized Water or Reactor Makeup systems. By manual transfer, emergency power is available for makeup pump operation.

Containment isolation valves are automatically closed on a safety injection signal. The component cooling water supply line to the excess letdown heat exchanger contains a check valve inside and a remotely operated valve outside the containment wall (the check valve is not credited for containment isolation purposes; see Tables 5.2-1 and 5.2-2). Except for the normally closed makeup line and equipment vent and drain lines, there are no direct connections between the component cooling water and other systems. Most of the equipment vent and drain lines outside the containment have manual valves which are normally closed unless the equipment is being vented or drained for maintenance or repair operations. Most of the vent lines are also capped when not in use as an additional safety feature.

Following a loss-of-coolant accident, one component cooling pump and one component cooling heat exchanger can accommodate the heat removal loads. If either a component cooling pump or component cooling heat exchanger fails, the redundant pump and the redundant heat exchanger provide 100% backup. Valves on the component cooling return lines from the safety injection, containment spray and residual heat removal pumps are normally open. Each of the component cooling inlet lines to the residual heat exchangers has a normally closed remotely operated valve. If one of the valves fails to open at initiation of long-term recirculation, the valve which does open supplies a heat exchanger with sufficient cooling capacity to remove the heat load.

If a break of a component cooling line occurs inside the containment, adequate valving is available outside the containment on the component cooling supply and return lines to isolate the leak, see Figures 10.4-4A and 10.4-4B (Figures 10.4-5A and 10.4-5B). None of the components inside the containment require component cooling water during recirculation. If a break occurs outside the containment, the leak could either be isolated and repaired, or the system could be shutdown for repairs depending on the position in the system at which the break occurred. During this period, no heat removal from the containment by the residual heat removal system is required since the fan coolers capability using cooling water exceeds decay heat generation.

Once the leak is isolated or the break has been repaired, makeup water is supplied from the plant makeup water system or by the reactor makeup system. A failure analysis of components is given in Table 10.2-4.

10.4.2.4 Inspection and Testing

Each pump was preoperationally tested during plant construction. The pumps are tested per the Technical Specifications. Preventive maintenance is performed on the pumps and driver systems at yearly intervals. Corrective maintenance is performed on the pumps and driver systems when necessary.

The preoperational test on each pump included tests to determine the suction head available and the shutoff head delivered. The pumps were initially operated for one hour and then operational data was recorded. The operational data recorded included suction head, discharge head, flow rates, pump speeds, bearing temperatures, and vibration data. The pumps performed as designed during the preoperational test.

10.4.3 Safeguards Chilled Water System

10.4.3.1 Design Basis

10.4.3.1.1 Function

The Safeguards Chilled Water System circulates chilled water to provide ambient air cooling to essential areas. These areas include the control room; safeguards switchgear (Unit 1 4160 VAC (4kV) and 480 VAC bus) rooms; the residual heat removal (RHR) pump pits; the relay room, including the old P-250 computer room; and the event monitoring (EM) equipment room. The system functions during both normal plant operations and accident conditions. Function of the system is to remove heat generated by safety related equipment and any accident condition. The system provides sufficient ventilation and cooling to maintain equipment operability. It does this by controlling temperatures within design ratings of the installed safety related systems. The system performs a safeguard function in that it cools critical equipment. It is Design Class III and QA Type I due to its support function of safety-related equipment.

10.4.3.1.2 Equipment Heat Release Rates

The chilled water temperature and flow rate were conservatively selected to remove the heat load in the various areas cooled. Conservative predictions of heat load in each room serviced by the safeguards chilled water system, during worst case scenarios, are approximately as follows (References 62 through 66):

Control Room	250,000 Btu/hr
Unit 1 4kV Room (safeguards)	26,000 Btu/hr
480 VAC Room 112, 122	59,000 Btu/hr
480 VAC Room 111, 121	89,000 Btu/hr
Relay Room (including old P-250 Computer Room)	245,000 Btu/hr
RHR Pit	74,000 Btu/hr
EM Room	59,700 Btu/hr

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10.4.3.2 <u>Description</u>

The safeguards chilled water system is a shared system between the two units. It consists of two separate, but normally cross-connected, closed 100% capacity loops, designated loop A and loop B, as shown on Figure 10.3-9. Each loop consists of a header with water chiller (two-stage centrifugal compressor, evaporator, water-cooled condenser, economizer with expansion tube), expansion tank to accommodate surge volume for temperature and volume variations, chilled water pump to supply the chilled water to air conditioning unit cooling coils and unit coolers as required, various unit coolers, and associated equipment, piping, valves instrumentation, and controls. The chilled water condenser is cooled by the cooling water system.

During normal operation, either water chiller and either chilled water pump can supply water to both headers and, thus, to all components of the system when cross-over valves are open. However, in the event of initiation of a safety injection signal these valves automatically close, splitting the two headers to form two separate loops, each header supplied by its associated chiller and pump. The signal also provides an automatic start signal to the idle water chiller unit(s) and the idle chilled water pump(s). These actions align the system to allow each loop to supply its own safeguards equipment and prevents a single failure from causing the loss of both chilled water loops. Loop A cools the equipment on electrical safeguards Train A and loop B similarly serves safeguards Train B. The cross-over valves must be manually reset at the valves to reopen.

The various areas cooled are supplied with chilled water fan coil unit coolers or air conditioners cooled by redundant loops. Constant flow valves are located in the discharge lines from each unit cooler. These valves maintain a constant loop total flow regardless of pressure difference across the valve. This eliminates the need for a pump minimum flow recirculation line. It also offers better control of the chiller outlet temperature since large changes in inlet temperature and flow are minimized.

Water flow through each unit cooler is controlled by a three-way divert valve. This valve is temperature controlled to pass through the cooler the amount of supply water flow needed to maintain the area temperature and bypass the remainder around the cooler.

The RHR pump motor unit coolers are unique in that they no longer have the three-way divert valve. Design Change 03ZH01 modified the safeguard chilled water supply to the unit coolers by removing the divert valves and associated temperature controllers, solenoid valves and bypass piping. In this configuration, the RHR pump motor coolers continually pass full flow of safeguard chilled water.

Power to the water chiller units, related chiller unit oil pumps, and the chilled water pumps is provided by the safety related 480 VAC miscellaneous auxiliaries electrical system.

A pneumatic source, compressed air, is required for the functioning of various valves throughout the safeguards chilled water system as well as the cooling water outlet control valves. Instrument air normally provides this pneumatic source. In a loss of instrument air event, the functions of the chillers are supported by a safety related backup compressed air system, provided by Design Change #97ZH02, that automatically supplies a 50 to 65 psig pneumatic source for a minimum 8 hr period. The system consists of two compressed air cylinders in parallel tied into the existing instrument air system. The system is capable of supplying a continuous pneumatic source providing the cylinders are maintained per plant operating procedures.

10.4.3.3 Performance Analysis

10.4.3.3.1 Unit Cooler Heat Removal Rates

Design unit cooler/air conditioner heat removal rates for the various rooms, during worst case room heat up scenarios, are as follows (References 67 through 70):

Control Room (2 Air Conditioners)	844,000 Btu/hr total
Unit 1 4kV Room (safeguards)	151,000 Btu/hr
480 VAC Room 112, 122	128,100 Btu/hr
480 VAC Room 111, 121	120,800 Btu/hr
Relay Room (4 Unit Coolers)	522,000 Btu/hr total
Computer Room	58,500 Btu/hr
RHR Pit	88,000 Btu/hr
EM Room (2 Unit Coolers)	256,650 Btu/hr total

These heat removal rates are greater than the design basis heat generation rates shown in Section 10.4.3.1.2.

10.4.3.3.2 Single Failure Considerations

If one loop of the safeguards chilled water system is assumed to be disabled (single active failure) in an accident scenario with safety injection signal, the other loop is available. In the affected loop, several rooms would suffer a loss of, or degraded, cooling.

Cooling would be lost to one Unit 1 4kV bus room. Ambient temperatures would be unacceptable for continued operation of the load sequencer in that bus. However, the load sequencer may have completed its function prior to the room temperature exceeding acceptable levels. In any case the other bus would still be available.

Cooling would be lost to two of the Unit 1 480V bus rooms (one train). Long term ambient temperatures may be unacceptable unless the doors to the rooms are opened and unnecessary heat loads identified and secured. Without operator intervention, though, the affected switchgear may not remain functional. However, the other two redundant buses would still be available.

Cooling would be lost to two of the four unit coolers in the relay room and, perhaps, the air conditioner in the computer room. Adequate cooling is maintained to the relay room with the remaining two unit coolers.

Cooling would be lost to one RHR pit in each unit. However, this will not make the uncooled train of RHR immediately inoperable. The limiting component, the RHR pump motor, is qualified for continuous operation without the unit coolers for up to one-half year following the worst case accident; all other components in the RHR pit are qualified to complete their operation time requirements even without accident or post accident chiller operation. In any case one RHR pit per unit would still have cooling.

One of the two control room air conditioning units would be lost. During normal plant operation, the control room environment is maintained with only one loop of the safeguards chilled water system operating and one operating control room ventilation system. Heat release rate in the control room during an accident scenario will not vary significantly from normal operation. Thus, the control room would not be adversely impacted.

At least one train of event monitoring equipment for both units would still have cooling.

Therefore, even if one loop of the safeguards chilled water system is assumed to be disabled (single active failure) in an accident scenario, at least one train of safeguards equipment serviced by the safeguards chilled water system would have cooling.

Even if both loops should fail, the results of room heatup analyses show that, while, without operator intervention, the control room and relay room cannot withstand a total system failure, operator actions are available in procedures to provide sufficient cooling to these rooms. The RHR pumps, 480V buses and 4kV buses can perform their functions without an immediate need for equipment heat removal and their long term operability is handled by procedures.

10.4.3.3.3 Seismic Adequacy

The safeguard chilled water system is Design Class III which implies that during the original design a specific detailed analysis for dynamic seismic loadings was not performed on the piping or unit cooler support system. However, it has been demonstrated that the safeguards chilled water system will maintain pressure boundary integrity during a seismic event (Reference 71).

Additional margin, normal load plus 50 percent, was built into the system piping supports. The majority of the piping is supported by rod-type hangers which will withstand the vertical seismic loading so that the piping will not fall from its supports. The piping is of a conventional and simple design. There is much previous experience and analysis with similar designs. The piping systems have been designed and erected in accordance with the piping code B31.1.0-1967. The design pressure is less than 150 psi. The normal operating temperature is expected to be less than 60°F. Therefore, the pressure stresses and temperature stresses are significantly below Code allowable stresses.

The unit coolers were purchased with seismic (operational basis earthquake) design and analyses specified. Since original construction, seismic supports have been provided for the unit coolers by Design Change Nos. 78L485 and 82Y230.

The safeguards chilled water system is supported by, and penetrates, Class I structures which are designed for the maximum postulated seismic event at Prairie Island. The Class I structure and the unit cooler support design provide confidence that the safeguards chilled water system will survive a seismic event without damage. The arrangement of the piping supports and the support provided by the unit coolers and wall penetrations do not allow for large piping movements in a seismic event. In general, the pipe routing in the rooms of concern will preclude it from being damaged through striking any adjacent equipment. Additionally, this piping arrangement essentially uncouples the piping from the building induced loads. The seismic inertial loads are minimized by this design.

Thus, a qualitative evaluation of the safeguards chilled water system concludes that the system is adequate for a seismic event. Therefore, engineering judgment provides reasonable assurance that the system would successfully withstand a seismic event.

In addition, an evaluation of system construction concluded that all piping and supports and associated equipment of the safeguards chilled water system, except for the chillers, is seismically adequate for the Prairie Island design basis earthquake to maintain its pressure boundary during and following the earthquake. Mechanical limit stops around the equipment mounting spring isolators for the chillers have been provided by Design Change #97ZH01 to limit the potential for lateral movement of the chillers during the Prairie Island design basis earthquake.

Therefore, based on the design and construction of the safeguards chilled water system piping, it is concluded that this piping would perform successfully during a seismic event.

10.4.3.4 Testing and Inspection

The safeguards chilled water system and its components are tested and inspected in accordance with the Prairie Island Nuclear Generating Plant ASME Section XI Inservice Testing Implementing Program.

10.4.3.5 Instrumentation and Control

Switches are provided in the control room to start and shut down the system remotely. Local switches are provided to operate the various pieces of rotating equipment.

The cross-over valves have local open-close push buttons. These valves close automatically on safety injection. They can only be opened by the local push buttons.

Control logic is provided to ascertain that cooling water is available and a flow path for the chilled water exists before starting the chiller unit.

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10.5 EQUIPMENT AND SYSTEM DECONTAMINATION

10.5.1 Design Basis

Activity outside the core could result from fission products from defective fuel elements, fission products from tramp uranium left on the cladding in small quantities during fabrication, products of n - γ or n - p reactions in the water or impurities in the water, and 98120
activated corrosion products. Fission products in the reactor coolant associated with normal plant operation and tramp uranium are generally removed with the coolant or in subsequent flushing of the system to be decontaminated. The products of water activation are not long lived and may be removed by natural decay during reactor cooldown and subsequent flushing procedures. Activated corrosion products are the primary source of the remaining activity.

The corrosion products contain radioisotopes from the reactor coolant which have been adsorbed on or have diffused into the oxide film. The oxide film, essentially magnetite (Fe_3O_4) with oxides of other metals including Cr and Ni, can be removed by chemical means presently used in industry.

Water from the primary coolant system and the spent fuel pit is the primary potential source of contamination outside of the corrosion film of the primary coolant system. The contamination could be spread by various means when access is required. Contact while working on primary system components could result in contamination of the equipment, tools and clothing of the personnel involved in the maintenance. Also, leakage from the system during operation or spillage during maintenance could contaminate the immediate areas and could contribute to the contamination of the equipment, tools, and clothing.

10.5.2 Methods of Decontamination

Surface contaminants which are found on equipment in the primary system and the spent fuel pit that are in contact with the water are removed by conventional techniques of flushing and scrubbing as required. Tools are decontaminated by flushing and scrubbing since the contaminants are generally on the surface only of non-porous materials. Personnel and their clothing are decontaminated according to the standard health physics requirements.

Those areas of the plant which are susceptible to spillage of radioactive fluids are painted with a sealant to facilitate decontamination that may be required. Generally, washing and flushing of the surfaces are sufficient to remove any radioactivity present.

The corrosion films generally are tightly adhering surface contaminants, and must be removed by chemical processes. The removal of these films is generally done with the aid of commercial vendors who provide both services and formulations. Since decontamination experience with reactors is continually being gained, specific procedures may change for each decontamination case. For corrosion films, the APAC (alkaline permanganate-diammonium citrate) treatment, or an organic acid variation of the APAC treatment is considered to be the most effective for removal. 98120

Portable components may be cleaned with a combination of chemical and ultrasonic methods if required.

10.5.3 Decontamination Facilities

Decontamination facilities on site consist of an equipment cleaning area inside containment and in the hot machine shop, a decontamination sink in the hot instrument laboratory as well as a cask decontamination pad located adjacent to the spent fuel storage pool. Fuel handling tools and other tools can be cleaned and decontaminated in the cask decontamination pad. These facilities are shared by Units 1 and 2.

In the cask decontamination pad, the outside surfaces of the shipping casks are decontaminated, as required by using water/detergent solutions and manual scrubbing to the extent required. When the outside of the casks are decontaminated, the casks may be removed with the auxiliary building crane. 98120

In the equipment cleaning areas, located in the hot machine shop and in the hot instrument small laboratories, equipment and tools can be decontaminated by using water, detergent solutions, and manual scrubbing to the extent required.

For personnel, a decontamination shower and washroom are located at Access Control. Personnel decontamination kits with instructions for their use are in the health physics station.

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20. Letter, R Clark (NRC) to D M Musolf (NSP), “Prairie Island Unit Nos. 1 and 2 Fire Protection Request for Exemption from a Requirement of Appendix R to 10 CFR Part 50, Section III.G”, February 2, 1983. (18347/1135)

21. Letter, D M Musolf (NSP) to Director of NRR, “Request for Relief from the Requirements of 10CFR50, Section 50.48(b) for Fire Areas No. 58, 59, 73 and 74”, March 11, 1983. (18347/1643)

22. Letter, R Clark (NRC) to D M Musolf (NSP), “Safety Evaluation Report on Appendix R Exemption Requests”, May 4, 1983. (18349/1242)

23. Letter, D M Musolf (NSP) to Director of NRR, “Clarifying Information in Support of Exemption Request for Fire Areas 58, 59, 73 and 74”, May 16, 1983. (18349/1382)

24. Letter, D M Musolf (NSP) to Director of NRR, “Supplementary Information to Exemption Request for Fire Areas 58, 59, 73 and 74”, September 2, 1983. (18351/0019)

25. Letter, J R Miller (NRC) to D M Musolf (NSP), “Approval of Appendix R Exemption Requests for Fire Areas 58, 59, 73 and 74”, January 9, 1984. (19896/0099)

26. Letter, D M Musolf (NSP) to Director of NRR, “Exemption Request to the Requirements of Appendix R to 10 CFR 50”, January 23, 1984. (19896/0442)

27. Letter, D M Musolf (NSP) to Director of NRR, “Information in Support of Exemption Requests Submitted January 23, 1984 and Request for Exemption from the Requirements of Section III.O of Appendix R to 10 CFR Part 50”, April 5, 1984. (19896/1274)

28. Letter, D G Eisenhut (NRC) to D M Musolf (NSP), “Exemption request of January 23, 1984 - Fire Protection Schedular Requirements of 10 CFR 50.48(c) - Prairie Island Nuclear Generating Plant, Units 1 and 2”, April 26, 1984. (19896/1457)

29. Letter, J R Miller (NRC) to D M Musolf (NSP), “Exemption from the Requirements of Appendix R to 10 CFR 50, Section III.G.2 in Fire Areas 1 and 71 and Section III.O”, July 31, 1984. (19897/1354)

30. (Not Used)

31. (Not Used)

32. (Not Used)

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34. Letter, D M Musolf (NSP) to the Director of NRR (NRC), “Fuel Rod Consolidation”, September 22, 1987. (30408/1024)

35. Letter, D M Musolf (NSP) to the Director of NRR (NRC), “Response to Generic Letter 88-14, Instrument Air Supply System Problems Affecting Safety-Related Equipment”, February 20, 1989. (1771/0215)

36. (Not Used)

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38. Generic Licensing Topical Report EDR-1(P)-A, “Ederer Nuclear Safety-Related Extra Safety and Monitoring (X-SAM) Cranes,” October 8, 1982.

39. “Criticality Analysis of the Prairie Island Units 1 & 2 Fresh and Spent Fuel Racks,” Westinghouse Commercial Nuclear Fuel Division, February 1993.

40. (Not Used)

41. Calculation No. ENG-ME-820, “CL Hydraulic Analysis - LOOP, LOCA, and Seismic Response.” 01457655

42. Calculation No. ENG-ME-244, Rev. 0, “CC Heat Exchanger TCV Position Range”, November 17, 1985. (3281/2019)

43. Design Change 95CL04, “CC HX TCV Travel Stops to Limit Cooling Water Flow through CC HX’s.” (3271/2245)

44. (Not Used)

45. WCAP 17400-NP, “Prairie Island Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis,” Rev. 0, July, 2011.

46. (Not Used)

47. (Not Used)

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49. Letter, K.C. Midock (Westinghouse) to T.L. Breene (NSP), “Northern States Power Company Prairie Island Units 1 & 2 Criticality Analysis F/A with Instrument Tube Removed”, June 27, 1997.

50. Operations Manual F5, Appendix F, “Fire Hazards Analysis”

51. Operations Manual F5, Appendix E, “Fire Protection Safe Shutdown Analysis Summary”

52. Operations Manual F5, Appendix K, “Fire Detection and Protection Systems”

53. (Not Used) 01457655

54. Letter, J P Sorenson to NRC, “Supplement 9 to License Amendment Request dated 1/29/97 Amendment to Cooling Water System Emergency Intake Design Bases”, June 30, 1997. (3404/0387)

55. Calculation No. ENG-ME-347, Rev. 1A, Minimum Required Intake Bay Volume.

56. Calculation No. ENG-355, Rev. 1A, Intake Canal Available Water Volume Comparison Analysis. (4471/2885)

57. Service Water System Self Assessment, question AQ-3, October 30, 1995.

58. (Not Used) 01457655

59. Letter, R. Clark (NRC) to L. O. Mayer (NSP), "Safety Evaluation Report and Environmental Impact Appraised on Increasing the Spent Fuel Storage Capability", May 13, 1981. (18309/0818)

60. IEEE Standard 279, "Proposed Criteria for Protection Systems for Nuclear Power Generating Stations," August 1968.

61. Letter, R. Clark (NRC) to D. M. Musolf (NSP), "Safety Evaluation Report on Spent Fuel Assembly Degradation", July 11, 1983. (18349/1865)

62. Calculation No. ENG-ME-177, Rev. 0, RHR Pump Pit Ventilation System Design. (2840/1425)

63. Calculation No. ENG-ME-185, Rev. 0, Unit 1 - 4160V Switchgear Room Ventilation System Design. (2840/1743)

64. Calculation No. ENG-ME-186, Rev. 0B, Unit 1 - 480V and Event Monitoring Room Ventilation System Design. (2839/2385) 01501543

65. Calculation No. ENG-ME-187, Rev. 1, Relay Room Ventilation System Design. (5513/1804)

66. Calculation No. ENG-ME-188, Rev. 0, Control Room Ventilation System Design. (2840/1841)

67. Drawing XH-520-1, "Relay Room Unit Cooler," Krack Corporation Part No. 2957 Drawing, Rev. 3, Model SMA-2100-CW, "Unit Cooler for Chilled Water Safeguard System", February 20, 1972.

68. Drawing XH-520-2, "RHR Pit Unit Cooler," Krack Corporation Part No. 2957 Drawing, Rev. 3, Model SMA-1700-CW, "Unit Cooler for Chilled Water Safeguard System", February 20, 1972.

69. Drawing XH-520-3, "Safe Guard Switch Gear Unit Cooler," Krack Corporation Part No. 2957 Drawing, Rev. 3, Model SMA-2100-CW, "Unit Cooler for Chilled Water Safeguard System", February 20, 1972.

70. NSP Prairie Island Technical Manual Number: XH-465-6, "Control Room Air Handlers."

71. Safety Evaluation No. 321 Rev. 2, "Safeguards Chilled Water System - Seismic Adequacy." (3381/1680)

72. Calculation No. ENG-ME-020, Latest Rev., D1/D2 and DDCLP Fuel Oil Storage Capacity.

73. Calculation No. ENG-ME-066, Latest Rev., D5/D6 Fuel Oil Storage Requirements.

74. Calculation No. ENG-ME-485, Evaluation of Prairie Island Spent Fuel Pool Thermal Transients, Rev. 0B.

75. Calculation No. ENG-ME-476, Spent Fuel Pool Thermal Heat Load, Rev. 1.

76. Calculation No. ENG-ME-477, Spent Fuel Pool Time to Boiling, Rev. 2.

77. Calculation No. ENG-ME-541, Rev 1, CC Hydraulic Model Proto Power Calc. 02-002.

78. (Not Used) 01457655

79. Generic Letter 96-06, Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions, dated September 30, 1996. [3883-0599]

80. NRC Letter to NMC, Prairie Island, Electric Power Research Institute (EPRI) Topical Report TR-113594, Resolution of Generic Letter 96-06 Waterhammer Issues, Volumes 1 and 2, dated April 24, 2002. [3994-2630]

81. Calculation ENG-ME-517, Determination of Potential Waterhammer Scenarios for Generic Letter 96-06. [4115-0109] 01450952

82. Calculation ENG-ME-518, Determination of Waterhammer Loads for Generic Letter 96-06. [4115-0232]

83. NRC Letter to NMC, Prairie Island, Final Closeout of Response to Generic Letter 96-06, dated March 21, 2003. [4071-1631]

84. Calculation PI-M-002, Safeguards Chiller Room Outside Air Duct Evaluation.

85. NEI 99-03, Rev. 1, Control Room Habitability.

86. NMC Response to Generic Letter 2003-01 for Prairie Island, Control Room Habitability.

87. Calculation ENG-ME-374, Addendum 2, Tracer Gas Inleakage Testing of the Control Room, dated Feb. 23, 2005.

88. F5, Appendix F, Fire Hazards Analysis.

89. Calculation ENG-EE-143, Rev. 0, “Review Elevated Temperature Effects for Auxiliary Building MCCs.”

90. Operations Manual H39, “Ventilation Filter Testing Program (VFTP).”

91. CHAMPS Work Order 9510143, Emergency Intake Line Flow Test.

92. Letter, G. Shibayama (PS & E) “Field Trip Report–Component Cooling System” February 12, 1973 (7202-1781).

93. Letter, D. E. Hanson and Wm. Deutsch “Notes of Field Visit Feb. 6, 7, & 8, 1973” February 14, 1973 (7202-1778).

94. (Not Used) 01457655

95. Letter, M D Wadley (NSP) to U.S. Nuclear Regulatory Commission (NRC), “Response to NRC Generic Letter 87-02, Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Issue(USI) A-46”, November 20, 1995. (4109/3981)

96. ENG-ME-254, Rev. 0, Safeguards Bay Supply Capacity.

97. Calculation ENG-ME-662, Unit 2 EDG Building Maximum Outside Air Temperature Evaluation. 01435305

98. EC 21789 Unit 2 RCP Seal Replacement Modification 01450457

99. EC 21790 Unit 1 RCP Seal Replacement Modification

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TABLE 10.2-1 FUEL HANDLING DATA

New Fuel Storage Pit	
Fuel assemblies storage capacity	74
Center-to-center spacing of assemblies, in.	21
Maximum k_{eff} with unborated water	0.95
Maximum k_{eff} with low density optimum moderation	0.98
Spent Fuel Storage Pool	
Fuel assemblies storage capacity	1582*
Number of space accommodations for spent fuel shipping cask	1
Center-to-center spacing of assemblies, in.	9.5
Maximum k_{eff} under normal operating conditions	0.95
Maximum k_{eff} with unborated water	<1.0
Miscellaneous Details	
Width of fuel transfer canal, ft.	4
Wall thickness for spent fuel storage pit, ft.	4-6
Weight of fuel assembly with RCC (dry), lb.	1404
Capacity of refueling water storage tank, each gal.	298,000
Quantity of water required for refueling, gal.	275,000

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* With the four storage racks in the southeast corner of pool #1 removed, a total of 1386 storage locations are provided.

TABLE 10.2-2 REACTOR AUXILIARY AND PLANT COOLING SYSTEMS CODE REQUIREMENTS

Component cooling heat exchangers	ASME III* Class C
Component cooling surge tank	ASME III, Class C
Component cooling system piping and valves	USAS*** B31.1
Residual heat exchangers	ASME III, Class C tube side ASME VIII** shell side
Residual heat removal system piping and valves	USAS B31.1
Spent fuel pool filter	ASME III, Class C (121) ASME VIII (122 & 123) 00-279
121 Spent fuel pool heat exchanger	ASME III, Class C tube side ASME VIII, shell side
122 spent fuel pool heat exchanger	ASME III, Class 3 (shell and tube side)
Spent fuel pool demineralizer	ASME III, Class C
Spent fuel pool cooling system piping and valves	USAS B31.1

* ASME III - American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Nuclear Vessels

** ASME VIII - American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section VIII

*** USAS B31.1 - Code for Pressure Piping, and special nuclear cases where applicable.

TABLE 10.2-3 SPENT FUEL POOL COOLING SYSTEM COMPONENT DATA

Pool Heat Load used in Evaluation of System Capability (Btu/hr)	
Normal (1362 normally discharged fuel assemblies)	16.72×10^6
Maximum (1362 normally discharged fuel assemblies plus 1 freshly offloaded core of 121 fuel assemblies)	30.54×10^6
121 Spent fuel pool heat exchanger	
Quantity	1*
Type	Shell and U-tube horizontal
Design heat transfer rate, Btu/hr	25.09×10^6
Shell side (component cooling water)	
Design Inlet temperature, °F	95
Design Outlet temperature, °F	121.6
Design flow rate, lb/hr	0.9×10^6
Design pressure, psig	150
Design temperature, °F	200
Material	Carbon steel
Tube side (spent fuel pool water)	
Design Inlet temperature, °F	171.2
Design Outlet temperature, °F	134.4
Design flow rate, lb/hr	0.6506×10^6
Design pressure, psig	150
Design temperature, °F	200
Material	Stainless steel

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* Shared by unit 1 and unit 2

TABLE 10.2-3 SPENT FUEL POOL COOLING SYSTEM COMPONENT DATA

122 Spent fuel pool heat exchanger	
Quantity	1*
Type	Shell and U-tube, horizontal
Design heat transfer rate, Btu/hr	25.09×10^6
Shell side (component cooling water)	
________ Design Inlet temperature, °F	95.0
________ Design Outlet temperature, °F	124.1°
________ Design flow rate, lb/hr	0.9×10^6
________ Design pressure, psig	150
________ Design temperature, °F	200
________ Material	Carbon Steel
Tube side (spent fuel pool water)	
________ Design Inlet temperature, °F	171.2
________ Design Outlet temperature, °F	138.5
________ Design flow rate, lb/hr	0.8×10^6
________ Design pressure, psig	150
________ Design temperature, °F	200
________ Material	Stainless Steel
Spent Fuel Pool Pump	
Quantity	2*
Type	Horizontal centrifugal
Design flow rate, gpm	1300
Total developed head, ft H_2O	150
Motor horsepower	60
Design pressure, psig	150
Design temperature, °F	200
Material	Stainless steel

TABLE 10.2-3 SPENT FUEL POOL COOLING SYSTEM COMPONENT DATA

Spent Fuel Pool Filters	
Quantity	3*
Type	Replacement cartridge
Internal design pressure of housing, psig	200 (121), 150 (122 & 123)
Design temperature, °F	250
Design flow rate, gpm	60
Maximum differential pressure across filter element at rated flow (clean cartridge), psi	5
Spent Fuel Pool Demineralizer	
Quantity	1*
Type	Flushable
Design pressure, psig	200
Design temperature, °F	250
Design flow rate, gpm	60
Resin volume, cu. ft.	20
Vessel volume, cu. ft.	27
Spent Fuel Pool Skimmers	
Quantity	2*
Design flow rate (each), gpm	50
Vertical fluctuation range:	
Floating, inch	4
Manual adjustment, feet	2
Spent Fuel Pool Skimmer Strainer	
Quantity	1*
Type	Basket
Design flow rate, gpm	100
Design pressure, psig	50
Design temperature, °F	200
Perforation, inch	1/8

TABLE 10.2-3 SPENT FUEL POOL COOLING SYSTEM COMPONENT DATA

Spent Fuel Pool Skimmer Filter	
Quantity	1*
Type	Replacement cartridge
Internal design pressure, psig	200
Design temperature, °F	250
Design flow rate, gpm	100
Spent Fuel Pool Skimmer Pump	
Quantity	1*
Type	Horizontal centrifugal
Design flow rate, gpm	100
Total developed head, ft H_2O	50
Design pressure, psig	150
Design temperature, °F	200
Material	Stainless steel
Spent Fuel Pool Cooling System Piping and Valves	
Design pressure, psig	150
Design temperature, °F	200
Spent Fuel Pool Skimmer System Piping and Valves	
Design pressure, psig	150
Design temperature, °F	200

* Shared by unit 1 and unit 2

TABLE 10.2-4 FAILURE ANALYSIS OF COMPONENTS

	Components	Malfunction	Comments and Consequences
1.	Component cooling water pumps	Rupture of a pump casing	The casing is designed for 150 psi and 200°F which exceeds maximum operating conditions. Pump is inspectable and protected against missiles. Rupture due to missiles is not considered credible. Each unit is isolable. The second unit can carry the total emergency heat load.
2.	Component cooling water pumps	Pump fails to start	One operating pump supplies sufficient water for emergency cooling.
3.	Component cooling water exchanger	Tube or shell rupture	Rupture is considered improbable because of low operating pressures. Each unit is isolable. The standby unit can carry total emergency heat load.
4.	Residual heat removal pumps	Rupture of a pump casing	The casing and shell are designed for 600 psi and 400°F. The pump is protected from overpressurization by two normally closed valves in the pump suction line and by an open relief line, containing a relief valve, back to the pressurizer relief tank. The pump is inspectable and is located in the auxiliary building protected against credible missiles. Rupture is considered unlikely but in any event the pump can be isolated.
5.	Residual heat removal pump	Pump fails to start	One operating pump furnishes half of the flow required to meet design cooldown rate. This increases the time necessary for plant cooldown.

