SECTION 6

ENGINEERED SAFETY FEATURES

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SECTION 6

ENGINEERED SAFETY FEATURES

6.1 GENERAL DESCRIPTION

The engineered safety features (ESF), together with the containment system (Section 5), serve as protection to the public in the unlikely event of a loss-of-coolant accident (LOCA). The ESF are designed to minimize the effects of a LOCA by performing five functions:

1. To supply borated water to the reactor coolant system (RCS) to cool the core, decrease reactivity, limit fuel rod cladding temperatures and metal-water reaction and ensure that the core remains intact.

2. To eliminate the driving potential (differential pressure) for leakage out of the containment.

3. To reduce the concentration of airborne fission products in the containment available for leakage.

4. To reduce the dose at the site boundary from any airborne fission products which leak from the containment.

5. To control the buildup in the containment of an explosive mixture of hydrogen and oxygen, by mixing of the containment atmosphere.

The first function is satisfied by the timely, continuous and adequate supply of borated water to the RCS and the reactor core. The second function is satisfied by the provision of:

1. Heat sinks for the condensation of steam released inside the containment

2. Depressurization of the containment following any LOCA

3. Means for maintaining the containment at near atmospheric conditions for an extended period of time.

The third function is satisfied by providing containment sump pH control: a chemical additive (sodium tetraborate decahydrate, denoted as “NaTB”) is provided for dissolution and transfer to the containment sump and recirculation spray to enhance the removal of radioactive iodine from the containment atmosphere in the event of a LOCA. The fourth function is provided by exhausting various areas contiguous to the containment to an elevated release point on top of the containment. The fifth function is satisfied by the containment depressurization system which provides the capability to thoroughly mix the containment atmosphere.

The ESF systems provided for satisfying these functions are:

1. A safety injection system, which injects borated water into the reactor coolant loops (Section 6.3).

2. Two separate low pressure safety injection subsystems which provide long term reactor core decay heat removal (Section 6.3).

3. The separate subsystems of the containment depressurization system (the quench spray subsystems and the recirculation spray subsystems) which, when operating together, reduce the containment pressure and temperature, remove heat from the containment and promote a mixed atmosphere required by 10 CFR 50.44. The recirculation spray subsystems transfer the heat from the containment to the river water system. A chemical additive (NaTB) provided for dissolution and transfer to the containment sump and recirculation spray subsystem reduces the concentration of airborne fission products in the containment (Section 6.4).

4. The supplementary leak collection and release system, which normally exhausts the structures contiguous to the containment (except the main steam valve cubicle), the fuel building and the waste gas shortage area to an elevated release point. Following a DBA or a waste gas storage accident, the exhaust from these areas is diverted to a filter prior to being discharged at the elevated release point (Section 6.6). The filters are not credited in the radiological analysis.

A composite schematic of the ESF systems is shown in Figure 6.1-1.

The safety injection system provides for the injecting of borated water to the RCS from the accumulators following a LOCA. The three accumulators are self contained and are designed to supply water as soon as the RCS pressure drops below accumulator pressure. Continued makeup is provided by the charging pumps and the low head safety injection pumps. Both the charging and low head safety injection pumps are located outside the containment and are electric motor-driven, capable of being energized and operated rapidly and powered by the emergency buses. The pumps also ensure an adequate supply of borated water for an extended period of time by recirculation of the water from the containment sump to the reactor core through two entirely separate flow paths.

The quench spray subsystems supply chilled borated water from the refueling water storage tank to the containment shortly after receipt of the containment isolation phase B (CIB) signal. The systems include two 100 percent design capacity electric motor-driven quench spray pumps which are located outside the containment and are supplied with power from the emergency buses.

Containment sump pH control consists of six covered metallic buffer baskets filled with a predetermined quantity of NaTB, mounted to the containment basement floor. The baskets are designed to allow dissolution of the NaTB buffer as a result of contact with water released from a RCS break inside containment and from the quench spray subsystem. The alkalated water then passes into the containment sump where it will be absorbed into the recirculation spray subsystem upon commencement of the recirculation phase of a LOCA.

The NaTB buffer baskets are passive components which increase the pH of the recirculation spray subsystem sprays to ensure effective removal and retention of radioactive iodine from the containment atmosphere following a LOCA. The baskets are periodically inspected for NaTB quantity and the buffer is tested for pH-raising effectiveness.

The recirculation spray subsystems recirculate the alkalated water from the containment sump through river water cooled recirculation spray coolers to the recirculation spray headers. Two 50 percent design capacity, motor-driven recirculation spray pumps are located inside the containment and two additional 50 percent design capacity, motor-driven recirculation spray pumps are located outside the containment. The recirculation spray coolers are located inside the containment and transfer containment heat to the river water system.

The recirculation spray subsystems, together with the quench spray subsystems, are capable of reducing the containment pressure.

The supplementary leak collection and release system has two 100 percent design capacity leak collection exhaust fans and two 100 percent design capacity main filter banks. Each main filter bank consists of roughing filters, charcoal filters and HEPA filters. The supplementary leak collection and release system normally exhausts structures contiguous to the containment, the fuel building and the waste gas storage area directly to the elevated release point without the air stream going through the filters. On a containment isolation phase A (CIA) signal, a leak collection area high-high radiation signal, a fuel building high-high radiation signal or a waste gas storage area high-high radiation signal, the exhaust from all areas served by the supplementary leak collection and release system is diverted to the filters prior to being discharged at the elevated release point. The filters are not credited in the radiological analysis.

The electrical requirements of all ESF are satisfied by normal outside a-c power provided from two independent emergency buses. Highly reliable onsite power is ensured by two emergency diesel generators, should all outside power sources fail. These systems are described in Section 8. All engineered safety features can be manually operated from the main control room. The minimum ESF equipment which can be started under emergency power conditions are:

1. One charging pump
2. One low head safety injection pump
3. Two recirculation spray pumps
4. One quench spray pump
5. One leak collection exhaust fan

An evaluation of these systems under various accident conditions is presented in Section 14.3.

Routine periodic testing of the ESF components is performed as discussed under the individual systems and in the Technical Specifications. Components of the ESF system are located in accessible areas for periodic visual inspection.

All ESF components (including duct work and piping) are designed as Seismic Category I (Appendix B).

6.2 EQUIPMENT - DESIGN AND CRITERIA

The containment depressurization system, emergency core cooling system (ECCS) and supplementary leak collection and release system are designed, fabricated, inspected and installed to meet the requirements as set forth by the applicable AEC General Design Criteria (Section 1.3.2 and Appendix IA). The above systems are considered to be essential to the prevention and/or mitigation of consequences of accidents which could affect the public health and safety.

6.2.1 Pumps and Valves

The safety classes and codes to which pumps and valves are designed are indicated in individual subsections. All welding and nondestructive testing performed on the pumps and valves are performed by personnel meeting the requirements of the American Society for Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section IX, and the American Society of Nondestructive Testing Specification SNT-TC-IA.

Valve packings are selected to minimize or eliminate leakage where necessary. For motor operated valves, the operators are selected to ensure reliable operation under accident conditions.

Periodic testing and inspections of the engineered safety features subsequent to installation supplement the design and fabrication criteria specified above. This testing and inspection is discussed in an individual section on each system and in the Technical Specifications.

6.2.2 Piping

Piping fabrication, installation and testing was in accordance with American National Standards Institute (ANSI) B31.1-1967[1], with supplemental requirements as necessary for use in nuclear applications. These are listed in Table 6.2-1. Pipe routing and supports are such that missiles generated from postulated events or the effects of loss-of-coolant accidents do not impair or adversely affect the operation of the containment depressurization system, the emergency core cooling system, and the supplementary leak collection and release system. The inservice inspection requirements for piping are controlled under the BVPS Inservice Inspection Plan in accordance with 10 CFR 50.55a(g).

Stresses and reaction forces/moments of piping and supports are computed for thermal gradients under normal conditions, with and without friction on sliding supports. The piping stress analysis work carried out on the BVPS-1 also includes these calculations under accident conditions. In addition, after the completion of a dead load hanger analysis, a dynamic study is made. Seismic loads are then calculated from the equivalent static loads and combined moments determined from the previously established dead load and pressure figures. All Seismic Category I systems are subjected to dynamic analysis (Appendix B.2.1).

6.2.2.1 Definition of Piping Classifications

The classifications listed below determine the fabrication, inspection and documentation requirements in the procurement and erection of piping. It should be noted that Table B.1-1 in Appendix B lists structures, systems and components which are safety related and which, therefore, are designed for seismic loadings. Some piping which would be identified from the following listing as Class II or III for fabrication, inspection and documentation purposes is identified as Seismic Category I for seismic design purposes in accordance with Appendix B.

The following definitions are intended to require computations, inspection and documentation consistent with the severity of the service. The definition of Class I conforms with the requirements stated in the Federal Register[(2)]. Line numbers on piping drawings for "Nuclear" systems will have a suffix, Q1, Q2 and Q3 to indicate the line's appropriate piping class. For a list of piping drawings that show piping class information see Table 6.2-2. Portions of systems which have no nuclear safety related function which may contain radioactive fluids and whose postulated failure would not result in conservatively calculated potential offsite dose that exceed 0.5 rem (in accordance with Regulatory Guide 1.26) to the whole body or its equivalent to any part of the body may be excluded from Q1, Q2 and Q3 classifications detailed below. Portions of systems which are not required to meet functional requirements (i.e., test lines) and are normally isolated from the system may be excluded from the following QI, Q2 and Q3 classifications:

Piping Class I (QI)

This classification encompasses the reactor coolant system and portions of auxiliary systems and emergency core cooling systems connected to the reactor coolant system. For piping of those auxiliary systems and emergency core cooling systems which penetrate containment, the piping Class I boundary extends to and includes the first containment isolation valve outside the containment capable of external actuation (simple check valves are excluded). For piping of those auxiliary systems which contain two valves, both of which normally are closed during normal operation, the piping Class I boundary extends to and includes the second of these valves. This second valve (excluding simple check valves) must be capable of external actuation, whether or not the system piping penetrates the containment. For piping of those emergency core cooling systems which does not penetrate the containment, the piping Class I boundary extends to and includes the second of two valves normally closed during normal operation. For piping of those auxiliary systems and emergency core cooling systems which contain a relief or safety valve, the piping Class I boundary extends to and includes the relief or safety valve.

Piping Class II (Q2)

This classification encompasses the following:

1. Residual heat removal system
2. Reactor coolant letdown and charging portions of chemical and volume control system
3. Portions of the emergency core cooling and containment depressurization systems that may recirculate reactor coolant

4. Portions of the main steam and feedwater systems extending from and including the secondary side of the steam generator up to and including the outermost containment isolation valves and connected piping up to and including the first isolation valve

5. Those portions of any other system used to effect isolation of containment

Piping Class III (Q3)

This classification encompasses the following:

1. Chemical and volume control system including piping from boric acid tanks to charging pumps but excluding portions defined above as Class II and portions defined as QA Category II in accordance with Appendix A, "Quality Assurance."
2. Containment depressurization system excluding those portions covered in Class II
3. Accumulator and refueling water supply subsystems of the emergency core cooling system
4. Auxiliary feedwater system
5. Portions of the component cooling and river water systems that transfer heat from systems for emergency core cooling, containment depressurization, residual heat removal and reactor coolant letdown
6. Vents and drains from Class I and Class II systems (except portions defined as QA Category II in accordance with UFSAR Appendix A, Quality Assurance)
7. Any portion of primary plant not classified as Class I or II
8. The reactor vessel flange leak detection lines up to and including valve SOV-RC-544.

6.2.2.2 Protection of Class I Piping

Class I piping systems are protected from damage due to whipping pipe from other systems by one of the following methods:

1. Class I pipe and equipment are shielded by a protective wall or enclosure
2. Class I pipe and equipment are separated from non-Class I systems that could cause damage by a distance that is sufficient to prevent damage.

Where a Class I system cannot be protected as in 1 or 2 above from a non-Class I system, the portion of the non-Class I system that threatens the Class I system is reclassified to Class I and is designed to the following criteria:

1. Failure of the piping is precluded due to conservative piping, hanger and support design

2. In the unlikely events of severance of a pipe, the pipe restraints are so placed that whipping is impossible, or, if that is impractical, so that the maximum length of whipping pipe is of a length that is insufficient to reach the systems being protected.

6.2.3 Motors

6.2.3.1 Motors Located Outside the Containment

Motor electrical insulation systems are in accordance with ANSI, Internal Electronic and Electrical Engineers (IEEE) and National Electrical Manufacturers Association (NEMA) standards and are tested as required by these standards. Temperature rise design is selected such that normal long life is achieved even under accident loading conditions. Periodic electrical insulation tests will be made during the lifetime of the equipment to detect deterioration, if any, of the insulation system.

6.2.3.2 Motors Located Inside the Containment

The motors are selected to ensure operation during DBA conditions. Motor electrical insulation systems are in accordance with ANSI, IEEE and NEMA standards. The motors are tested as required by these standards. Winding insulation systems are provided which can operate at temperatures in excess of those calculated to occur under DBA conditions. Prototype tests are performed on each motor type required for operation after a DBA. Periodic electrical insulation tests will be made during the life of the equipment to detect any deterioration of the insulation to ensure that motors remain in a reliable operating condition.

6.2.4 Ventilation

Safeguards areas are ventilated by a cooled recirculation type ventilation system designed to limit ambient air temperature buildup and to provide suitable environment for personnel and equipment.

An exhaust system for these rooms is provided by the supplementary leak collection and release system described in Section 6.6.

Engineered safety features equipment contained in the ventilated spaces includes the outside recirculation spray pumps, low head safety injection pumps, auxiliary feedwater pumps and quench spray pumps.

Design air rates are based on maintaining a maximum space temperature of 120 F when the pumps are running.

References for Section 6.2

1. "ANSI Power Piping Code", ANSI B31.1-1967, The American National Standards Institute.

2. Federal Register, Vol. 34, No. 226, Tuesday, November 25, 1969 - p. 18823.

6.3 EMERGENCY CORE COOLING SYSTEM

6.3.1 Design Bases

6.3.1.1 Performance Objectives

The design bases for the emergency core cooling system (ECCS) are:

1. To protect the station personnel and the public by maintaining clad integrity, thus minimizing the release of fission products from the fuel during the unlikely event of a loss-of-coolant accident (LOCA).

2. To protect the core for a range of possible mishaps, evaluated as less unlikely, thereby, minimizing financial loss and loss of power generation capability.

The ECCS is designed to cool the reactor core, as well as to, provide additional shutdown capability following initiation of the following accident conditions:

1. Pipe breaks and spurious relief or safety valve lifting in the reactor coolant system (RCS) which cause a discharge larger than that which can be made up by the normal system, up to and including the instantaneous circumferential rupture of the largest pipe in the RCS.

2. Rupture of a control rod drive mechanism causing a rod cluster control assembly (RCCA) ejection accident.

3. Pipe breaks and spurious relief or safety valve lifting in the steam system, up to and including the instantaneous circumferential rupture of the largest pipe in the steam system.

4. A steam generator tube rupture.

The acceptance criteria for the consequences of each of these accidents is described in Section 14 in the respective accident analyses sections.

The primary function of the ECCS for the ruptures described above is to remove the stored and fission product decay heat from the core such that fuel damage, to the extent that would impair effective cooling of the core, is prevented. This implies that the core remain intact and in place, with its essential heat transfer geometry preserved. To ensure effective cooling of the core, limits on peak clad temperature and local metal-water reaction, as defined in Section 14, will not be exceeded.

For any rupture of a steam pipe and the associated uncontrolled heat removal from the core, the ECCS will perform such that:

1. In the event of an uncontrolled steam release resulting from any single active failure in the main steam system (such as the opening with failure to close, of any single steam relief, control or bypass valve) there is no return to criticality after reactor trip.

2. In the event of a steam line break together with the combined effects of any single control rod remaining stuck out of the core (after reactor trip) and the most restrictive single active failure in the engineered safety features (ESF), the core shall remain in place and intact with its essential heat transfer geometry preserved.

The ECCS meets the intent of 1967 AEC General Design Criteria, as noted in Section 1.3.2 and discussed below:

1. The ECCS objectives are met even though loss of normal station power has occurred coincident with the accident.

2. Any single active failure during injection does not prevent the accomplishment of safety injection system objectives. The ECCS meets the single failure criterion discussed in Section 1.3.1.

3. Critical parts of the ECCS and of the RCS may be physically inspected.

4. Active components of the ECCS may be tested periodically to ensure that each component is operable.

5. An integrated ECCS test of active components may be performed during shutdown without introducing flow into the RCS.

6. Maintenance outages of active components is permitted for limited periods of time.

7. It is assumed that the highest worth control rod assembly remains stuck out of the core upon reactor trip.

8. Components exposed to the accident environment are designed to operate in that environment for the length of time required.

6.3.1.2 ECCS Single Failure Criterion Compliance

Active Failure Criteria

The ECCS is designed to accept a single active failure following the incident without loss of its protective function. The system design will tolerate the failure of any single active component in the ECCS itself or in the necessary associated service systems at any time during the period of required system operations following the incident.

A single active failure analysis is presented in Table 6.3-1 and demonstrates that the ECCS can sustain the failure of any single active component in either the short or long term and still meet the level of performance for core cooling.