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TABLE 10.2-4 FAILURE ANALYSIS OF COMPONENTS

	Components	Malfunction	Comments and Consequences
6.	Remote operated valves inside containment in the residual heat removal pump suction lines	Valve fails to open in one of the suction lines	One suction path is adequate for plant cooldown, however, the time required to cooldown may increase slightly.
7.	Remote operated valves inside containment on the residual heat removal pump discharge line	Valve fails to open	Injection of Residual Heat Removal System discharge through the safety injection nozzles in response to a LOCA is classified as an emergency condition and is assumed to occur only once during reactor lifetime. When the Residual Heat Removal System is placed into operation for cooldown from 350°F, pump discharge pressure instrumentation would show pump shutoff head indicating no flow if the normal loop return valve is closed. The low head safety injection line valves may be opened and utilized to direct flow through the nozzles provided a reactor coolant pump is in operation. The most extreme condition on the nozzles in this case is when a minimum of water temperature of 290°F contacts the nozzles at a maximum temperature of 350°F. Due to the low temperature of the nozzles, the small change in temperature and the classification of the occurrences, no thermal stress analysis is necessary.
8.	Residual heat exchanger	Tube or shell rupture	Rupture is considered unlikely, but in any event the faulty heat exchanger may be isolated.
9.	Spent fuel pool pumps	Rupture of a pump casing	The casing and shell are designed for 150 psi and 200°F which exceeds maximum operating conditions. The pump is inspectable and is located in the auxiliary building protected against credible accidents. Rupture is considered unlikely; however, pump can be isolated.

TABLE 10.2-4 FAILURE ANALYSIS OF COMPONENTS

Components	Malfunction	Comments and Consequences
10. Spent fuel pool pumps	Pump stops running and cannot be re-started	Pool temperature does not exceed 200°F. See Section 10.2.2.3 for details
11. Spent fuel pool pump	Suction strainer plugs	There is sufficient time to stop the pumps and clean the strainer before the spent fuel pool temperature becomes too high.
12. Spent fuel pool skimmer pump	Pump stops running and cannot be restarted	Spent fuel assemblies continue to be cooled by spent fuel pool pump. Surface of pool may become slightly murky possibly decreasing visual observations until pump is restored to service. Spent fuel pool water is clarified by passing spent fuel pool water through spent fuel pool demineralizer.

TABLE 10.2-5 CHEMICAL AND VOLUME CONTROL SYSTEM CODE REQUIREMENTS

Regenerative heat exchanger	ASME III*, Class C
Letdown heat exchanger	ASME III, Class C ASME VIII, shell side
Mixed bed demineralizers	ASME III, Class C
Reactor coolant filter	ASME III, Class C
Volume control tank	ASME III, Class C
Seal water heat exchanger	ASME III, Class C, tube side ASME VIII, shell side
Excess letdown heat exchanger	ASME III, Class C, tube side ASME VIII, shell side
Chemical mixing tank	No code stamp
Deborating demineralizers	ASME III, Class C
Cation bed demineralizer	ASME III, Class C
Seal water injection filters	ASME III, Class C
Holdup tanks	API*** - 620
Boric acid filter	ASME III, Class C
Evaporator condensate demineralizers	ASME III, Class C
Concentrates filter	ASME III, Class C

* ASME III - American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

*** American Petroleum Institute Code.

TABLE 10.2-5 CHEMICAL AND VOLUME CONTROL SYSTEM CODE REQUIREMENTS

Evaporator feed ion exchanger	ASME III, Class C
Ion exchanger filter	ASME III, Class C
Condensate filter	ASME III, Class C
Chg Pmp Pulse Dampener	ASME VIII
Chg Pmp suction Stabilizer	ASME III, Class C
Piping and valves	USAS B31.1**

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** USAS B31.1 - Code for Pressure Piping and special nuclear cases where applicable.

TABLE 10.2-6 PRINCIPAL COMPONENT DATA SUMMARY

Page 1 of 2

	Quantity[1]	Heat Transfer Btu/hr	Letdown Flow lb/hr	Letdown ΔT °F	Design Pressure psig	Design Temperature °F
Heat Exchangers					Shell/tube	Shell/tube
Regenerative	1	5.46x10^{6}	19,760	254.5	2485/2735	650/650
Letdown	1	10.2x10^{6}	19,760	166.4	150/600	200/400
Seal water	1	1.19x10^{6}	79,940	16	150/150	250/200
Excess letdown	1	4.6x10^{6}	12,350	349	150/2485	200/650

	Quantity[1]	Type	Nominal Capacity Each gpm	Head ft	Design Pressure psig	Design Temperature °F
Pumps						
Charging	3	Pos. displ.	60.5	2385psi	3000	200
Boric acid transfer	2	Centrifugal	40	235	150	250
Holdup tank recirc.	1*	Centrifugal	500	100	150	200
Reactor makeup water	4**	Centrifugal	80	192	150	125
Monitor tank	2*	Centrifugal	100	150	150	200
Concentrates holding tank transfer	2*	Centrifugal	40	150	150	200
Gas stripper feed	3*	Canned and Centrifugal	15	250	150	200

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TABLE 10.2-6 PRINCIPAL COMPONENT DATA SUMMARY

Page 2 of 2

	Quantity[1]	Type	Volume, Each Gal.	Design Pressure psig	Design Temperature °F
Tanks					
Volume Control	1	Vertical	200 ft^3	75Int/15Ext	200
Boric acid	3*	Vertical	5000	Atmos.	250
Chemical mixing	1	Vertical	5.0	150	200
Batching	1*	Jacket	800	Atmos.	250
Holdup	3*	Vertical	8800 ft^3	14	200
Reactor makeup water	4**	Vertical	46000	Atmos.	125
Concentrates holding	1*	Vertical	2000	Atmos.	250
Monitor	3*	Vertical	10,000	Atmos.	150
Chg Pump Pulse Dampener	3	Horizontal	5.0	2735	200
Chg Pump Suct. Stabilizer	3	Vertical	5.0	150	200

	Quantity[1]	Type	Resin Volume ft^3	Flow gpm	Design Pressure psig	Design Temperature °F
Demineralizers						
Mixed bed	2	Flushable	30	80	200	250
Cation bed	1	Flushable	12	40	200	250
Evaporator feed ion exchanger	4*	Flushable	20	15	200	250
Evaporating condensate	2*	Fixed	20	40	200	250
Deborating	2	Fixed	30	40	200	250

[1] Quantity per unit unless otherwise specified shared or capable of being shared by Unit 1 and Unit 2; items not marked are duplicated for each unit.

* Share Component Plant Total #

** Two per unit

TABLE 10.2-7 CHEMICAL AND VOLUME CONTROL SYSTEM PERFORMANCE REQUIREMENTS*

Original Plant design life, years	40	
Seal water supply flow rate, normal, gpm	16	
Seal water return flow rate, normal, gpm	5	
Normal letdown flow rate, gpm	40	
High flow letdown (purification) flow rate, gpm	80	
Normal total charging pump flow, gpm	46	
Normal charging line flow, gpm	30	
Maximum rate of boration with one boric acid transfer pump and one charging pump, ppm/min (EOL)	6.7	67 pcm**/min
Equivalent cooldown rate to above rate of boration, °F/min	2.0	
Maximum rate of boron dilution at refueling (two charging pumps), ppm/min	4.9	49 pcm**/min
Two-pump rate of boration (at EOL), using refueling water, ppm/min	6.0	60 pcm**/min
Equivalent cooldown rate to above rate of boration, °F/min	1.8	
Design temperature of reactor coolant entering system at full power, °F	544.5	
Design temperature of coolant return to Reactor Coolant System at full power, °F	487.6	
Normal coolant discharge temperature to holdup tanks, °F	127.0	
Amount of 12% boric acid solution required (per unit) to meet cold shutdown requirements shortly after full power operating, approximate gallons (including consideration for one stuck rod)	3100	

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* All volumetric flow rates in gpm are based on 127°F and 15 psig

** 1 pcm = 10^{-5} $\Delta k/k$

TABLE 10.2-8 MALFUNCTION ANALYSIS OF CHEMICAL AND VOLUME CONTROL SYSTEM

	Component	Failure	Comments and Considerations
1)	Letdown line	Rupture in the line inside the reactor containment	The remote air-operated valves located near the main coolant loop are closed on low pressurizer level to prevent supplementary loss of coolant through the letdown line rupture. The containment isolation valve in the letdown line outside the reactor containment and also the orifice block valves are automatically closed by the containment isolation signal initiated by the concurrent loss-of-coolant accident. The closure of the isolation valve outside the reactor containment prevents any leakage of the reactor containment atmosphere to the outside atmosphere.
2)	Charging line	See above	The check valve located near the main coolant loop will prevent supplementary loss of coolant through the line rupture. The remote operated control valve with its bypass line check can also be used to isolate the reactor coolant system from the rupture. The check valve located at the boundary of the reactor containment would prevent any leakage of the reactor containment atmosphere outside the reactor containment. Should the check valve leak, any leakage would be collected in the closed system outside of containment which serves as a secondary containment boundary. 00043

TABLE 10.2-8 MALFUNCTION ANALYSIS OF CHEMICAL AND VOLUME CONTROL SYSTEM

	Component	Failure	Comments and Considerations
3)	Seal water return line	See above	The motor-operated isolation valves located inside and outside of the containment are manually closed or are automatically closed by the containment isolation signal initiated by the concurrent loss-of-coolant accident. The closure of the inside valve would normally isolate the rupture from the outside atmosphere. Should the inside valve fail or the rupture be located between the inside valve and the containment wall, the outside valve would prevent any leakage of the reactor containment atmosphere outside the reactor containment.

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TABLE 10.2-9 RESIDUAL HEAT REMOVAL SYSTEM COMPONENT DATA
Page 1 of 2

Reactor coolant temperature at startup of residual heat removal, °F	350
Time to cool reactor coolant system from 350°F to 140°F, hr (all equipment operational)	20
Decay heat generation at 20 hrs after shutdown condition Btu/hr	32.94×10^4
H_3B0_3 concentration in refueling water storage tanks, ppm boron	2600-3500 01-091
Residual heat removal pumps	
Quantity (per unit)	2
Type	Vertical centrifugal
Design flow rate (each), gpm	2000
Total developed head, ft H_2O	280
Motor horsepower, hp	200
Material	SS
Design pressure, psig	600
Design temperature, °F	400
Residual heat removal pump room sump pumps	
Quantity (per unit)	2
Type	Self-priming sump pump
Capacity, gpm	60
Head, ft	60
Motor horsepower	3.5
Material (wetted surface)	316 stainless steel

TABLE 10.2-9 RESIDUAL HEAT REMOVAL SYSTEM COMPONENT DATA

Residual Heat Exchangers

Quantity (per unit)	2
Type	Shell and U-tube vertical
Design heat transfer, Btu/hr	26.0×10^6
Shell side (component cooling water)	
Design Inlet temperature, °F	95
Design Outlet temperature, °F	116.1
Design flow rate, lb/hr	1.25×10^6
Design pressure, psig	150
Design temperature, °F	350
Material	Carbon steel
Tube side (reactor coolant)	
Design Inlet temperature, °F	160
Design Outlet temperature, °F	133.5
Design flow rate, lb/hr	1.0×10^6
Design pressure, psig	600
Design temperature, °F	400
Material	Stainless Steel

TABLE 10.2-10 REACTOR MAKEUP WATER DEOXYGENATION SYSTEM COMPONENT DATA

Degasifier	
Number	1
Design Capacity	80 gpm
Design Pressure	full vacuum/150 psig
Design Temperature	35-150°F
Design Code	ASME Sect. VIII, Division I
Transfer Pumps	
Design Capacity	80 gpm
Design Temperature	120°F
Vacuum Pumps	
Design Capacity	26.5 SCFM @ .60″ Hg A.

01400506

TABLE 10.3-1 SAMPLING SYSTEM CODE REQUIREMENTS

Sample heat exchanger	ASME III*, Class C, tube side ASME VIII, shell side
Piping and valves	ANSI B31.1**

* ASME III - American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

** ANSI B31.1 - Code for Pressure Piping and special nuclear cases where applicable.

TABLE 10.3-2 SAMPLING SYSTEM COMPONENTS

Sample Heat Exchanger

General

Number	10 (both units)
Type	Shell and coiled-tube 98120
Design heat transfer rate (duty for 652.7°F saturated steam to 127°F liquid), each Btu/hr	1.05×10^5

Shell

Design pressure, psig	150
Design temperature, °F	200
Component cooling water flow, gpm	6.0
Pressure loss at 6.0 gpm, psi	10
Operating cooling water temperature, in (maximum), °F	95
Operating cooling water temperature, out (maximum), °F	130
Material	Carbon Steel

Tubes

Design pressure, psig	2500
Design temperature, °F	680
Sample flow, normal, each, lb/hr	200
Maximum pressure loss	40 psi
Operating sample temperature, in (maximum), °F	653
Operating sample temperature, out (maximum), °F	125
Material	Austenitic stainless steel

TABLE 10.3-2 SAMPLING SYSTEM COMPONENTS

Sample Cylinders	
Number, total	8
Volume, 4 supplied, cc	75
Volume, 4 supplied, cc	500
Design pressure, psig	1800
Design temperature, °F	Amb
Manual Pressure Reducing Valves	
Normal operating temperature, °F	120-130
Design pressure, psig	6000
Body design temperature, °F	Amb
Piping	
Design pressure, psig	2500
Design temperature, °F	650

TABLE 10.3-3 MALFUNCTION ANALYSIS OF SAMPLING SYSTEM

Sample Chains	Malfunction	Comments and Consequences
Pressurizer steam space sample, pressurizer liquid space sample, hot leg sample, or steam generator blowdown sample.	Remote operated sampling valve inside reactor containment fails to close. (Inside containment isolation valve)	Remotely operated outside containment isolation valve closes on containment isolation signal. 01-056
Any of the above sample chains.	Sample line break inside containment.	Remotely operated containment isolation valves close on containment isolation signal. 01-056
Any of the above sample chains.	Sample line break outside containment.	Remotely operated containment isolation valves close on containment isolation signal. 01-056

TABLE 10.4-1 COOLING WATER REQUIREMENTS FOR SINGLE UNIT OPERATION IN GPM

Note: These are nominal flow rates, not the output of a calculation.

	Mode 1, Power Operation or Mode 2, Startup	**Mode 3, Hot Standby**	**Normal Cooldown**	**Slow Cooldown**
Component Cooling Heat Exchangers	4,500	4,500	9,000	4,500
Fan Coil Units	3,600	3,600	3,600	3,600
Diesel Generator	- -	- -	- -	- -
Auxiliary Feedwater Pump	- -	- -	- -	- -
Air Compressor	100	50	50	50
Control Room Air Conditioning	280	280	280	280
Administration Building Air Conditioning	280	280	280	280
Cooling Water Pump Cooling	20	20	20	20
Feedwater Pump Oil Coolers	100	100	100	100
Filter Water, Strainer Backwash and Screen Backwash	250	250	250	250
Steam Generator Blowdown Heat Exchanger	100*	- -	- -	- -
Miscellaneous Equipment Ventilation	1,160	810	810	810
Turbine:				
Oil Coolers	1,290	520	520	520
Hydrogen Coolers (Unit 1)	4,350	4,350	4,350	4,350
Hydrogen Coolers (Unit 2)	4,095	4,095	4,095	4,095
Hydrogen Seal Oil Coolers (Unit 1)	170	170	170	170
Hydrogen Seal Oil Coolers (Unit 2)	150	150	150	150
Bus Duct Coolers	110	- -	- -	- -
Exciter Air Coolers (Unit 1)	400	400	400	400
Exciter Air Coolers (Unit 2)	220	220	220	220
Total Flow Required (GPM) (Unit 1)	16,710	15,330	19,830	15,330
Total Flow Required (GPM) (Unit 2)	16,255	14,875	19,375	14,875

01435983

* Only used when SGB Heat Exchanger 11(21) is in operation.

TABLE 10.4-2 COMPONENT COOLING SYSTEM COMPONENT DATA

Component Cooling Pumps		
Quantity (per unit)	2	
Type	Centrifugal	
Design flow rate (each), gpm	4000	
Total developed head, ft H_2O	175	
Motor horsepower, hp	250	
Casing material	ASTM A216 Grade WCB	
Design pressure, psig	150	
Design temperature, °F	200	
Component Cooling Heat Exchangers		
Quantity (per unit)	2	
Type	Straight Tube, Horizontal	00079
Design heat transfer, Btu/hr.	29×10^6	
Shell side (component cooling water)		
Design Inlet Temp., °F	109.5	
Design Outlet Temp., °F	95	
Design flow rate, lb/hr	2×10^6	
Design temperature, °F	200	
Design pressure, psig	150	
Material	Carbon Steel	
Tube side (cooling water)		
Design Inlet temperature, °F	85	
Design Outlet temperature, °F	97.9	
Design flow rate, lb/hr	2.25×10^6	
Design temperature, °F	200	
Design pressure, psig	150	
Material	ASTM A249 TP 304	

TABLE 10.4-2 COMPONENT COOLING SYSTEM COMPONENT DATA

Component Cooling Surge Tank	
Quantity (per unit)	1
Volume, gal.	2000
Normal water volume, gal.	1000
Design pressure (internal), psig	150
Design pressure (external), psig	Atmospheric
Material	ASTM A-515 Grade 70 Pressure Vessel Quality
Component Cooling System Piping and Valves	
Design pressure, psig	150
Design temperature, °F	250

TABLE 10.4-2a COMPONENT COOLING SYSTEM, NOMINAL MODE FLOW RATES AND DUTY (Note 1) 01291437 01320893

UNIT 1 OR UNIT 2

	Start-up		Mode 1, Power Operation		Mode 2, Startup		Mode 3, Hot Standby		Mode 4, Hot Shutdown		Mode 5, Cold Shutdown	
	GPM	**BTU/HR**	**GPM**	**BTU/HR**	**GPM**	**BTU/HR**	**GPM**	**BTU/HR**	**GPM**	**BTU/HR**	**GPM**	**BTU/HR**
1. Letdown HX	690	10.2x10^{6}	225	3.3x10^{6}	225	3.3x10^{6}	690	10.2x10^{6}	225	3.3x10^{6}	- -	- -
2. Excess Letdown HX	235	4.6x10^{6}	- -	- -	235*	4.6x10^{6}*	235*	4.6x10^{6}	235	4.6x10^{6}	- -	- -
3. Seal Water HX	95	1.19x10^{6}	95	1.19x10^{6}	95	1.19x10^{6}	95	1.19x10^{6}	95	1.19x10^{6}	- -	- -
4. Spent Fuel Pit HX	1800	7.89x10^{6}	1800	7.89x10^{6}	1800	7.89x10^{6}	1800 [A]	7.89x10^{6} [A]	1800[B]	7.89x10^{6} [B]	1800[B]	7.89x10^{6} [B]
5. RHR HX	- -	- -	- -	- -	- -	- -	- -	- -	2500	26x10^{6}	2500	26x10^{6}
6. RHR Pumps	30	- -	30	- -	30	- -	30	- -	30	0.5x10^{6}	30	- -
7. Safety Injection Pumps	50	- -	50	- -	50	- -	50	- -	50	- -	50	0.5x10^{6}
8. Reactor Coolant Pumps	400	2.4x10^{6}	400	2.4x10^{6}	400	2.4x10^{6}	400	2.4x10^{6}	200	1.2x10^{6}	- -	- -

01344967

TABLE 10.4-2a COMPONENT COOLING SYSTEM, NOMINAL MODE FLOW RATES AND DUTY (Note 1) 01291437 01320893

UNIT 1 OR UNIT 2

	Start-up		**Mode 1, Power Operation**		**Mode 2, Startup**		**Mode 3, Hot Standby**		**Mode 4, Hot Shutdown**		**Mode 5, Cold Shutdown**	
	GPM	**BTU/HR**	**GPM**	**BTU/HR**	**GPM**	**BTU/HR**	**GPM**	**BTU/HR**	**GPM**	**BTU/HR**	**GPM**	**BTU/HR**
9. Waste Gas Comp	75	$1.15x10^6$	75	$1.15x10^6$	75	$1.15x10^6$	75	$1.15x10^6$	75	$1.15x10^6$	75	$1.15x10^6$
10. Sampling HX	205	$1.05x10^6$	205	$1.05x10^6$	205	$1.05x10^6$	205	$1.05x10^6$	205	$1.05x10^6$	- -	- -
11. Containment Spray Pumps	20	- -	20	- -	20	- -	20	- -	20	- -	20	- -
Totals												
Design:												
Duty	3600	$2.848x10^7$	2900	$1.698x10^7$	3135	$2.158x10^7$	3600	$2.848x10^7$	5435	$4.688x10^7$	4475	$3.554x10^7$ 01344967
Pumps Avail.	2	2	2	2	2	2	2	2	2	2	2	2
Mandatory:												
Duty	3600	$2.848x10^7$	2900	$1.698x10^7$	2900	$1.698x10^7$	3365	$2.848x10^7$	5435	$4.688x10^7$	4475	$3.554x10^7$ 01344967
Pumps Avail.	1	1	1	1	1	1	1	1	1	1	1	1

LEGEND

* Not Mandatory

A. Start to transfer to Unit 2

B. Available on Unit 2

1. Assuming no more than one unit undergoing cooldown.
2. Assuming only one Unit under DBA, the other Unit in Mode 2, Startup.
3. The Flow Rates and Heat Removal Rates presented in this Table are nominal and not intended to be inputs into the safety analysis. 01291437

TABLE 10.4-2b COMPONENT COOLING SYSTEM, NOMINAL DBA MODE FLOW RATES AND DUTY (Note 2 & 3)

01291437 01320893 01344967

	UNIT 1						UNIT 2	
	DBA-Immediate		DBA-Interim		DBA-Long Term		Mode 2, Startup	
	GPM	BTU/HR	GPM	BTU/HR	GPM	BTU/HR	GPM	BTU/HR
1. Letdown HX	- -	- -	- -	- -	- -	- -	225	$3.3x10^6$
2. Excess Letdown HX	- -	- -	- -	- -	- -	- -	235*	$4.6x10^6$ *
3. Seal Water HX	- -	- -	- -	- -	- -	- -	95	$1.19x10^6$
4. Spent Fuel Pit HX	- -	- -	- -	- -	- -	- -	1800	$7.89x10^6$
5. RHR HX	- -	- -	2500	$26x10^6$	2500	$26x10^6$	- -	- -
6. RHR Pumps	30	$0.5x10^6$	30	$0.5x10^6$	30	$0.5x10^6$	30	- -
7. Safety Injection Pumps	50	$0.5x10^6$	50	$0.5x10^6$	50	$0.5x10^6$	50	- -
8. Reactor Coolant Pumps	- -	- -	- -	- -	- -	- -	400	$2.4x10^6$
9. Waste Gas Comp	- -	- -	- -	- -	- -	- -	75	$1.15x10^6$ *
10. Sampling HX	- -	- -	- -	- -	- -	- -	205	$1.05x10^6$

01344967

TABLE 10.4-2b COMPONENT COOLING SYSTEM, NOMINAL DBA MODE FLOW RATES AND DUTY (Note 2 & 3) 01291437 01320893

		UNIT 1						UNIT 2		01344967
		DBA-Immediate		**DBA-Interim**		**DBA-Long Term**		**Mode 2, Startup**		
		GPM	**BTU/HR**	**GPM**	**BTU/HR**	**GPM**	**BTU/HR**	**GPM**	**BTU/HR**	
11. Containment Spray Pumps		20	0.4×10^6	20	0.4×10^6	20	0.4×10^6	20	0.4×10^6	01320893
Design:	Duty	100	1.400×10^6	2600	2.740×10^7	2600	2.740×10^7	3135	2.198×10^7	01344967
	Pumps	2	2	2	2	2	2	2	2	
Mandatory:	Duty	100	1.400×10^6	2600	2.740×10^7	2600	2.740×10^7	2825	1.623×10^7	01344967
	Pumps	1	1	1	1	1	1	1	1	

LEGEND

* Not Mandatory

A. Start to transfer to Unit 2

B. Available on Unit 2

1. Assuming no more than one unit undergoing cooldown.
2. Assuming only one Unit under DBA, the other Unit in Mode 2, Startup.
3. The Flow Rates and Heat Removal Rates presented in this Table are nominal and not intended to be inputs into the safety analysis. 01291437