Since the operation of the active components of the ECCS following a steam line rupture is identical to that following a LOCA, the same analysis is applicable and the ECCS can sustain the failure of any single active component and still meet the level of performance for the addition of shutdown reactivity.

Passive Failure Criteria

The following philosophy provides for necessary redundancy in component and system arrangement to meet the intent of the AEC General Design Criteria on single failure as it specifically applies to failure of passive components in the ECCS. Thus, for the long term, the system design is based on accepting either a passive or an active failure. The long term is defined in Section 1.3.1.

Redundancy of Flow Paths and Components for Long Term Emergency Core Cooling

In the design of the ECCS, the following criteria are utilized:

1. During the long term cooling period following a LOCA, the emergency core cooling flow paths shall be separable into two subsystems, either of which can provide minimum core cooling functions and return spilled water from the floor of the containment back to the RCS.

2. Either of the two subsystems can be isolated and removed from service in the event of a leak outside the containment.

3. Adequate redundancy of check valves is provided to tolerate failure of a check valve during the long term as a passive component.

4. Should one of these two subsystems be isolated in this long term period, the other subsystem remains operable.

5. Provisions are also made in the design to detect leakage from components outside the containment, collect this leakage and to provide for maintenance of the affected equipment.

Thus, for the long term emergency core cooling function, adequate core cooling capacity exists with an open flow path removed from service whether isolated due to a leak, because of blocking of one flow path, or because failure in the containment results in a spill of the delivery of one subsystem.

Subsequent Leakage from Components in Engineered Safety Features Subsystems

With respect to piping and mechanical equipment outside the containment, considering the provisions for visual inspection and leak detection, leaks will be detected before they propagate to major proportions. A review of the equipment in the system indicates that the largest sudden leak potential would be the sudden failure of a pump shaft seal. Evaluation of leak rate assuming only the presence of a seal retention ring around the pump shaft showed flow less than 50 gpm would result. Piping leaks, valve packing leaks or flange gasket leaks have been of a nature to build up slowly with time and are considered less severe than the pump seal failure.

Larger leaks in the ECCS are prevented by the following:

1. The piping is designed in accordance with ANSI B31.1-1967[6]

2. The seismic design for piping, equipment and supports ensures no loss of function for the Design Basis Earthquake

3. The system piping is located within a controlled area on the site

4. The piping system receives pressure tests and is accessible for periodic visual inspection

5. The piping is austenitic stainless steel which, due to its ductility, can withstand severe distortion without failure.

6.3.1.3 Codes and Classifications

Table 6.3-2 tabulates the codes and standards to which the ECCS components are designed.

6.3.2 System Design and Operation

6.3.2.1 System Description

Adequate emergency core cooling following a LOCA is provided by the ECCS shown in Figures 6.3-1, and 6.3-8 or 6.3-9. A simplified composite flow diagram of the entire ECCS is given as part of Figure 6.1-1, which is a composite flow diagram of the ESF. The system components operate in the following possible modes:

1. The injection mode in which any reactivity increase following the postulated accidents is terminated, initial cooling of the core is accomplished and coolant lost from the primary system in the case of a LOCA is replenished.

2. The recirculation mode in which long term core cooling is provided during the accident recovery period.

The initiation signal for core cooling by the safety injection charging pumps and the low head safety injection pumps is the safety injection signal (SIS) which is to be actuated by any of the following:

1. Low pressurizer pressure (two-out-of-three)

2. High containment pressure (two-out-of-three)

3. Low steam line pressure (two-out-of-three detectors in any one main steam line)

4. Manual actuation (1/2).

Injection Phase

The principle components of the ECCS which provide emergency core cooling immediately following a loss of coolant are the three accumulators (one for each loop), two of the three safety injection charging pumps (which perform the charging functions during normal operations) and the two low head safety injection pumps. The safety injection charging pumps are located in the auxiliary building. The two low head safety injection pumps are located in the safeguards area alongside the containment structure.

The accumulators, which are passive components, discharge into the cold legs of the reactor coolant piping when RCS pressure decreases below accumulator pressure thus ensuring rapid core cooling for large breaks. They are located inside the containment and are protected against possible missiles.

The SIS opens the boron injection header isolation valves and starts the safety injection charging pumps. The accumulator isolation valves also receive the SIS, even though these valves are normally open.

The safety injection charging pumps deliver borated water to the three cold legs of the reactor coolant loops during the injection phase. These pumps provide for the makeup of coolant and add negative reactivity following a small break which does not immediately depressurize the RCS to the accumulator discharge pressure. For large breaks, they start delivery through separate lines after the accumulators start their discharge.

The suction of the safety injection charging pumps is diverted from their normal suction at the volume control tank to the refueling water storage tank by the SIS. The pumps feed a common injection header. The injection header contains a boron injection tank on the discharge side of the safety injection charging pumps. The safety injection charging pumps' discharge is isolated by the redundant normally closed parallel boron injection tank outlet valves. The valves open upon receipt of an SIS and refueling water flows from the discharge of the safety injection charging pumps through the boron injection tank and into the RCS cold legs.

For large pipe ruptures, the RCS is depressurized and voided of coolant rapidly and a high flow rate of emergency coolant is required to quickly cover the exposed fuel rods and limit possible core damage. This high flow is provided by the passive accumulators, followed by the charging pumps and the low head safety injection pumps which discharge into the cold legs of the RCS.

Manual valves in the ECCS are mainly those required for maintenance, refueling or test operations. Those manual valves that, if improperly positioned, would have an adverse effect on the performance of the ECCS, are administratively controlled in the correct position. The high head safety injection/charging pump recirculation line combined isolation valve (MOV-CH373) is administratively controlled in the deenergized and locked open position to prevent spurious closure. Manual throttle valves in the injection branch lines are properly adjusted by flow tests during system start-up testing.

A detailed listing of the instrumentation readouts of the main control board which the operator can monitor during the initial injection is given in Table 6.3-3.

Valves of the safety injection systems which are remotely operated and are normally in their ready position but do not receive a SIS have their positions indicated on a common portion of the control board. At any time during operation when one of these valves is not in the ready position for injection, it is shown visually on the board. In addition, an audible alarm alerts the operator to the condition.

Safety Injection Recirculation Phase

After the injection operation and when low level is reached in the refueling water storage tank, the reactor coolant and the injected refueling water spilled from the break and the water from the containment depressurization system (Section 6.4) collect in the containment sump and part is returned to the RCS by the low head safety injection pump(s). The balance flows to the recirculation spray subsystems.

Because the injection phase of the accident is terminated before the refueling water storage tank is completely emptied, all pipes are kept filled with water before recirculation is initiated. The extreme low level setpoint on the RWST is set such that the containment sump level will be sufficient to provide adequate NPSH for the LHSI pumps when utilizing the sump suction path. Adequate NPSH for the LHSI pumps when utilizing the RWST suction path is available at the extreme low level setpoint. Two out of the four RWST level signals coincident with a safety injection signal will automatically initiate the transfer from injection to recirculation. The transfer occurs as described in Section 6.3.3.9.

If the break is large, depressurization of the RCS occurs rapidly due to the large rate of mass and energy loss through the break to the containment. For small breaks, the depressurization of the RCS by the ECCS and by the rupture during the injection phase can be augmented by secondary steam dump and auxiliary feedwater addition. Operator action to dump steam augments RCS depressurization but credit is not taken for this in the safety analysis in Section 14.

For the expected accident progression, no manual operator actions are required until after the symptoms have been diagnosed and the specific type of accident is identified. If abnormal conditions are present, the operating procedures will permit actions for feedwater and RCS temperature control prior to identification of the type of accident. The procedures also provide guidance in differentiating between the various types of accidents.

When the necessary depressurization has been accomplished, the low head safety injection pumps take suction from the containment sump and return the coolant to the reactor. If depressurization of the RCS proceeds slowly, recirculation of spilled coolant can be accomplished by aligning the discharge of the low head safety injection pumps with the suction of the safety injection charging pumps.

The redundant features of the ECCS recirculation loop include one pump in each of two separable and redundant trains with crossover capability at the discharge of each pump. Each pump takes suction through separate cross-connected lines from the containment sump.

The design of the containment sump and piping configuration from the containment sump to the low head safety injection pumps is illustrated in Figure 6.1-1.

After one day, the spray water collected is cold enough to reduce the temperature of the combined mass sufficiently for recirculation without flashing. All heat removal is through the recirculation spray subsystem. There are no heat exchangers in the ECCS.

Those portions of the ECCS located outside of the containment which are designed to circulate, under post accident conditions, radioactively contaminated water collected in the containment meet the following requirements:

1. Shielding to maintain radiation levels within the guidelines set forth in 10 CFR 50.67.

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

3. Means to detect and control radioactivity leakage into the environs, to the limits consistent with guidelines set forth in 10 CFR 50.67.

This criterion is met by minimizing leakage from the system. Recirculation loop leakage is discussed in Section 6.3.3.

Changeover from Injection Phase to Recirculation Phase

During the injection phase, the RWST level decreases to the low level set point at which time sufficient water is delivered to the containment sump via the containment depressurization system, the ECCS and from the RCS through the break to provide the required NPSH of the low head safety injection pumps to change to recirculation. The automatic transfer sequence is provided in Section 6.3.3.9.

Steam Break Protection

A large break of a main steam system pipe causes an uncontrolled removal of heat which rapidly cools the reactor coolant causing insertion of reactivity into the core. Compensation is provided by injection of boric acid solution from the RWST through the boron injection tank into the cold legs. This system was originally designed and installed to maintain a nominal 12 weight percent concentration of the boric acid solution in the boron injection tank. However, the boron injection tank is no longer credited as a source of borated water for the accident analysis. Therefore, the boron concentration in the boron injection tank no longer needs to be sampled.

6.3.2.2 Components

All associated components, piping, structures and power supplies of the ECCS are designed to conform with Seismic Category I criteria (Appendix B). All components inside the containment are capable of withstanding or are protected from differential pressure which may occur during the rapid pressure rise to containment design pressure.

Accumulators

The accumulators are pressure vessels filled with borated water and pressurized with nitrogen gas. During normal operation, each accumulator is isolated from the RCS by two check valves in series. If the RCS pressure falls below the accumulator pressure, the check valves open and borated water is forced into the RCS. Mechanical operation of the swing-disk check valves by means of differential pressure is the only action required to open the injection path from the accumulators to the core via the cold leg.

The level of borated water in each accumulator tank is adjusted remotely as required during normal station operation. Makeup water from the refueling water storage tank is added using the positive displacement hydrotest pump.

Water level is reduced by draining to the primary drain transfer tank. Samples of the solution in the accumulators are taken at the sampling station for periodic checks of boron concentration.

The accumulators are passive engineered safety features because the nitrogen gas pressure forces injection; no external source of power or signal transmission is needed to obtain fast acting, high flow capability when the need arises. One accumulator is connected to each of the cold legs of the RCS.

The accumulators are carbon steel, clad with stainless steel and designed to ASME Boiler and Pressure Vessel Code, Section III, Class C requirements. Redundant level and pressure indicators are provided with readouts on the control board. Each channel is equipped with high and low level alarms. The margin between the minimum operating pressure and design pressure provides a range of acceptable operating conditions within which the accumulator system meets its design core cooling objectives. The band is sufficiently wide to permit the operator to minimize the frequency of adjustments in the amount of contained gas or liquid to compensate for leakage. Limiting conditions for operation and surveillance are set forth in the Technical Specifications.

The accumulator design/operation parameters are listed in Table 6.3-5.

Environmental qualification of ECCS equipment, inside the containment, which is required to operate following a LOCA is discussed in Section 7.3.2.1.2.

The quality standards of all ECCS components are tabulated in summary form in Table 6.3-4.

Boron Injection Tank

The boron injection tank is constructed of carbon steel and clad with stainless steel for corrosion resistance.

Table A.1-1 identifies the portions of the equipment employed with the BIT which are designed as Category I.

The design parameters are presented in Table 6.3-6.

Pumps

Charging pumps, which are also used as high head safety injection pumps, supply borated water to the RCS. The pumps are of the horizontal centrifugal type, driven by electric motors. Minimum operating requirements are provided in the periodic operating surveillance tests for these pumps. Procedure references are provided to ensure a safety evaluation is performed when pump performance requirement changes could affect accident analysis assumptions.

The low head safety injection pumps are a vertical centrifugal pump equipped with a turbulence limiter and anti-vortexing devices (cruciforms and grating) to reduce vortexing and a false bottom pump-can to lower pump vibrations. These pumps supply borated water to the RCS during the safety injection and recirculation mode. The head-flow curve for the low head pumps is given in Figure 6.3-5.

All pressure containing parts of the pumps were chemically and physically analyzed and the results were checked to ensure conformance with the applicable ASTM specification. In addition, all pressure containing parts of the pump were liquid penetrant inspected in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII, Appendix VIII. The acceptance standard for the liquid penetrant test is ANSI B31.1, Case N-10. Parts of the pump in contact with borated water are stainless steel or equivalent corrosion resistant material.

The pressure containing parts of the pumps are castings conforming to ASTM A-351[7] Grade CF8 or CF8M, or equivalent.[32] Stainless steel forgings were procured per ASTM A-182[8] Grade F 304 or F 316 or ASTM A-336,[9] Class F8 or F8M, or equivalent.[32] Stainless steel plate was constructed to ASTM A-240[10] Type 304 or 316, or equivalent.[32] All bolting material conforms to ASTM A-193[11] or equivalent.[32]

Materials such as weld-deposited Stellite, Chromaloy or equivalent[32] are used at points of close running clearances in the pumps to prevent galling and to ensure continued performance capability in high velocity areas subject to erosion.

Pump design is reviewed with special attention to the reliability and maintenance aspects of the working components. Specific areas include evaluation of the shaft seal and bearing design to determine that adequate allowances have been made for shaft deflection and clearances between stationary parts. Table 6.3-7 lists the design parameters for the safety injection charging and low head safety injection pumps.

Where welding of pressure containing parts was necessary, a welding procedure including joint detail was submitted for review and approval. The procedure includes evidence of qualification necessary for compliance with the ASME Boiler and Pressure Vessel Code, Section IX. This requirement also applies to any repair welding performed on pressure containing parts.

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

Each pump is given a complete shop performance test in accordance with Hydraulic Institute Standards. The pumps were run at design flow and head, shutoff head and three additional points to verify performance characteristics. Where NPSH was critical, this value was established at design flow by means of adjusting suction pressure.

Valves

All parts of valves used in the ECCS in contact with borated water are austenitic stainless steel or equivalent corrosion resistant material. The motor operators on the injection line isolation valves are capable of rapid operation.

All valves required for initiation of safety injection or isolation of the system have remote position indication in the main control room.

Exceptional tightness is specified for all valves and, where possible, such as instrument valves, packless diaphragm valves are used. All valves, except those which perform a control function, are provided with backseats which are capable of limiting leakage to less than one cubic centimeter per hour per inch of stem diameter, assuming no credit taken for valve packing. Those valves which are normally open are backseated. Normally closed globe valves are installed with pressure under the seat to prevent leakage of recirculated water through the valve stem packing. Relief valves are installed with pressure under the seat to prevent leakage of recirculated water through the valve stem packing. Relief valves are totally enclosed. All modulating control valves of all sizes, which are exposed to normally radioactive fluid, are provided with double-packed stuffing boxes and stem leakoff connections which are piped to the vent and drain system.

The check valves which isolate the ECCS from the RCS are installed adjacent to the reactor coolant piping to reduce the probability of a safety injection line rupture causing a LOCA.

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

The specified leakage across the valve disk required to meet the equipment specification and hydrotest requirements for conventional globe valves, gate valves, motor operated valves, check valves and accumulator check valves is 3 cubic centimeters per hour per inch of nominal pipe size.

Motor-Operated Valves

The pressure containing parts (body, bonnet and disks) of the valves employed in the safety injection system were designed according to criteria established by the ANSI B16.5(12) or Manufacturers Standardization Society Standard Practice (MSS) SP-66(13) specifications. The materials of construction for these parts were procured as per ASTM A-182, F-316(14) or A-351, Gr CF8M or CF8, or equivalent.(32)

Radiographic inspection was conducted in accordance with the procedure outlined in ASTM E-94(15). Radiographic acceptance standards were outlined in ASTM E-71(16), E-186(17), or E-280(18) whichever was applicable, and meet the requirements of severity level 2 except that D, E, F and G defects were not permissible. The body, bonnet and disk were liquid penetrant inspected in accordance with ASME Boiler and Pressure Vessel Code, Section III, Paragraph N-627.

When a gasket was employed, the body-to-bonnet joint was designed as per ASME Boiler and Pressure Vessel Code, Section VIII or ANSI B16.5 with a fully trapped, controlled compression, spiral wound asbestos or equivalent(32) gasket with provisions for seal welding, or of the pressure seal design with provisions for seal welding. The body-to-bonnet bolting and nut materials were procured as per ASTM A-193(11) and A-194,(19) respectively. These body-to-bonnet bolting and nut materials may have been replaced with materials equivalent(32) to the requirements of ASTM A-193(11) and A-194.(19)

The entire assembled unit was hydrotested in accordance with MSS SP-61(20) with the exception that the test was maintained for a minimum period of 30 minutes. Failure of the test was cause for rejection. The seating design was of the Darling parallel disk design, the Crane flexible wedge design or the equivalent. These designs have the feature of releasing the mechanical holding force during the first increment of travel. Thus, the motor operator has to work only against the frictional component of the hydraulic unbalance on the disk and the packing box friction. The disks are guided throughout their full travel to prevent chattering and provide ease of gate movement. The seating surfaces are hard faced (Stellite No. 6 or equivalent) to prevent galling and reduce wear.