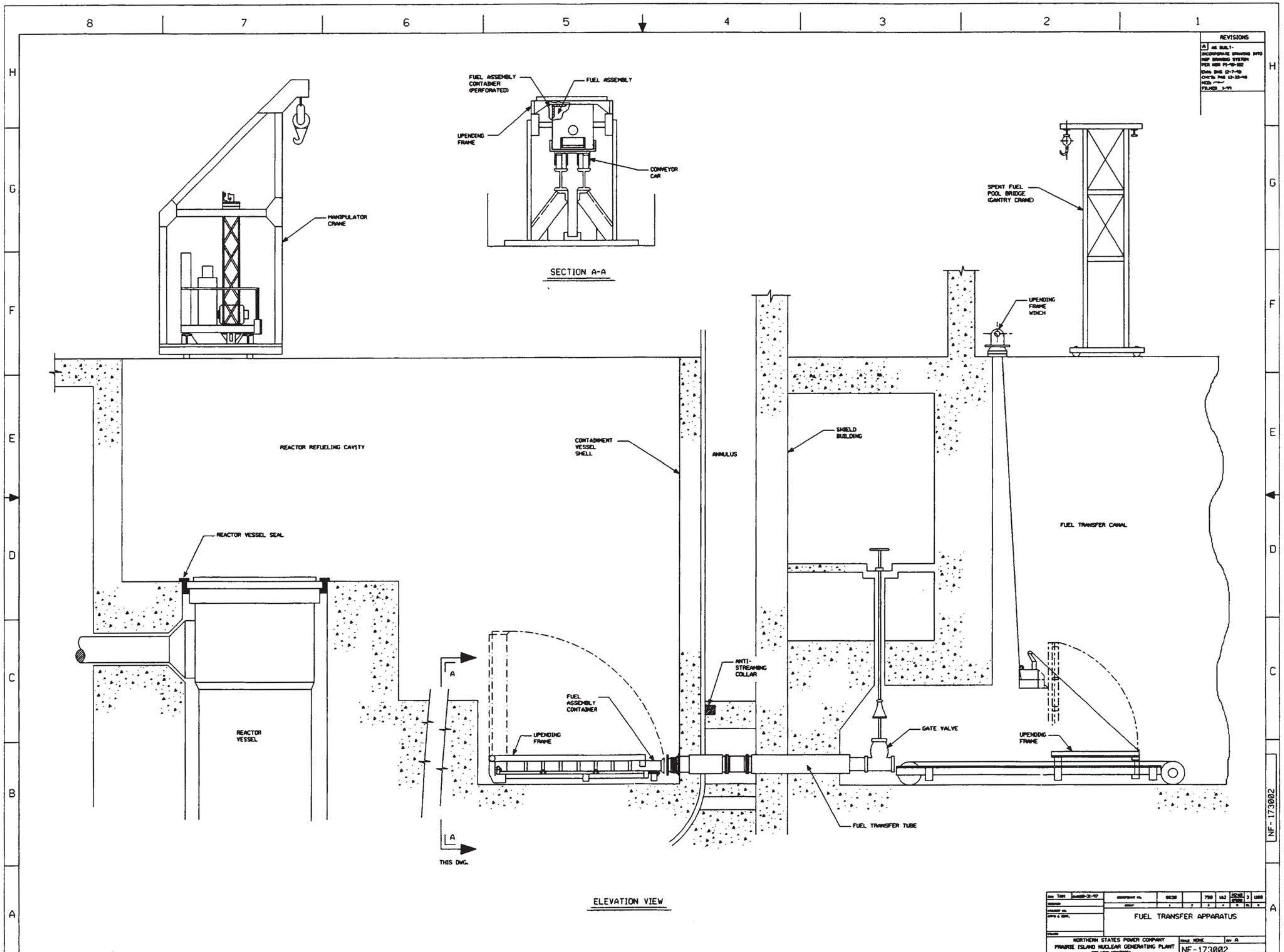
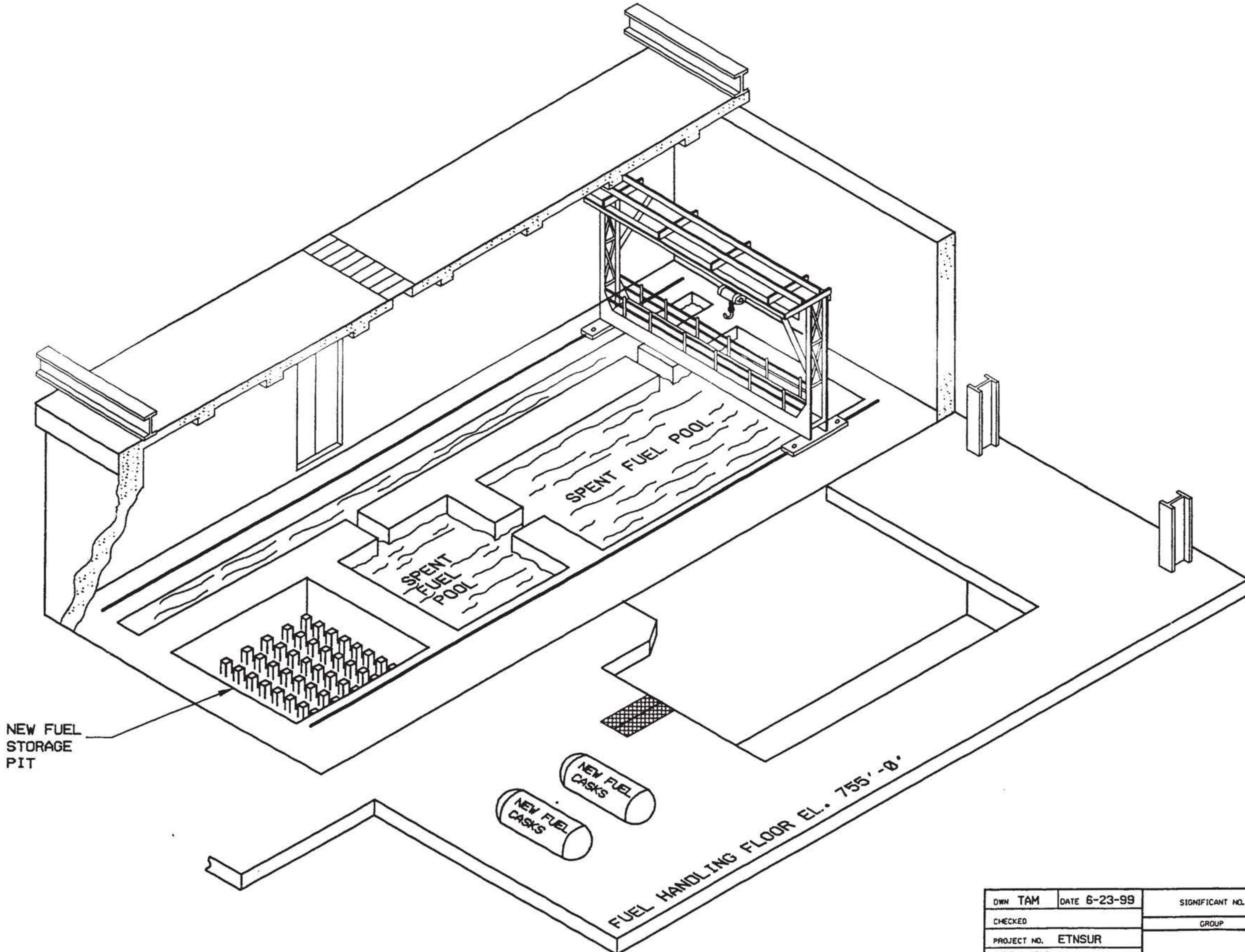


FIGURE 10.2-1 REV. 20

REVISIONS



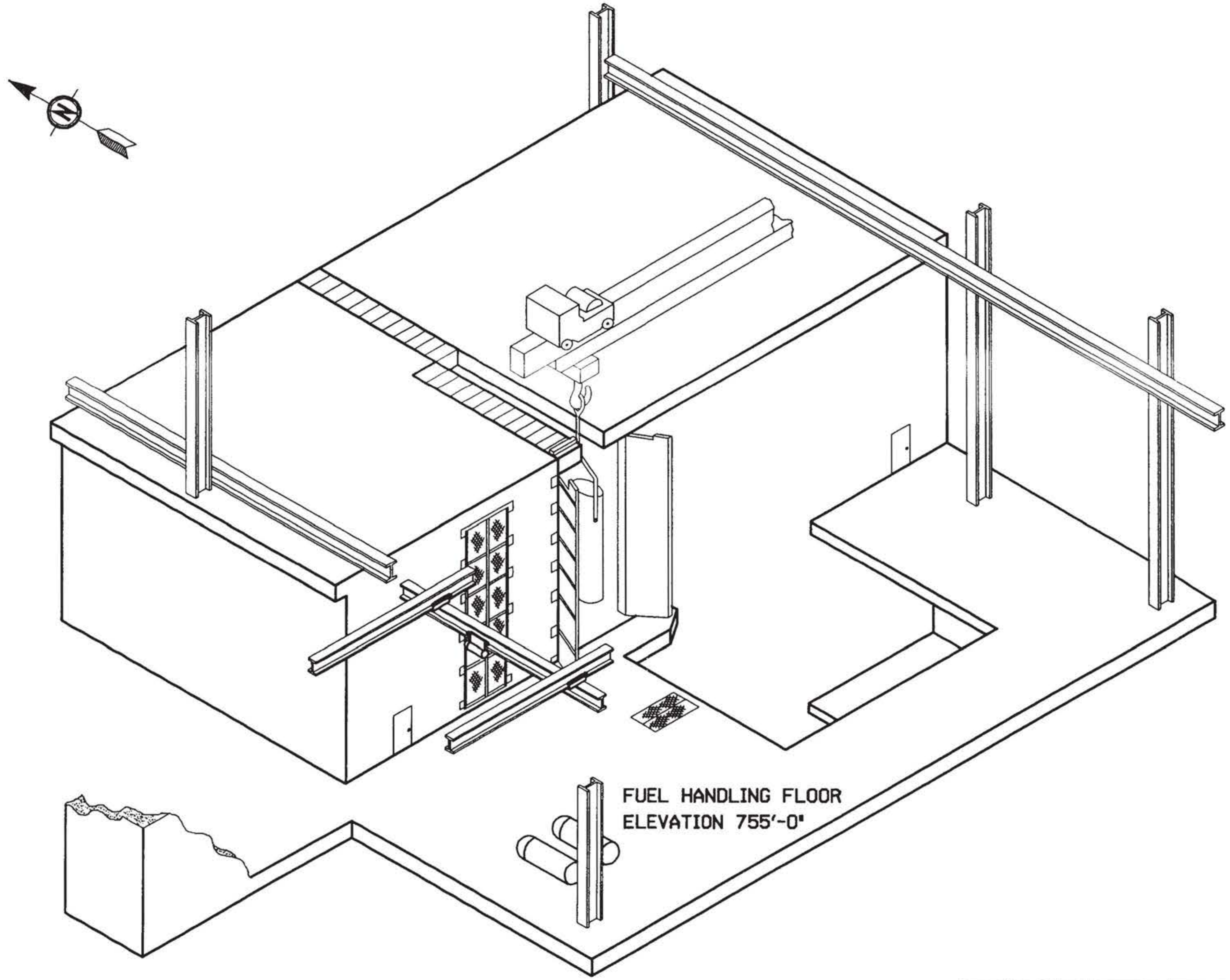
DWN TAM	DATE 6-23-99	SIGNIFICANT NO.							
CHECKED		GROUP	1	2	3	4	5	CL	6
PROJECT NO. ETNSUR									
APP'D & CERT.									
CAD FILE U10202.DGN									

SPENT FUEL POOL ENCLOSURE
INTERNAL

NORTHERN STATES POWER COMPANY
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RED WING MINNESOTA

SCALE NONE

FIGURE 10.2-2 REV. 18

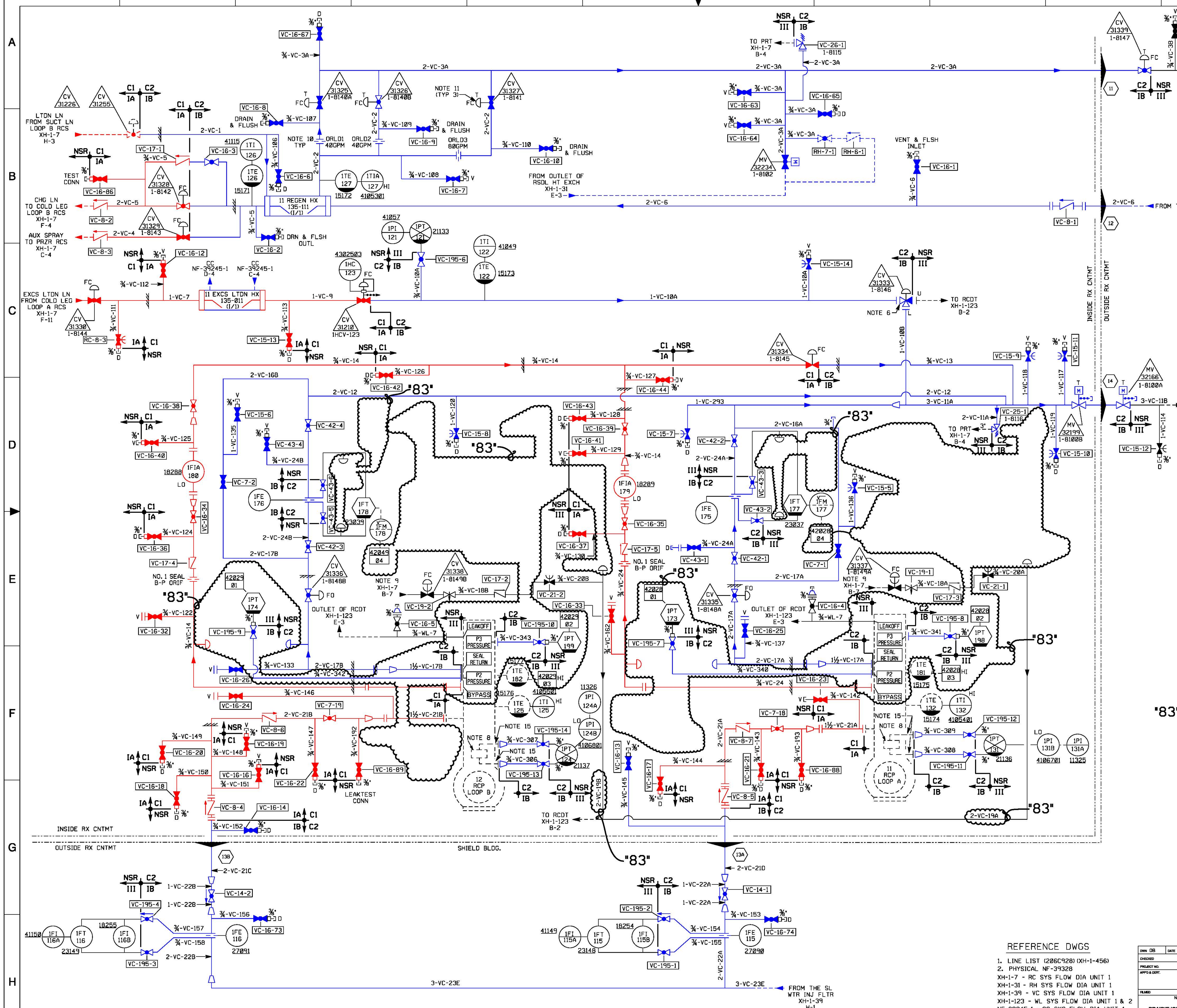


REVISIONS

DWN TAM	DATE 6-23-99	SIGNIFICANT NO.							
CHECKED		GROUP	1	2	3	4	5	CL	6
PROJECT NO. ETNSUR									
APP'D & CERT.		SPENT FUEL POOL ENCLOSURE EXTERNAL							
CAD FILE 17300302.DGN									
NORTHERN STATES POWER COMPANY PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING MINNESOTA			SCALE NONE						
			FIGURE 10.2-3 REV. 18						

FIGURE 10.2-4 REV. 34

CAD FILE: U10204.DGN



LEGEND

C2 NSR / IB III — CLASS BREAK / QA TYPE

- C1 - SAFETY- RELATED; ISI CLASS 1
- C2 - SAFETY-RELATED; ISI CLASS 2
- C3 - SAFETY-RELATED; ISI CLASS 3
- SR - SAFETY-RELATED FLUID BOUNDARY; NON-CODE CLASS
- NSR - NON-SAFETY-RELATED

IA,IB,IC,IIA,IIB & III DENOTES PIPING QA TYPES.

NOTES:

1. FOR VALVE POSITIONS SEE OPS MANUAL C PROCEDURE CHECKLIST.
2. UNLESS OTHERWISE SHOWN, CLASS BOUNDARIES STOP AT VENT AND DRAIN VALVES WITH DOWNSTREAM ITEMS (e.g., PIPE, CAPS, ETC.) CLASSED AS NON-SAFETY-RELATED.
3. DENOTES CATEGORY I VENTILATION ZONE
4. FOR SYMBOLS SEE XH-1-105, NF-39214.
5. I/1 DENOTES EQUIPMENT QA TYPE/ DESIGN CLASS.
6. VALVE FAILS WITH THE FLOW TO VOLUME CONTROL TANK.
7. "83"
8. REACTOR COOLANT PUMP THERMAL BARRIER.
9. FROM PRESSURIZER RELIEF TANK, REACTOR MAKE-UP WATER LINE.
10. LETDOWN ORIFICES ORLD 1, ORLD 2 UPSTREAM OF ISOLATION VALVES 8140A, 8140B RESPECTIVELY ARE DESIGNED TO PASS 40 GPM AND THIRD ORIFICE ORLD 3 UPSTREAM OF ISOLATION VALVE 8141 IS DESIGNED TO PASS 80 GPM.
11. INSTALL A NEEDLE VALVE IN THE AIR LINE BETWEEN THE SOLENOID VALVE AND THE ORIFICE ISOLATION VALVE OPERATOR, MAKING THE OPENING TIME APPROXIMATELY 10 SECONDS.
12. FOR ROUTING OF VALVE STEM LEAKOFF (INSIDE CONTAINMENT) SEE NF-39250
13. T - DENOTES CNTMT ISOLATION TRIP SIGNAL
14. DENOTES CNTMT PENETRATION NUMBER
15. 1/8" HOLE DRILLED AT THERMAL BARRIER FLANGE LAB SEAL DP XMTR TAPS.

REFERENCE DWGS

1. LINE LIST (2060920) (XH-1-456)
2. PHYSICAL NF-39328

XH-1-7 - RC SYS FLOW DIA UNIT 1
XH-1-31 - RH SYS FLOW DIA UNIT 1
XH-1-39 - VC SYS FLOW DIA UNIT 1
XH-1-123 - WL SYS FLOW DIA UNIT 1 & 2
NF-39245-1 - CC SYS FLOW DIA UNIT 1

REVISIONS

81 AS BUILT- CAP CHANGED TO VENTED CAPS ON VALVES. (VC-16-4 & VC-16-5)
PER DRR PI-15-028
DWN: MEA 2-17-15
CHK'D: MJN 2-18-15
MOD#: EC-25202
APP'D: CMR 2-18-15

82 AS BUILT- VALVES CHANGED FROM NC TO NO. (VC-16-4 & VC-16-5)
PER DRR PI-15-029
DWN: MEA 3-10-15
CHK'D: RWC 3-16-15
MOD#: EC-25203
APP'D: KJW 3-16-15

83 ISSUED FOR CONSTRUCTION
REMOVED STANDPIPES & EXISTING XMTRS 1PT-173, 1PT-174, 1PT-175, 1PT-176 & ASSOC DEVICES & PIPING. ADDED NEW XMTRS 1PT-173, 1PT-174, 1PT-198, 1PT-199 ASSOC DEVICES & PIPING. REMOVED SEAL LINE 2 DEVICES & PIPING. DELETED NOTE 7.
DWN: JMB 3-11-14
CHK'D: FJH 3-11-14
MOD#: EC-21790
REVIEWED: RW 3-11-14
PE CERT: T.A. MAHER 3-11-14 PE# 51207
APP'D: TAM 3-11-14
ISSUED FOR CONSTRUCTION INCORP. PLANNER COMMENTS, REDESIGNATED PREVIOUSLY OMITTED LINES 2VC-19A & 2VC-19B.
DWN: JMB 7-28-14
CHK'D: FJH 9-16-14
MOD#: EC-21790
REVIEWED: RWA 9-16-14
PE CERT: PE#
APP'D:
AS BUILT- INCORPORATED A/E's REV. 83 PER DRR PI-15-005 CERTIFIED REV. 83, TRANSFERRED TO RECORD TRACING.
DWN: MEA 4-17-15
CHK'D: RSA 6-1-15
MOD#: EC-21790
APP'D: KJW 6-1-15

This map / document is a tool to assist employees in the performance of their jobs. Your personal safety is provided for by using safety practices, procedures and equipment as described in safety training programs, manuals and SPAR's.

DWN DB	DATE 8-12-10	SIGNIFICANT NO.	8630			1	M430 9071	1	4320
CHECKED		GROUP	1	2	3	4	5	CL	6

FLOW DIAGRAM
CHEMICAL AND VOLUME CONTROL
UNIT 1

NORTHERN STATES POWER COMPANY
Xcel Energy
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RED WING, MINNESOTA

SCALE NONE — XH-1-38 — REV 83

XH-1-38

01516979

FIGURE 10.2-5 REV. 34

CAD FILE: U10205.DGN

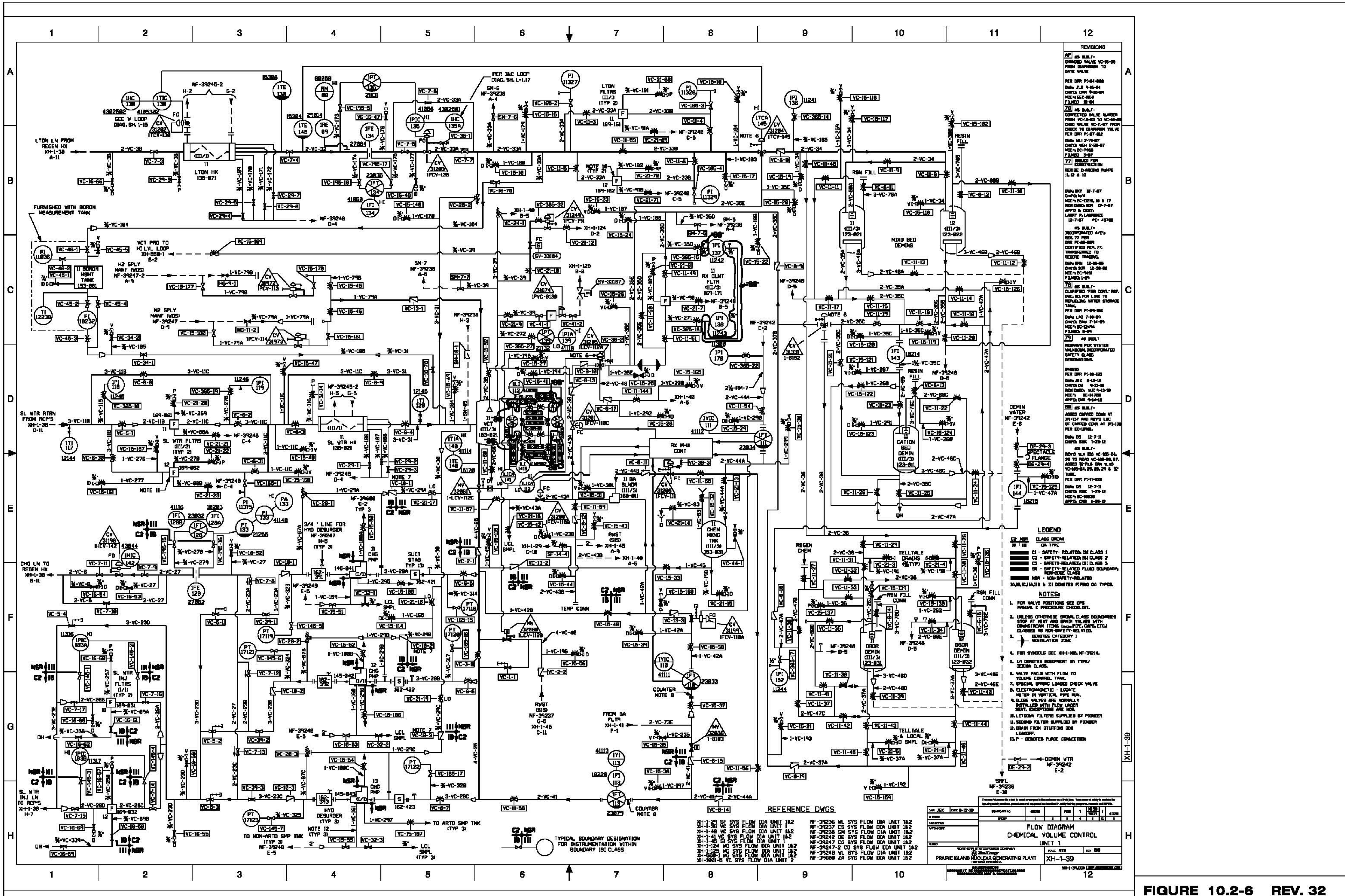


FIGURE 10.2-6 REV. 32

01352784

CAD FILE: U10206.DGN

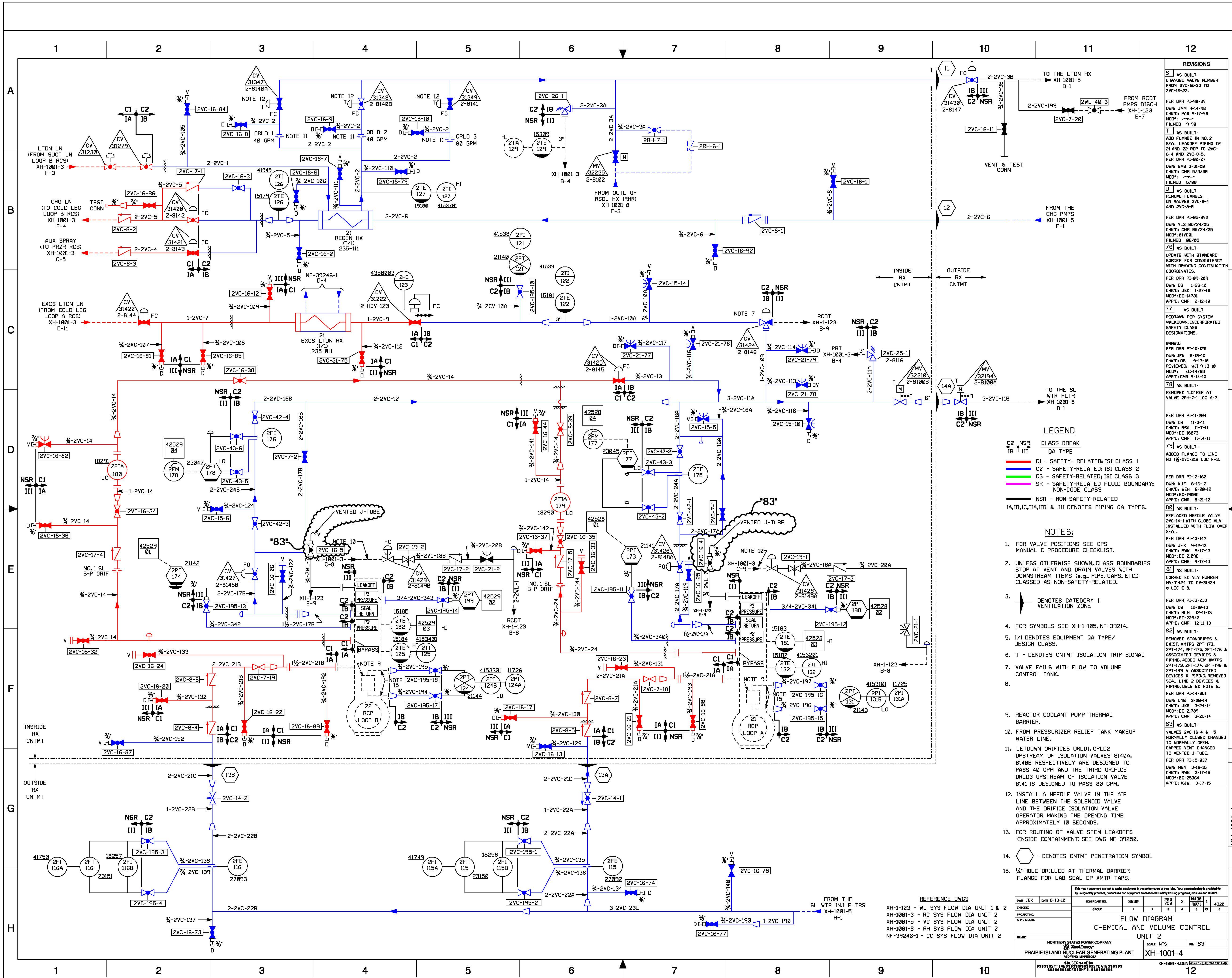


FIGURE 10.2-7 REV. 34

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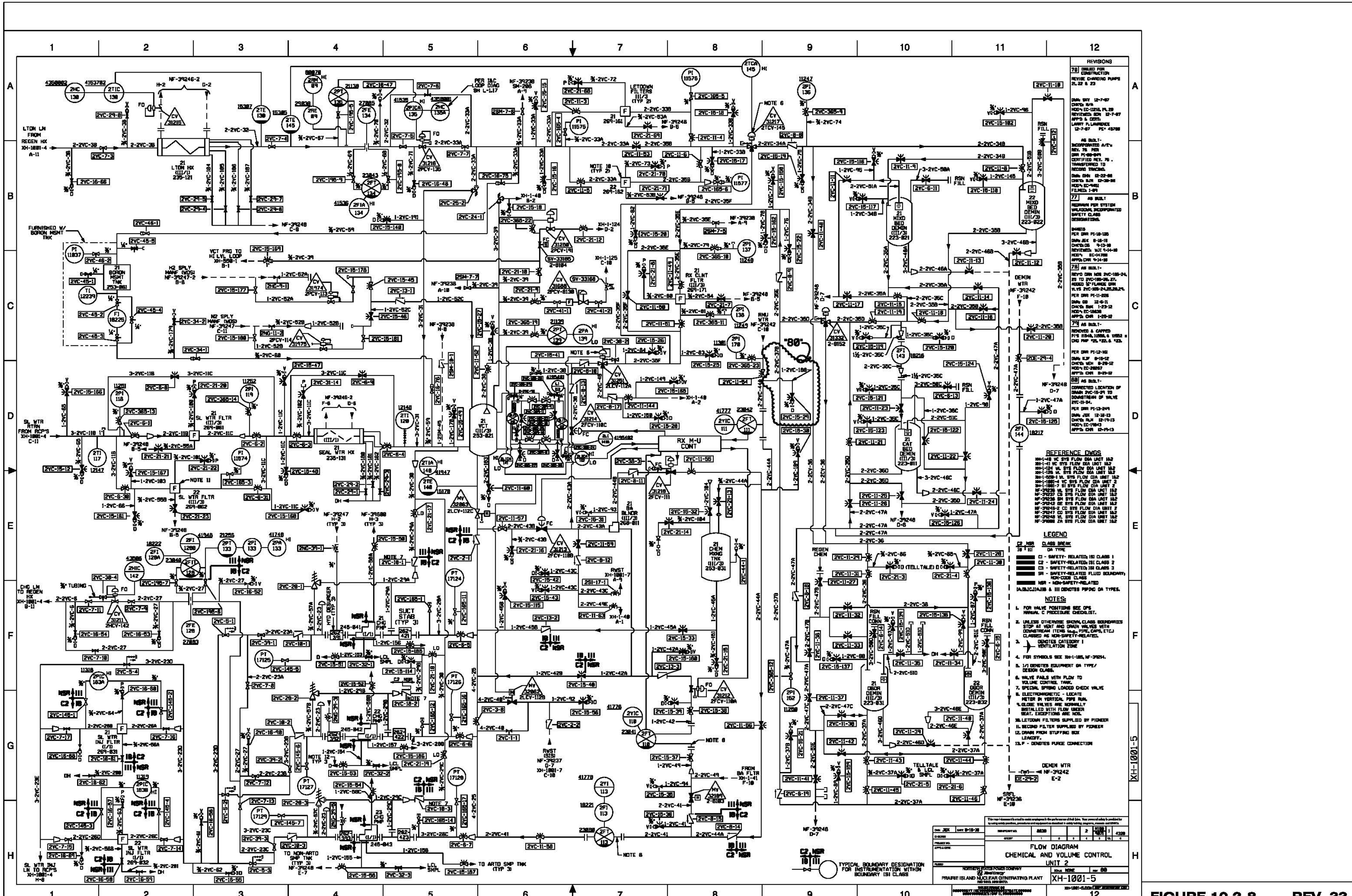