The stem material was ASTM A-276(21) Type 316 condition B or precipitation hardened 17-4 PH stainless steel procured and heat treated to Westinghouse specifications or equivalent(32) stainless steel.

These materials were selected because of their corrosion resistance, high tensile properties and their resistance to surface scoring by the packing. The valve stuffing box was designed with a lantern ring leak-off connection with a minimum of a full set of packing below the lantern ring and a maximum of one-half of a set of packing above the lantern ring; a full set of packing was defined as a depth of packing equal to 1.5 times the stem diameter. The experience with this stuffing box design and the selection of packing and stem materials has been very favorable in both conventional and nuclear power stations.

The motor operator is extremely rugged. Some units incorporate a "lost motion" feature that allows the motor to come up to operating speed prior to being fully loaded. These units also have a "hammer blow" feature that allows the stem nut threads of the actuator to be impacted away from the stem threads. This is necessary for these units since the stem nut lock nut arrangement is a rigid two piece system. For units that do not incorporate the "lost motion" feature and the "hammer blow" feature, the actuator utilizes a spring compensator to alleviate excessive closing thrusts due to inertia by allowing the stem nut to indirectly float against the compensator spring pack.

The valve was assembled, hydrostatically tested, seat-leakage tested (fore and back), operationally tested, cleaned and packaged as per specifications. In some cases, extension stems were used on motor-operated valves. These valves were operationally tested initially without the extension stems. The valves were then operationally tested with the extension stems, after installation in the unit. All manufacturing procedures employed by the valve supplier such as hard facing, welding, repair welding and testing were submitted to Westinghouse for approval.

The following generic design requirements were applied for the originally supplied valves. These values were not based on accident analyses. For those valves which must function on the SIS, the valve operator completes its cycle from one position to another in 10 seconds maximum (for valves up to and including 8 inches) and 49 inches per minute operation (for valves over 8 inches). For all other valves in the system, (for valves up to and including 8 inches) 12 inches per minute operation and (for valves larger than 8 inches) the valve operator completes its cycle from one position to another in 120 seconds maximum. Current valve stroke time requirements listed in the Licensing Requirements Manual may differ from these values.

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

Manual Valves

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

The carbon steel valves were built to conform with ANSI B16.5. The materials of construction of the body, bonnet and disk conform to the requirements of ASTM A-105[22] Grade II, A-181[23] Grade II, A-216[24] Grade WCB or WCC, or their equivalent.[32] The carbon steel valves pass only nonradioactive fluids and were subjected to hydrostatic tests as outlined in MSS SP-61, except that the test pressure was maintained for at least 30 minutes per inch of wall thickness. Since the fluid controlled by the carbon steel valves was not radioactive, the double packing and seal weld provisions were not provided.

Accumulator Check Valves

The pressure containing parts of this valve assembly are designed in accordance with MSS SP-66. All parts in contact with the operating fluid are of austenitic stainless steel or of equivalent corrosion resistant materials procured to applicable ASTM or Westinghouse Nuclear Energy Systems specifications. The cast pressure containing parts are radiographed in accordance with the procedure outlined in ASTM E-94. Radiographic acceptance standards are as outlined in ASTM E-71, E-186 or E-280, whichever is applicable and meet the requirements of severity level 2, except that D, E, F and G defects are not permissible. The cast pressure containing parts, machined surfaces, finished hard facings and gasket bearing surfaces are liquid penetrant inspected as per ASME Boiler and Pressure Vessel Code, Section VIII and the acceptance standard is as outlined in ANSI B31.1 Code, Case N-10. The final valve is hydrotested in accordance with MSS SP-66, except that the test pressure is maintained for at least 30 minutes. The seat leakage test is conducted in accordance with the manner prescribed in MSS SP-61, except that the acceptable leakage is three cubic centimeters per hour per inch of nominal pipe diameter.

The valves were designed with a low pressure drop configuration with all operating parts contained within the body, which eliminates those problems associated with packing glands exposed to boric acid. The clapper arm shaft was manufactured from 17-4 PH stainless steel heat treated to Westinghouse specifications or equivalent.[32] The clapper arm shaft bushings were manufactured from Stellite No. 6 or equivalent[32] hardened material. The various working parts were selected for their corrosion resistant, tensile and bearing properties.

The disk and seat rings were manufactured from a forging. The mating surfaces are hard faced with Stellite No. 6 or equivalent[32] wear resistant material to improve the valve seating life. The disk is permitted to rotate, providing a new seating surface after each valve opening.

The valves are intended to be operated in the closed position with a normal differential pressure across the disk of approximately 1,600 psi. The valves remain in this position, except for testing and accumulator discharge. Since the valve is not required to normally operate in the open condition, and hence be subjected to impact loads caused by sudden flow reversal, it is expected that this equipment will satisfactorily perform its required functions indefinitely with minimal maintenance.

When the valve is required to function, a differential pressure of less than 25 psig will shear any particles that may attempt to prevent the valve from functioning. Although the working parts are exposed to the boric acid solution contained within the reactor coolant loop, a boric acid "freeze up" is not expected with this low a concentration.

The experience derived from the check valves employed in the emergency injection systems indicate that the system is reliable and workable; check valve leakage has not been a problem. This was substantiated by the satisfactory experience obtained from operation of the Ginna Nuclear Power Plant and subsequent plants where the usage of check valves is identical to this application.

Accumulator Isolation Valve

The isolation valve at each accumulator is closed when the reactor is intentionally depressurized. The only other time the isolation valve is closed is for testing or maintenance purposes, for which a time limitation is specified in the Technical Specifications. The valve is designed to operate with full system differential pressure. For a further discussion of these valves, see Section 6.3.3.7.

Accumulator check valves are tested for leakage after entry into Mode 5 during each refueling outage, prior to reactor startup. Additional accumulator check valve leakage testing may be required after other plant outages in accordance with the Technical Specifications. The check valves are tested when the RCS is pressurized with a minimum of 150 psi differential across the valves. This test confirms the seating of the disk and whether or not there has been an increase in the leakage since the last test. Upon RCS heatup following the outage, there should be no increase in leakage since increasing reactor coolant pressure increases the seating force and decreases the probability of leakage.

Relief Valves

The accumulator relief valves are sized to pass nitrogen at a rate in excess of the accumulator gas fill line delivery rate. The relief valves also pass water in excess of the expected leak rate, but this is not necessary because the time required to fill the gas space gives the operator ample opportunity to correct the situation. For an inleakage rate 15 times the manufacturing test rate, there are about 1,000 days before water reaches the relief valves. Prior to this, level and pressure alarms would have been actuated. Table 6.3-8 lists all of the system's relief valves with their capacities and setpoints.

Leakage Limitations of Valves

Exceptional tightness is specified for all valves and where possible, such as instrument valves, packless diaphragm valves are used.

Normally open valves have backseats which limit leakage to less than one cubic centimeter per hour per inch of stem diameter assuming no credit for packing in the valve. Normally closed globe valves are installed with pressure under the seat to prevent stem leakage from the more radioactive fluid side of the seat.

The specified leakage across the valve disk required to meet the equipment specification and hydrotest requirements is as follows:

1. Conventional globe valves - three cubic centimeters per hour per inch of nominal pipe size

2. Gate valves - three cubic centimeters per hour per inch of nominal pipe size

3. Motor-operated gate valves - three cubic centimeters per hour per inch of nominal pipe size

4. Check valves - three cubic centimeters per hour per inch of nominal pipe size; ten cubic centimeters per hour per inch for 300 and 150 lb ANSI standard.

5. Accumulator check valves - two cubic centimeters per hour per inch of nominal pipe size.

Leakage from components of the recirculation loop including valves is tabulated in Table 6.3-9.

Piping

All ECCS piping in contact with borated water is austenitic stainless steel. Piping joints are welded, except where flanged connections are required for instrumentation, test connections, or to facilitate equipment removal for maintenance.

The piping beyond the accumulator stop valves is designed for RCS conditions (2,485 psig and 650°F). All other piping connected to the accumulator tanks is designed for 700 psig and 400°F.

The safety injection charging pump suction piping from the refueling water storage tank is designed for low pressure losses to meet NPSH requirements of the pumps.

The safety injection high pressure branch lines designed for high pressure losses to limit the flow rate out of the branch line which may have ruptured at the connection to the reactor coolant loop. The system design incorporates the ability to isolate the safety injection charging pumps on separate headers such that full flow from at least one pump is ensured should a branch line break.

The piping was designed and fabricated to meet the minimum requirements set forth in the ANSI B31.1, ANSI B36.10[25] and ANSI B36.19,[26] ASTM Standards, supplementary standards and additional quality control measures. Table 6.2-1 provides a summary of required examination and tests for the different classifications of nuclear piping.

Minimum wall thicknesses were determined by the ANSI Code formula found in Section 1 of B31.1. This minimum thickness was increased to account for the manufacturer's permissible tolerance of 12.5 percent of the nominal wall and 8 percent allowance for bending. Purchased pipes and fittings have a specified nominal wall thickness that was no less than the sum of that required for pressure containment, pipe bending, mechanical strength and manufacturing tolerance.

Special attention is directed to the piping configuration at pumps with the objective of minimizing pipe imposed loads at the suction and discharge nozzles.

Piping is supported to accommodate expansion due to temperature changes and hydraulic forces during an accident.

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

1. Check analyses were performed on both the purchased pipes and fittings.

2. Pipe branch lines between the reactor coolant pipes and the isolation stop valves conform to ASTM A-376[27] or its equivalent[32] and meet the supplementary requirement S6 for ultrasonic testing.

3. Fittings conform to the requirements of ASTM A-403[28] or its equivalent.[32]

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

Welds for pipes sized 2.5 inches and larger were butt welded except for those welds in the LHSI pump suction piping within the valve pit, which were made when removing the two 12 inch manually operated maintenance valves at the suction of each of the LHSI pumps and reinstalling them as MOV-SI862A and B (see Figure 6.3-8 or 6.3-9). These piping welds were made using slip on couplings, fillet welded to the joined components. A piping flexibility analysis has confirmed the acceptability of these welds in lieu of butt welds. Reducing tees were used where the branch size exceeds one-half of the header size. Branch connections of sizes that were equal to or less than one-half of the header size were of a design that conforms to the ANSI rules for reinforcement set forth in the ANSI B31.1. Bosses for branch connections were attached to the header by means of full penetration welds.

All welding was performed by welders and welding procedures qualified in accordance with the ASME Boiler and Pressure Vessel Code, Section IX. The shop fabricator was required to submit all welding procedures and evidence of qualifications for review and approval prior to release for fabrication. All welding materials used by the shop fabricator must have prior approval.

All high pressure piping butt welds containing radioactive fluid, at greater than 600°F temperature and 600 psig pressure, were radiographed. The remaining piping butt welds were randomly radiographed. The technique and acceptance standards were those outlined in N-624.2 and N-625.3 of the ASME Boiler and Pressure Vessel Code, Section III. In addition, butt welds were liquid penetrant examined in accordance with the procedure of ASME Boiler and Pressure Vessel Code, Section III, Paragraph N-627.2 and the acceptance standard as defined in Paragraph N-627.3. Finished branch welds were liquid penetrant examined on the outside and the root passes were examined for sizes 6 inches and larger and for schedule 80S and heavier for all sizes.

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

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

Packaging of the piping subassemblies for shipment was done so as to preclude damage during transit and storage. Openings were closed and sealed with tight fitting covers to prevent entry of moisture and foreign material. Flange facings and weld end preparations were protected from damage by means of wooden cover plates and securely fastened in position. The packing arrangement proposed by the shop fabricator was subject to approval.

Pump and Valve Motors

Motors Outside the Containment

Motor electrical insulation systems were supplied and tested in accordance with ANSI, IEEE and NEMA standards.

Temperature rise design selection was such that normal long life is achieved even under accident loading conditions.

Motors Inside the Containment

Motors for those valves inside containment which must operate during and/or following the LOCA were designed for continuous service in the environmental conditions following the accident. This assures that the valves can perform their required safety function during the recovery period. Periodic operations of the motors and tests of the insulation ensure that the motors remain in a reliable operating condition. The only motors of the ECCS which must operate inside the containment are valve motors.

Although these motors, which are provided only to drive ESF equipment, are normally run only for test, the design loading and temperature rise limits are based on the accident conditions.

Normal design margins are specified for these motors to make sure the expected lifetime includes allowance for the occurrence of accident conditions.

6.3.2.3 Electrical Supply

Details of the normal and emergency power source for the ECCS are presented in Section 8.

6.3.2.4 Protection Against Dynamic Effects

The high head safety injection lines penetrate the containment adjacent to the auxiliary building.

For most of the routing, these lines are outside each reactor coolant loop cubicle and hence are protected from missiles originating within these areas. Each line penetrates the cubicle as near the injection point to the reactor coolant pipe as possible. In this manner, maximum separation and hence protection is provided in the coolant loop area.

All hangers, stops and anchors were designed in accordance with ANSI B31.1 and American Concrete Institute (ACI) 318[29] Building Code Requirements for Reinforced Concrete which provide minimum requirements on materials, design and fabrication with ample safety margins for both dead and dynamic loads over the life of the equipment. Specifically, these standards required the following:

1. All materials used were in accordance with ASTM specifications which establish quality levels for the manufacturing process, minimum strength properties and for test requirements which ensure compliance with the specifications.

2. Qualification of welding processes and welders for each class of material welded for types and positions of welds.

3. Maximum allowable stress values were established which provide an ample safety margin on both yield strength and ultimate strength.

In the event of the hypothetical double-ended severance of a reactor coolant pipe, the functional integrity of the ECCS connections to the remaining reactor coolant loops is not impaired. This integrity is established and maintained by the application of the following design criteria:

1. The reactor vessel, steam generators and pumps are supported and restrained to limit their movement under pipe break conditions (including a double-ended reactor coolant pipe rupture) to a maximum amount which ensures the integrity of the main steam and feedwater piping. The safety injection piping to the intact loops were designed to accommodate the limited movement of the loop components without failure.

2. The safety injection piping serving each loop is anchored at the missile barrier in each loop area to restrict incident damage to that portion of piping downstream of this point. The anchorage is designed to withstand without failure the thrust force on the safety injection branch line severed from the reactor coolant pipe discharging safety injection flow to the containment, and to withstand a bending moment equivalent to that which produces failure of the safety injection piping under the action of free and end discharge or motion of the broken reactor coolant pipe to which the safety injection piping is connected. This anchorage prevents possible failure upstream from the support point where the branch line ties into the safety injection piping header.

6.3.2.5 Environmental Effects Protection

All of the ECCS operating equipment, with the exception of the motor-operated valves in the accumulator injection lines, are located outside the containment, and hence are not required to operate in the steam-air environment produced by the accident.

All motors, instruments, transmitters and their associated cables and penetrations located inside the containment are designed to function under the post accident temperature, pressure, radiation and humidity conditions for the length of time required.

6.3.3 Design Evaluation

6.3.3.1 Range of Core Protection

The measure of effectiveness of the ECCS is the ability of the pumps and accumulators to keep the core flooded or to reflood the core rapidly where the core has been uncovered for postulated large area ruptures. The result of this performance is to sufficiently limit any increase in clad temperature below a value where emergency core cooling objectives are met as discussed in Section 6.3.1. The core protection over the range of break sizes provided by the minimum components of the ECCS is presented in Section 14.

The following analyses are performed to ensure that the limits on core behavior following a RCS pipe rupture are met by the ECCS operating with minimum design equipment:

1. Large pipe break analysis.

2. Small line break analysis.

3. Recirculation cooling.

The flow delivered to the RCS by the ECCS as a function of reactor coolant pressure with the operation of minimum design equipment is analyzed in Section 14.3.

The design basis performance characteristic is derived from the specified performance characteristic for each pump.

The performance characteristic utilized in the accident analysis is derived in the same manner except that margin is included for uncertainties. If the rupture is assumed to be in an injection line, the injection curve utilized in the analysis is modified to account for the loss of injection water through the broken line.

Large Break Analysis

The large pipe break analysis is used to evaluate the initial core thermal transient for a spectrum of pipe ruptures up to the double-ended rupture of the largest pipe in the RCS.

The injection flow from active components is required to control the cladding temperature subsequent to accumulator injection, complete reactor vessel refill and eventually return the core to a subcooled state. The results indicate that the maximum cladding temperature attained at any point in the core is such that the limits on core behavior as specified in Section 14.3 are met.

Small Pipe Break Analysis

The small pipe break analysis is used to evaluate the initial core thermal transient for a spectrum of pipe rupture from 3/8 inch up to and including the rupture of a 6 inch diameter pipe. For breaks 3/8 inch or smaller, the charging system can maintain the pressurizer level at the RCS operating pressure and the ECCS would not be actuated.

The results of the small pipe break analysis indicate that the limits on core behavior are adequately met, as shown in Section 14.

Recirculation Cooling

Core cooling during recirculation can be maintained by the flow from one low head pump if RCS pressure is low. If RCS pressure remains high, one safety injection charging pump operating in series with the low head pump provides the added head capacity needed to maintain adequate cooling.