FIGURE 10.2-8 REV. 33

CAD FILE: U10208.DGN

01429088

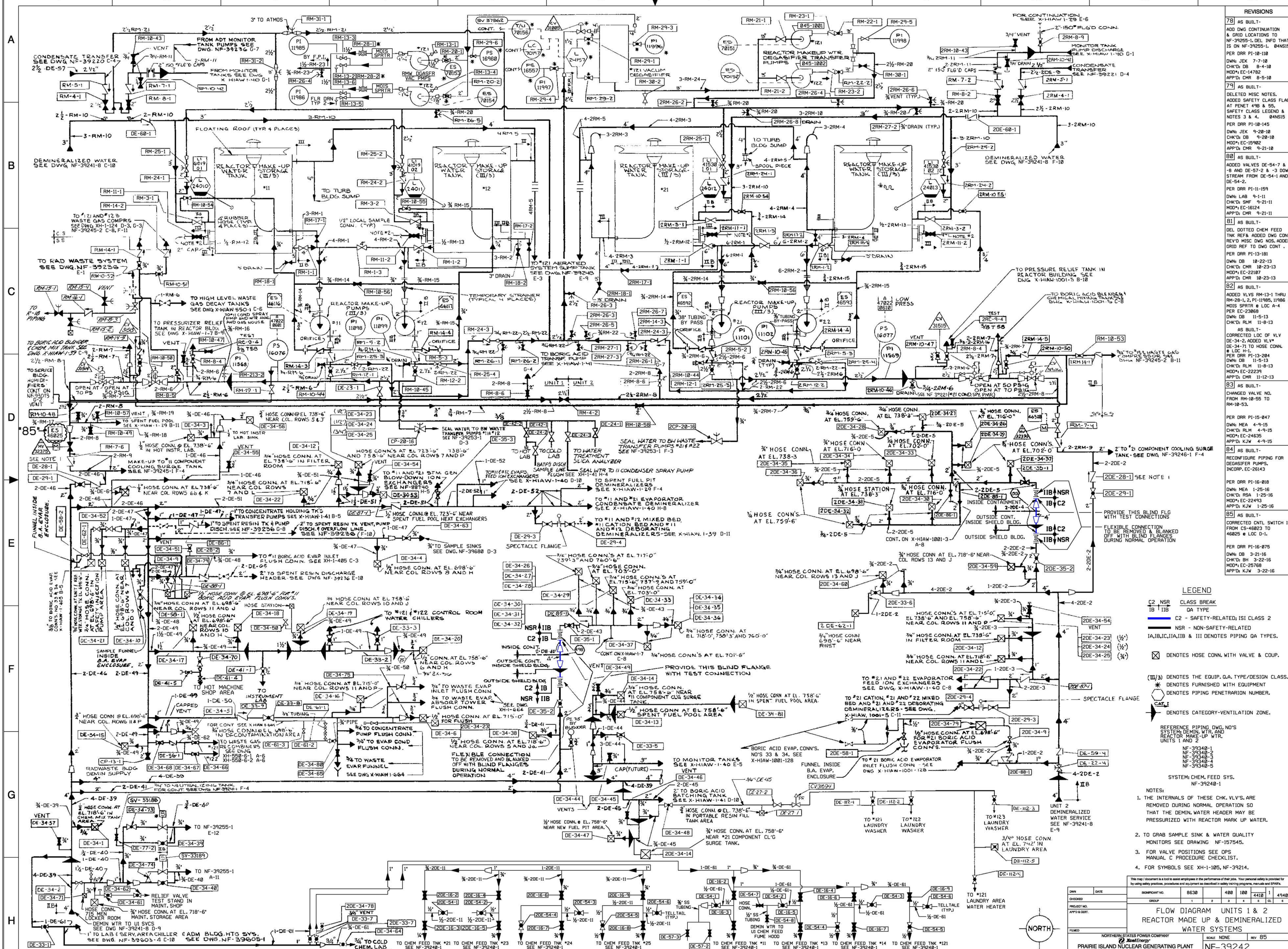
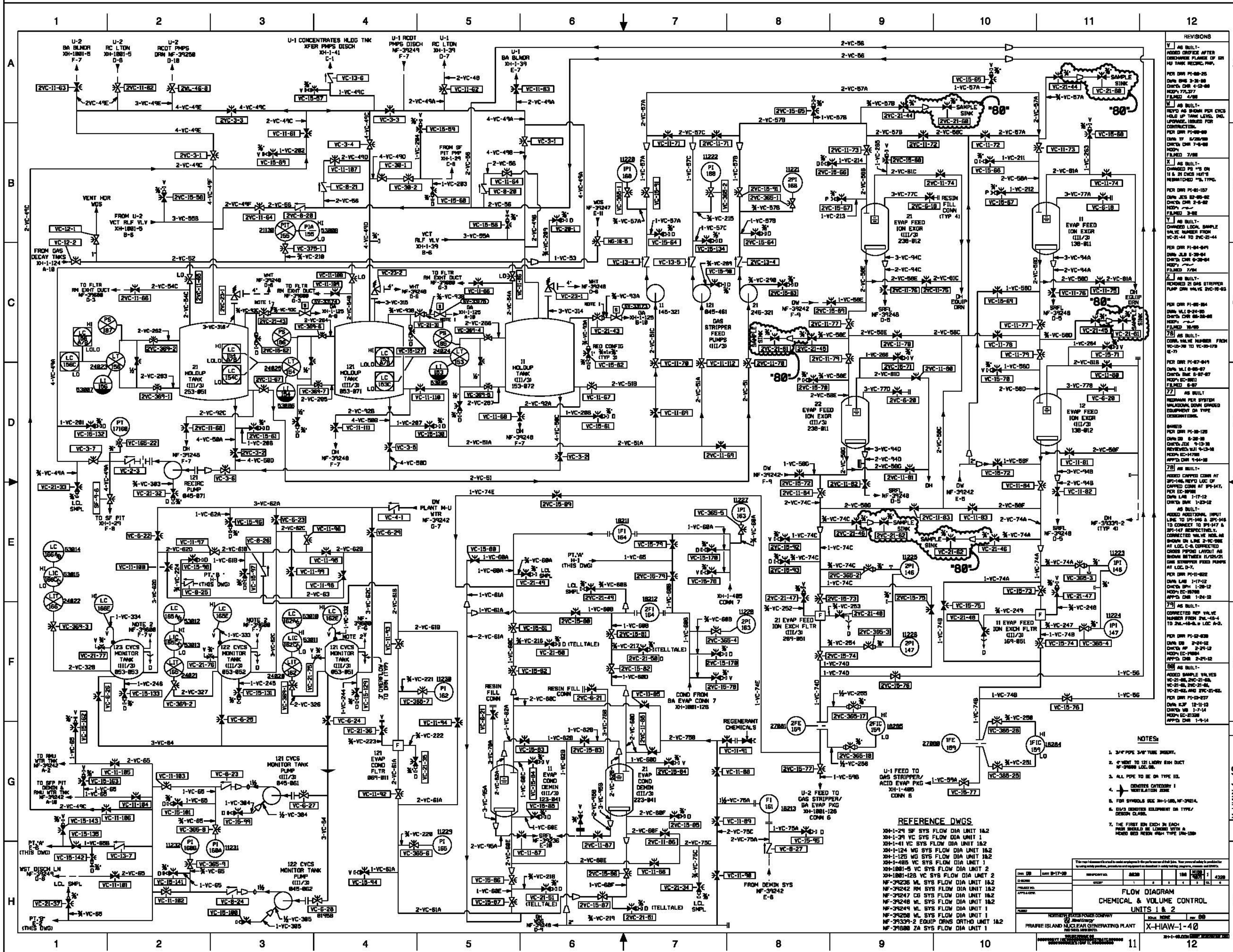


FIGURE 10.2-9 REV. 34

01516979

CAD FILE: U10209.DGN



01429088

FIGURE 10.2-10A REV. 33

CAD FILE: U10210A.DGN

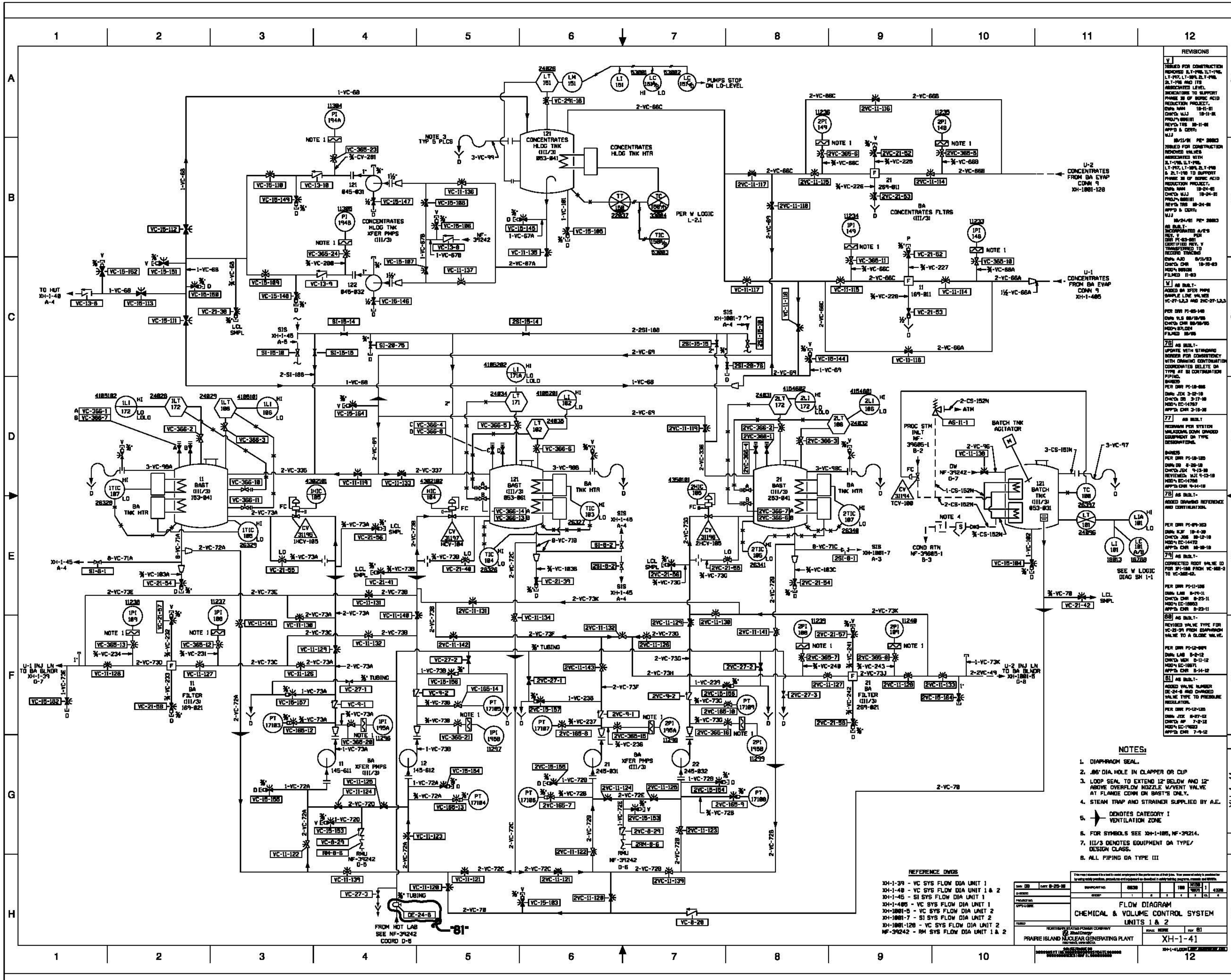


FIGURE 10.2-10B REV. 32

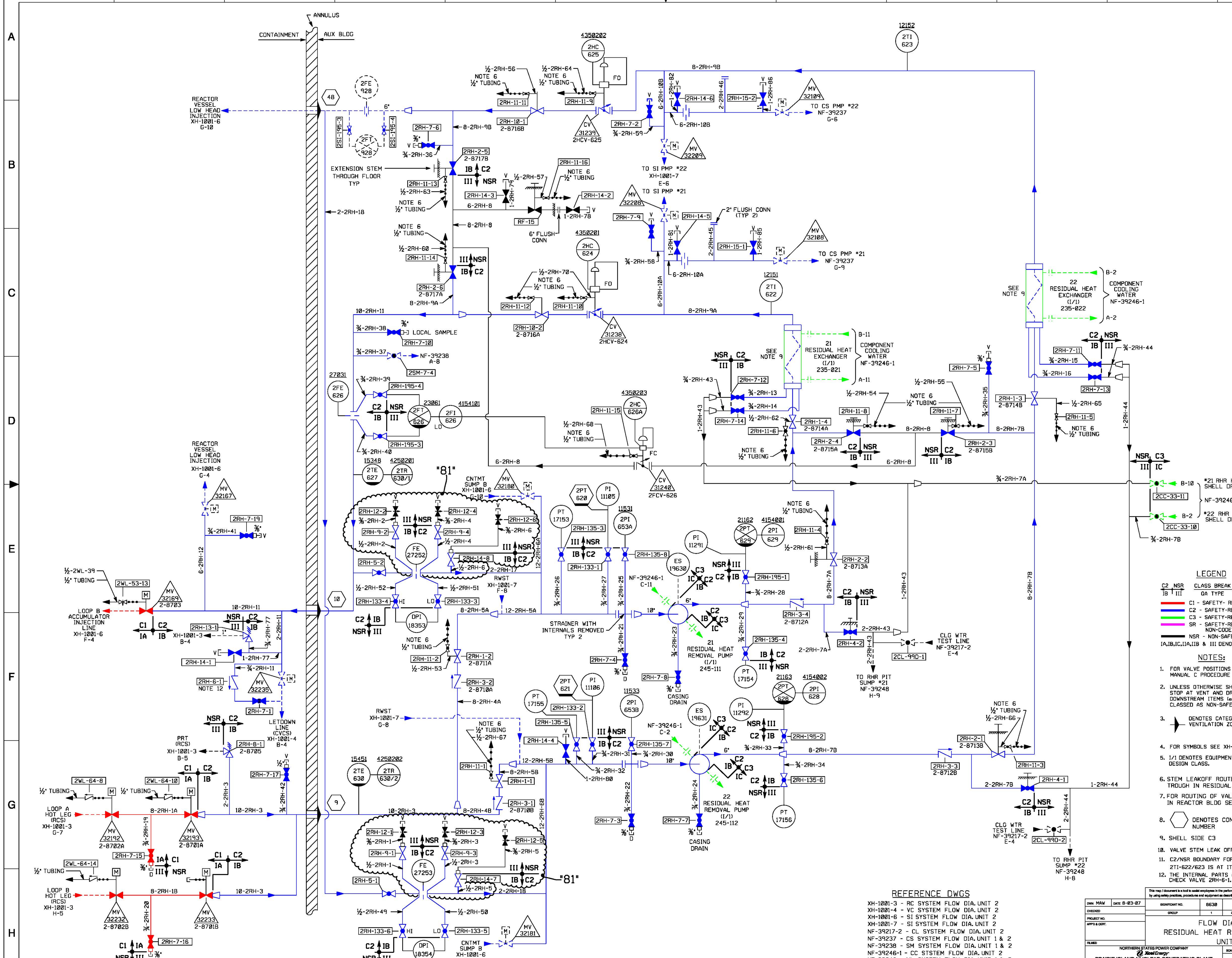
01352784

CAD FILE: U10210B.DGN

FIGURE 10.2-11 REV. 34

01516979

CAD FILE: U10211.DGN



LEGEND

C2 NSR / IB III — CLASS BREAK / QA TYPE

- C1 - SAFETY- RELATED; ISI CLASS 1
- C2 - SAFETY-RELATED; ISI CLASS 2
- C3 - SAFETY-RELATED; ISI CLASS 3
- SR - SAFETY-RELATED FLUID BOUNDARY; NON-CODE CLASS
- NSR - NON-SAFETY-RELATED

IA,IB,IC,IIA,IIB & III DENOTES PIPING QA TYPES.

NOTES:

1. FOR VALVE POSITIONS SEE OPS MANUAL C PROCEDURE CHECKLIST.
2. UNLESS OTHERWISE SHOWN, CLASS BOUNDARIES STOP AT VENT AND DRAIN VALVES WITH DOWNSTREAM ITEMS (e.g., PIPE, CAPS, ETC.) CLASSED AS NON-SAFETY-RELATED.
3. DENOTES CATEGORY I VENTILATION ZONE
4. FOR SYMBOLS SEE XH-1-105, NF-39214.
5. I/1 DENOTES EQUIPMENT QA TYPE/ DESIGN CLASS.
6. STEM LEAKOFF ROUTED TO COLLECTION TROUGH IN RESIDUAL HEAT PIT.
7. FOR ROUTING OF VALVE STEM LEAK OFFS IN REACTOR BLDG SEE DWG NF-39250.
8. DENOTES CONTAINMENT PENETRATION NUMBER
9. SHELL SIDE C3
10. VALVE STEM LEAK OFF LINES ARE NSR.
11. C2/NSR BOUNDARY FOR 2TE-627/630 & 2TI-622/623 IS AT ITS THERMOWELL.
12. THE INTERNAL PARTS ARE REMOVED FROM CHECK VALVE 2RH-6-1.

REFERENCE DWGS

XH-1001-3 - RC SYSTEM FLOW DIA. UNIT 2
XH-1001-4 - VC SYSTEM FLOW DIA. UNIT 2
XH-1001-6 - SI SYSTEM FLOW DIA. UNIT 2
XH-1001-7 - SI SYSTEM FLOW DIA. UNIT 2
NF-39217-2 - CL SYSTEM FLOW DIA. UNIT 2
NF-39237 - CS SYSTEM FLOW DIA. UNIT 1 & 2
NF-39238 - SM SYSTEM FLOW DIA. UNIT 1 & 2
NF-39246-1 - CC STSTEM FLOW DIA. UNIT 2
NF-39248 - WL SYSTEM FLOW DIA. UNIT 1 & 2

DWN MAW DATE 8-03-07 | SIGNIFICANT NO. 8630 | 2 | J104 9070 | 1 | 4822

FLOW DIAGRAM
RESIDUAL HEAT REMOVAL SYSTEM
UNIT 2

NORTHERN STATES POWER COMPANY
Xcel Energy
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RED WING, MINNESOTA