Initially the recirculated water would be heated to the saturation temperature at the prevailing reactor coolant pressure when recirculated to the reactor. The system pressure at the time recirculation is initiated is dependent on the initiating break size. Heat removal from the recirculated sump water via operation of the recirculation spray coolers ensures a high degree of subcooling of the water from the sump and serves to limit the duration and extent of core boiling.

6.3.3.2 Borated Water Injection Chemistry

During the injection of emergency cooling water into the RCS following a LOCA, the concentration of boron will vary depending on the depressurization history of the reactor. If depressurization were slow, the high head pumps would inject boric acid solution which would be diluted by the coolant remaining in the system. Rapid depressurization would bring about early injection of water from the accumulators. When recirculation begins, the average concentration of boric acid is (and will remain) about 1.4 weight percent. With chemical additive in the containment spray, the recirculated water will have a pH between 7.0 and 8.28.

The concentrations of other materials, including chlorides, are quite low in this solution, corrosion products being generally insoluble in a basic solution. The principle fission product in solution (assuming a gross core failure) would be iodine at about 3.5 ppm, corresponding to 50 percent of the maximum core inventory. The temperature of the sump water is reduced below 150°F, under normal operating conditions with a minimum of two recirculation spray coolers in operation, after a relatively short period of time (i.e., a few hours) and below 150°F chloride stress corrosion does not constitute a problem.

6.3.3.3 System Response

Time Response

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

Operation with two of the three available accumulators delivering their contents to the reactor vessel (one accumulator spilling through the break after termination of bypass) will meet the criteria stated in Section 6.3.1. The function of the safety injection pumps is to complete the refill of the vessel and ultimately return the core to a subcooled state.

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

The starting sequence of the safety injection charging pumps, the low head safety injection pumps and the related emergency power equipment is designed so that delivery of the full rated flow is reached within 27 seconds after the process parameters reach the setpoints for the injection signal. Reference is made to Section 8.5.

For the small break analysis, an additional delay time is allowed for the SIS to activate when the appropriate setpoint is reached.

Piping Restraints

Following a SIS, the maximum velocity of water is approximately 110 fps in the accumulator discharge lines and 40 fps in the high head safety injection system lines (assuming two pumps deliver through 3 inch common cold injection lines). When the RCS pressure falls below the accumulator pressure, the check valve opens and borated water is forced into the coolant system. The resultant hydraulic forces are controlled by supports, including snubbers and limit stops located at various points in the piping system. Supports in the accumulator discharge lines do not eliminate the potential for water hammer, however, water hammer is not expected, since the relative low pressure of the low-head safety injection pumps would not rapidly close the check valves.

Mechanical and hydraulic snubbers are utilized to limit piping movements resulting from dynamic loadings while permitting normal thermal expansion/contraction. Snubbers are designed for both tension and compression.

Limit stops are also used. They perform the same function as a snubber in that they permit cyclic thermal movement. In a dynamic event, Limit Stops 'limit' pipe movement. Limit stops are a passive device and require no piston or piston valve arrangement to activate. Limit stops are designed for tension and compression.

The valves of the high head safety injection system open or close in ten seconds (in contrast to accumulator check valves that open almost instantaneously); therefore, the hydraulic force should present no problem. The relatively slow closure time of these valves should present no water hammer problems.

6.3.3.4 Single Failure Analysis

A single active failure analysis is presented in Table 6.3-1. Credible active system failures are considered. The analyses of the LOCA presented in Section 14 is consistent with the single failure analysis, based on a single failure in the ECCS.

The analysis shows that the failure of any single active component does not prevent fulfilling the design function; also, operator action is not required to correct the malfunction.

In addition to the single active failure capability, an alternate flow path is available through the high head safety injection pumps should any part of the flow path from the low head pumps to RCS cold legs become unavailable. This feature ensures that core cooling would be maintained in the event of a piping failure in the ECCS. It also should be noted that the reactor cold legs are fed via a common header in the injection mode and it is not possible to isolate the broken leg and prevent spilling under the present design.

In the particular case of a pump being out for maintenance, an additional active or passive failure is not considered. The maximum period that operation would be continued with two pumps out for maintenance is specified in the Technical Specifications.

Failure analysis of the emergency power supply under LOCA conditions are described in Section 8.5.

Inadvertent Closure of Normally Open Motor-Operated Valves

The ESF systems are designed to tolerate, without loss of their protective function, a single failure during the period of recovery following an incident such as a LOCA. The period of recovery includes both the injection phase and the recirculation phase. Compliance with the single failure criterion is discussed in Section 6.3.1.2.

In order to assure that the accumulator isolation valves will be open during power operation when availability of the accumulator is required, the valve control circuit has the following features:

1. The valves receive an "open" signal upon initiation of safety injection. Upon receipt of a safety injection signal, the valve is blocked from closing.

2. The valves automatically receive an "open" signal when the pressurizer pressure exceeds a given pressure (safety injection unblocked).

3. The valves have redundant position indication in the control room (Vertical Board and Bench Board) operated by independent limit switches in the motor operator.

4. The valves have audible alarms, with the alarm circuits actuated by the diverse position sensors in the motor operator and stem mounted limit switches for each valve, to alarm whenever the valve is not fully open and reactor pressure is above a preset value.

5. The audible alarm is timed to reactivate at approximately one hour intervals as long as the valve is not fully open and system pressure is above the preset value.

6. The valve control circuit has power lock out jacks and automatic indication of grounding or shorting of the lock out jack as described below.

Each accumulator isolation valve control circuit also contains a power lock out jack. Removal of the power lock out jack will remove power from the valve actuating contacts, but will permit the operation of the valve position indicating lights.

The power lock out jacks are installed only during plant heat up and cool down. As the plant pressure exceeds a specified pressure, the accumulator isolation valves are opened and the power lock out jacks are removed thus preventing inadvertent closure of the valves. Similarly, the lock out jacks are not installed during plant cooldown until the plant pressure is below the specified pressure and the valves are required to be closed. The installation of the power lock out jacks and operation of the valves is covered by the Technical Specifications. The power lock out jack circuits also contain an automatic indication of grounding or shorting of at least one of the two controls in the power isolation jack.

A failure analysis has been done to investigate the probability that such a valve would be closed during the short time required following a LOCA. Using the fault tree technique, the various combinations of equipment faults and human errors leading to spurious valve closure were evaluated and the probability of such an occurrence was determined to be 0.65×10^{-7} per demand for a single valve. Failure rates for the various equipment faults were taken from data in References 1, 2, 3 and 4. Conservative engineering judgement was applied where specific data on identical components in required failure modes were unavailable. Refer to Table 6.3-11 and Figure 6.3-7.

Based on this analysis it can be stated that this situation of spurious valve closure can indeed be considered extremely unlikely.

6.3.3.5 Reliance on Interconnected System

During the injection phase, the safety injection charging pumps are not dependent on any portion of other systems with the exception of the suction line from the refueling water storage tank. During the recirculation phase of the accident for small breaks, suction to the safety injection charging pumps is provided by the low head safety injection pumps. An installed bypass line connects the outside recirculation spray pumps to the safety injection charging pumps. This bypass line is not intended nor is it assumed to function in any design basis event. It serves only as a tool to address beyond design basis events such as multiple LHSI system failures. Administrative control of the locked closed manual valves ensures strict compliance with this intent. (See Figure 6.1-1)

To limit post accident pressure and temperature in containment to less than design values and provide a means for long term containment heat removal, spray recirculation and cooling must be continued following a LOCA. Initially spray recirculation is continuous, but as the core residual heat level decreases, the recirculation rate is reduced and, eventually, the system may be operated intermittently.

Since the heat removal from the containment must be accomplished initially through the recirculation spray subsystems and since this represents a more than adequate heat removal mechanism for the containment, the use of heat exchangers in the low head safety injection system for cooling is not required. The low head safety injection system operates to provide long term core cooling with no heat exchangers in the system by using water from the containment sump.

6.3.3.6 Normal/Accident Function Evaluation

Table 6.3-12 is an evaluation of the main components, which have been previously discussed, and a brief description of how each component functions during normal operation and during the accident.

6.3.3.7 Passive Systems

The accumulators are a passive safety feature in that they will perform their design function in the total absence of an actuation signal or power source. The only moving parts in the accumulator injection train are in the two check valves.

The working parts of the check valves are exposed to fluid of relatively low boric acid concentration. Even if some unforeseen deposition accumulated, a reversed differential pressure of about 25 psi would shear any particles in the bearing that tended to prevent valve functioning. This is demonstrated by calculation.

The isolation valve at each accumulator is normally open with power to the motor operator locked out via banana type lock out jack, located on the main control board. Redundant position indicating lights located at the control room switch are provided for each valve. In addition, an indicating light is provided on each control circuit to show grounding or shorting of the lock out jack. The isolation valve is closed when the RCS is intentionally depressurized or to test the check valves in the line to the accumulator while the RCS is pressurized.

An alarm annunciator point is activated by both a valve motor operator limit switch and by a valve position limit switch activated by stem travel, whenever an accumulator valve is not fully open for any reason with the system at pressure (the pressure at which the safety injection block is unblocked). A separate annunciator point is used for each accumulator valve. This alarm will be recycled at approximately one hour intervals to remind the operator of the improper valve lineup.

The check valves operate in the closed position with a nominal differential pressure across the disk of approximately 1,600 psi. They remain in this position, except for testing or when called upon to function. Since the valves operate normally in the closed position and, therefore, are not subject to the abuse of flowing operation or impact loads caused by sudden flow reversal and seating, they do not experience any wear of the moving parts, and therefore, will function with reliability.

Accumulator check valves are tested for leakage after entry into Mode 5 during each refueling outage, prior to reactor startup. Additional accumulator check valve leakage testing may be required after other plant outages in accordance with the Technical Specifications. The check valves are tested when the RCS is pressurized with a minimum of 150 psi differential across the valves. This test confirms the seating of the disk and whether or not there has been an increase in the leakage since the last test. Upon RCS heatup following the outage, there should be no increase in leakage since increasing reactor coolant pressure increases the seating force and decreases the probability of leakage.

The accumulators are able to accept leakage from the RCS without any effect on their availability. Table 6.3-13 indicates what inleakage rates, over a given time period, require readjusting the level at the end of the time period. The manufacturing acceptance test is 36 cubic centimeters per hour (i.e., three cubic centimeters per hour per inch of pipe diameter).

The accumulators are located inside the reactor containment and protected from the RCS piping and components by missile barriers. When the accumulators are completely discharged, the pressurizing gas exhausts to the containment. Release of the gas charge in the accumulators would cause an increase in the containment pressure, but occurs after containment peak pressure has been reached. This release of gas has been included in the containment pressure analysis for the LOCA Section 14.3.

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

6.3.3.8 External Recirculation Loop Leakage

Table 6.3-9 summarizes the nominal (design) leakage from the leak sources of the recirculation loop which goes through the low head safety injection pumps and through the safety injection charging pumps if required. The recirculation loop is shown in Figure 6.3-9. The nominal (design) leakage resulting from all sources is 2,382 cubic centimeters per hour to the auxiliary building atmosphere and 60 cubic centimeters per hour to the waste drain tanks. Leakage specified in Table 6.3-9 are nominal (design) values and limiting values are specified in Section 14.3.5.2.

During recirculation, a significant margin exists between the design and operating conditions of the low head safety injection system components. In view of these margins, it is considered that the leakage rates tabulated in Table 6.3-9 are conservative.

Leakage detection exterior to the containment is achieved through the use of sump level detection (Section 9.7). The auxiliary building sump pumps start automatically in the event that liquid accumulates in the sump and an alarm in the main control room indicates that water has accumulated in the sump. Valving is provided to permit the operator to isolate individually the low head safety injection pumps.

6.3.3.9 Pump NPSH Requirements

<u>Low Head Safety Injection Pumps</u>

The NPSH of the low head safety injection pumps is evaluated for both the injection and recirculation phase operation of the design basis accident. The limiting available NPSH occurs at the initiation of the recirculation phase following a LOCA. A transient calculation of the available NPSH is performed with the MAAP-DBA computer code described in Section 14.3 for the containment evaluation. However, the available NPSH transient analysis maximizes the energy addition to the containment sump instead of the atmosphere as was done in the Section 14.3 analysis. The major differences are:

1. The pressure flash model is used for break fluid flashing.

2. The release streams are mixed prior to flashing to direct higher enthalpy water to the sump.

As shown in Figure 6.3-10, the minimum available NPSH is 16.0 ft and occurs at the initiation of LHSI recirculation mode following a pump suction double-ended rupture. The minimum NPSH required is 10.6 feet for the LHSI pumps. Lines diverting cold discharge water from the quench spray headers assist in ensuring this NPSH requirement is met during DBA conditions.

The limiting single failure is the loss of one emergency diesel. This results in the loss of two recirculation spray heat exchangers and therefore minimizes heat removal from the sump water before the LHSI pump suctions are switched to the sump.

The LHSI pumps will be automatically throttled during all operating modes (initial injection, cold leg recirculation and hot leg recirculation) using cavitating venturies. The cavitating venturi is designed to reduce the pressure below the fluid vapor pressure and thus choke the flow. Pressure recovers downstream of the venturi near its initial value.

Safety Injection Charging Pumps

The NPSH for the safety injection charging pumps is calculated for both the injection and recirculation phase operation of the design basis accident.

The end of injection phase operation gives the limiting NPSH requirement and the NPSH available is determined from the elevation head and vapor pressure of the water in the refueling water storage tank, and the pressure drop in the suction piping from the tank to the pumps. The required NPSH for these pumps is 39.5 ft and the minimum available NPSH is 53.5 ft.

Automatic Transfer from Safety Injection to Recirculation

The transfer from the safety injection mode to the recirculation mode will automatically take place on an extreme low level signal from the refueling water storage tank (RWST). The transfer to recirculation actuation signal will be from a "two out of four energize to trip" logic circuit coincident with a safety injection signal. Three of the level transmitters will have indication in the control room and the fourth will be recorded. Extreme low level, via any one of the four channels, is annunciated in the main control room. At the termination of the injection phase:

1. The suction of the low head safety injection pumps is realigned from the RWST to the containment sump
2. The four LHSI pumps' minimum recirculation valves are closed
3. The four valves in the charging pumps' suction are realigned from the RWST to the discharge of the LHSI pumps.

Transfer Description

The following paragraphs are a description of the transfer from the injection mode to the recirculation mode. Refer to Figure 6.1-1 for valve locations.

Upon initiation of a safety injection signal, the LHSI pumps start and recirculate to the RWST until the reactor coolant pressure drops to a pressure commensurate with their discharge head. The safety injection accumulators discharge automatically as the reactor coolant pressure decreases. The charging pumps' normal suction and discharge paths are isolated and a suction path is opened from the RWST. A new discharge path through the boron injection tank is simultaneously opened. As the water level in the RWST decreases to the designated extreme low level initiation point, any two out of the four low level transmitters, coincident with a safety injection signal, will energize and initiate the automatic transfer sequence.

Immediately upon receipt of the transfer signal, valves MOV-SI885A, B, C and D on the LHSI pump minimum flow and test lines begin to close. Miniflow is the difference between flow for a given pump head and the measured flow developed at the header. This ensures that potentially radioactive containment sump water is not returned to the RWST.

A simultaneous signal is sent to open valves MOV-SI863A and B, which ensures a continuous LHSI pump discharge flow path even if reactor coolant pressure is still relatively high (i.e., a small break).

Additionally, valves MOV-SI860A and B in each of the two suction lines from the containment sump are signaled to open. These valves are normally closed during plant operation and an alarm occurs when they are not fully closed.

Valves MOV-CH115B and D are normally closed during plant operation but were opened on either low-low level in the volume control tank or on the initial safety injection signal. In order to ensure a continued suction source for the charging pumps, these valves remain open after the initiation of the transfer signal. However, after a 30 second time delay, the valves are signaled to close to provide redundant (with check valve SI-27) isolation of the LHSI pump discharge from the RWST.

In order to guarantee the continued availability of borated water to the LHSI pumps, valves MOV-SI862A and B remain open following initiation of the transfer signal. After a 2 minute time delay, the valves are signaled to close to preclude sump water from reaching the RWST.

The system flow paths and valve positions are shown in the safety injection mode and after completion of the transfer to cold leg recirculation in Figures 6.3-8 and 6.3-9, respectively. Table 6.3-1 contains a single failure analysis.

Approximately 6.5 hours after the loss of coolant accident, the cold leg only recirculation mode will be terminated and the simultaneous cold leg and hot leg recirculation mode initiated.

The change-over is accomplished by realigning the LHSI pumps to deliver to the hot legs. During the change-over, the HHSI pumps will continue injection to the cold legs.

6.3.4 Inspections and Tests

6.3.4.1 Inspection

Quality standards of safety injection system components are presented in Table 6.3-4.

All components of the ECCS system are inspected periodically to demonstrate system operability.

The pressure containing components are inspected for leaks from pumps seals, valve packing, flanged joints and safety valves during system testing.

6.3.4.2 Tests

Preoperational Component Testing

Preoperational performance tests of the components were performed in the manufacturer's shop. The pressure containing parts of the pumps were hydrostatically tested in accordance with ASME, Boiler and Pressure Vessel Code, Paragraph UG-99. Each pump was given a complete shop performance test in accordance with Hydraulic Institute Standards. The pumps were run at design flow and head, shutoff and head and at additional points to verify performance characteristics. NPSH was established at design flow by means of adjusting suction pressure for a representative pump. This test was witnessed by qualified Westinghouse and licensee personnel.

The remote operated valves in the ECCS are motor or air operated. Shop tests for each valve included a hydrostatic pressure test, leakage tests, a check of opening and closing time and verification of torque switch and limit switch settings. The ability of the operator to move the valve with the design differential pressure across the gate was demonstrated by opening the valve with an appropriate hydrostatic pressure on one side of the valve.

The recirculation piping and accumulators are initially hydrostatically tested at 150 percent of design pressure.

Pre-operational System Testing

After hot functional testing and prior to initial fuel loading, the ECCS was operationally tested. These tests included individual pump full flow tests, accumulator operation and complete system operational flow tests, with the reactor head removed. The purpose of this test was to demonstrate the proper functioning of the instrumentation and actuation circuits and to evaluate the dynamics of placing the system in operation. Water was supplied from the refueling water storage tank for this series of tests. Actuation of the pressurizer low level and pressure signals initiated the automatic startup of the ECCS.

The operability of the accumulators was checked by closing the stop valve, raising the pressure in the accumulator and then opening the stop valve and observing the accumulator level change to provide indication of system delivery.

For those portions of the ECCS and CVCS containing concentrated boric acid, the preoperational test of the heat tracing system included review of manufacturers' design and field testing of all areas, using thermocouples to verify the adequacy of the installations.

A complete check was also made of the controls and alarms of each redundant circuit.

Tests During Refueling Shutdowns

Testing can be conducted to demonstrate proper automatic operation of the ECCS. The test of the safety injection charging pumps can use the minimum flow recirculation lines which return to the volume control tank and can also be tested at any time during station operation due to their dual charging and ECCS function. A test of the low head safety injection pumps can use the recirculation lines which return to the refueling water storage tank.

The operation of the remote stop valves in the accumulator discharge line are tested by opening the remote test valves in the test line between the remote stop valves and the check valves. Flow through the test line is measured and the opening and closing of the discharge line stop valves is verified by the flow instrumentation. The test line can also be used to check leakage through both the check valves and to ascertain that these valves seat whenever the reactor system pressure is raised.

Since the isolation valves are open with power removed during plant operation, and required periodic surveillances verify that power to the valve motor operators is removed, the valves are open, and no position alarms are present, the possibility of inadvertent closure of the isolation valves is eliminated. Because the valves are energized for only brief intervals to change valve position during startup or shutdown, the automatic features to assure the valves will open when required serve no useful function. Thus, the auto open and blocking of closing on an SI signal and auto open on RCS pressure greater than 2000 psig are not required to be tested.

Testing During Normal Operation

Each active component of the ECCS may be individually actuated at any time during operation of the unit to demonstrate operability.

The CVCS charging pumps serve as the high head safety injection pumps. As such, the operability of at least one pump is demonstrated by continuous charging operation while the unit is at power. Demonstration tests can be performed at other times on the other two pumps while charging with the third by employing the minimum flow recirculation line which returns to the volume control tank.

The tests of the low head safety injection pumps employ the minimum flow recirculation test line which returns to the refueling water storage tank. Remotely operated valves are exercised and actuation circuits are tested during this flow test.

The accumulator pressure and level are continuously monitored during station operation and any RCS leakage past the accumulator check valves can be checked at any time using test lines.

The accumulators are maintained full of borated water while the unit is in operation. The accumulators are refilled with borated water from the refueling water storage tank by using the positive displacement hydrotest pump.

The concentration in the accumulators is checked periodically by sampling.

On January 11, 2008, the NRC issued Generic Letter 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems." Generic Letter 2008-01 requested licensees to evaluate the licensing basis, design, testing, and corrective action programs for the emergency core cooling, decay heat removal, and containment spray systems.

As a result, the company performed evaluations that included the review of gas susceptible piping locations, the development of activities to monitor various piping locations as appropriate based on industry experience and plant specific experience, and acceptance of some generic locations that normally accumulate voids that do not adversely affect the design function(s) of the system.

The company established a gas accumulation prevention and management program to ensure that gas accumulation is reasonably prevented or maintained less than the amount that challenges the functionality of the applicable systems and that appropriate action is taken when conditions adverse to quality are identified.

References for Section 6.3

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17. "ASTM Reference for Radiographs for Heavy-Walled [2 to 4 1/2 inch (51 to 114 mm)] Steel Castings," ASTM E-186, The American Society for Test Materials.

18. "ASTM Reference Radiographs for Heavy-Walled [4 1/2 to 12 inch (114 to 305 mm)] Steel Castings," ASTM E-280, The American Society for Test and Materials.

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28. "ASTM Specification for Wrought Austenitic Stainless Steel Piping Fittings," ASTM A-403, The American Society for Test and Materials.

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32. The term "Equivalent" is described in UFSAR Section 1.8.2, "Equivalent Materials."

6.4 CONTAINMENT DEPRESSURIZATION SYSTEM

6.4.1 Design Bases

The containment depressurization system is composed of two groups of subsystems: the quench spray subsystems and the recirculation spray subsystems. These systems are designed to provide the necessary cooling and depressurization of the containment after any LOCA.

The subsystems are designed in accordance with 1971 General Design Criteria 38 through 43 of Appendix A to 10CFR50. Operating together, these subsystems cool and depressurize the containment to less than ½ of the peak calculated pressure within 24 hours following the Design Basis Accident (DBA), assuming the operation of at least minimum engineered safety features as defined in Section 14.3.

In addition, the recirculation spray subsystems are capable of reducing and maintaining the containment pressure in the containment for an extended period following the DBA.

Equipment in the containment depressurization system are designed according to the code criteria and earthquake criteria specified in Section 6.2 and 2.5, respectively.

6.4.2 Description

The quench and recirculation spray subsystems are sized to satisfy the following design bases:

1. The containment shall be depressurized (to less than ½ of the peak calculated pressure within 24 hours) following a LOCA.

2. The LOCA to be considered is a double-ended rupture of a reactor coolant pump suction line.

3. The temperature of the RWST is in the range of 45°F to 65°F.

4. The inlet temperature of the cooling (river) water for the recirculation cooler is at a maximum of 90°F.

5. The minimum engineered safeguards configuration (one 360 degree quench spray header and two 180 degree recirculation spray headers) shall be used.

6. Initiation of a recirculation spray subsystem shall be delayed to ensure adequate water in the containment sumps and to provide high containment back pressures to enhance ECCS effectiveness during reflood.

7. The quantity of buffer provided for dissolution and transfer to the recirculation spray subsystem shall be sufficient to maintain the containment pH of 7 or greater during the recirculation phase of a LOCA response.

The function of the sprays is to remove heat from the containment atmosphere by convection and condensation. Heat is transferred out of the containment by means of the recirculation coolers. Table 6.4-2 lists the design parameters for components of the heat removal system.

All parts in contact with the pumped fluids are made of austenitic stainless steel. The tube side fouling factor is assumed to be 0.0005. The shell side fouling factor is assumed to be zero, as the shell will be laid up dry. Any shell side fouling occurring during service is more than compensated for since the heat duty required of the cooler decreases rapidly with time.

The fluid flow paths through a recirculation spray cooler is shown in Figure 6.4-4.

Quench Spray Subsystem

Two quench spray subsystems, shown in Figure 6.4-1A, are made up of two separate parallel subsystems, each of 100 percent capacity. Each of these subsystems draws water independently from the refueling water storage tank and supplies it to a quench spray header located inside the containment. In addition, quench spray water is also directed to the suction of the outside and inside recirculation spray pump at a nominal flow rate of 300 gpm and 150 gpm respectively. Component design data for the quench spray subsystems are given in Table 6.4-1.

During design of the containment spray subsystem, it was assumed that the quench spray pumps became effective momentarily after receipt of a containment isolation phase B signal. The quench spray delay was intended to provide time for diesel generator startup, pump startup, discharge valve opening time and piping fill time. Analysis of the quench spray subsystem showed that up to 43.9 seconds may be required from receipt of a CIB signal to achieve rated flow through the nozzles.

The quench spray will have a pH of no lower than 4.6 based on a maximum boron concentration of 2600 ppm. The final pH in the containment after a LOCA, considering acid production, the contents of the RWST, and the sodium tetraborate decahydrate buffer baskets, is 7 or greater.

A piping loop seal has been added to each of the quench spray flow paths which are maintained full during normal operation. This prevents air ingestion into the suction of the Recirculating Spray and Low Head Safety Injection pumps.

The ultimate pH of the containment sump is determined from the boric acid concentration of the reactor coolant system, RWST, boric acid addition tank and the three safety injection accumulators, plus the dissolution of sodium tetraborate decahydrate buffer.

No provisions for monitoring or adjusting the pH of the sump water are necessary. The only way in which the pH of the sump water could be altered would be leakage of river water at the recirculation coolers. This is prevented by maintaining the recirculation water pressure greater than the river water pressure. Any possible radioactive outleakage through a heat exchanger would be detected by the use of the four recirculation spray heat exchanger river water radiation monitors.

The RWST is a vertical cylindrical tank with flat bottom and hemispherical top, mounted on and secured to a reinforced concrete foundation. Maximum tank volume is approximately 441,000 gallons. The tank is fabricated of ASTM A-240, Type 304L, or equivalent,[(6)] stainless steel plates. Operational parameters for the RWST are provided in the Technical Specifications and Licensing Requirements Manual.

The water in the RWST is cooled prior to initial unit startup, during unit operation, and following refueling operations, to a temperature corresponding to the requirements of the Technical Specifications and Licensing Requirements Manual by circulating it through a heat exchanger which uses chilled water to remove heat. The tank is insulated to limit the temperature rise of the water to 1/2°F or less per 24 hour period. During normal plant operation, after the water in the tank has been cooled initially by the RWST coolers, it is maintained at the required temperature by circulation through either of two mechanical refrigeration units. Freezing in the RWST during cold weather periods is prevented by the RWST insulation and by maintaining the RWST temperature above the minimum temperature specified in the Technical Specifications and Licensing Requirements Manual. Heat tracing is provided on all lines exposed to the weather.

During normal operation, RWST water may be processed through the fuel pool cooling and purification system to meet water chemistry requirements and/or to reduce radiation levels.

The tank is provided with a manhole for access into the tank for inspection during the refueling periods.

The tank is designed as a Seismic Category I component to withstand design seismic loadings. An evaluation is made to establish that there is no loss of function following the design earthquake conditions. The connecting piping is also designed to withstand seismic loading, to ensure the functioning of the system.

Figure 1.2-1 shows the refueling water storage tank (RWST) in relation to adjacent facilities. The following protective measures have been taken to prevent the tank from being functionally degraded by missiles generated by rotating equipment, fires, explosions and the failure of adjacent nonseismic structures:

1. The RWST has concrete shielding completely surrounding the tank to El. 762 ft 6 inches. The chemical addition tank pump and the chemical injection pumps now abandoned-in-place, were the only rotating equipment in this area. Therefore, the need for missile protection from rotating equipment has been eliminated.

2. Degradation of the RWST from fires is considered highly unlikely. The transformer located west of the RWST has a sprinkler system provided to handle accidental fire. No combustible materials are stored in this area. The concrete shielding, metal covering and insulation provide protection of the tank from fire.

3. The nonseismic structures in the vicinity of the RWST are the service building and warehouse. These structures will collapse in place during a seismic disturbance. The distance from these structures and the concrete shielding provides adequate protection.

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Lines connected to the RWST that are essential in attaining and maintaining a safe shutdown during accident conditions are 12"-SI-1-153W-Q3, 12"-QS-1-153B-Q3 and 12"-QS-2-153B-Q3. All essential piping from the RWST is routed through missile protected pipe trenches before entering the safeguards or the cable vault structures. All lines in unheated spaces have insulation and heat tracing provided to prevent freezing of the pipes. All lines connected to the RWST have valves that can be either manually or remote manually closed in case of line rupture. Redundant level alarms in the main control room will annunciate when water level drops below normal liquid elevation.

Each quench spray pump alone is capable of supplying about 2,000 gpm of borated water to separate 360 degree quench spray ring headers located approximately 96 ft. above the operating floor in the curved section of the dome. The quench spray ring headers have a centerline diameter of about 67 ft. and a total of 196 nozzles per header.

Each of the quench spray headers has 118 Spraying Systems Co. Type 1/2-B40 nozzles. These nozzles produce a relatively fine hollow cone spray pattern. These nozzles are positioned to spray downward and have a spray angle of 70-80 degrees. There are also 40 quench spray nozzles that are plugged on each header. The remaining 78 nozzles, on each quench spray header, are Spraying Systems Co. Type 1713A nozzles. These nozzles are positioned to spray upward at an angle of 22.5 degrees from horizontal. The spray nozzles are positioned so that there are two nozzles side by side with a 10.75 inch distance between them. A 21.5 inch space is then provided on the header before the next two nozzles are located. This symmetrical configuration is utilized the entire distance of the quench spray header.

The mean equivalent diameter of the quench spray droplets is less than 1,000 microns. The quench spray pumps are located adjacent to both the containment structure and the RWST. Each quench spray pump discharge line contains a weight-loaded check valve inside the containment. One-half inch drain lines are located immediately after the check valves which are provided with normally closed double isolation valves. These lines provide flow path to containment sump, normally used during refueling outage. A stainless steel strainer is provided in the discharge of each quench spray pump.

The quench spray pumps shall be flow tested at the frequency specified in ASME Section XI. A deviation from flow rates and discharge pressures, as previously determined during preoperational tests, will indicate either particulate buildup in the strainers or clogging of the test spray nozzles (smallest size nozzles orifice) in the refueling water storage tank. Strainers and test nozzles are easily removable if cleaning is required. The strainers located on the discharge of the quench spray pumps have a 4 mesh per inch (0.063 inch diameter wire) outer basket and a 20 mesh per inch (0.014 inch diameter wire) inner basket.

Recirculation Spray Subsystem

Each of the four recirculation spray subsystems, shown in Figure 6.4-1B, consists of a recirculation spray pump and a recirculation spray cooler and feeds a 180 degree spray ring header located approximately 80 ft above the operating floor. Component design data for the recirculation spray subsystems is given in Table 6.4-2. Analysis of the recirculation spray subsystem showed that approximately 65 seconds may be required from receipt of a containment isolation phase B signal in conjunction with a RWST low level signal to achieve rated flow through the nozzles. This reflects the time required to fill the system piping and deliver rated flow. The total delay in the recirculation spray effective time was selected to provide sufficient sump level to submerge the containment sump strainers.

Two of the recirculation spray ring headers have a radius of 49 ft-3 inches. The other two have a radius of 50 ft-3 inches. Each of the headers has 195 fittings for spray nozzles, with each fitting having two spray nozzles. Ninety-eight of these fittings have two Spraying Systems Co. Type 1/2-B60 nozzles. These nozzles are similar to the quench spray 1/2-B40 nozzles in having a relatively fine spray, but have a larger orifice. One nozzle per fitting is positioned to spray vertically downward while the other is positioned to spray horizontally toward the center of the containment.

Each of the remaining 97 fittings per header is equipped with one Spraying Systems Co. Type 1/2-B60 nozzle and either a Spraying Systems Co. Type 1713A nozzle or a plug. Per header, there are 64 Spraying Systems Co. Type 1713A nozzles and 33 plugs. The 1/2-B60 nozzle is positioned to spray horizontally toward the center of the containment while the 1713A nozzles are positioned to spray upward at an angle of 45 degrees to the horizontal toward the center of the containment. The 1713A nozzles on the lower headers are provided with 9 inch extensions so that the spray can clear the upper header. The two nozzle arrangements are positioned alternately on each recirculation spray ring header. The mean equivalent diameter of the recirculation spray droplets is less than 1,000 microns.

The four recirculation spray pumps take suction from the containment sump, which is protected from debris by a strainer assembly. The strainer assembly consists of two strings of strainer modules which provide approximately 3400 square feet of strainer area through the use of 1/16" perforations. The two strings of modules provide clean containment water flow to a channel box. The channel boxes are connected to each of the module strings and are connected to a common channel box which enters the suction box surrounding the containment sump. The containment sump strainer assembly provides protection for both trains of the recirculation spray pumps. A passive single failure is not assumed credible for the strainer assembly as it meets the requirements of Section 1.3.1 for passive failure exclusion criteria. Figure 6.4-2 shows the containment sump strainer assembly.

The recirculation spray subsystem provides for containment sump pH control. Prior to entering the containment sump, water in the containment basement infiltrates six sodium tetraborate decahydrate baskets to dissolve a predetermined quantity of buffer. The baskets are passive devices which permit the post-LOCA containment pH to be maintained at 7 or greater by virtue of the subsystem's recirculation action. Table 6.4-2 provides data for the buffer baskets.

Two of the recirculation spray pumps and associated motors are located outside the containment and two pumps and motors are located inside the containment. The four pumps are of the vertical deep-well type. The outside pumps have shaft extensions to permit locating the pump suctions at a level below the containment sump, with the motors at an elevation slightly above grade. Each pump has a capacity of approximately 3,300 gpm.

The recirculation spray pump motor supports are located at El. 702 ft 10 5/8 inch inside containment. The maximum post accident water level in the containment is at El. 698 ft 9 inch.

The recirculation spray pump motors have been selected to ensure their operation under accident conditions.