SCALE NONE REV 81
XH-1001-8

XH-1001-8.DGN NSRP GENERATION CAD

FIGURE 10.2-12 REV. 34

01516979

CAD FILE: U10212.DGN

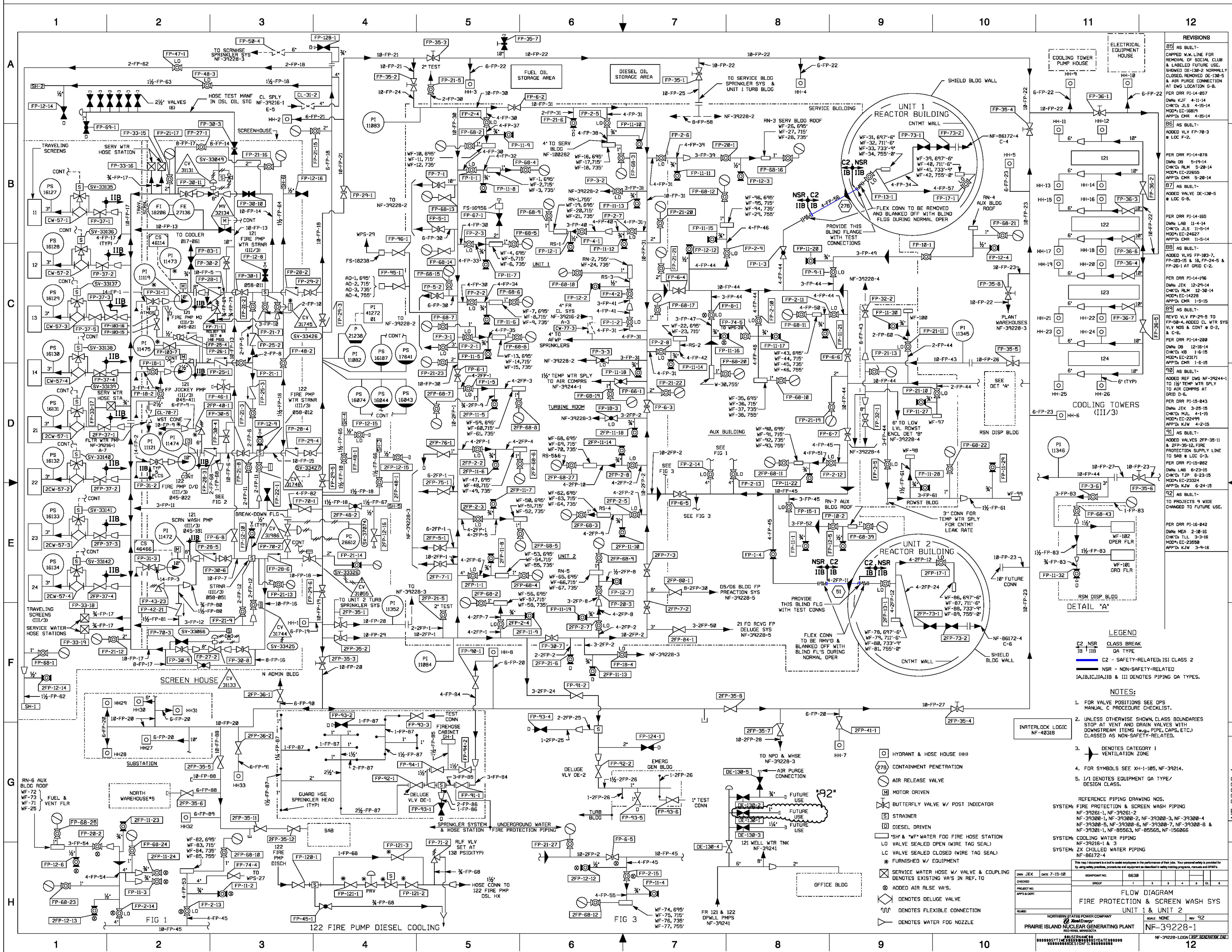


FIGURE 10.3-1 REV. 34

01516979

CAD FILE: U10301.DGN

FIGURE 10.3-2 REV. 34

CAD FILE: U10302.DGN

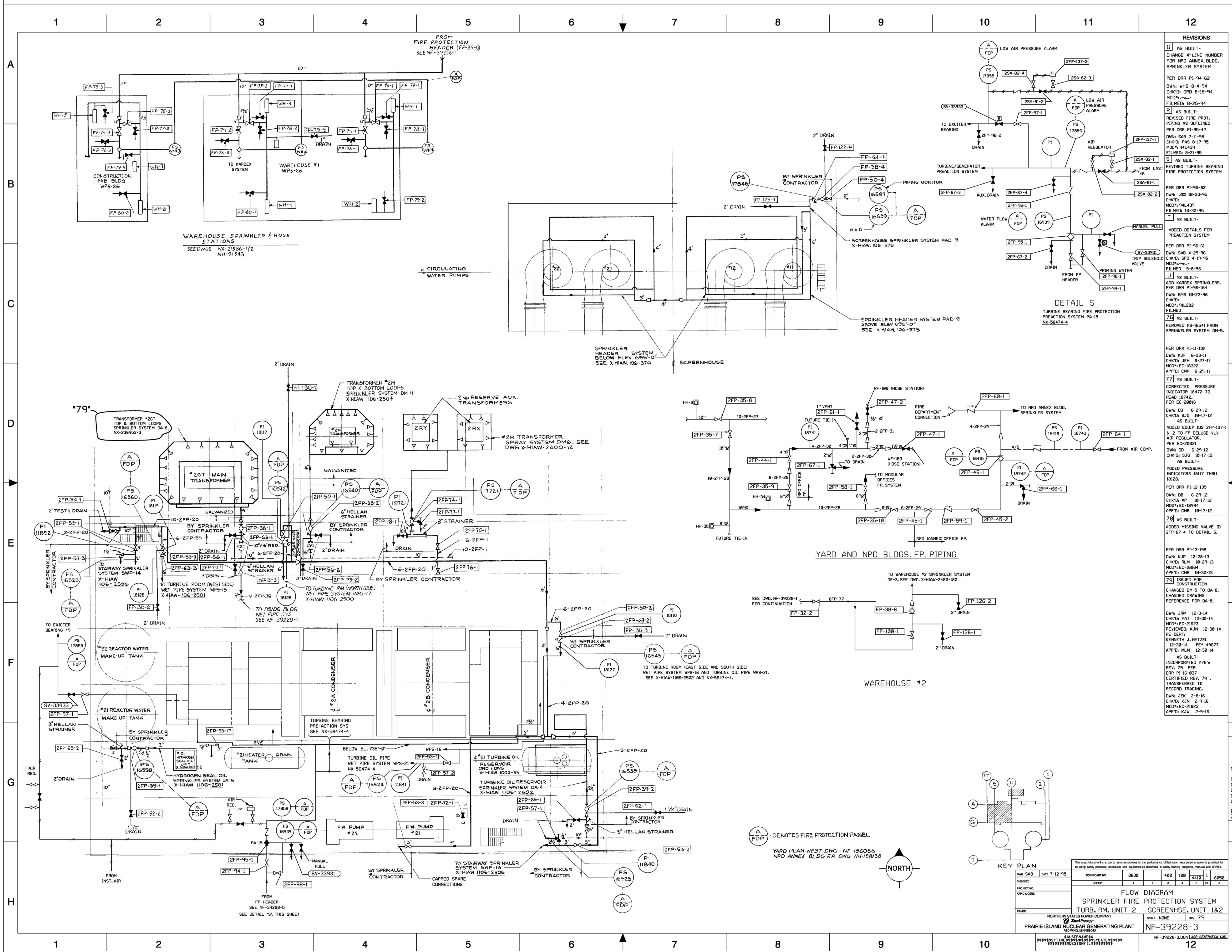


FIGURE 10.3-3 REV. 34

CAD FILE: U10303.DGN

01516979

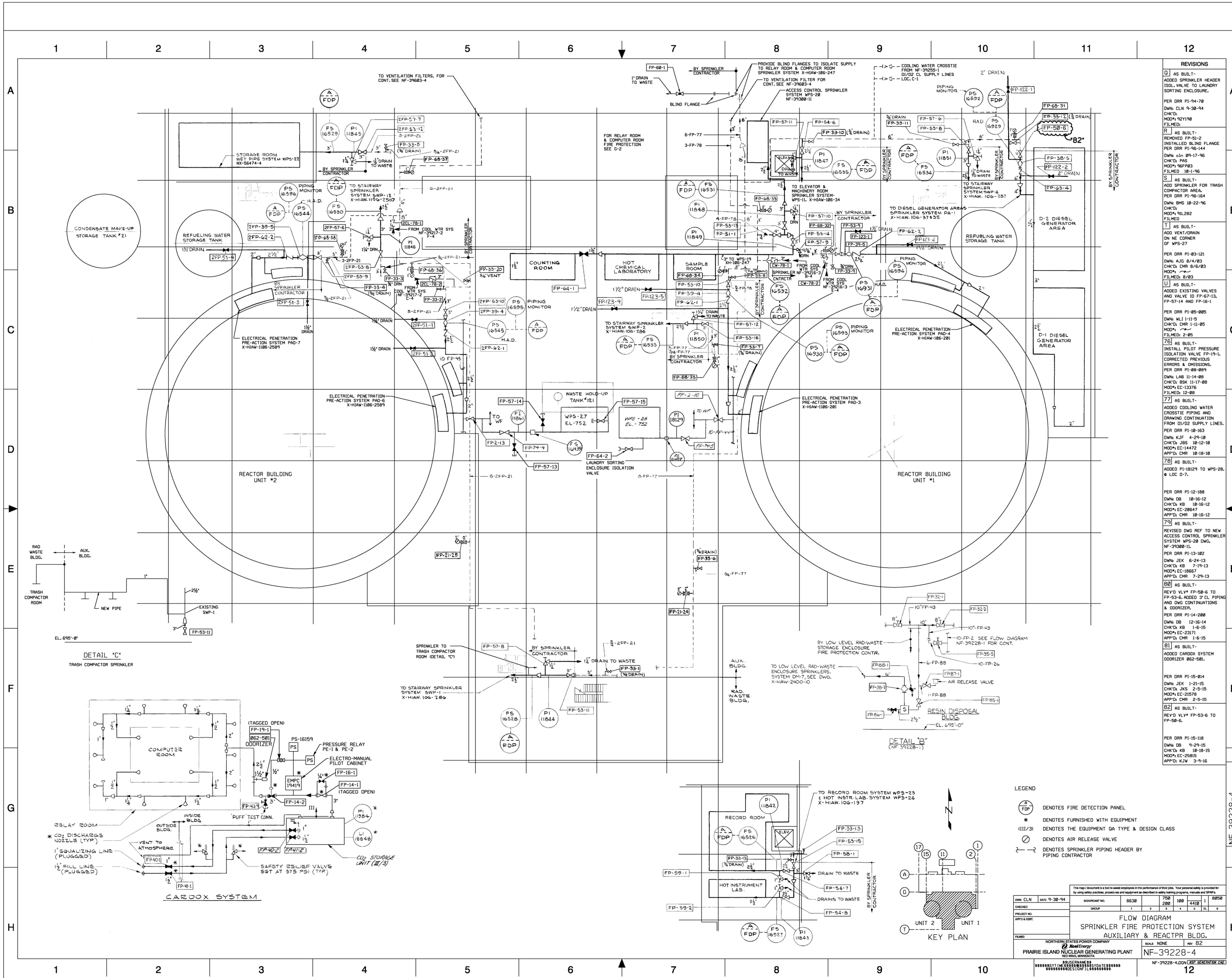


FIGURE 10.3-4 REV. 34

01516979

CAD FILE: U10304.DGN

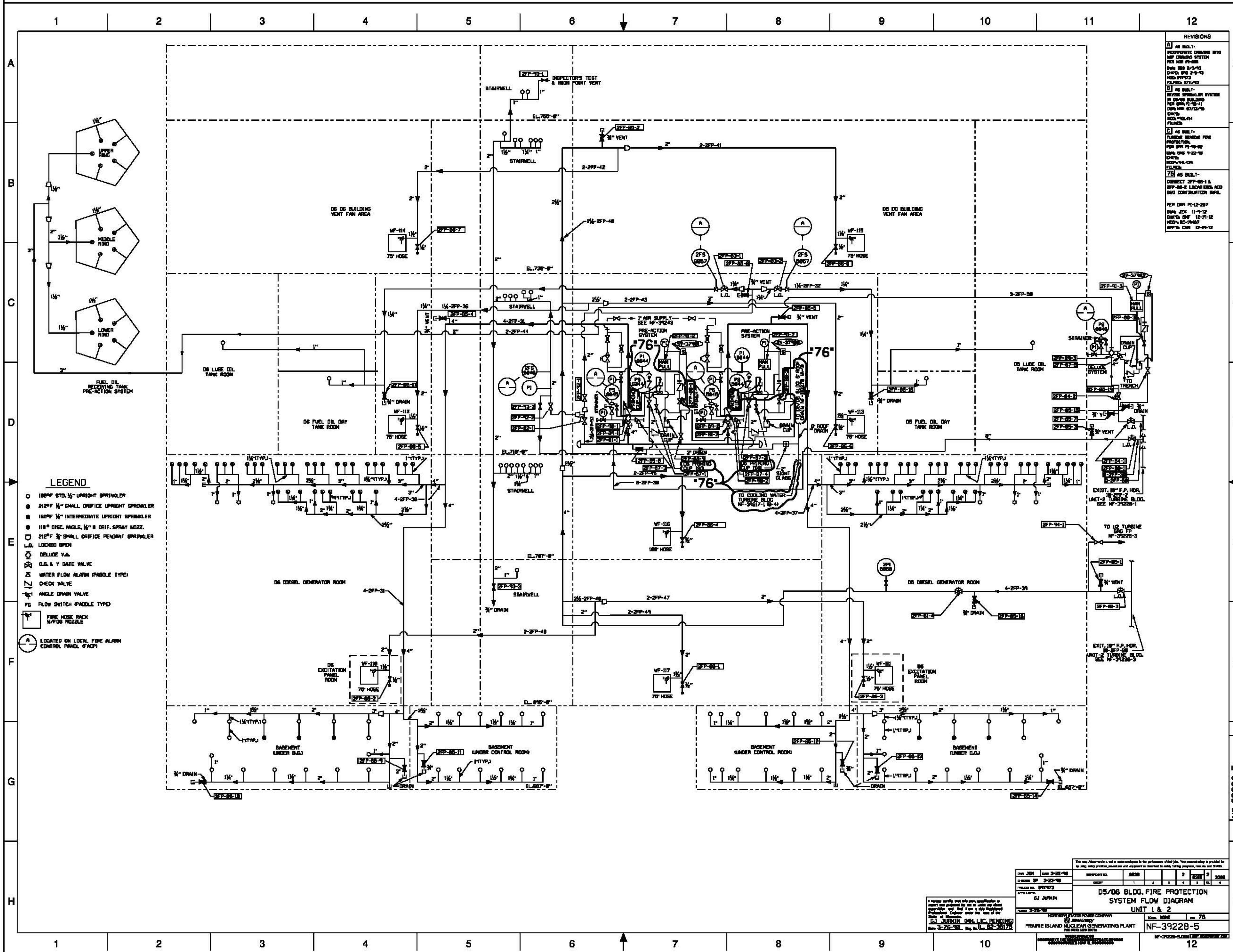


FIGURE 10.3-5 REV. 33

CAD FILE: U10305.DGN

01429088

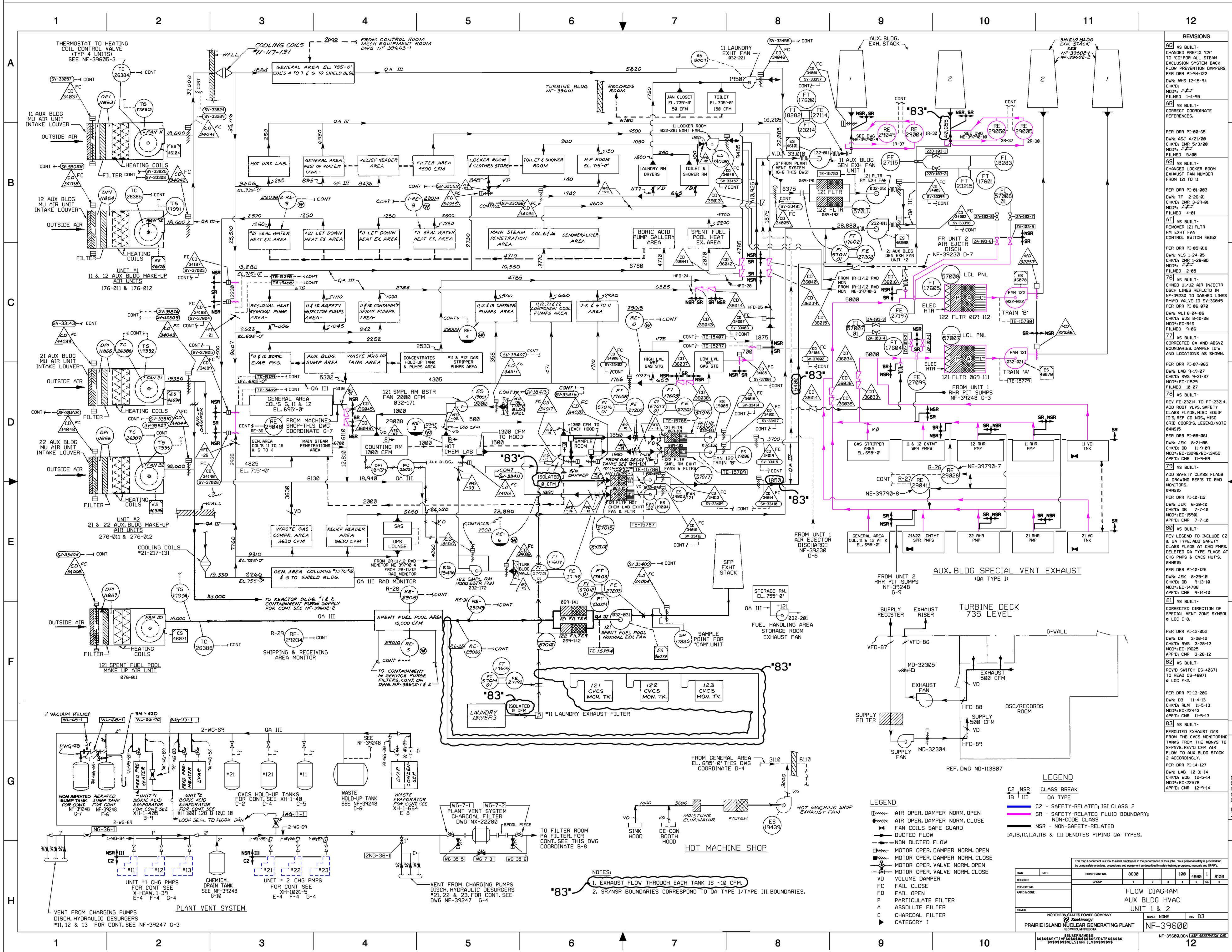


FIGURE 10.3-6 REV. 34

01516979

CAD FILE: U10306.DGN

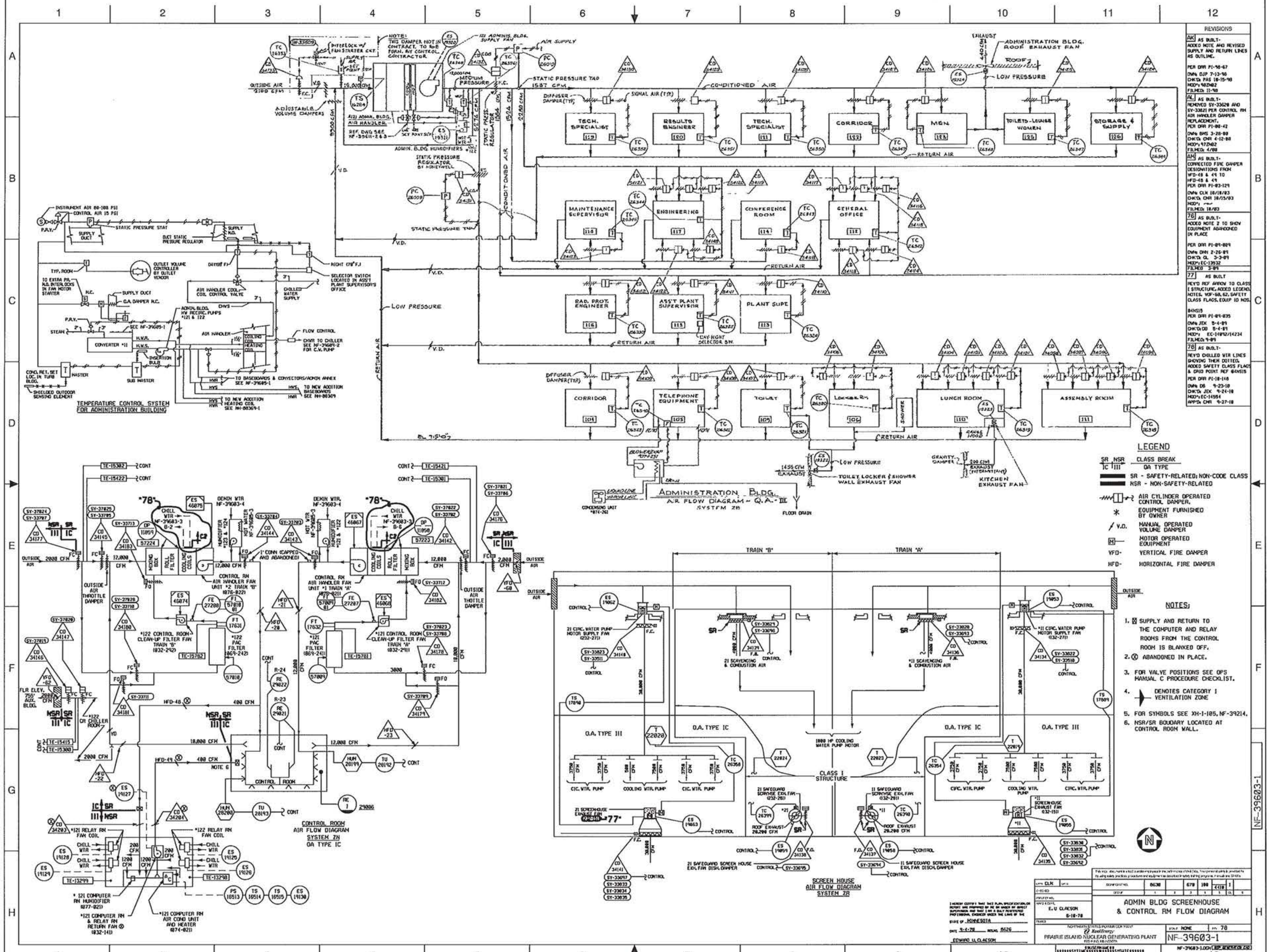


FIGURE 10.3-7 REV. 32

CAD FILE: U10307.DGN

01352784

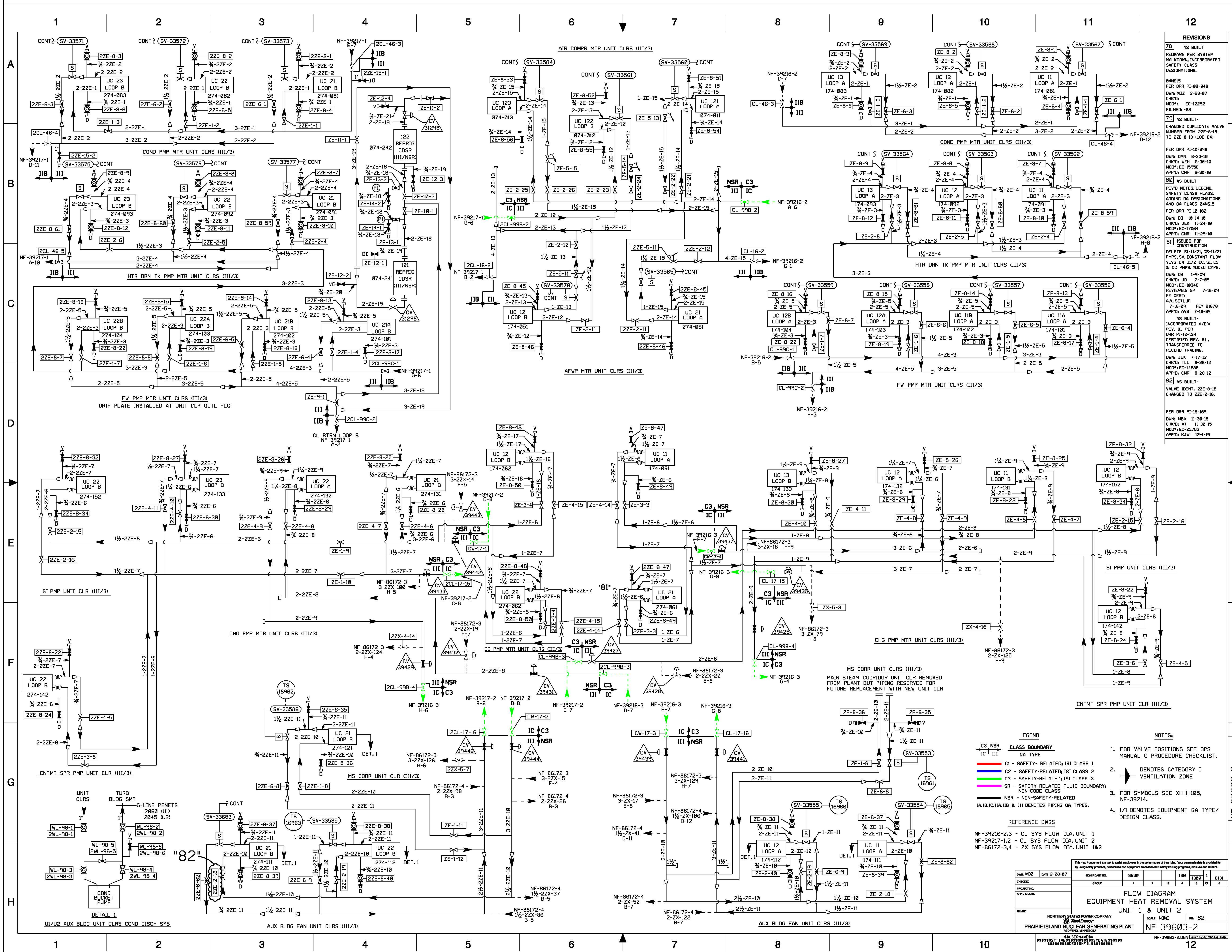


FIGURE 10.3-8 REV. 34

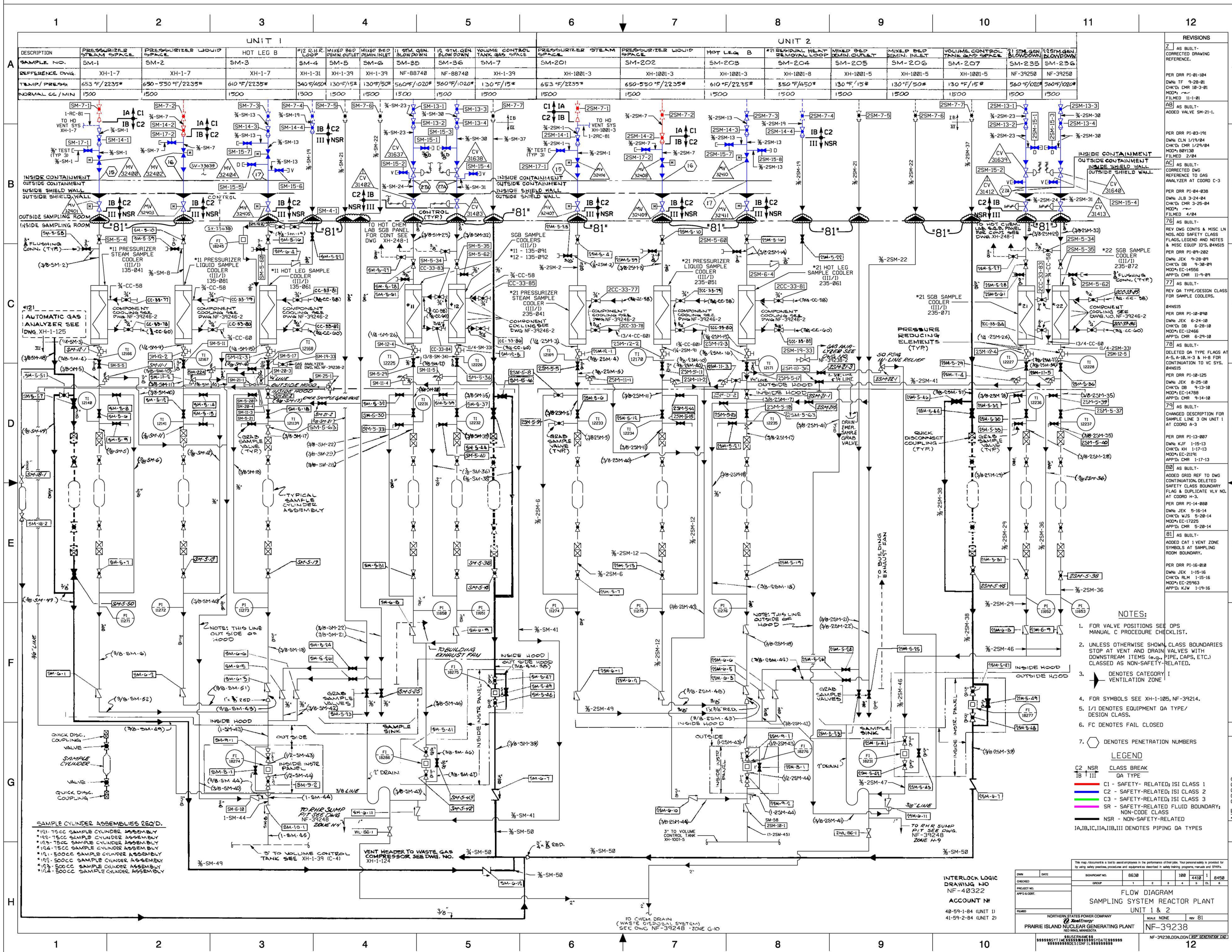
CAD FILE: U10308.DGN

01516979

FIGURE 10.3-9 REV. 34

01516979

CAD FILE: U10309.DGN



UNIT 1

DESCRIPTION	PRESSURIZER STEAM SPACE	PRESSURIZER LIQUID SPACE	HOT LEG B	#12 R.H.R. LOOP	MIXED BED DEMIN. OUTLET	MIXED BED DEMIN. INLET	11 STM. GEN. BLOWDOWN	12 STM. GEN. BLOWDOWN	VOLUME CONTROL TANK GAS SPACE
SAMPLE NO.	SM-1	SM-2	SM-3	SM-4	SM-5	SM-6	SM-35	SM-36	SM-7
REFFERENCE DWG.	XH-1-7	XH-1-7	XH-1-7	XH-1-31	XH-1-39	XH-1-39	NF-88740	NF-88740	XH-1-39
TEMP./PRESS.	653 °F/2235#	650-550 °F/2235#	610 °F/2235#	340 °F/450#	130 °F/15#	130 °F/50#	560 °F/1020#	560 °F/1020#	130 °F/15#
NORMAL CC/MIN	1500	1500	1500	1500	1500	1500	1500	1500	1500

UNIT 2

DESCRIPTION	PRESSURIZER STEAM SPACE	PRESSURIZER LIQUID SPACE	HOT LEG B	#21 RESIDUAL HEAT REMOVAL LOOP	MIXED BED DEMIN. OUTLET	MIXED BED DEMIN. INLET	VOLUME CONTROL TANK GAS SPACE	21 STM. GEN. BLOWDOWN	22 STM. GEN. BLOWDOWN
SAMPLE NO.	SM-201	SM-202	SM-203	SM-204	SM-205	SM-206	SM-207	SM-235	SM-236
REFFERENCE DWG.	XH-1001-3	XH-1001-3	XH-1001-3	XH-1001-8	XH-1001-5	XH-1001-5	XH-1001-5	NF-39250	NF-39250
TEMP./PRESS.	653 °F/2235#	650-550 °F/2235#	610 °F/2235#	350 °F/450#	130 °F/15#	130 °F/50#	130 °F/15#	560 °F/1020#	560 °F/1020#
NORMAL CC/MIN	1500	1500	1500	1500	1500	1500	1500	1500	1500