The outside recirculation spray pumps are fitted with a tandem mechanical seal arrangement. The tandem mechanical seal arrangement provides a positive seal against leakage of radioactive fluid from the seals of the outside recirculation spray pumps. The seal arrangement also provides adequate lubrication and cooling for the seals. Level alarms are provided to ensure that adequate accumulator water volume is available and to indicate the failure of either seal.

The seal arrangement consists of two mechanical face seals (Items No. 2 and No. 7), arranged to enclose a non-radioactive seal fluid. After the pump is started, the pressure between the seals is maintained higher than the pressure outside either seal by an accumulator with a weight loaded diaphragm. The accumulator is fitted with two sealed level switches and alarms for annunciation in the main control room. The seal fluid is cooled by being pumped by the pumping ring (Item No. 6) through an air cooler. The accumulator is conservatively sized to allow for sufficient volume of demineralized water outleakage during operation of the pump.

The total volume of the seal water system accumulator is 2.6 gallons. The volume in the lower section of the seal chamber between level alarms is 1,085 ml. The normal outleakage rate, obtained from Crane Packing Company, is 0.26 ml per hour. Based on the normal outleakage rate and the volume of the lower section of the accumulator between level alarms, the recirculation spray pump could be operated for 5.8 months without requiring a supply of makeup water to the accumulator. A seal water accumulator low level alarm at this point will warn the operator of the pending loss of seal water (1.75 months remaining of seal water). A valved hose connection is provided on the seal piping in order to fill the accumulator and associated piping from nearby demineralized water hose connections. Operating the pumps without seal water will cause eventual seal failure and subsequent loss of radioactive fluid to the safeguards area. However, a recirculation spray pump will not be operated without seal water. Twenty-four hours after a DBA only one recirculation spray pump is required for intermittent operation in order to remove decay heat being generated in the reactor core to maintain near atmospheric conditions in the containment. Sufficient time is available to allow for refill of one outside recirculation spray accumulator or use of one of the other three recirculation spray pumps.

The seal water accumulator and piping are constructed of stainless steel and designed for service at peak operating conditions. The accumulator is designed to ASME Boiler and Pressure Vessel Code, Section III, Class C.

The space between the seal faces is maintained at a pressure greater than the recirculation water with demineralized water as seal fluid, thus preventing leakage of the recirculation water, which may be radioactive.

The recirculation water flows through recirculation spray coolers, where it is cooled by river water (Section 9.9).

The entire containment depressurization system is constructed of corrosion resistant materials, primarily stainless steel. However, other materials are used where suitable, such as brass for the quench spray nozzles. The system has a 150 psig design pressure.

6.4.3 Evaluation

The containment depressurization system consists of two 100 percent capacity quench spray subsystems and four 50 percent capacity recirculation spray subsystems. The use of separate spray headers on the discharge of each pump allows for the selection of spray nozzle sizes (0.516 inch, 0.360 and 0.375 inch) to give optimum combination of small spray particles for maximum heat transfer and larger particles for better coverage toward the center and sides of the containment.

The components of the containment depressurization system have been selected so that the conditions of service (pressure, temperature and fluid composition) do not prevent the system from performing its intended function.

The methods of preventing the plugging of spray nozzles in the two systems differ. For the quench spray subsystems, the materials of construction, as well as the pump discharge strainer, prevent nozzle plugging. A method of nozzle testing is provided in the refueling water storage tank to ensure that no particulates which could plug the quench spray nozzles collect in the refueling water storage tank. A temporary strainer is provided in the quench spray pumps suction piping for initial system cleanout. Despite this and regardless of screen opening, some type of particle could conceivably pass lengthwise through the screen and cause clogging of a spray nozzle. However, since the final screen opening is smaller than the smallest spray nozzle size, which is 0.360 inch, such an occurrence is considered to be highly improbable. For the recirculation spray subsystems, the screen assemblies in the containment sumps are arranged with a common containment sump strainer assembly (Section 6.4.2). The containment sump strainer is designed in response to NRC issued Generic Letter 2004-02. Sufficient area has been provided to insure that the specific BVPS-1 debris strainer loading does not impair system operation. The containment sump strainer provides protection for both trains of the Recirculation Spray System. A passive single failure is not assumed credible for the strainer assembly as it meets the requirements of Section 1.3.1 for passive failure exclusion criteria.

During normal unit operation, the recirculation spray coolers are dry on the shell side and filled with river water with corrosion preventatives on the tube side as described in Section 9.9.2. For long term operation, on the order of weeks, there may be some fouling of the tubes on the river water side, with resultant loss in heat transfer capability. The loss of heat transfer capability will be more than offset by the decrease in necessary heat load due to decreasing decay heat production. One day after a loss-of-coolant accident, the drop in decay heat is such that one pump and heat exchanger has sufficient heat-removal capacity to hold the containment depressurized. With an expected maximum river water temperature of 90°F, the recirculation spray subsystem design is conservative, with a minimum 100 percent backup capacity at the onset of an accident. Within one day after the LOCA, the backup capacity exceeds 400 percent.

The recirculation spray coolers have welded construction at all points where there is a potential for leakage of radioactive recirculation water into the river water. The maximum pressure differential which can occur between the river water and the recirculation water is 150 psi; under these conditions, leakage flow from the recirculation spray subsystems is toward the river water system. The river water is monitored for leakage by means of radiation monitors. The defective subsystems are shut down if leakage above the allowable values (within the limits of 10CFR20) is detected.

The recirculation spray coolers low point piping or shell drain valve caps have 1/8 inch holes to permit drainage of any liquid buildup, thereby eliminating any stagnant boric acid and potential stress corrosion cracks. The addition of 1/8 inch holes to the recirculation spray coolers drains has been analyzed and determined to have no effect on the spray function of the recirculation spray system.

Because two of the recirculation spray pumps and associated valves and piping are outside the containment, a potential danger of radioactive leakage exists. Valves have been selected to reduce this leakage potential to a negligible amount. To protect against large leaks the following is done:

1. On the discharge of the recirculation spray pumps, gross leaks are detected by variations in the pump discharge pressure readings in the main control room.

2. Large leaks in the suction piping in the safeguards area are detected by liquid level measuring devices. The floor of the safeguards area is provided with baffles, dividing the floor into two sections. Thus, leakage from recirculation suction lines is detected by the increased liquid level on the affected section of the floor. Upon detection, the operator in the main control room remote-manually isolates the leaking pipe, leaving three recirculation spray loops operating. In the case of small leaks, detection and isolation of leaks are not possible; however, the gaseous contents of the structure (Figure 6.4-3) enclosing the piping outside the containment are normally discharged to the atmosphere via the elevated release point in the supplementary leak collection and release system (Section 6.6).

Consistent with letters from the ACRS[1] concerning vital piping which must function during a LOCA, passive failure of the recirculation spray pump suction piping during a LOCA is not considered credible during the injection phase of the LOCA.

The valves located in the safeguards valve pit area are the outside recirculation spray pump suction isolation valves, outside recirculation spray bypass isolation valve and the low head safety injection pump suction isolation valves.

The safeguards area suction valve pit is shown in Figure 6.1-1 and in Figure 5.1-5 at El. 687 ft-11 inches. Valve operators are located at El. 747 ft in the safeguard area structure. Valve extensions are totally enclosed and water tight.

The valve pit structure is attached directly to the reactor containment mat and lower portion of the shell. Porous concrete is placed below the pit and against its three sides. The pit is designed for seismic and hydrostatic loading in addition to dead and live loads. The pit pump casings are completely enclosed by the butyl waterproof membrane surrounding the reactor containment, up to El. 730 ft. where it terminates between the bottom mat of the safeguard building and the containment shell.

The concrete which surrounds the safeguards area is sealed to prevent entry of ground water into the safeguards area. The seal membrane also serves to prevent leakage of any recirculation water from the safeguards area into the earth backfill between the cofferdam and the containment.

The inside and outside recirculation spray pumps are capable of meeting NPSH requirements under LOCA conditions.

A transient analysis of the available NPSH is performed with the MAAP-DBA computer code described in Section 14.3. Some modifications are made to the assumptions in the Section 14 containment pressure analysis to provide conservatism in the NPSH analysis. Generally, these assumptions tend to maximize sump water temperature and minimize containment pressure.

Inputs and assumptions are also chosen to minimize the available NPSH and are based on the results of a sensitivity study which determines the direction of limiting bias for key inputs.

The major model differences are as follows:

1. Break fluid flashing is modeled by a pressure flash.
2. Spray thermal effectiveness is assumed to be 100 percent.
3. Maximum engineered safety features are assumed operable.

As shown in Figure 6.4-10, the minimum available NPSH is 14.7 ft. for the inside recirculation spray pumps and 14.4 ft. for the outside recirculation spray pumps. The minimum available NPSH occurs following a hot leg double-ended rupture at approximately 400 seconds after the pumps start. The deep well pumps require approximately 9.8 ft NPSH at a flow of 3400 gpm. Lines diverting cold discharge water from the quench spray headers are provided to assist in ensuring the NPSH requirement is met during DBA conditions.

The containment depressurization system provides the capability to thoroughly mix the containment atmosphere and minimize the possibility of hydrogen accumulating in the containment subcompartments and escaping the mixing action of the spray water. The internal design of the containment structures allows air to circulate freely. All cubicles and compartments within the containment are open at the top and allow air circulation. Convective mixing in conjunction with containment spray assures a uniform mixture of hydrogen in the containment. BVPS thus complies with 10 CFR 50.44(b)(1).

Provisions for sampling the containment atmosphere following a LOCA are provided.

A failure mode analysis for the components of the containment depressurization system is included in Table 6.4-3.

The air recirculation cooling system may be used to limit the buildup of pressure in the containment structure for small reactor coolant system (RCS) and steam line leaks which are not sufficient in magnitude to automatically initiate operation of the containment depressurization system.

The containment depressurization system is automatically initiated upon receipt of a containment isolation phase B signal. The operator does not have the option of preventing the actuation of containment spray system through the use of the air recirculation cooling system.

Portions of the containment depressurization system can be bypassed for testing.

Containment depressurization system pumps and valves are assigned to ESF actuation slave relays and are tested as described in Section 7.3. Assignments of the containment depressurization system components to these relays, which are individually tested one at a time, are made so that the containment depressurization is not inadvertently initiated due to testing and also so that redundant devices remain functional during the testing.

Post test requirements followed to ensure availability for activation for the components grouped on common test-table relays are established in approved BVPS test procedures.

The requirement for performance of these tests is that minimum engineered safety features are available at all times during testing.

Electrical interlocks which prevent the operator from tripping the spray pumps inadvertently or prematurely from the main control room are accomplished by use of a control switch trip action blocking contact. Upon the receipt of a containment isolation phase B signal, a contact from the pump motor starting timer produces this signal instantaneously.

The containment spray pumps suction and discharge valves are protected from inadvertent closure by a control room operator during pump operation by either a normally closed pump switchgear auxiliary contact placed in the closing circuit of the motor-operated valve, or by access covers placed over the benchboard control switch.

To deactivate any containment spray pump would require two operator actions: (1) manually reset the containment isolation phase B train signal associated with the equipment and (2) place the control switch for the pump in the stop position.

Above the operating floor (El. 767 ft-10 inches), the quench sprays cover a volume of approximately 431,102 cu ft and the recirculation sprays cover a volume of approximately 529,625 cu ft. The total free volume above the operating floor is approximately 1,030,000 cu ft. Because of the forced circulation set up by the sprays, the entire volume above the operating floor is considered to be uniformly mixed and scrubbed by the sprays.

Below the operating floor, the quench sprays cover approximately 55,443 cu ft and the recirculation sprays cover approximately 219,711 cu ft. The total free volume below the operating floor is approximately 724,000 cu ft.

All of the subcompartments, with the exception of the volume below the reactor vessel, the refueling cavity and the incore instrumentation passage, are well vented and scrubbed by the sprays. The following regions are not directly covered by sprays:

Region	Area (cubic ft)
Incore instrumentation passage	6,500
Volume below reactor vessel	2,000
Volume below refueling cavity	21,750

Note that this volume is a portion of the base mat floor. As such, it has no side walls, is not enclosed and is subject to good mixing with the main spray volume.

The spray patterns for the quench and recirculation sprays are depicted in Figure 6.4-8. These spray patterns are based on minimum engineered safeguards.

The degree of mixing between the regions covered by the quench and recirculation sprays can be estimated by the amount of overlap of coverage by the quench and recirculation sprays and from the forced circulation set up by the containment sprays. The volume above the operating floor which is covered by overlapping quench and recirculation sprays is 253,878 cubic ft. The mixing from forced circulation induced by the sprays may be visualized by inspection of Figure 6.4-9, which shows a simplified diagram of the air flow in the containment set up by the sprays.

Consider that the containment volume is split into two concentric cylindrical regions. In the outer region, air is entrained by sprays and forced downward. In the inner portion, the air moves upward. This circulation pattern is augmented by rising air currents from hot components, such as the reactor pressure vessel, steam generators and pressurizer. The effectiveness of the sprays in mixing the containment atmosphere has been verified by BNWL[2] in which measurements, made at different points in the containment systems experiment vessel (similar to the BVPS-1 containment) for iodine and noble gases, showed that iodine was completely distributed throughout the containment, even at the high points in the vessel.

The design of the containments and containment spray systems for BVPS-1 and BVPS-2, are similar to those of the Surry Power Station, Units 1 and 2, for which it has been concluded by the Atomic Energy Commission[3] that the containment spray systems provide adequate mixing of the containment atmosphere in the post LOCA environment.

A measure of the mixing between subcompartments below the operating floor (such as the steam generator cubicles and the pressurizer cubicle), that are covered by the sprayed volume, may be determined by the natural circulation set up by the difference in air/steam densities between the subcompartment volumes and the containment volume covered by the sprays. The flow due to "stack effect" can be computed from the following equation:[4]

$$Q = 7.2\,A\,\sqrt{h(t_i - t_o)} \qquad (6.4\text{-}1)$$

where:

- Q = air flow, cubic feet per minute
- A = free area of inlets or outlets (assumed equal; if different, the smaller of the two is used), sq ft
- h = height from inlets to outlets, ft
- t_i = average temperature of inside air, °F
- t_o = temperature of outside air, °F
- 7.2 = constant of proportionality, including a value of 50 percent for effectiveness of openings

From the above equation, circulation rates ranging from 1,500 cfm to 61,000 cfm were computed.

In addition to the mixing of the containment atmosphere with the sprays due to natural and forced convection, there will be direct coverage by the sprays in the form of "rain" falling through the opening in the operating floor into the volumes below. No credit is taken in the analysis for iodine removal below the operating floor by sprays falling in the form of rain.

The design pressure drop across the quench spray nozzles is 40 psi.

Estimated Leakage During Recirculation

A periodic inspection is made of all potential points of leakage. Table 6.4-4 summarizes the potential leakage from the recirculation spray subsystems.

Leakage of a pumped fluid from the outside recirculation spray pumps cannot occur, due to the manner in which the pump shafts are sealed. Two mechanical seals are arranged in tandem with a seal fluid between them. The seal fluid is supplied from a reservoir arranged in such a manner that the pressure of the seal fluid is slightly above (about 1 psi) the pumped fluid pressure at the inboard side of the inboard seal. This arrangement means that, assuming a single seal failure, seal fluid leaks through the failed seal and the other seal remains available to prevent escape of pumped fluid to the atmosphere. A level alarm on the reservoir provides indication of seal failure.

6.4.4 Tests and Inspections

After the quench spray subsystems have been completely installed, temporary connections are made on the spray headers and pipe plugs placed in the spray nozzle sockets and header drain lines. The quench spray pumps are started and tested, circulating water through the spray header supply lines to the spray headers and out the temporary test connections. This is to provide a system capability test to ensure that the system meets flow requirements. It also provides a complete flush of the system to remove any particulate matter which could conceivably result in plugging of the spray nozzles at a future time. At the completion of this test, the temporary test connections are removed, the pipe plugs are removed and the spray nozzles are installed. The subsystems then are ready for operation.

With a complete system flush to remove all particulate matter prior to the installation of spray nozzles and with corrosion resistant nozzles and piping, it is not considered credible that a significant number of nozzles could become plugged during the life of the unit to reduce the effectiveness of the subsystems, therefore, no means are provided for intermittent testing of the nozzles using water flow. Nozzle tests are performed following maintenance that results in the potential for nozzle blockage as described below.

Means are provided for intermittent testing of the quench spray pumps. This testing is performed periodically by opening the normally closed valves on the recirculation line returning water to the refueling water storage tank. Initiation of the subsystem allows the pumps to operate and recirculate water back to the tank. The purpose of the recirculation is to test pump operation and to detect the presence of particulate matter in the refueling water storage tank that could clog spray nozzles.

Periodic activation of the containment depressurization signals ensures operability of the initiation signals and control circuitry.

Prior to unit startup, an initial full flow system test with water is performed on the recirculation spray system, as follows:

1. With the nozzle openings blocked off, temporary connections are made between the nozzle headers and the containment sumps.

2. Sufficient water is then added to the containment sump so that a recirculation spray pump can recirculate water up through its respective cooler and header.

3. The full flow test through the shell side initially ensures that the required recirculation spray for containment depressurization is achieved.

4. Upon completion of the above system test, the water is drained from each recirculation spray cooler, the pumps, the headers and the sumps. The temporary connections between the headers and sumps are removed and the nozzles installed.

Since the system is left in a dry, ready condition after the initial full flow tests, no further testing with water flow through the shell side of the recirculation spray heat exchangers is deemed necessary to ensure system capability.

Periodic flow testing of the outside recirculation spray pumps is performed as follows:

1. The pump discharge containment isolation valves are closed

2. The pump casing is filled with water

3. The test line valves are opened

4. The pump is started, taking suction from the casing, discharging through the test line with its installed flow instrument and returning to the casing.

Testing to verify that spray nozzles are not obstructed is performed following maintenance that results in the potential for nozzle blockage. A blind flange in the piping upstream of each spray header is provided for this purpose. Each nozzle is tested individually to verify that there is no flow blockage. Maintenance with the potential for nozzle blockage is determined by engineering evaluation as required by Technical Specifications.

The volume and pH-raising effectiveness of the sodium tetraborate decahydrate buffer is surveilled each refueling outage to ensure the radiological and equipment qualification functions of the containment depressurization system will be fulfilled in the event of a LOCA.

On January 11, 2008, the NRC issued Generic Letter 2008-01, “Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems.” Generic Letter 2008-01 requested licensees to evaluate the licensing basis, design, testing, and corrective action programs for the emergency core cooling, decay heat removal, and containment spray systems.

As a result, the company performed evaluations that included the review of gas susceptible piping locations, the development of activities to monitor various piping locations as appropriate based on industry experience and plant specific experience, and acceptance of some generic locations that normally accumulate voids that do not adversely affect the design function(s) of the system.

The company established a gas accumulation prevention and management program to ensure that gas accumulation is reasonably prevented or maintained less than the amount that challenges the functionality of the applicable systems and that appropriate action is taken when conditions adverse to quality are identified.

<u>References for Section 6.4</u>

1. Letter from Stephen H. Hanauer, Chairman, Advisory Committee on Reactor Safeguards, to Honorable Glenn T. Seaborg, Chairman, U.S.A.E.C., dated May 15, 1969, subject: "Report on Edwin I. Hatch Nuclear Plant" and letter from Stephen H. Hanauer, Chairman, Advisory Committee on Reactor Safeguards, to Honorable Glenn T. Seaborg, Chairman, U.S.A.E.C., dated May 15, 1969, subject: "Report on Brunswick Steam Electric Plant".

2. Hilliard, et al, "Removal of Iodine Particles from Containment Atmosphere by Sprays", BNWL-1244, Battelle Pacific Northwest Laboratory (February 1970).

3. Safety Evaluation by the Division of Reactor Licensing, Atomic Energy Commission in the Matter of Virginia Electric and Power Company, Surry Power Station - Units 1 and 2, Docket No. 50-280 and 50-281 (February 23, 1972).

4. ASHRAE Handbook of Fundamentals, page 344 (1972).

5. "ASTM Specification for Heat-Resisting Chromium and Chromium-Nickel Stainless Plate, Sheet, and Strip for Fusion-Welded Unfired Pressure Vessels", ASTM A-240, The American Society for Testing and Materials.

6. The term "equivalent" is described in UFSAR Section 1.8.2, "Equivalent Materials."

6.5 POST DBA HYDROGEN CONTROL SYSTEM

6.5.1 Hydrogen Recombiner and Purge System

10 CFR 50.44 was revised in 2003 to eliminate the requirement for hydrogen recombiners and hydrogen purge systems. Elimination was based on improved understanding of combustible gas behavior during severe accidents and confirmation that the hydrogen release postulated from a design-basis LOCA was not risk-significant because it was not large enough to lead to early containment failure, and that the risk associated with hydrogen combustion was from beyond design-basis accidents. The revised rule retained the requirement that containments for all currently licensed power plants ensure a mixed atmosphere. Also, it acknowledged the existing "industry-wide commitment" to include combustible gas control strategies in the severe accident management guidelines (SAMGs) as the basis for not including a requirement related to purging and/or venting capabilities. Section 14.3.4.4 addresses the mixed atmosphere requirement and plant SAMGs provide combustible gas control strategies that do not rely on the hydrogen recombiners or hydrogen purge blowers for beyond design-basis accidents. Therefore, there is no requirement to maintain the hydrogen recombiner and purge system.

6.5.2 Containment Hydrogen Analyzer System

The containment hydrogen analyzer system consists of two redundant hydrogen monitors energized from separate Class IE Power supplies to provide protection against single failure and single loss of power. Containment samples are obtained through independent sample lines for each monitor. Indication is provided for each hydrogen analyzer, on the vertical board in the main control room, with an indicating range of 0-10 percent hydrogen. A recorder is provided to record the Train A hydrogen level. The hydrogen analyzer system is designed to provide a continuous positive indication of the containment hydrogen concentration within 30 minutes after the initiation of safety injection. The containment hydrogen analyzer system was added as a requirement of an NRC Order[1] in response to NUREG 0737 TMI Issue II.F.1.

The hydrogen analyzer units are located in the cable vault areas (Train "A" in the west cable vault and Train "B" in the east cable vault). The hydrogen analyzer units are of the thermal conductivity type. Each hydrogen analyzer has sampling lines running to two containment locations. One sample line from each analyzer connects to a dome snorkle line which runs to the top of the containment dome. The second sample line for each analyzer runs to a location high within the pressurizer cubicle. Each of the four sample lines contains silver zeolite filters which remove iodine from the air sample prior to leaving containment. All sample lines are sloped to loop seal drains which allow condensate to drain while preventing air from being drawn in through the drain. Each analyzer has one line for returning samples to containment. There are two normally closed (fail closed) solenoid operated valves on each sample and return line. One valve is located in containment and the other outside containment in the cable vault. The solenoid valves are remotely operated from the main control room with key switches. Valve position indicating lights (open/closed) are provided for each solenoid valve.

The analyzer control panels located in the switchgear area contain supporting electronics, signal conditioning, digital readouts, recorder outputs, caution and high alarm setpoint controls, span gas controls, zero and span adjustments and main system power controls. Span gases are supplied directly to the analyzers to allow for periodic calibration of the units.

Reference for Section 6.5

1. S. A. Varga, Jr. (USNRC) BVPS-1 Order Modifying License, letter to J. J. Carey (BVPS) (July 10, 1981).

6.6 SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM

6.6.1 Design Bases

The supplementary leak collection and release system is designed according to the following criteria:

1. The maintenance of 0.125 inch water gauge negative pressure in most areas contiguous to the containment (with the exception of the main steam valve area, the steam generator blowdown room, the purge air duct area, east cable vault and west cable vault), the waste gas storage area and the fuel building.

2. The filtration by impregnated charcoal of contaminated air for radioactive iodine removal with individual charcoal cells sized for 600 cfm. The filters are not credited in the radiological analysis.

3. The discharge of exhaust air at the SLCRS Vent.

4. The use of redundant filter banks and exhaust fans with the fans operable/functional on emergency power.

5. Equipment and system capability of withstanding the design basis earthquake without loss of function.

6. Removal of heat from areas containing safety related equipment following a design basis accident with loss of offsite power.

7. Maintain the containment contiguous areas below 10 CFR 50.49 radiological limits.

6.6.2 Description

The elements of the supplementary leak collection and release system are shown in Figure 6.6-1. The design data is provided in Table 6.6-1.

One function of the system is to ensure that radioactive leakage from recirculating ECCS systems following a DBA is filtered for iodine removal prior to discharge to the atmosphere at the SLCRS Vent. The filters are not credited in the dose consequence analysis.

Another function of the system is the removal of heat from areas contiguous to containment where equipment important to safety is located (i.e., the charging pump cubicles, the low head safety injection pump cubicles, the recirculation spray pump cubicles, the auxiliary feedwater pump room, the east and west cable vaults, the MCC room, and the safeguards pipe tunnel). The heat removal function is also credited for minimizing the radiation level in these areas. Following a loss of offsite power, SLCRS is the only available means of assuring that components in these areas do not exceed the design temperatures and following an accident, do not exceed the radiological environmental levels associated with the Equipment Qualification (EQ) radiation zones. Temperature switches are provided to open SLCRS dampers when area temperatures exceed 110°F.

The supplementary leak collection and release system consists of two 100 percent capacity leak collection exhaust fans with a design capacity of 50,000 cfm each. Air is exhausted from the fuel building, waste gas storage area auxiliary building, blowdown tank room, personnel access hatch area, purge air duct area, east cable vault, pipe tunnel and north and west safeguards areas and, during refueling, the containment.

The capacity of each fan is in excess of the estimated air inleakage to the containment contiguous areas, fuel building and waste gas storage area auxiliary building. An intake damper is provided at the suction header of the fans to provide makeup air. The excess capacity of the fan ensures a minimum of the required negative pressure in the exhausted areas. Provision is made in the various exhausted areas to permit confirmation that they are maintained at the required negative pressure.

The provision made to confirm the maintenance of required negative pressure areas served by the supplementary leak collection and release system consists of the use of differential pressure gauges. These gauges are portable instruments which indicate pressure in inches water gauge either above or below the prevailing atmospheric pressure. Periodic observation of gauge pressure readings will confirm that the subject areas are maintained at the required negative pressure.

In order to limit air leakage into the above structures to the capacity of a leak collection exhaust fan, penetration pipes, ducts and cables are sealed as required at or near the point where they pass from the contiguous structure to some other structure, e.g., the auxiliary building. Inleakage of air at doors in exhausted structures is limited by gasketing. Doors are either locked or are self-closing and are under administrative control.

One of the leak collection exhaust fans is normally used to exhaust the various areas. The other fan is used as a standby.

During normal operation, the exhaust to the fan does not go through filters. On a containment isolation phase A signal or a high-high radiation signal from monitors in the ventilation exhausts from the fuel building, the waste gas storage area, or from areas contiguous to the containment with the exception of the main steam valve cubicle, the leak collection system exhaust is diverted so that it first flows through one of the two parallel main filter banks before flowing to the leak collection exhaust fans. The other main filter bank is used as a backup. Each main filter bank consists of roughing filters, charcoal filters and pleated glass fiber type HEPA filters. The roughing filters remove large particulates to prevent excessive pressure drop buildup on the charcoal and HEPA filters. The charcoal filters are effective for radioactive iodine removal, and the HEPA filters, at a rated efficiency of 99.97 percent where tested with 0.30 micron diameter of particle, remove particulates and charcoal fines. Each charcoal filter is rated at 600 cfm. Each roughing and HEPA filter is rated at 1,200 cfm. It is of flat parallel tray design, containing approximately 44 lb of charcoal in two 2 inch beds having a total of about 8 sq ft of face area. The media is new impregnated activated coconut shell charcoal. The media particle size is from 8 to 16 U.S. Sieve mesh with not more than 5.0 percent retained on 8 mesh nor more than 5.0 percent passing through 16 mesh. Charcoal cells are leak tested in accordance with Technical Specifications.

Charcoal filter efficiency is based on the attenuation of radioactive iodine with a minimum composite efficiency of 95 percent.

A heat detection and alarm and automatically actuated water spray deluge system is provided for the charcoal cells to maintain cooling of the media and prevent ignition in the event of decay heat buildup.

The leak collection exhaust fans discharge through a duct to the SLCRS Vent. The SLCRS Vent is located on the top of the containment structure. The containment extends 144 ft above grade. The duct and supporting structure is designed to accommodate seismic forces. The release above the containment is a high-velocity discharge at an elevation of 150 ft above grade.

During refueling, the fuel building is normally maintained at negative pressures of 0.125 inches of water column, by the operating leak collection exhaust fan.

During Mode 6, SLCRS ventilation damper VS-D-4-18A may be closed and the upstream containment contiguous areas may not be ventilated.

If there should be a fuel building high-high radiation signal, the exhaust from the fuel building along with the rest of the leak collection system exhaust, may be diverted to one of the main filter banks in the leak collection release system before being released at the SLCRS Vent.

6.6.3 Evaluation

The supplementary leak collection and release system incorporates redundant 100 percent capacity leak collection exhaust fans and main filter banks. In addition, there are redundant dampers where required. The redundant fans and dampers are connected to different emergency buses, which are capable of being supplied either from normal or emergency power sources. Thus, there is sufficient redundancy in the system to ensure system reliability. Proper operation of the system is further ensured by the capability for testing the system periodically.

The instrumentation, control and electrical equipment of the supplemental leak collection system are in accordance with IEEE 279[(1)]-1971 and IEEE 308[(2)]-1971 with one exception.

IEEE 279-1971 requires that the "protection system shall, with precision and reliability, automatically initiate appropriate protective action whenever a condition monitored by the system reaches the preset level". It also requires that this be accomplished with a single failure.

The exhausts from the contiguous areas, (with the exception of the main steam valve cubicle), the fuel building and the waste gas storage area auxiliary building are each monitored by a radiation monitor which automatically diverts the exhaust through a filter path when the radiation reaches a preset level. Should any of these systems fail, a redundant backup radiation monitor, located in the SLCRS Vent duct, would indicate and alarm to the operator that the preset level had been reached, which would alert the operator to manually divert the exhaust through the filters. The automatic and manual systems are redundant and on separate power supplies.

The post accident dose from radioactivity released from the contiguous and waste gas storage areas is expected to be much less than specified by the guidelines described in Section 14.3.5 and Section 11.2.3.4, respectively.

The supplementary leak collection and release system collects, filters and releases at an elevated point (SLCRS Vent) the leakage from the containment following a DBA. Essentially, all the leakage from the containment following a DBA flows into those containment contiguous areas which house the various containment penetrations and the engineered safeguards equipment circulating radioactive water.

Although a negative pressure of .125 inch water gauge in the areas contiguous to Containment is adequate to prevent exfiltration under DBA conditions, no credit has been taken for the collection of containment leakage through any electrical penetration in the LOCA analysis of Section 14.3. Such leakage is modeled as the release of .1% volume per day directly to the environment without regard to the specific location of the containment isolation failure.

Similarly, no credit is taken for collection and filtration by the SLCRS of leakage from ESF piping systems and components that recirculate containment sump water outside the containment. However, a negative pressure of .125 inch water gauge is required for these areas.

6.6.4 Tests and Inspections

All systems are tested and inspected as separate components and as an integrated system. The testing and visual inspection of all equipment and systems is in accordance with the Technical Specifications. Anemometer or velometer readings are taken to ensure that all air systems are balanced to exhaust the required air quantities at design conditions, for the purposes of maintaining negative pressure and removing heat from safety related equipment.

Materials and construction methods for ventilation systems are inspected for compliance with National Fire Protection Association (NFPA) Standard 90A[3] and NFPA Standard 91.[4]

Capacity and performance of centrifugal fans conform to the required conditions and are tested or rated for compliance with Advanced Material Concept Agency (AMCA) test codes and certified ratings program. Filter cells are individually tested after fabrication and again as an integrated system after installation. Filter banks are tested for leakage while in place using methods outlined in the operating Technical Specifications. Defective cells are replaced where test requirements are not satisfied. The frequency of and requirements for the tests are given in the Technical Specifications.

References for Section 6.6

1. "IEEE Criteria for Protection Systems for Nuclear Power Generating Stations," IEEE-279-71, The Institute of Electrical and Electronic Engineers.

2. "IEEE Standard Criteria for Class IE Electric Systems for Nuclear Power Generating Stations," IEEE-308-71, The Institute of Electrical and Electronic Engineers.

3. "NFPA Standard for The Installation of Air Conditioning and Ventilation Systems," NFPA Std. 90A, The National Fire Protection Association.

4. "NFPA Standard for the Installation of Blower and Exhaust Systems for Dust Stock and Vapor Removal or Conveying," NFPA Std. 91, The National Fire Protection Association.

BVPS UFSAR UNIT 1

TABLES FOR SECTION 6

Table 6.2-1

SUMMARY OF REQUIRED EXAMINATION AND TESTS FOR NUCLEAR PIPING

Material	Group Q1	Group Q2	Group Q3
Identification (May be in addition to requirements of the ASTM material specification)			
1. Specification and grade	Required for each length or part.	Required for each length or part.	Required for each length or part.
2. Heat number and piping classification number	Required for each length or part.	Required for each length or part.	Required for each length or part.
Examination and Test			
1. Seamless and welded (without filler metal) tubular products and fittings	In accordance with material specification and complete examination by either of the following: 1. Ultrasonic examination. 2. Eddy Current examination. 3. Radiography plus magnetic particle or liquid penetrant examinations on accessible surfaces. 4. Magnetic particle or liquid penetrant examination on all surfaces, both inside and outside. 5. A combination of these methods.	In accordance with material specification.	In accordance material specification.

Table 6.2-1 (CONT'D)

SUMMARY OF REQUIRED EXAMINATION AND TESTS FOR NUCLEAR PIPING

Examination and Test	Group Q1	Group Q2	Group Q3
2. Welded (with filler metal) tubular products and fittings	In accordance with material specification and examinations as follows: 1. All longitudinal or girth welds made in accordance with ASTM A-155 or A-358 for pipe and the WPW grades of ASTM A-234, A-403 and A-420 for fittings shall be completely examined by radiography. 2. The accessible surface of the welds and heat- affected zone of the base metal adjacent to the welds shall be examined by magnetic particle or by liquid penetrant methods.	In accordance with material specification ASTM A-144 Class 1 or A-358 Class 1 do not require supplementary examination. All welds made with filler metal in fittings as per WPW grades of A-234, A-403 and A-420 shall be completely radiographed.	In accordance with material specification
3. Plates used in fabrication of pipe and fittings	In accordance with material specification and complete examination by ultrasonic method in accordance with ASTM A-435 to cover entire surface of plate.	In accordance with material specification.	In accordance with material specification.