01516979

FIGURE 10.3-10 REV. 34

CAD FILE: U10310.DGN

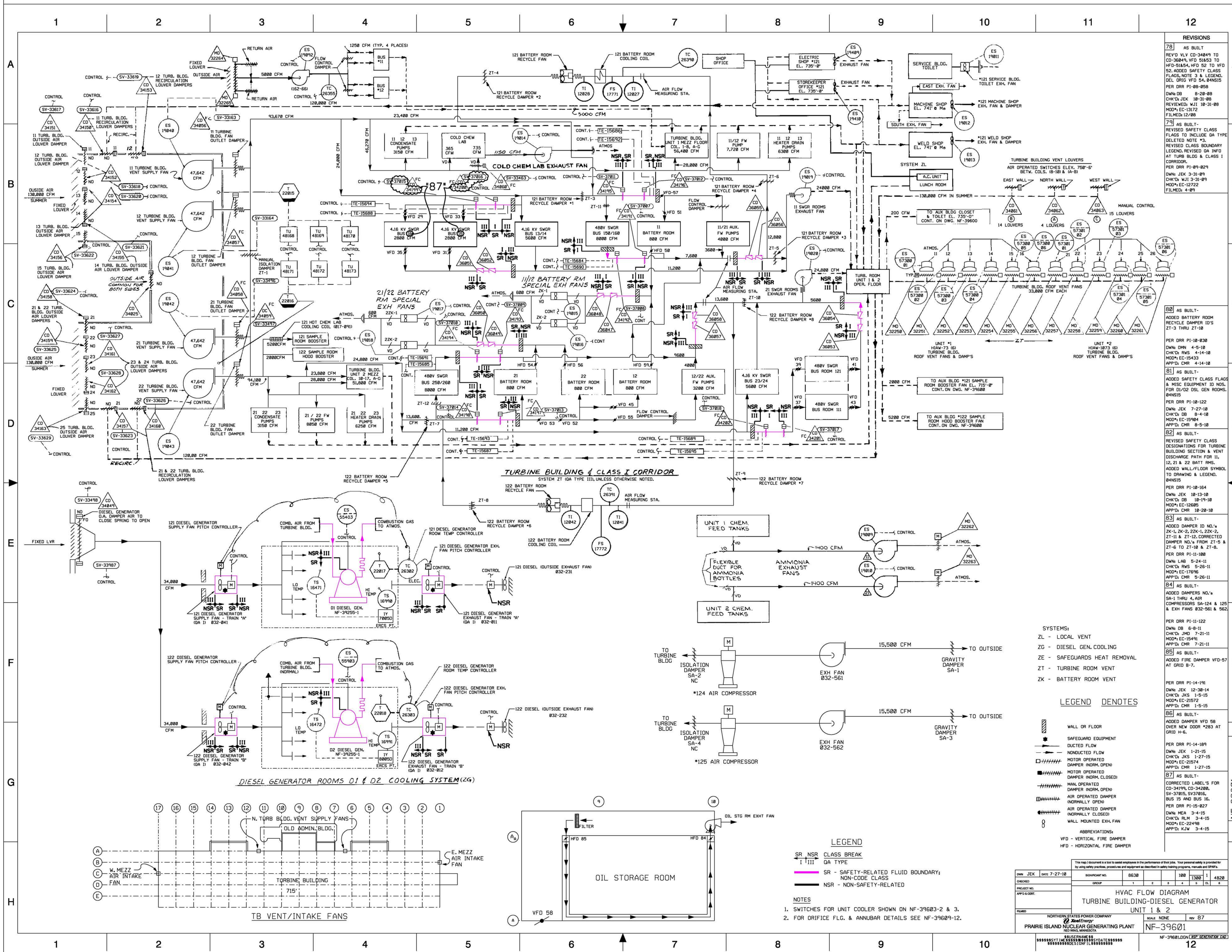


FIGURE 10.3-11 REV. 34

01516979

CAD FILE: U10311.DGN

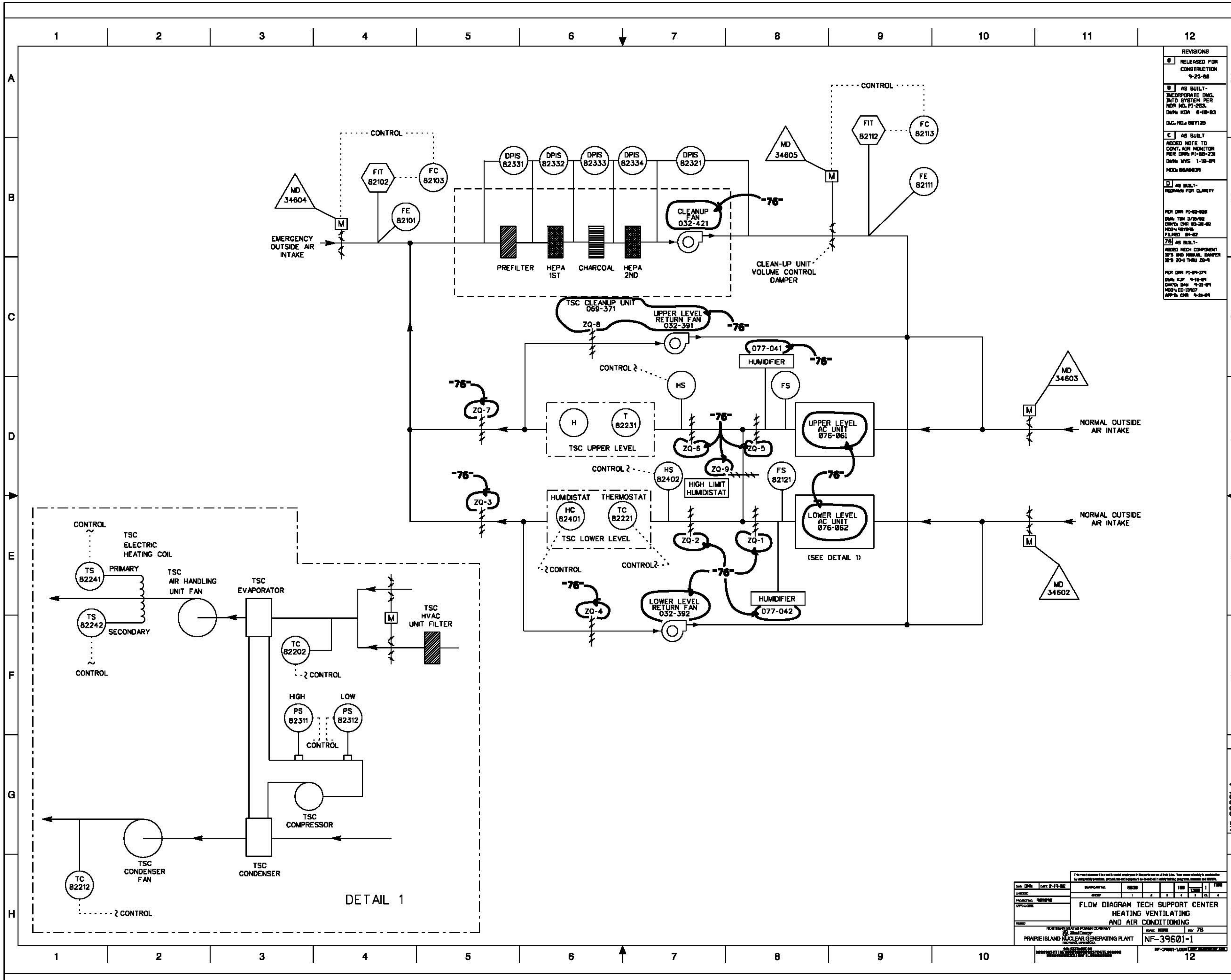


FIGURE 10.3-12 REV. 32

CAD FILE: U10312.DGN

01352784

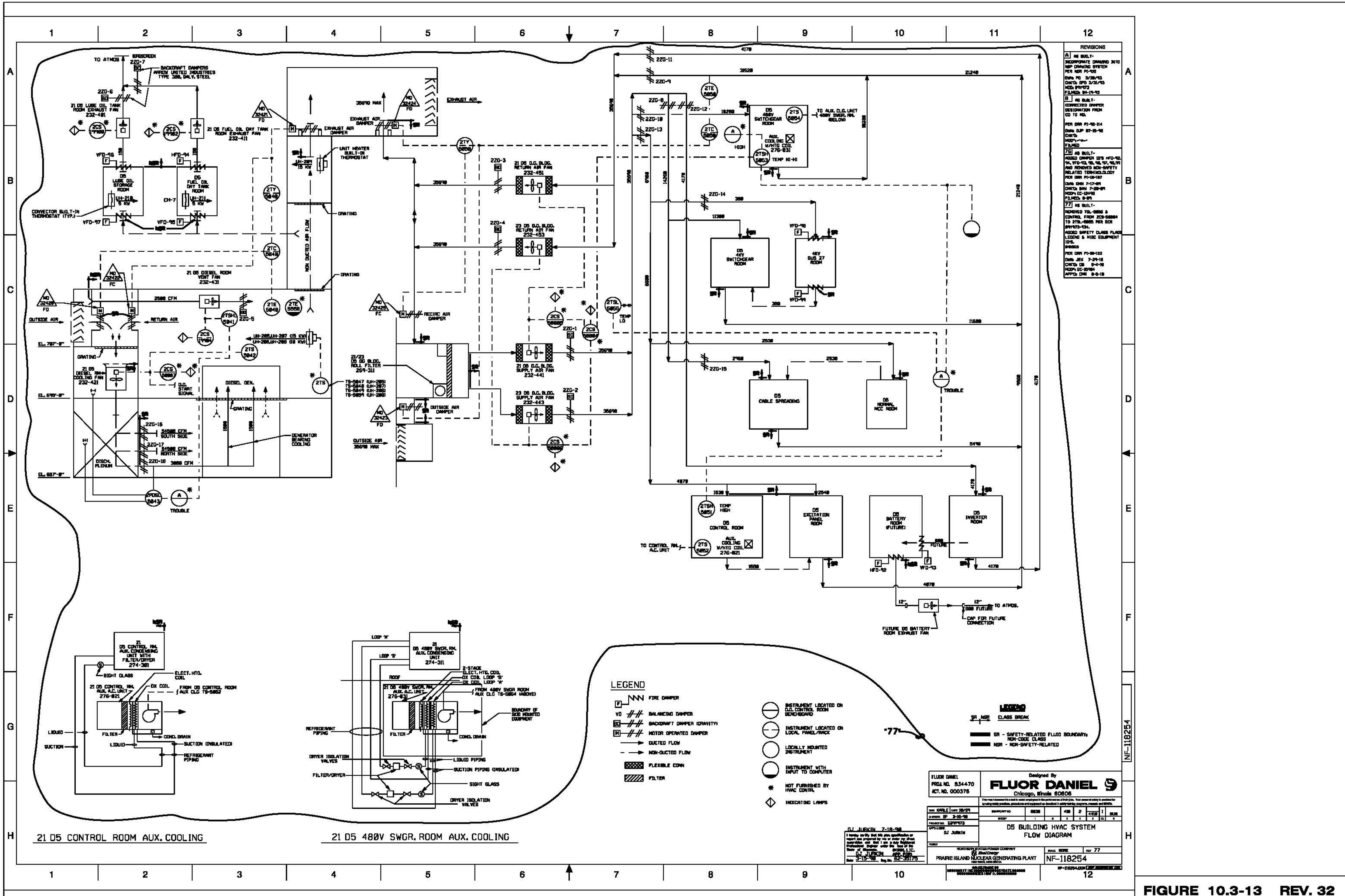


FIGURE 10.3-13 REV. 32

01352784

CAD FILE: U10313.DGN

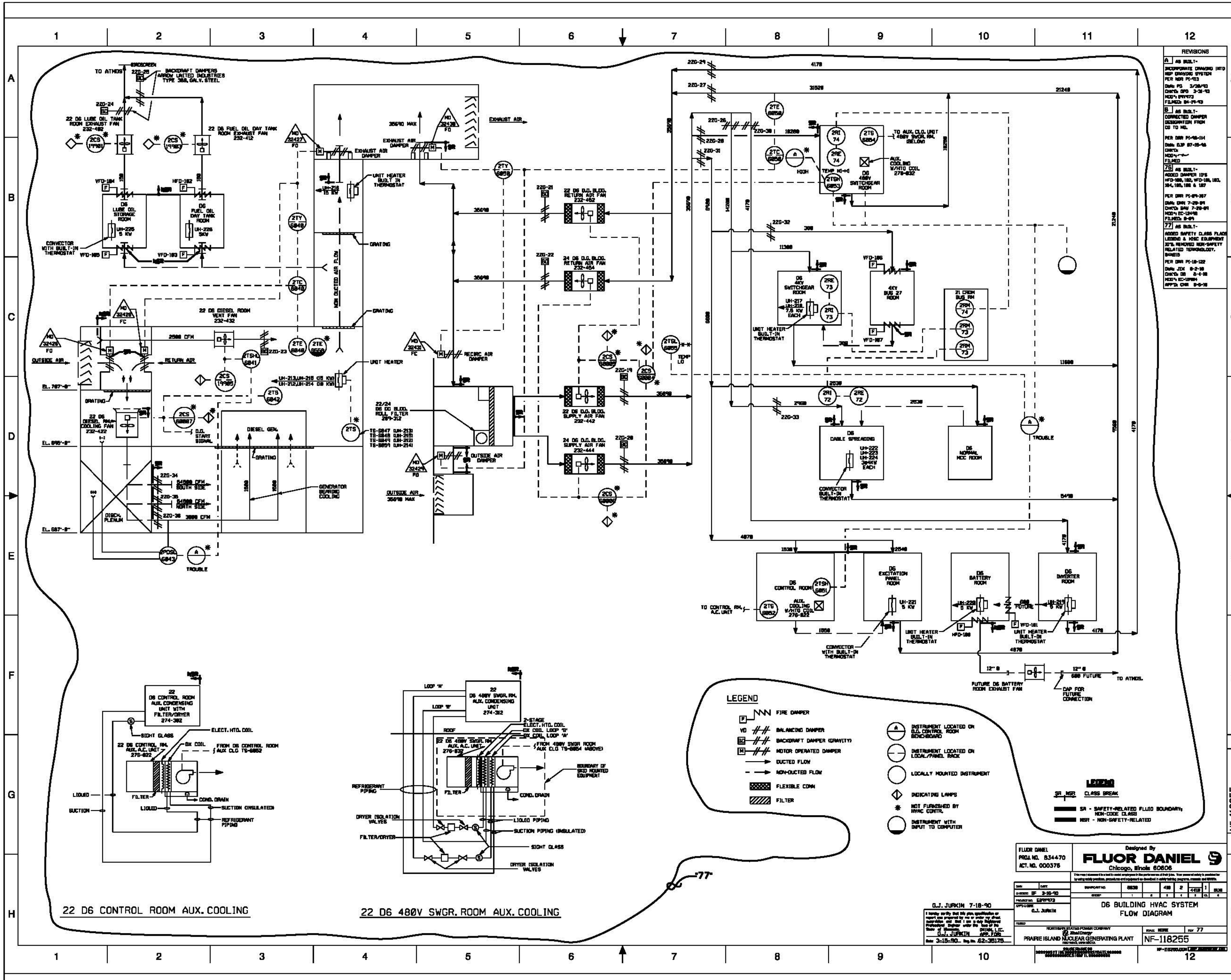
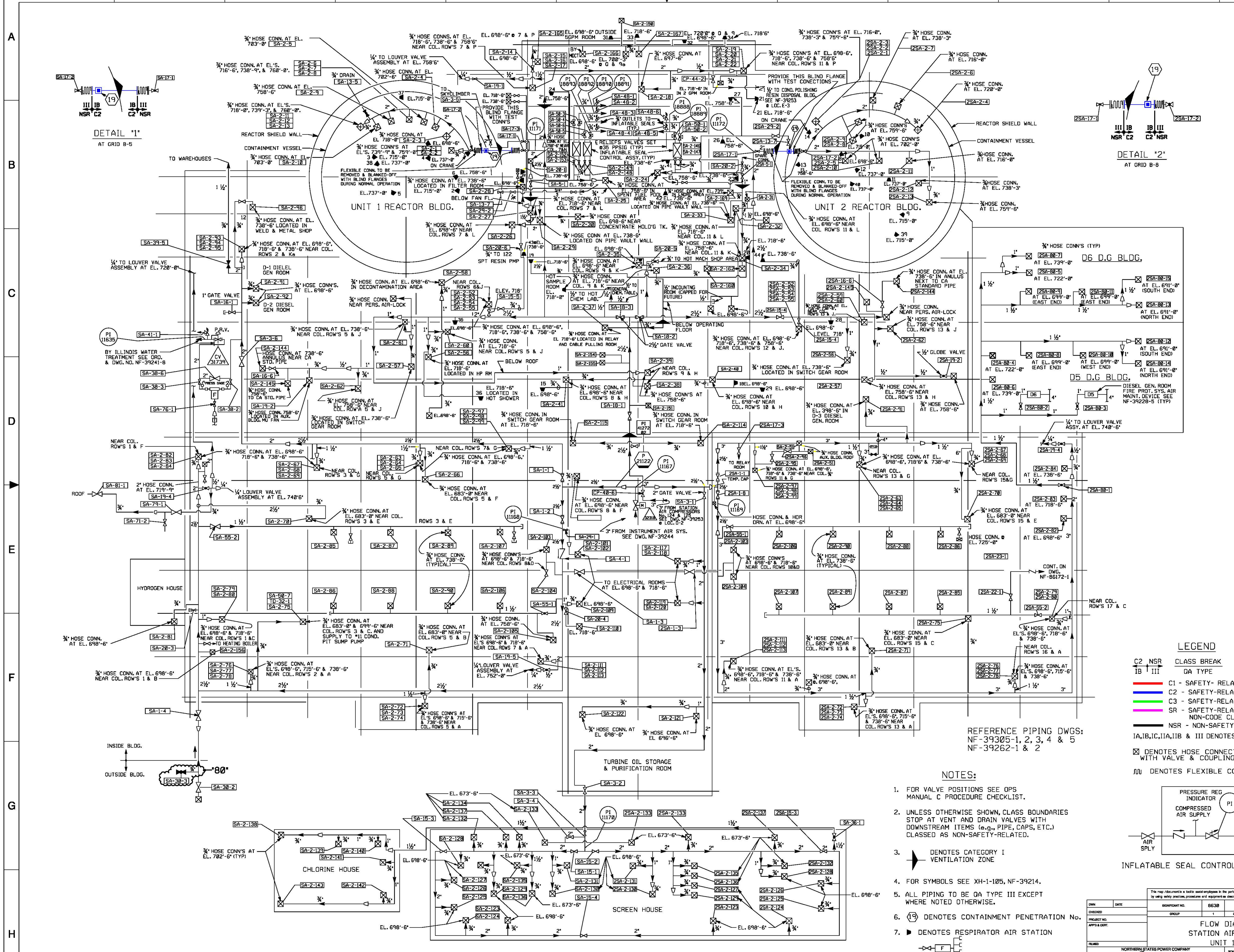


FIGURE 10.3-14 REV. 32

CAD FILE: U10314.DGN

01352784



REVISIONS

76 AS BUILT-
CHANGED DUPLICATE VALVE ID SA-80-1 TO SA-71-2 GRID E-2
PER DRR PI-08-046
DWN: DMN 7-29-08
CHK'D: TS 7-29-08
MOD#: EC-12264
FILMED: 8-08

77 AS BUILT-
REMOVED CONTINUATION OF LINE AFTER SA-16-1 GATE VALVE AND ADDED LINE CAP
PER DRR PI-09-167
DWN: KJF 9-21-09
CHK'D: BAW 9-21-09
MOD#: EC-13858
APP'D: CMR 9-21-09

78 AS BUILT-
ADD DETAILS 1 & 2, SAFETY CLASS FLAGS, NOTES & LEGEND.
04NS15
PER DRR PI-09-233
DWN: JEK 11-16-09
CHK'D: DB 11-23-09
MOD#: EC-14629
APP'D: CMR 11-23-09

79 AS BUILT-
CORRECTED VALVE ID FROM 25A-55-2 TO 2SA-55-2 AT DRAWING COORDINATE E-10.
PER DRR PI-11-003
DWN: KJF 1-11-11
CHK'D: BWK 1-13-11
MOD#: EC-17078
APP'D: CMR 1-14-11

80 AS BUILT-
ADDED VALVE SA-30-3 @ GRID LOC G-2
PER DRR PI-15-211
DWN: DB 12-22-15
CHK'D: RLM 1-5-16
MOD#: EC-21830
APP'D: KJW 1-6-16

REFERENCE PIPING DWGS:
NF-39305-1, 2, 3, 4 & 5
NF-39262-1 & 2

LEGEND

C2 NSR CLASS BREAK
IB III QA TYPE

C1 - SAFETY- RELATED; ISI CLASS 1
C2 - SAFETY-RELATED; ISI CLASS 2
C3 - SAFETY-RELATED; ISI CLASS 3
SR - SAFETY-RELATED FLUID BOUNDARY; NON-CODE CLASS
NSR - NON-SAFETY-RELATED

IA,IB,IC,IIA,IIB & III DENOTES PIPING QA TYPES.

⊠ DENOTES HOSE CONNECTION WITH VALVE & COUPLING.

DENOTES FLEXIBLE CONNECTION

PRESSURE REG INDICATOR
PI
COMPRESSED AIR SUPPLY
AIR SPLY

INFLATABLE SEAL CONTROL ASSY (TYP)

NOTES:

1. FOR VALVE POSITIONS SEE OPS MANUAL C PROCEDURE CHECKLIST.
2. UNLESS OTHERWISE SHOWN, CLASS BOUNDARIES STOP AT VENT AND DRAIN VALVES WITH DOWNSTREAM ITEMS (e.g., PIPE, CAPS, ETC.) CLASSED AS NON-SAFETY-RELATED.
3. DENOTES CATEGORY I VENTILATION ZONE
4. FOR SYMBOLS SEE XH-1-105, NF-39214.
5. ALL PIPING TO BE QA TYPE III EXCEPT WHERE NOTED OTHERWISE.
6. (19) DENOTES CONTAINMENT PENETRATION No.
7. ▶ DENOTES RESPIRATOR AIR STATION

DWN	DATE	SIGNIFICANT NO.	8630			100	4410	1	
CHECKED		GROUP	1	2	3	4	5	CL	6
PROJECT NO.									
APP'D & CERT.									
FILMED									

FLOW DIAGRAM
STATION AIR SYSTEM
UNIT 1 & 2

NORTHERN STATES POWER COMPANY
Xcel Energy
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RED WING, MINNESOTA

SCALE NONE REV 80
NF-39243

$$USERNAME$$
$$$$$$SYTIME$$$$$$$$$$$$$SYDATE$$$$$$
$$$$$$$$$DESIGNFIL$$$$$$$$$

NF-39243.DGN NSP GENERATION CAD

NF-39243

01516979

FIGURE 10.3-15 REV. 34

CAD FILE: U10315.DGN

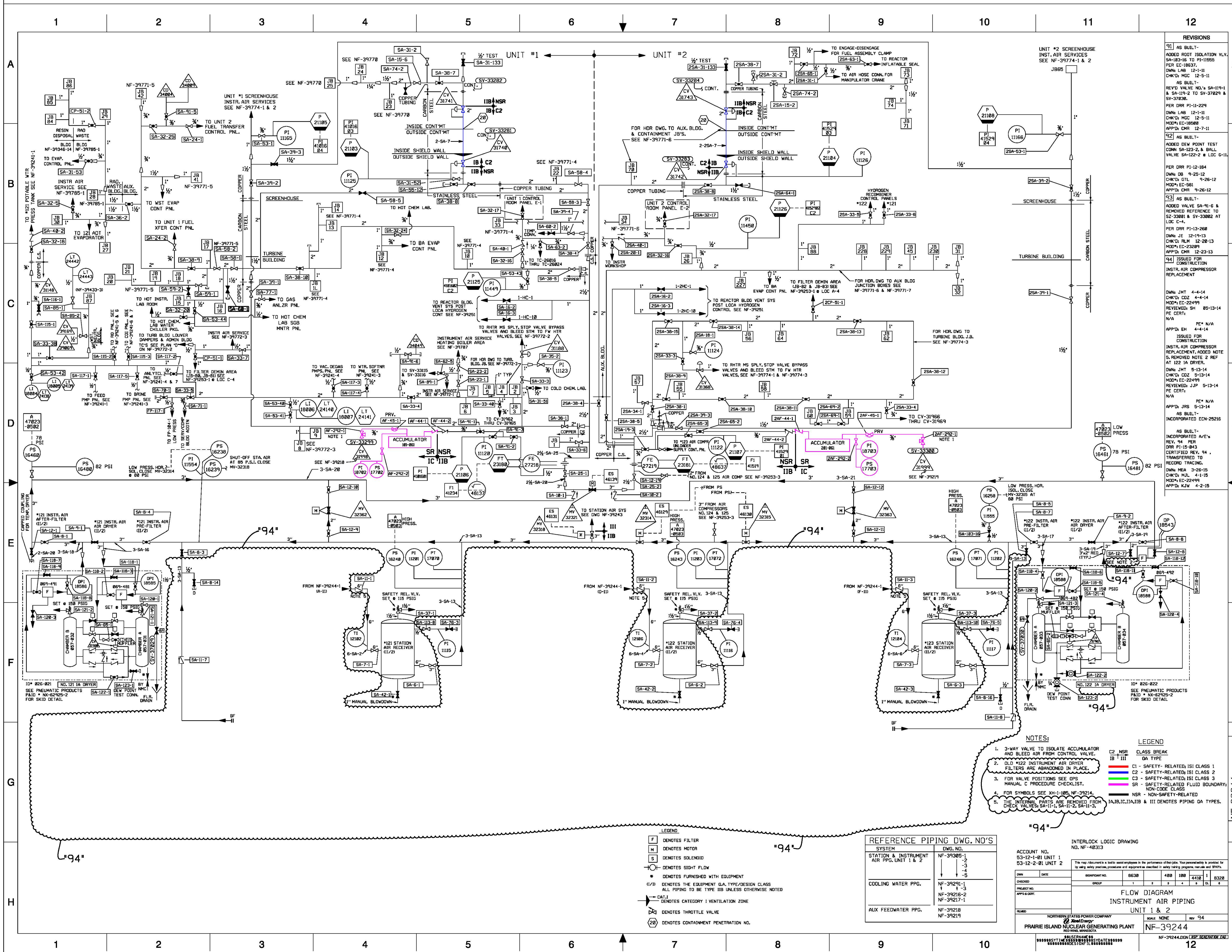


FIGURE 10.3-16 REV. 34

CAD FILE: U10316.DGN

01516979

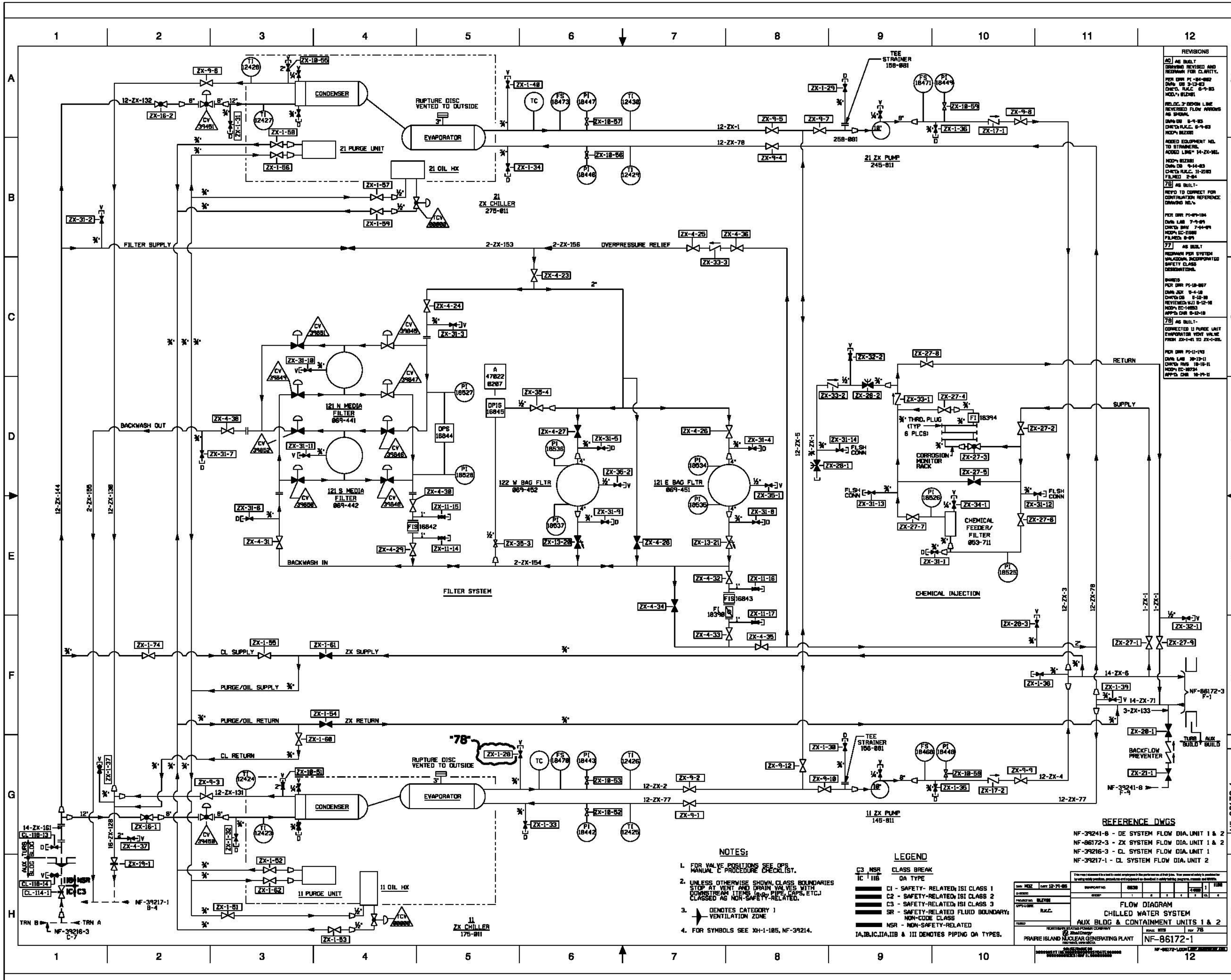


FIGURE 10.3-17 REV. 32

01352784

CAD FILE: U10317.DGN

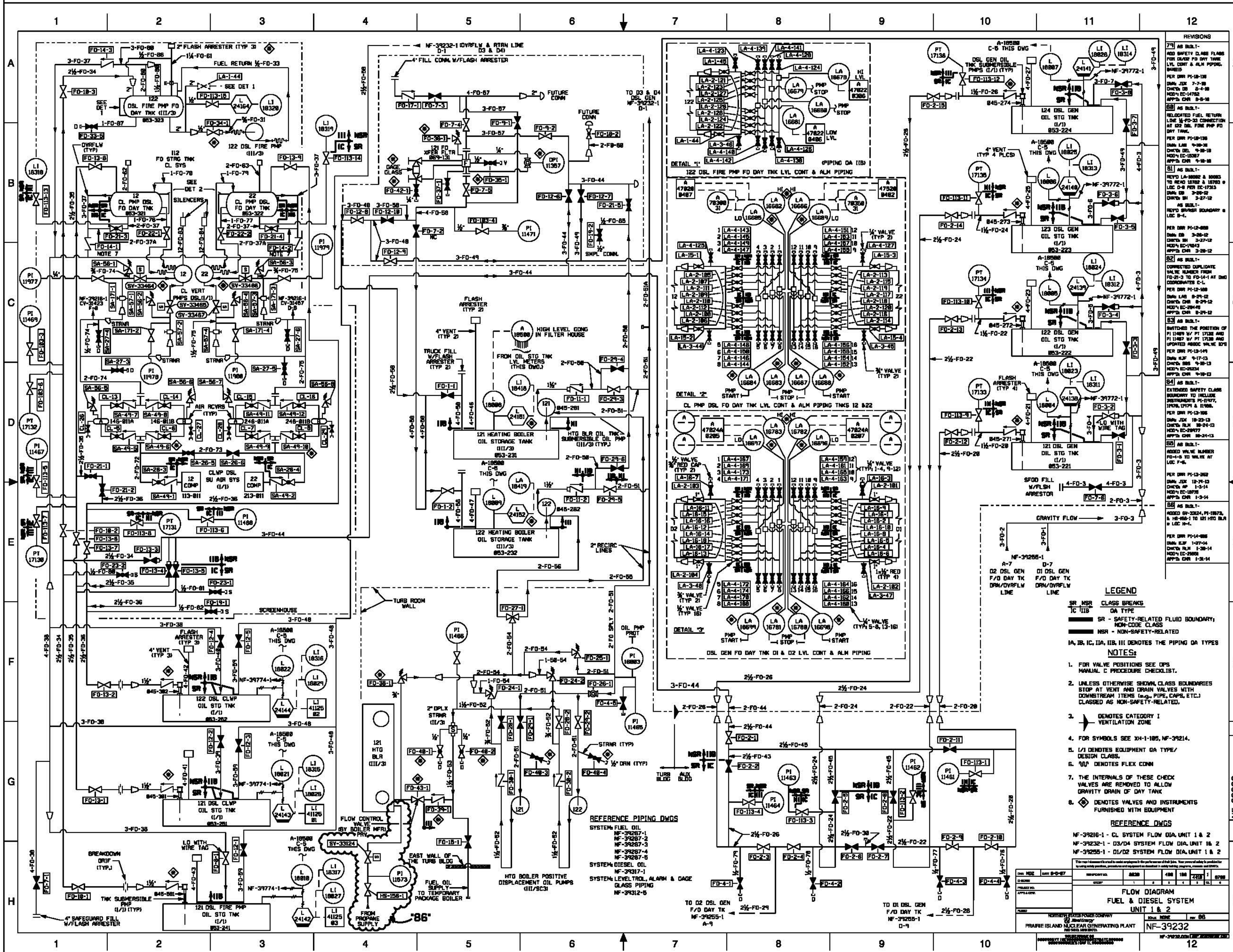
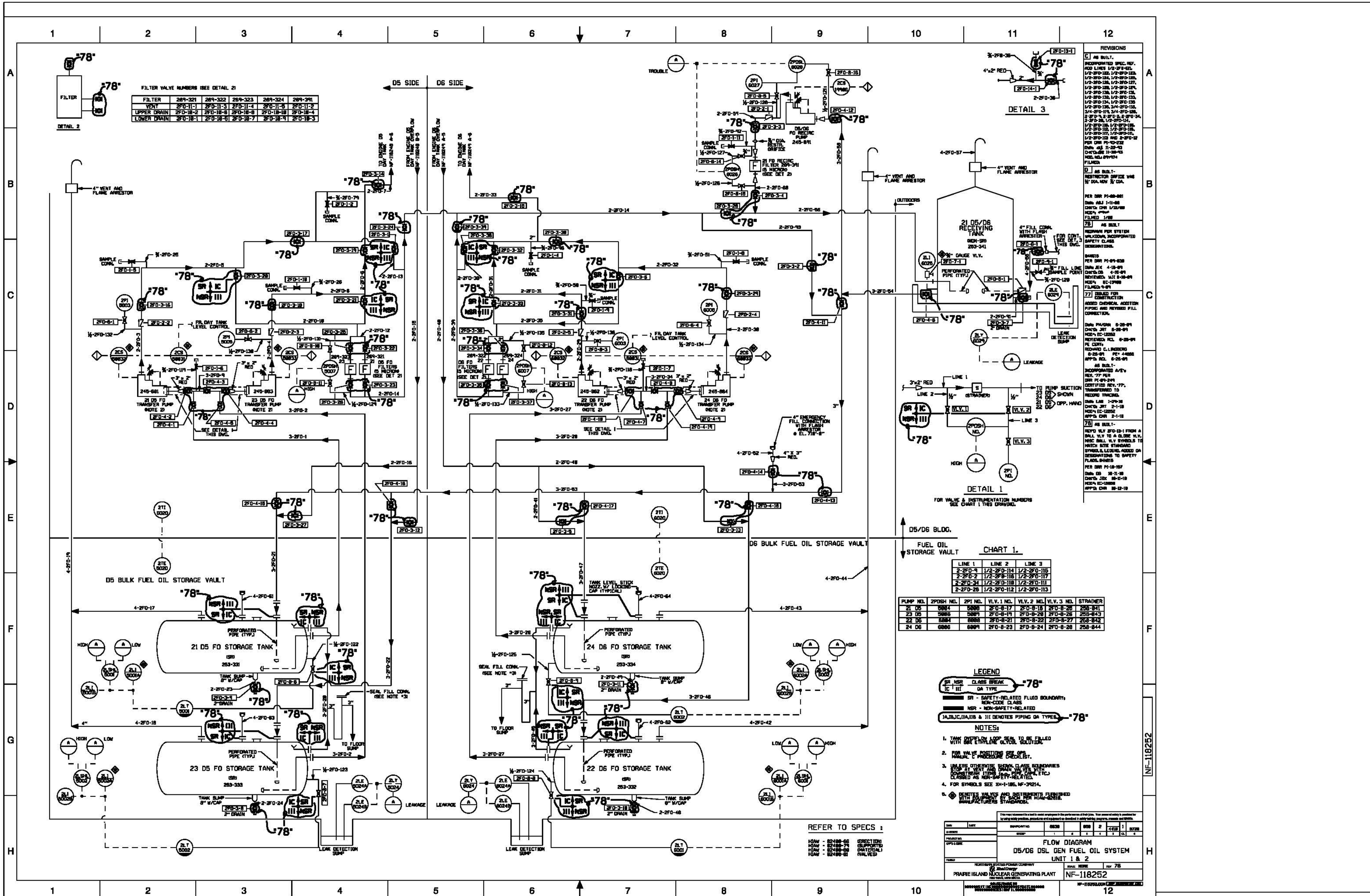


FIGURE 10.3-18 REV. 33

CAD FILE: U10318.DGN

01429088



FILTER VALVE NUMBERS (SEE DETAIL 2)

FILTER	269-321	269-322	269-323	269-324	269-391
VENT	2FO-11-1	2FO-11-3	2FO-11-4	2FO-11-5	2FO-11-2
UPPER DRAIN	2FO-10-2	2FO-10-6	2FO-10-8	2FO-10-10	2FO-10-4
LOWER DRAIN	2FO-10-1	2FO-10-5	2FO-10-7	2FO-10-9	2FO-10-3

CHART 1.

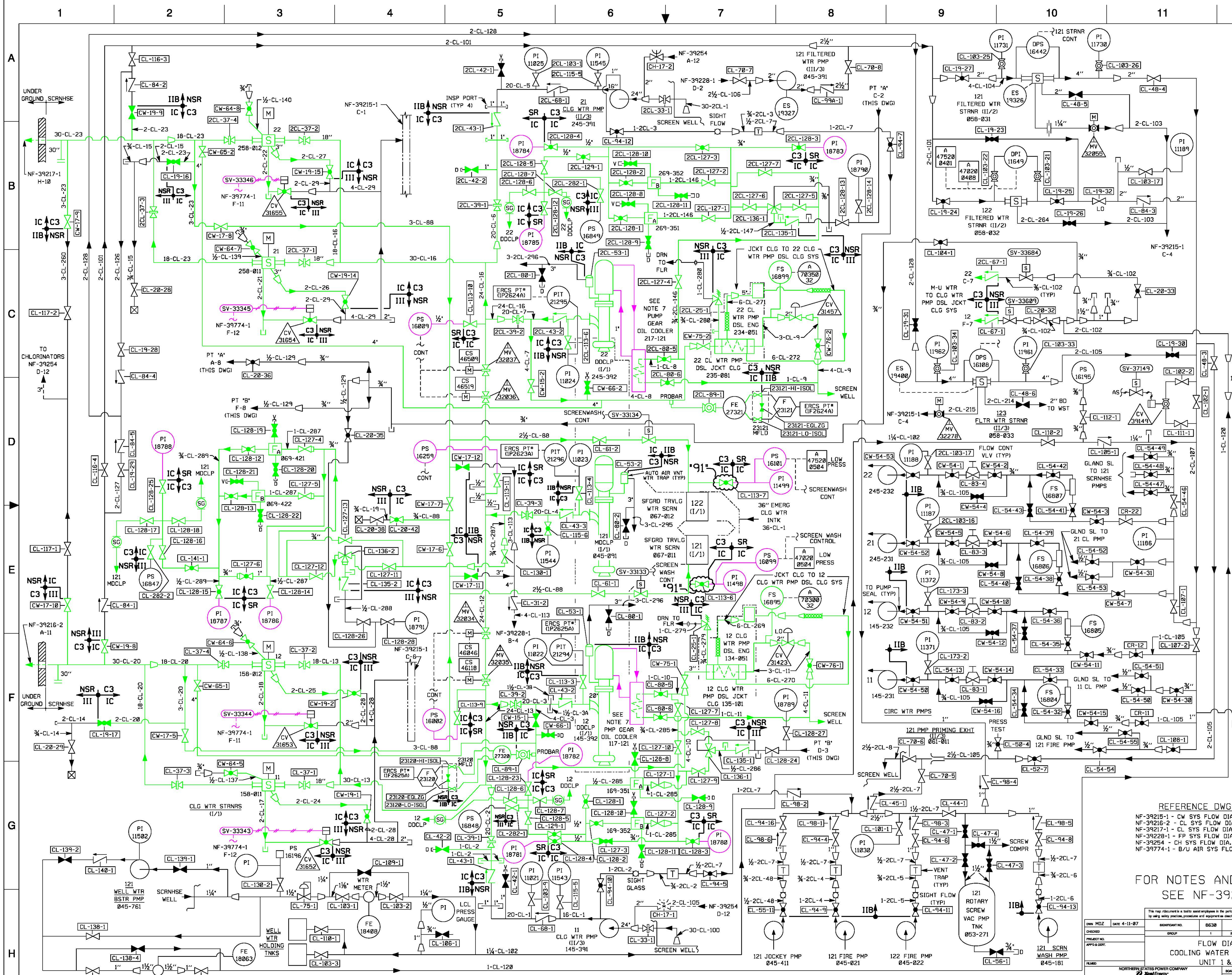
LINE 1	LINE 2	LINE 3
2-2FO-9	1/2-2FO-114	1/2-2FO-115
2-2FO-2	1/2-2FO-116	1/2-2FO-117
2-2FO-34	1/2-2FO-110	1/2-2FO-111
2-2FO-28	1/2-2FO-112	1/2-2FO-113

PUMP NO.	2PDSH NO.	2PI NO.	VLV. 1 NO.	VLV. 2 NO.	VLV. 3 NO.	STRAINER
21 D5	5004	5008	2FO-8-17	2FO-8-18	2FO-8-25	258-841
23 D5	5006	5009	2FO-8-19	2FO-8-20	2FO-8-26	258-843
22 D6	6004	6008	2FO-8-21	2FO-8-22	2FO-8-27	258-842
24 D6	6006	6009	2FO-8-23	2FO-8-24	2FO-8-28	258-844

FIGURE 10.3-19 REV. 32

01352784

CAD FILE: U10319.DGN



REVISIONS

83 AS BUILT-
CORRECTED VALVE NUMBER FROM CL-54-31 TO CW-54-31.
PER DRR PI-11-244
DWN: JEK 12-13-11
CHK'D: MGC 1-3-12
MOD#: EC-18785
APP'D: CMR 1-11-12

84 AS BUILT-
CHANGED FE-18202 TO FE-18063 AT WTR METER FROM 121 WELL WTR BSTR PMP.
PER DRR PI-12-007
DWN: JEK 2-2-12
CHK'D: SK 4-7-12
MOD#: EC-19422
APP'D: CMR 4-9-12

85 AS BUILT-
REMOVED STUB PIPE, LINES 1 1/2-CL-7A & 1 1/2-CL-7B.
PER DRR PI-12-239
DWN: LAB 12-14-12
CHK'D: KH 12-15-12
MOD#: EC-21095
APP'D: CMR 12-17-12

86 AS BUILT-
REMOVED LINE 1 1/2-CL-13 DNSTRM OF MV-32035 AT COORD F-5
PER DRR PI-13-009
DWN: KJF 1-15-13
CHK'D: KH 1-17-13
MOD#: EC-21134
APP'D: CMR 1-17-13

87 AS BUILT-
REPLACED FLOW SWITCHES 16847 & 16864, 16848 & 16862, 16849 & 16863 WITH SIGHT FLOW INDICATORS & PRESSURE SWITCHES 16847, 16848 AND 16849. REV'D VALVE CW-15-1 TO NC.
PER DRR PI-14-009
DWN: LAB 9-30-14
CHK'D: GTL 10-6-14
MOD#: EC-21558
APP'D: CMR 10-6-14

88 AS BUILT-
ADDED LINE NO'S & SAFETY CLASSIFICATION TO AUTO AIR VENT WATER TRAP PIPING AS SHOWN.
PER DRR PI-14-164
DWN: LAB 11-18-14
CHK'D: RSA 12-15-14
MOD#: EC-24180
APP'D: CMR 12-24-14

89 AS BUILT-
ADDED PIPING CLASSIFICATION FROM VALVE TO PRESS. SENSOR.
PER DRR PI-15-102
DWN: MEA 8-3-15
CHK'D: RSA 8-4-15
MOD#: EC-25706
APP'D: KJW 8-5-15

90 AS BUILT-
CORRECTED CW-66-1 FROM VENT TO DRAIN. SHOW PIPE FROM CL-127-7 CONNECTED TO RIGHT OF CL-80-5. SHOW 4" LINE FROM CW-65-1 TO CW-15-1 NOT CONNECTED TO 24-CL-12.
PER DRR PI-16-002
DWN: JEK 1-15-16
CHK'D: RLM 1-15-16
MOD#: EC-21242
APP'D: KJW 1-19-16

91 AS BUILT-
CHANGED VALVE SYMBOL FOR CL-113-6 & CL-113-7 FROM GATE VALVE TO GLOBE VALVE.
PER DRR PI-16-022
DWN: JEK 1-27-16
CHK'D: RLM 1-27-16
MOD#: EC-25161
APP'D: KJW 1-27-16

REFERENCE DWGS

NF-39215-1 - CW SYS FLOW DIA. UNIT 1 & 2
NF-39216-2 - CL SYS FLOW DIA. UNIT 1
NF-39217-1 - CL SYS FLOW DIA. UNIT 2
NF-39228-1 - FP SYS FLOW DIA. UNIT 1 & 2
NF-39254 - CH SYS FLOW DIA. UNIT 1 & 2
NF-39774-1 - B/U AIR SYS FLOW DIA. UNIT 1 & 2

FOR NOTES AND LEGEND SEE NF-39216-2

This map /document is a tool to assist employees in the performance of their jobs. Your personal safety is provided for by using safety practices, procedures and equipment as described in safety training programs, manuals and SPAR's.

SIGNIFICANT NO.	8630			1/2	4410	1	8548
GROUP	1	2	3	4	5	CL	6

DWN MDZ DATE 4-11-07
CHECKED
PROJECT NO.
APP'D & CERT.
FILMED

FLOW DIAGRAM
COOLING WATER - SCREENHOUSE
UNIT 1 & UNIT 2

NORTHERN STATES POWER COMPANY
Xcel Energy
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RED WING, MINNESOTA

SCALE NONE REV 91
NF-39216-1

$$USERNAME$$
$$$$$$SYTIME$$$$$$$$$$$$SYDATE$$$$$$
$$$$$$$$$DESIGNFIL$$$$$$$$$

NF-39216-1.DGN NSP GENERATION CAD

NF-39216-1

01516979

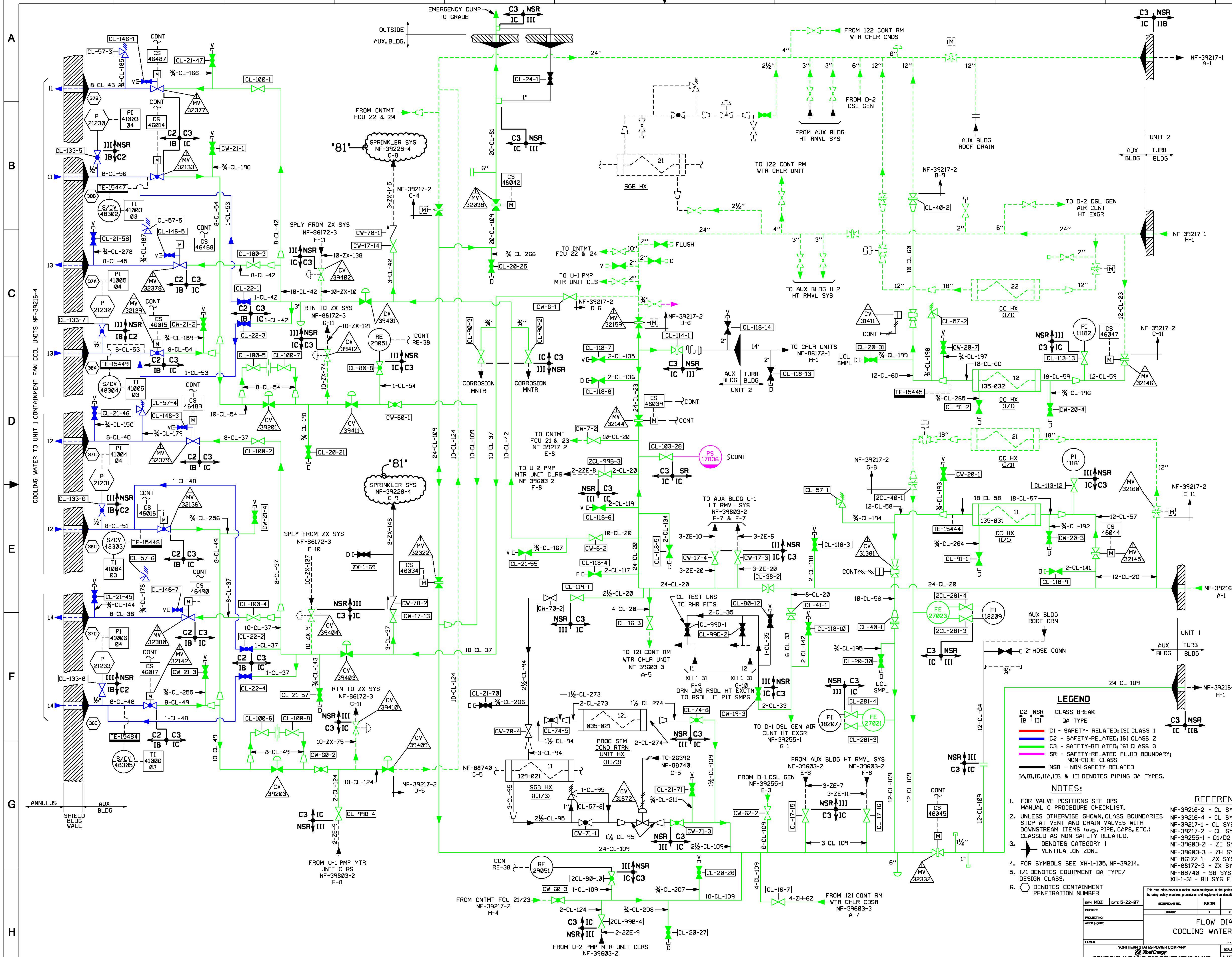
FIGURE 10.4-1A REV. 34

CAD FILE: U10401A.DGN

FIGURE 10.4-1B REV. 34

01516979

CAD FILE: U10401B.DGN



REVISIONS

76 AS BUILT-
ADDED VENT VALVES CL-146-1, CL-146-3, CL.146-5, CL-146-7.
PER DRR PI-09-208
DWN: DMN 10-27-09
CHK'D: BWK 10-28-09
MOD#: EC-691
APP'D: CMR 10-28-09

77 AS BUILT
REDRAWN PER SYSTEM WALKDOWN, INCORPORATED SAFETY CLASS DESIGNATIONS.
04NS15
PER DRR PI-10-048
DWN: DB 4-13-10
CHK'D: JEK 5-11-10
REVIEWED: WJI 5-11-10
MOD#: EC-14552
APP'D: CMR 5-11-10

78 AS BUILT-
ADDED DRAIN LINE & DWG. CONT. & COORDINATES FOR 121 AIR HANDLER EQUIPMENT.
PER DRR PI-09-163
DWN: KJF 8-16-10
CHK'D: JBS 10-12-10
MOD#: EC-14472
APP'D: CMR 10-18-10

79 AS BUILT-
CORRECTED VALVE ID FROM CW-40-1 TO CL-40-1.
PER DRR PI-12-134
DWN: JEK 6-29-12
CHK'D: SBS 7-11-12
MOD#: EC-18147
APP'D: CMR 7-16-12

80 AS BUILT-
REVISED DWG TO REFLECT CONT RM AIR HANDLER DRAIN PATH REMOVAL.
PER DRR PI-13-236
DWN: KJF 12-10-13
CHK'D: CMR 12-11-13
MOD#: EC-22967
APP'D: CMR 12-11-13

81 AS BUILT-
ADDED DWG CONT REF @ LOC E-4, B-4.
PER DRR PI-14-200
DWN: DB 12-18-14
CHK'D: KB 1-6-15
MOD#: EC-23171
APP'D: CMR 1-6-15

LEGEND

C2 NSR / IB III — CLASS BREAK / QA TYPE

- C1 - SAFETY- RELATED; ISI CLASS 1
- C2 - SAFETY-RELATED; ISI CLASS 2
- C3 - SAFETY-RELATED; ISI CLASS 3
- SR - SAFETY-RELATED FLUID BOUNDARY; NON-CODE CLASS
- NSR - NON-SAFETY-RELATED

IA,IB,IC,IIA,IIB & III DENOTES PIPING QA TYPES.

NOTES:

1. FOR VALVE POSITIONS SEE OPS MANUAL C PROCEDURE CHECKLIST.
2. UNLESS OTHERWISE SHOWN, CLASS BOUNDARIES STOP AT VENT AND DRAIN VALVES WITH DOWNSTREAM ITEMS (e.g., PIPE, CAPS, ETC.) CLASSED AS NON-SAFETY-RELATED.
3. DENOTES CATEGORY I VENTILATION ZONE
4. FOR SYMBOLS SEE XH-1-105, NF-39214.
5. I/1 DENOTES EQUIPMENT QA TYPE/ DESIGN CLASS.
6. DENOTES CONTAINMENT PENETRATION NUMBER

REFERENCE DWGS

NF-39216-2 - CL SYS FLOW DIA. UNIT 1
NF-39216-4 - CL SYS FLOW DIA. UNIT 1
NF-39217-1 - CL SYS FLOW DIA. UNIT 2
NF-39217-2 - CL SYS FLOW DIA. UNIT 2
NF-39255-1 - D1/D2 SYS FLOW DIA. UNIT 1/2
NF-39603-2 - ZE SYS FLOW DIA. UNIT 1/2
NF-39603-3 - ZH SYS FLOW DIA. UNIT 1/2
NF-86172-1 - ZX SYS FLOW DIA. UNIT 2
NF-86172-3 - ZX SYS FLOW DIA. UNIT 1/2
NF-88740 - SB SYS FLOW DIA. UNIT 1
XH-1-31 - RH SYS FLOW DIA. UNIT 1

DWN MDZ	DATE 5-22-07	SIGNIFICANT NO.	8630			1	4410	1	8540
CHECKED		GROUP	1	2	3	4	5	CL	6
PROJECT NO.									
APP'D & CERT.									
FILMED									

FLOW DIAGRAM
COOLING WATER - AUX BLDG
UNIT 1

NORTHERN STATES POWER COMPANY
Xcel Energy
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RED WING, MINNESOTA

SCALE NONE — REV 81

NF–NF-39216-3

$$USERNAME$$
$$$$$$SYTIME$$$$$$$$$$$$SYDATE$$$$$$
$$$$$$$$$DESIGNFIL$$$$$$$$$

NF-39216-3.DGN NSP GENERATION CAD

NF–NF-39216-3

01516979

FIGURE 10.4-1C REV. 34

CAD FILE: U10401C.DGN

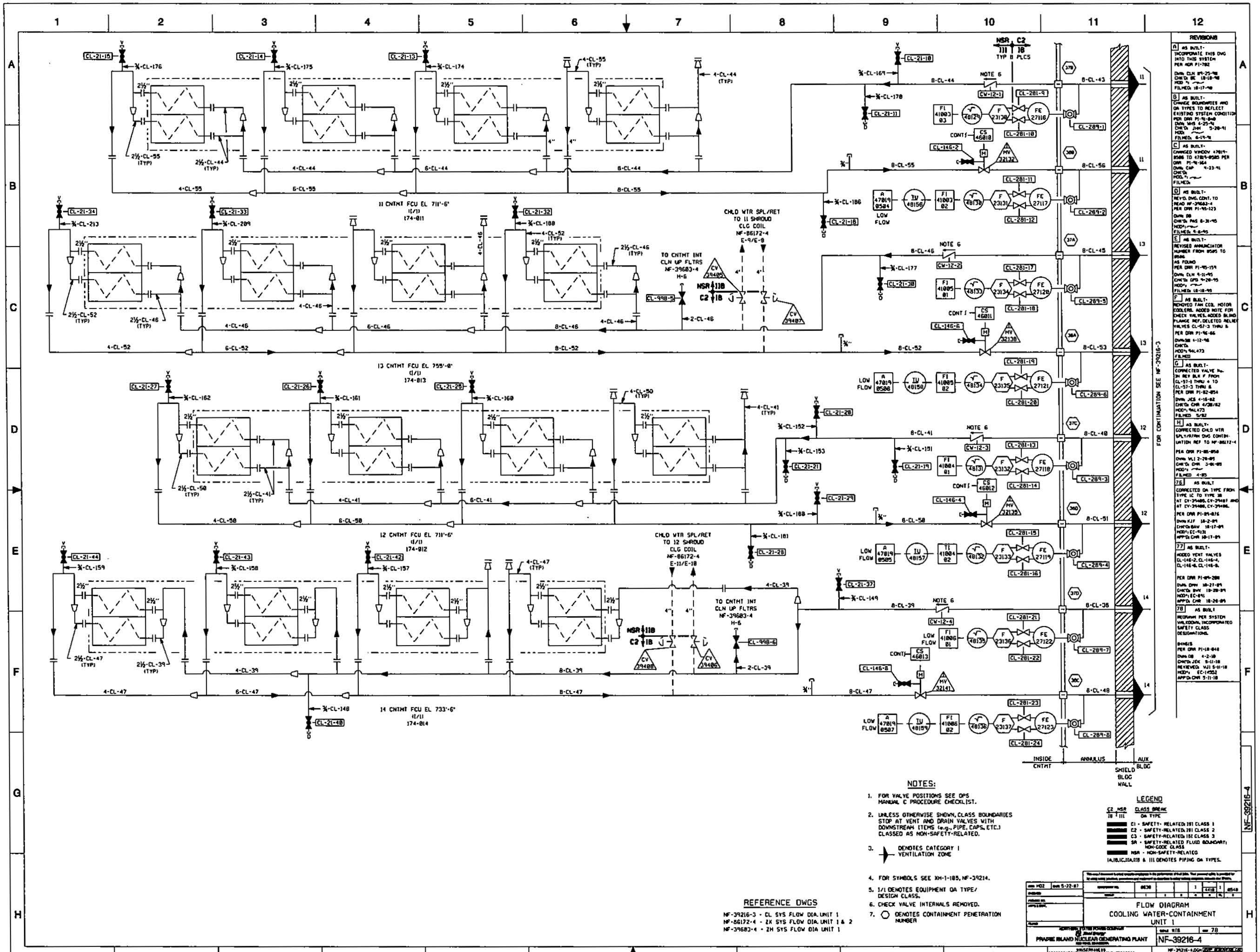


FIGURE 10.4-1D REV.31

01248495

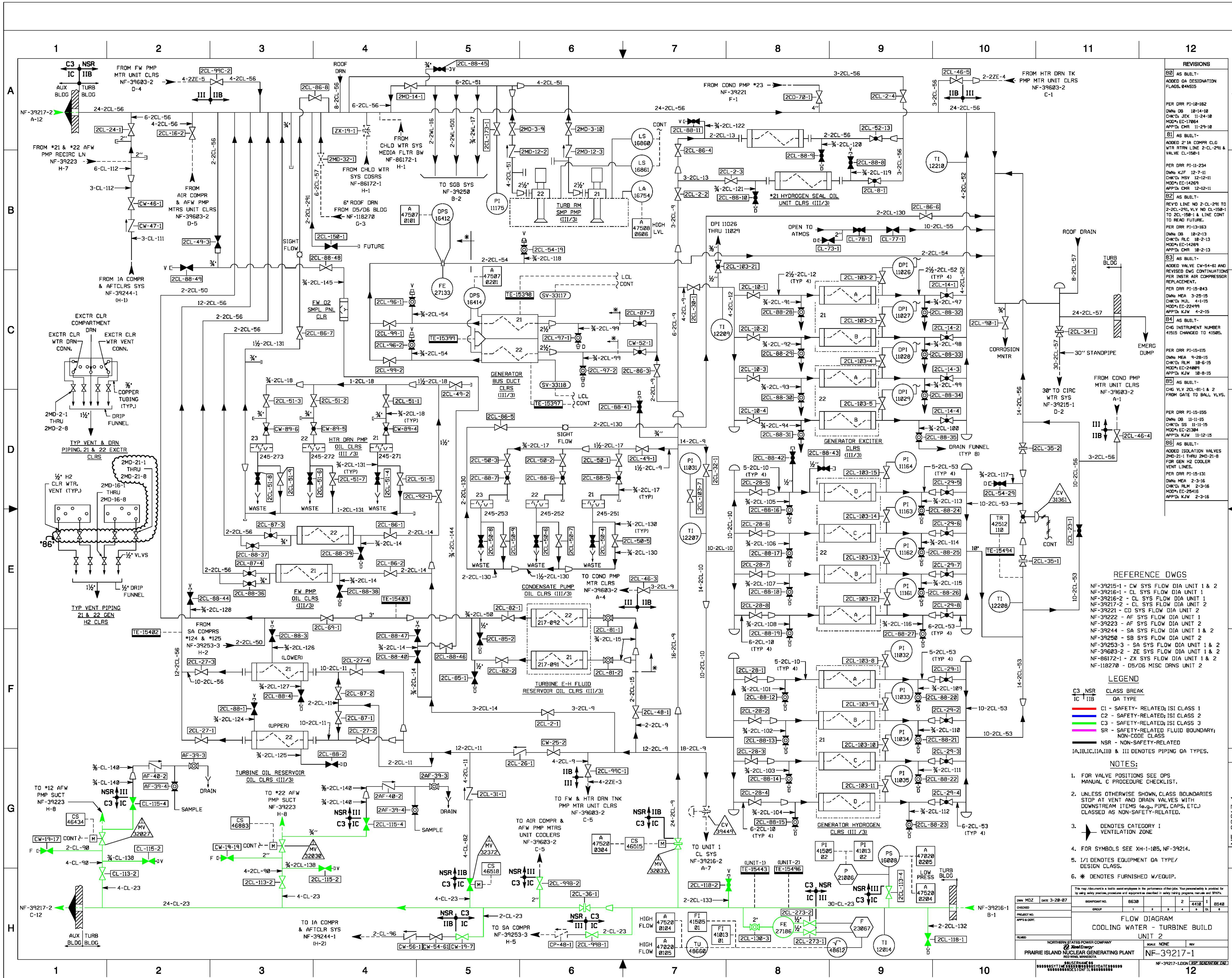
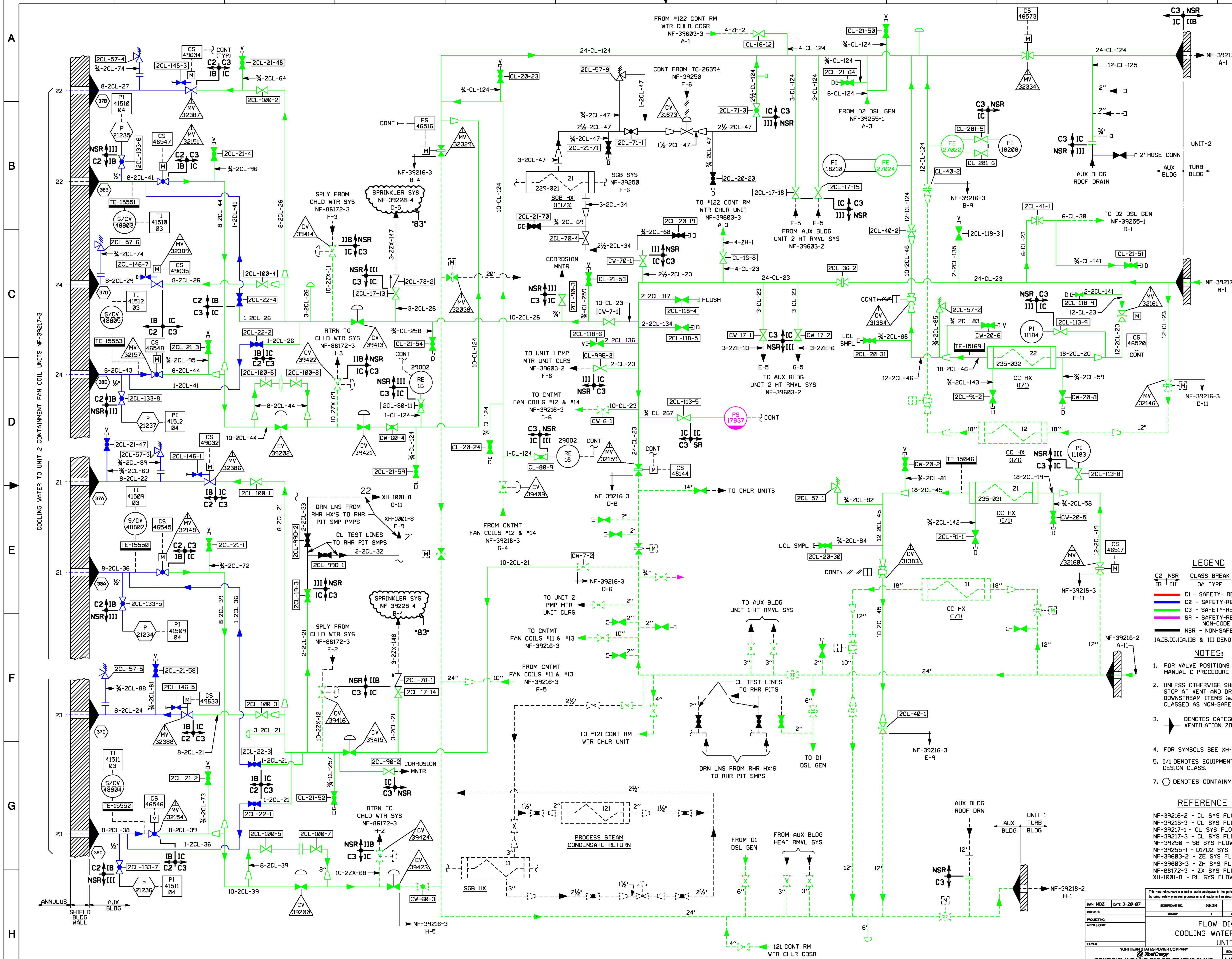


FIGURE 10.4-2A REV. 34

01516979

CAD FILE: U10402A.DGN



REVISIONS

76 AS BUILT-
ADDED VENT VALVES 2CL-146-1, 2CL-146-3, 2CL-146-5, 2CL-146-7.
PER DRR PI-06-068
DWN: JLB 10-1-08
CHK'D: BWK 10-17-08
MOD#: EC-691
FILMED 11/08

77 AS BUILT-
UPDATE WITH STANDARD BORDER FOR CONSISTENCY WITH DRAWING CONTINUATION COORDINATES
PER DRR PI-09-209
DWN: DB 1-26-10
CHK'D: JEK 1-27-10
MOD#: EC-14781
APP'D: CMR 2-12-10

78 AS BUILT
REDRAWN PER SYSTEM WALKDOWN, INCORPORATED SAFETY CLASS DESIGNATIONS.
04NS15
PER DRR PI-10-048
DWN: JEK 3-31-10
CHK'D: DB 5-11-10
REVIEWED: WJI 5-11-10
MOD#: EC-14552
APP'D: CMR 5-12-10

79 AS BUILT-
ADDED DWG CONT AND COORDINATES FOR 122 CR AIR HANDLER. UPDATED PPG. ID'S FOR D2 RTN LN
PER DRR PI-09-163
DWN: KJF 8-16-10
CHK'D: JBS 10-12-10
MOD#: EC-14472
APP'D: CMR 10-18-10

80 AS BUILT-
REV'D 3" LINES FROM HT RMVL SYS FROM NSR TO C3, REVERSED QA DESIGNATIONS (@ C-5, D-6). 04NS14
PER DRR PI-10-182
DWN: DB 10-14-10
CHK'D: JEK 11-24-10
MOD#: EC-17064
APP'D: CMR 11-29-10

81 AS BUILT-
REV'D 8" LINE NUMBER 8-2CL-39 TO 8-2CL-38. @ LOC G-1.
PER DRR PI-13-160
DWN: DB 10-7-13
CHK'D: CMR 10-7-13
MOD#: EC-22106
APP'D: CMR 10-7-13

82 AS BUILT-
REVISED DWG TO REFLECT CONT RM AIR HANDLER DRAIN PATH REMOVAL.
PER DRR PI-13-236
DWN: KJF 12-10-13
CHK'D: CMR 12-11-13
MOD#: EC-22967
APP'D: CMR 12-11-13

83 AS BUILT-
ADDED DWG CONT REF. @ LOC B-4, E-4
PER DRR PI-14-200
DWN: DB 12-18-14
CHK'D: KB 1-6-15
MOD#: EC-23171
APP'D: CMR 1-6-15

LEGEND

C2 NSR / IB III — CLASS BREAK / QA TYPE

- C1 - SAFETY- RELATED; ISI CLASS 1
- C2 - SAFETY-RELATED; ISI CLASS 2
- C3 - SAFETY-RELATED; ISI CLASS 3
- SR - SAFETY-RELATED FLUID BOUNDARY; NON-CODE CLASS
- NSR - NON-SAFETY-RELATED

IA,IB,IC,IIA,IIB & III DENOTES PIPING QA TYPES.

NOTES:

1. FOR VALVE POSITIONS SEE OPS MANUAL C PROCEDURE CHECKLIST.
2. UNLESS OTHERWISE SHOWN, CLASS BOUNDARIES STOP AT VENT AND DRAIN VALVES WITH DOWNSTREAM ITEMS (e.g., PIPE, CAPS, ETC.) CLASSED AS NON-SAFETY-RELATED.
3. DENOTES CATEGORY I VENTILATION ZONE
4. FOR SYMBOLS SEE XH-1-105, NF-39214.
5. I/1 DENOTES EQUIPMENT QA TYPE/ DESIGN CLASS.
7. ⬡ DENOTES CONTAINMENT PEN NUMBER

REFERENCE DWGS

NF-39216-2 - CL SYS FLOW DIA. UNIT 1
NF-39216-3 - CL SYS FLOW DIA. UNIT 1
NF-39217-1 - CL SYS FLOW DIA. UNIT 2
NF-39217-3 - CL SYS FLOW DIA. UNIT 2
NF-39250 - SB SYS FLOW DIA. UNIT 2
NF-39255-1 - D1/D2 SYS FLOW DIA. UNIT 1 & 2
NF-39603-2 - ZE SYS FLOW DIA. UNIT 1 & 2
NF-39603-3 - ZH SYS FLOW DIA. UNIT 1 & 2
NF-86172-3 - ZX SYS FLOW DIA. UNIT 1 & 2
XH-1001-8 - RH SYS FLOW DIA. UNIT 2

DWN MDZ	DATE 3-20-07	SIGNIFICANT NO.	8630						
CHECKED		GROUP	1	2	3	4	5	CL	6
PROJECT NO.									
APP'D & CERT.									
FILMED									

FLOW DIAGRAM
COOLING WATER - AUX BLDG
UNIT 2

NORTHERN STATES POWER COMPANY
Xcel Energy
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RED WING, MINNESOTA

SCALE NONE | REV 83

NF-39217-2

$$USERNAME$$
$$$$$$SYTIME$$$$$$$$$$$$SYDATE$$$$$$
$$$$$$$$$DESIGNFIL$$$$$$$$$

NF-39217-2.DGN NSP GENERATION CAD

NF-39217-2

FIGURE 10.4-2B REV. 34

CAD FILE: U10402B.DGN

01516979

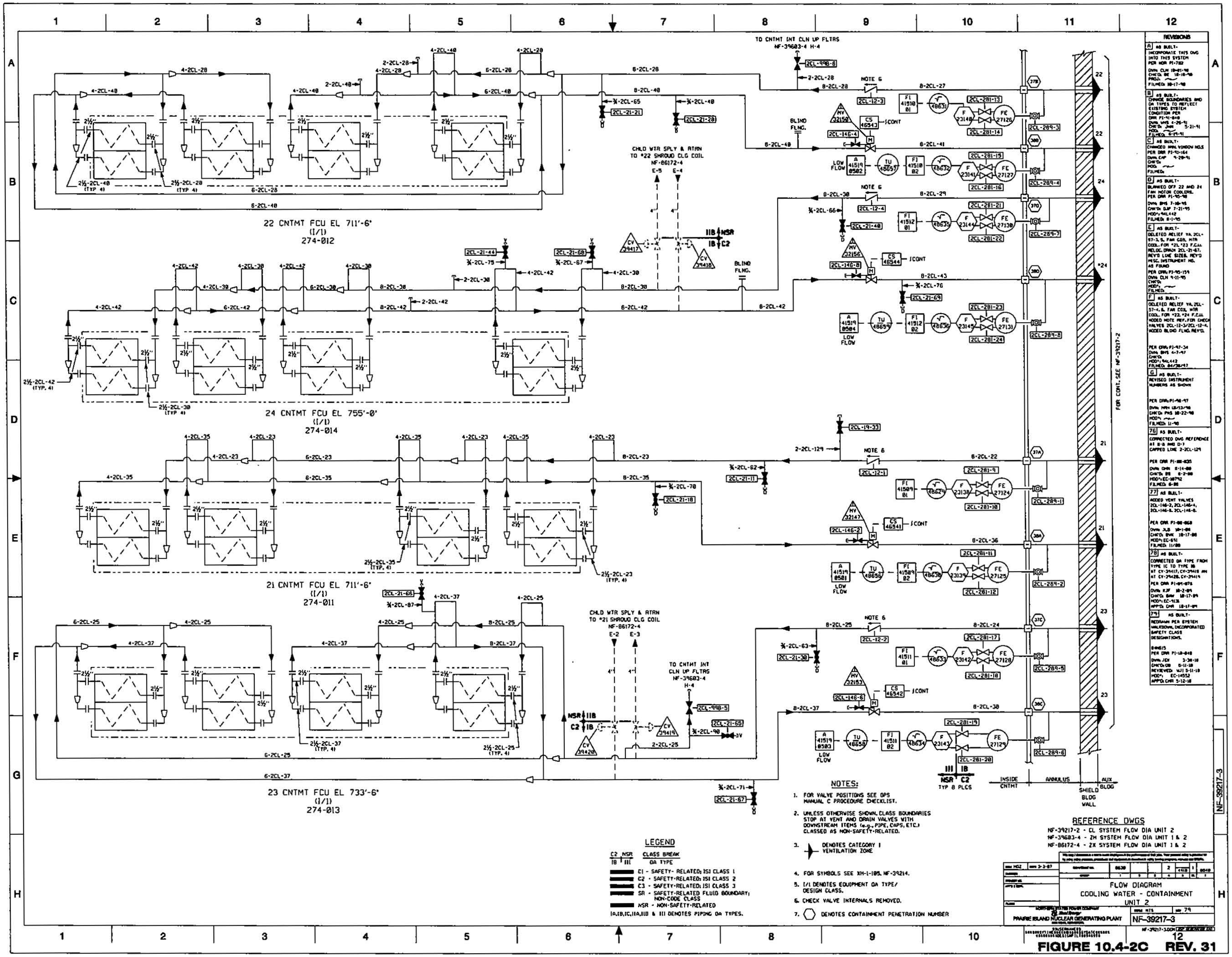


FIGURE 10.4-2C REV. 31

01248495

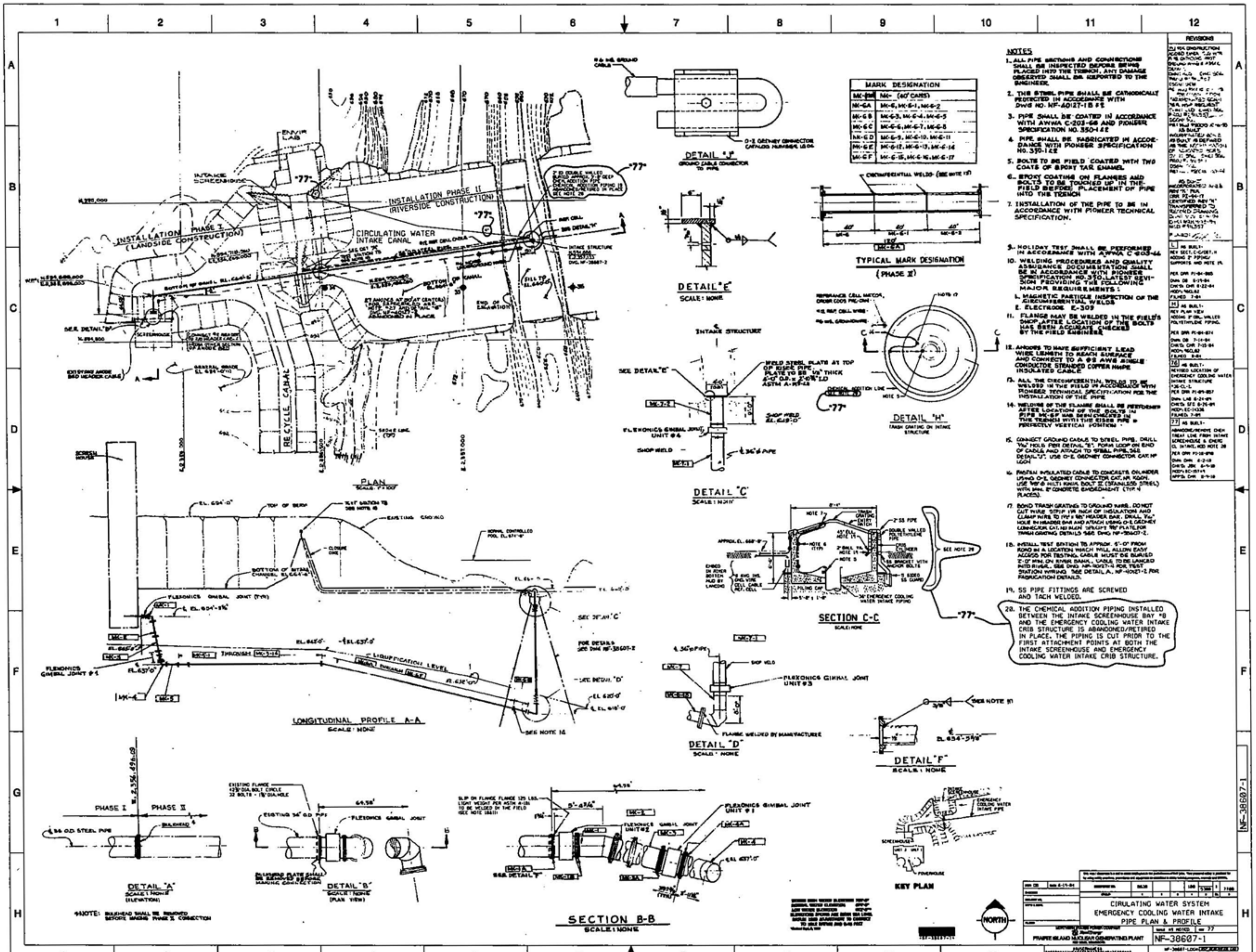


FIGURE 10.4-3A REV. 31

01248495

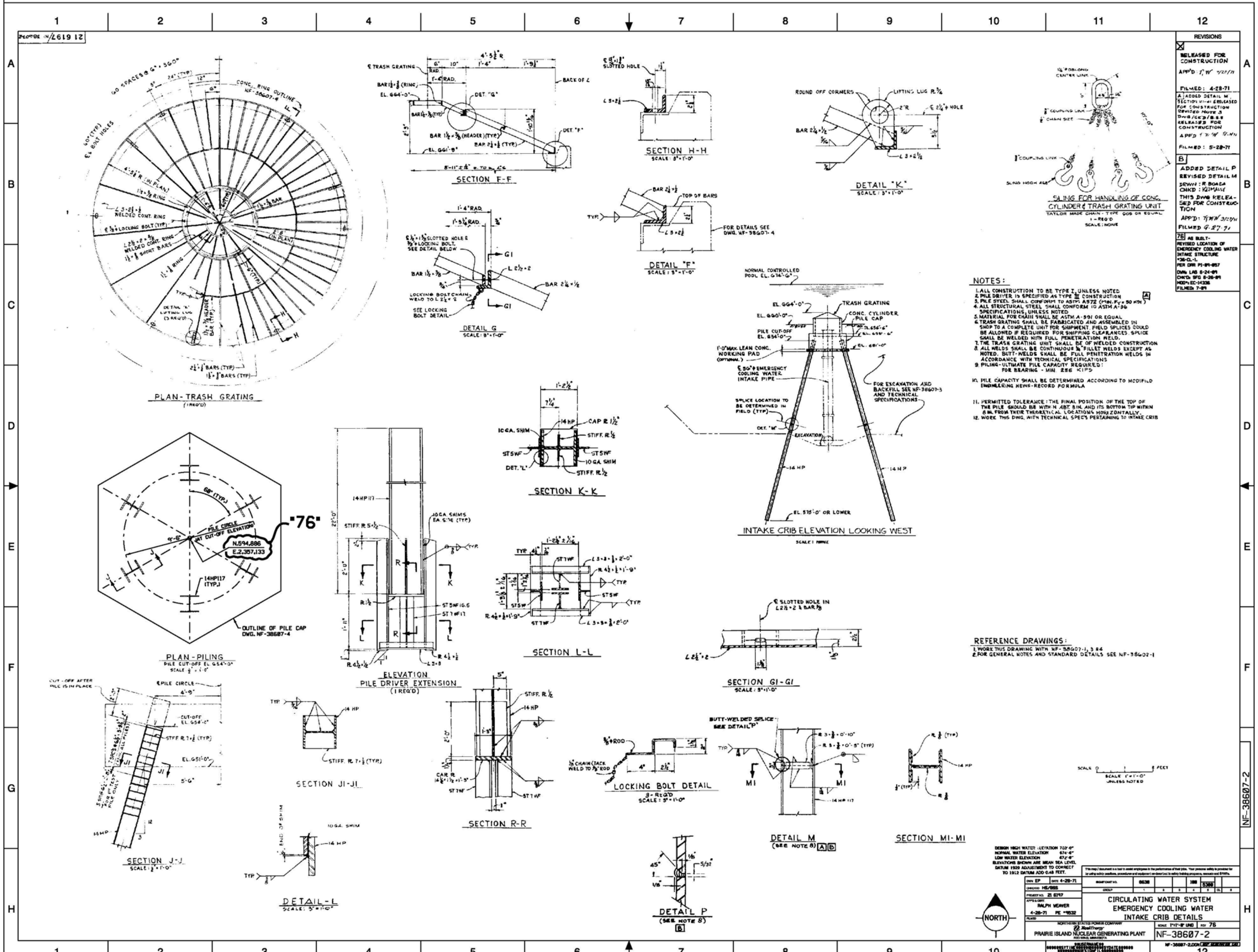


FIGURE 10.4-3B REV. 32

CAD FILE: U10403B.DGN

01352784

FIGURE 10.4-4A REV. 34

CAD FILE: U10404A.DGN

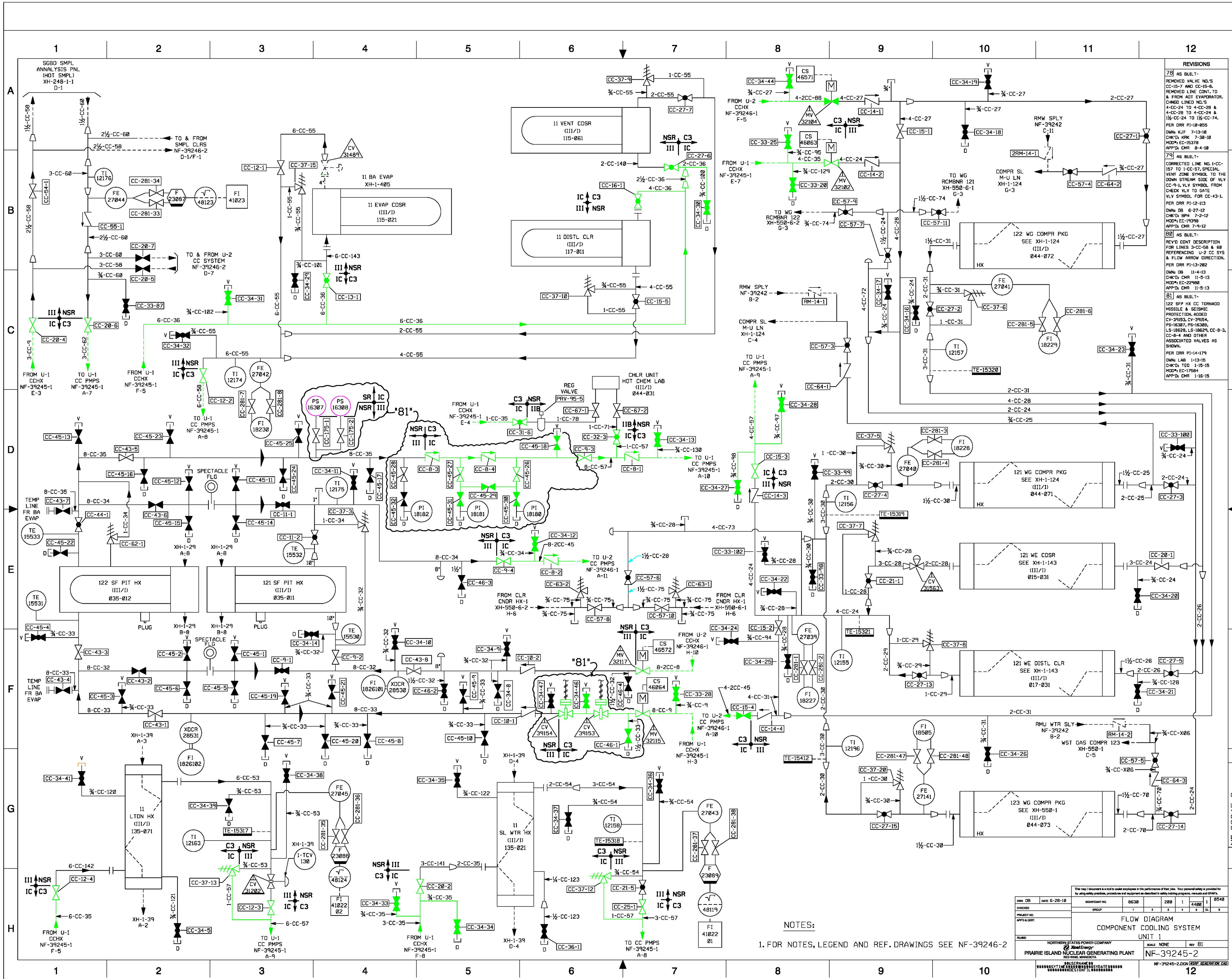


FIGURE 10.4-4B REV. 34

01516979

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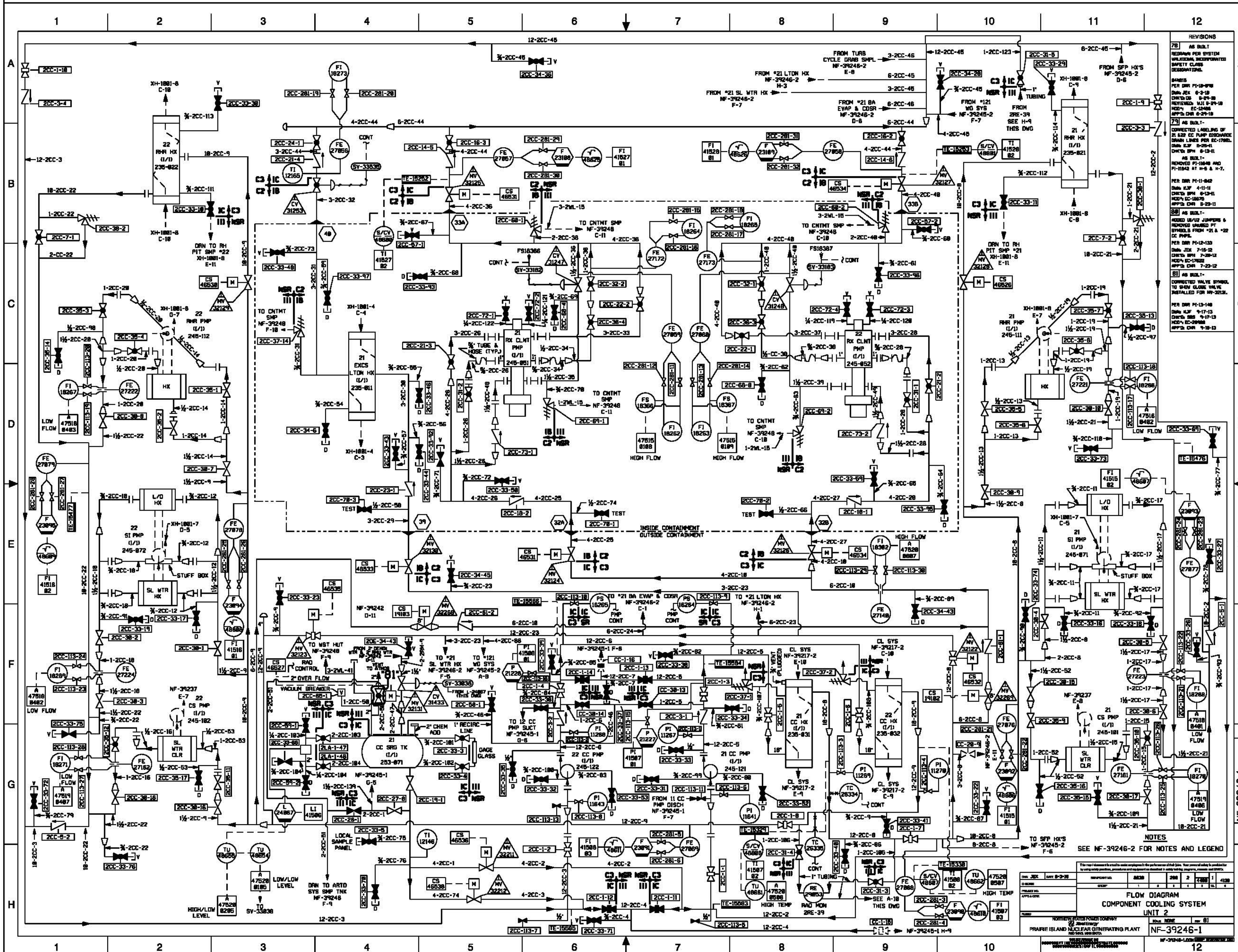

FIGURE 10.4-5A REV. 33

CAD FILE: U10405A.DGN

01429088

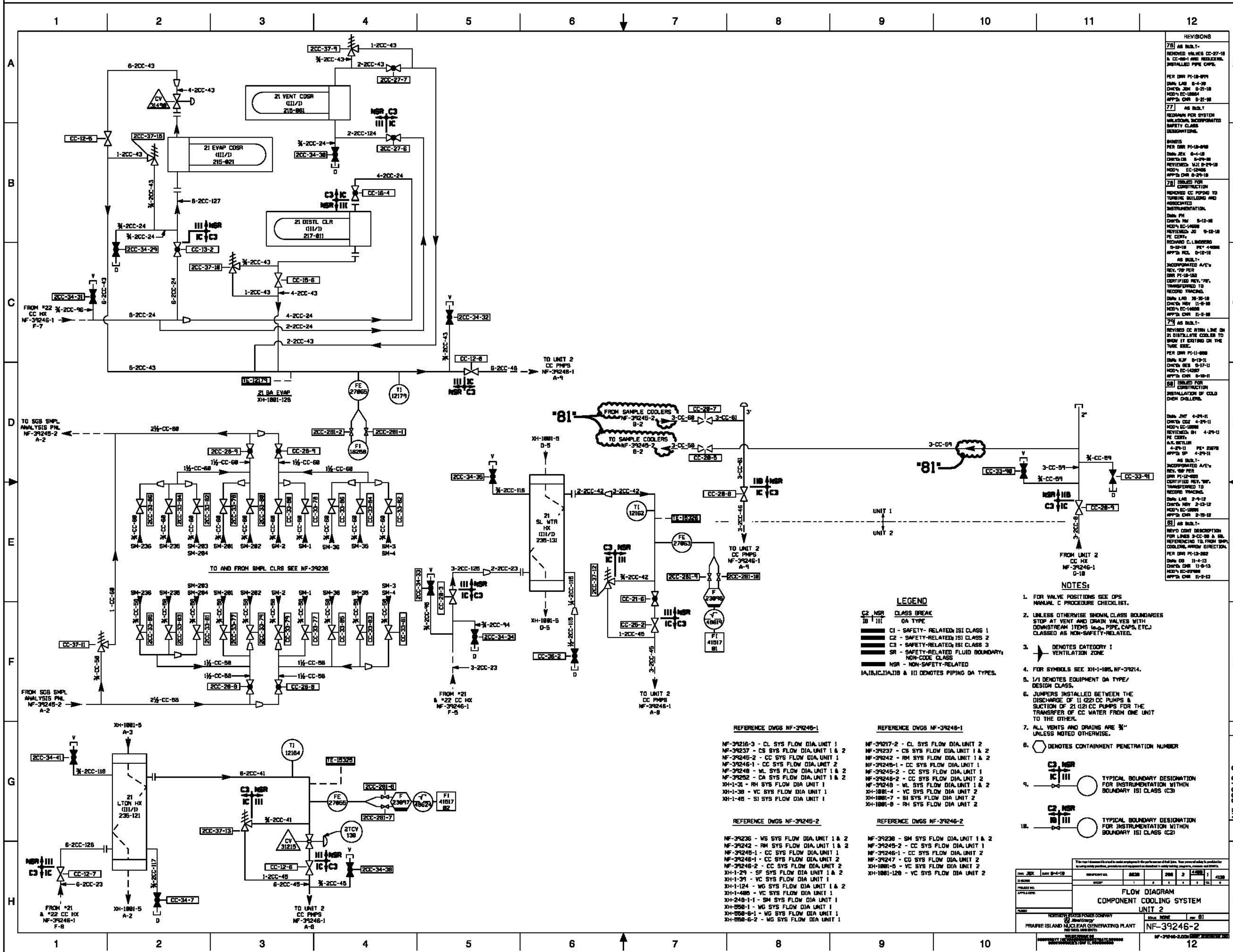

FIGURE 10.4-5B REV. 33

01429088

CAD FILE: U10405B.DGN