Table 6.2-1 (CONT'D)

SUMMARY OF REQUIRED EXAMINATION AND TESTS FOR NUCLEAR PIPING

Examination and Test	Group Q1	Group Q2	Group Q3
4. Forgings and bar used in fabrication of flanges and fittings (over 2 inches in size)	In accordance with material specification and examination by the ultrasonic method in the as furnished or finished condition and by the magnetic particle or liquid penetrant method in the finished condition.	In accordance with material specification.	In accordance with material specification.
5. Static or centrifugally cast pipe and fittings	In accordance with material specification. Statically cast items shall be examined by radiographic methods. Centrifugally cast products may be examined by ultrasonic methods in lieu of radiography if it can be demonstrated that a meaningful examination can be made. All ferritic steel castings over 12 inches thick shall be examined by both radiographic and ultra- sonic methods. In addition all cast products shall be examined by either magnetic particle or liquid penetrant methods.	In accordance with material specification. Examination to be same as for Group Q1.	In accordance with material specification.

Table 6.2-1 (CONT'D)

SUMMARY OF REQUIRED EXAMINATION AND TESTS FOR NUCLEAR PIPING

Examination and Test	Group Q1	Group Q2	Group Q3
6. Bolting	Visual examination shall be applied after final heat treatment to the areas of threads, shanks and heads of final machined parts. Harmful discontinuities such as laps, seams or cracks that would be detrimental to the intended service are unacceptable. All bolts, studs and nuts greater than 1 inch nominal bolt size shall be examined by the magnetic particle method in accordance with ASTM E-109 and E-138 or the liquid penetrant method in accordance with ASTM E-165. This examination shall be performed on the finished component after threading or on the material stock at approximately the finished diameter before threading and after heading (if involved). Linear nonaxial indications are unacceptable. Linear axial indications greater than 1 inch in length are unacceptable.		

Table 6.2-1 (CONT'D)

SUMMARY OF REQUIRED EXAMINATION AND TESTS FOR NUCLEAR PIPING

Welding	Group Q1	Group Q2	Group Q3
Girth Butt Welds			
a. Radiography	100% required.	100% required.	Not required.
b. Magnetic Particle c. Liquid Penetrant	100% required of (b) or (c).	Not required. Not required.	100% required over 4 inch pipe size of (b) or (c).
Longitudinal Butt Welds			
a. Radiography	100% required.	100% required.	Not required.
b. Magnetic Particle c. Liquid Penetrant	100% required of (b) or (c).	Not required. Not required.	100% required over 4 inch pipe size of (b) or (c)
Fillet and Socket Welds			
a. Radiography	Not required.	Not required.	Not required.
b. Magnetic Particle c. Liquid Penetrant	100% required of (b) or (c).	100% required of (b) or (c).	100% required over 4 inch pipe size of (b) or (c).
Seal Welds			
a. Radiography	Not required.	Not required.	Not required.
b. Magnetic Particle c. Liquid Penetrant	100% required of (b) or (c).	100% required of (b) or (c).	Not required. Not required.

Table 6.2-1 (CONT'D)

SUMMARY OF REQUIRED EXAMINATION AND TESTS FOR NUCLEAR PIPING

Welding	Group Q1	Group Q2	Group Q3
Welded Branch Connections			
a. Radiography	100% required over 4 inch pipe size.	100% required over 4 inch pipe size.	Not required.
b. Magnetic Particle c. Liquid Penetrant	100% required of (b) or (c).	100% required on 4 inch and under of (b) or (c).	100% required over 4 inch pipe size of (b) or (c).
Attachment Welds - Supports, Legs, Anchors and Guides			
a. Radiography	Not required.	Not required.	Not required.
b. Magnetic Particle c. Liquid Penetrant	100% required of (b) or (c).	100% required of (b) or (c).	Not required. Not required.
Miscellaneous Requirements			
1. Welding and brazing materials	Shall conform to requirements of Section IX of the ASME Boiler and Pressure Vessel Code.	Same as for Group Q1.	Same as for Group Q1.

Table 6.2-1 (CONT'D)

SUMMARY OF REQUIRED EXAMINATION AND TESTS FOR NUCLEAR PIPING

Miscellaneous Requirements	Group Q1	Group Q2	Group Q3
	Materials not incorporated in Section IX may be used if a procedure qualification test is successfully made in accordance with Section IX. Materials shall be clearly identified on the package or container and shall remain legible until the material is actually consumed in the process. Identification shall include material specification grade and classification number, supplier's name and trade designation and the control or heat number.		
2. Erected Pipe System			
a. Hydrostatic Test	All installed piping shall be tested by a hydrostatic test at test pressures specified in line designation tables prior to initial operation to demonstrate leak tightness.	Same as Group Q1.	Same as Group Q1.

Table 6.2-1 (CONT'D)

SUMMARY OF REQUIRED EXAMINATION AND TESTS FOR NUCLEAR PIPING

Miscellaneous Requirements	Group Q1	Group Q2	Group Q3
b. Pneumatic Test	A pneumatic test may be substituted where a hydrostatic test is not practicable. There shall be no leakage for satisfactory performance of the tests.		
c. Mass Spectrometer and Halide Leak Test	Systems with conditions of operation and design that require leak tightness of a greater degree of sensitivity than can be obtained by a hydrostatic or pneumatic test shall be leak tested by other methods, such as helium mass spectrometer leak test or halide leak test, which have the required sensitivity.		

Table 6.2-2

DRAWINGS SHOWING PIPING CLASS INFORMATION

The piping class information shown on the station drawings listed below is incorporated by reference into the UFSAR. Other information presented on these drawings is not considered part of the UFSAR. The contents of these station drawings are controlled by station procedure.

SYSTEM	STATION DRAWING
Reactor Coolant System	RM-37A RM-37B
Containment Vacuum & Leakage Monitoring System	RM-412-1
Safety Injection System	RM-411-1 RM-411-2
Containment Depressurization System	RM-513-1 RM-513-2
Post DBA Hydrogen Control System	RM-546-1
Post DBA Hydrogen Analyzer System	RM-546-2
Chemical and Volume Control System	RM-39A RM-39B
Boron Recovery System	RM-29A RM-29B
Residual Heat Removal System	RM-410-1
Primary Plant Component Cooling Water System	RM-22A RM-22B RM-22C RM-22D RM-22E
Fuel Pool Cooling and Purification System	RM-420-1
Sample System	RM-32A RM-32B
Vent and Drain System	RM-34A RM-34B
Compressed Air System	RM-434-1
Instrument Air & Containment Instrument Air System	RM-434-2
River Water System (Flow Diagram Intake Structure)	RM-430-1 RM-430-2 RM-430-3 RM-430-4

Table 6.2-2 (CONT'D)

DRAWINGS SHOWING PIPING CLASS INFORMATION

SYSTEM	**STATION DRAWING**
Fuel Oil System for Emergency Diesel Generators	RM-53A
Main Steam System	RM-14A
Auxiliary Steam and Air Removal System	RM-16A
Feedwater System	RM-18A
Steam Generator Blowdown	RM-50A
Gaseous Waste Disposal System	RM-43A RM-43B

Table 6.3-1

SINGLE ACTIVE FAILURE ANALYSIS ECCS

Component	Malfunction	Comments
A. Accumulator (injection phase)	Delivers to broken loop	Totally passive system with one accumulator per loop. Evaluation based on two accumulators delivering to the RCS and one spilling from ruptured loop after termination of bypass.
B. Pump: (injection phase)		
1) Safety injection charging	Fails to start	Three provided. Evaluation based on operation of one.
2) Low head safety injection	Fails to start	Two provided. Evaluation based on operation of one.
C. Automatically Operated Valves: (one on safety injection signal) - (injection phase)		
1) Boron injection tank outlet isolation valves at discharge of high head safety injection pumps. (cold leg injection)	Fails to open	One of two parallel valves is required to open.
2) Accumulator stop valves	Fails to open	One valve per accumulator, normally open, or opened if initially closed. Analysis assumes all three accumulator stop valves are open.
3) Refueling water storage tank to charging pump suction valves	Fails to open	Two valves; only one valve in either line is required to open.
D. Valves automatically closed on Safety Injection Signal		

SINGLE ACTIVE FAILURE ANALYSIS ECCS

1) Charging line injection	Fails to close	Two valves in series; only one valve required to close.
2) Volume control tank outlet	Fails to close	Two valves in series; only one valve required to close.
E. Valves Operated on Automatic Transfer From Safety Injection to Recirculation		
1) MOV-SI862A or B (Close to realign LHSI pump suction from the refueling water storage tank)	Fails to close	The affected LHSI pump would continue to take suction from the RWST until pump NPSH was lost. However, operation of the LHSI pump on the other parallel suction path would be unaffected. Since minimum safeguards analysis is based on only one LHSI pump operation, the single failure criteria is satisfied.
2) MOV-SI860A or B (Open to realign LHSI pump suction to the containment sump)	Fail to open	The affected LHSI pump would lose NPSH as its associated RWST valve closed. However, operation of the LHSI pump on the other parallel suction path would be unaffected. Since minimum safeguards analysis is based on only one LHSI pump operating, the single failure criterion is not violated. Closure of RWST valve prevents containment sump water from entering the RWST.
3) MOV-SI885 A, B, C or D (Close to prevent LHSI pump discharge back to the RWST)	Failure to Close	There are redundant (2) valves in series in each of the LHSI pump minimum flow bypass lines to the RWST. Should one of these valves fail to close the other valve in the associated line would close, thus meeting the single failure criterion.
4) MOV-SI863A or B (Open to align LHSI pump discharge to charging pump suction)	Failure to Open	The LHSI pumps supply a common charging pump suction header via independent parallel paths. Should one of the paths be blocked, the other LHSI pump would adequately supply the suction header, thus meeting single failure criterion.
5) MOV-CH115B or D (Close to realign charging pump suction from RWST)	Failure to Close	A check valve (SI-27) on the common line supplying both parallel valves will prevent flow returning to the RWST should either valve fail to close. Therefore, the single failure of either of the valves will not permit return flow to the RWST.

Table 6.3-2

ECCS - CODE REQUIREMENTS

Component	Code
Accumulators	ASME Section III Class C
Boron Injection Tank	ASME Section III Class C
Valves	ANSI B16.5
Piping	ANSI B31.1

Table 6.3-3

READOUTS ON THE CONTROL BOARD WHICH OPERATOR CAN MONITOR DURING INJECTION PHASE

VALVES(1)

FSAR Figure	Actuation Position on Injection	Valve No.	Description
6.3-8	Normally Closed	MOV-SI836	ALT. HI-HEAD COLD LEG ISOLATION
6.3-8	Open	MOV-SI867 C, D	B.I.T. OUTLET ISOLATION VALVES
6.3-1	Normally Open	MOV-SI865 A, B, C(2)(3)	S.I. ACCUM ISOLATION VALVES
6.3-8	Normally Open	MOV-SI890 C(2)(3)	LOW-HEAD S.I. ISOLATION TO COLD LEGS
6.3-8	Normally Open	MOV-SI862 A, B	LOW-HEAD S.I. PUMP RWST ISOLATION
6.3-8	Normally Closed	MOV-SI860 A, B	CONTAINMENT SUMP ISOLATION VALVES
6.3-8	Normally Closed	MOV-SI863 A, B	LOW-HEAD TO HI-HEAD S.I. PUMP ISOLATION
6.3-8	Normally Open	MOV-SI864 A, B	LOW-HEAD S.I. PUMP DISCHARGE CROSS-CONNECT ISOLATION

Table 6.3-3 (CONT'D)

READOUTS ON THE CONTROL BOARD WHICH OPERATOR
CAN MONITOR DURING INJECTION PHASE

VALVES[1]

6.3-8	Normally Open	MOV-SI885 A, B, C, D[2]	LOW-HEAD S.I. PUMP MINIFLOW ISOLATION
6.3-8	Normally Closed	MOV-SI869 A, B	HI-HEAD S.I. HOT LEG
6.3-8	Normally Closed	MOV-SI890 A, B	LOW-HEAD, HOT LEG POST ACCIDENT RECIRC. ISOLATION
6.3-8	Closed	MOV-CH289	NORMAL CHARGING ISOLATION OUTSIDE CONTAINMENT
6.3-8	Closed	MOV-CH310	NORMAL CHARGING ISOLATION INSIDE CONTAINMENT
6.3-8	Closed	MOV-LCV-CH 115 C, E	VCT TO CHARGING PUMP ISOLATION
6.3-8	Open	MOV-LCV-CH 115 B, D	RWST TO HI-HEAD S.I./ CHARGING PUMP ISOLATION

Table 6.3-3 (CONT'D)

READOUTS ON THE CONTROL BOARD WHICH OPERATOR
CAN MONITOR DURING INJECTION PHASE

INSTRUMENTS

FSAR Figure	Instrument No.	Description
6.3-8	FI-SI946	Low Head S.I. Pump B Flow
6.3-8	FI-SI945	Low Head S.I. Pump A Flow
6.3-8	FI-SI943	Cold Leg Hi-Head Inj. Header Flow
6.3-8	FI-SI940	Hot Leg Hi-Head Recirculation Header Flow
6.3-8	FI-SI960	Loop 2 Hot Leg Hi-Head Recirculation Flow
6.3-8	FI-SI961	Loop 1 Cold Leg Hi-Head Inj. Flow
6.3-8	FI-SI962	Loop 2 Cold Leg Hi-Head Inj. Flow
6.3-8	FI-SI963	Loop 3 Cold Leg Hi-Head Inj. Flow
6.3-8	FI-SI932	Loop 3 Hot Leg Hi-Head Recirculation Flow
6.3-8	FI-SI933	Loop 1 Hot Leg Hi-Head Recirculation Flow

Table 6.3-3 (CONT'D)

READOUTS ON THE CONTROL BOARD WHICH OPERATOR
CAN MONITOR DURING INJECTION PHASE

PUMPS

FSAR Figure	Pump
6.3-8	Low Head Safety Injection
6.3-8	Safety Injection Charging
6.4-1A	Quench Spray
6.4-1B	Recirculation Spray

Notes:

(1) Individual position lights are provided to indicate the full open or full closed position of each valve.

(2) Valves actuate a combination light and alarm if not fully open.

(3) If valve position is not corrected after alarm defined by note two occurs, an hourly reflash will occur.

Table 6.3-4

QUALITY STANDARDS OF ECCS COMPONENTS

PUMPS

A. Tests and Inspection

1. Performance test
2. Dye penetrant of pressure retaining parts
3. Hydrostatic test

B. Special Manufacturing Process Control

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

ACCUMULATORS AND BORON INJECTION TANK

A. Tests and Inspections

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

B. Special Manufacturing Process Control

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

VALVES

I. Tests and Inspections

A. 200 psig and 200 F or below (stainless steel valves only)

1. Dye penetrant test
2. Hydrostatic test
3. Seat leakage test

B. Above 200 psig and 200 F

1. Forged valves

a. UT of billet prior to forging
b. Dye penetrant check 100 percent of accessible areas after forging
c. Hydrostatic test
d. Seat leakage test

Table 6.3-4 (Cont'd)

QUALITY STANDARDS OF ECCS COMPONENTS

2. Cast valves

   a. Radiograph 100 percent[(1)]
   b. Dye penetrant check all accessible areas[(1)]
   c. Hydrostatic test
   d. Seat leakage test

C. Performance Tests Required for:

   1. Motor-operated valves
   2. Auxiliary relief valves
   3. Air-operated valves

D. Auxiliary relief valves (these valves are not included in the above categories)

   1. 100 percent dye penetrant check of nozzles and disks
   2. 100 percent dye penetrant check of bodies(1)
   3. Hydrotest bodies, nozzles and disks
   4. Seat leakage test
   5. Operational tests

II. Special Manufacturing Process Control

A. Weld, NDT, performance testing, assembly and inspection procedure review

B. Surveillance of suppliers quality control and product

C. Special weld process procedure qualification (e.g. hard facing)

Notes: (1) For valves with radioactive service only.

Table 6.3-5

ACCUMULATOR DESIGN/OPERATING PARAMETERS

Number	3
Type	Stainless steel lined/ carbon steel
Design pressure, psig	700
Design temperature, F	300
Operating temperature, F	100 to 150
Normal Operating Pressure, psig	685
Minimum Operating Pressure, psig	611
Total Volume, ft^3	1,450
Water Volume at Operating Conditions, gallons	6,681 to 7,645[(2)]
Minimum Boron Concentration (as boric acid), ppm	2,300
Relief Valve Setpoint, psig[(1)]	700

(1) The relief valves have soft seats and are designed and tested to ensure zero leakage at the normal operating pressure.

(2) The volumes do not include the tank undeliverable volume below the elevation of the discharge nozzle. The discharge piping volume is also not included in this value.

Table 6.3-6

TANK DATA

BORON INJECTION TANK

Number	1
Total Volume, gallons	900
Design Pressure, psig	2,735
Design Temperature, F	300
Operating Pressure, nominal, psig	100
Material	Stainless steel clad/ carbon steel

Table 6.3-7

PUMP PARAMETERS

Safety Injection Charging Pumps	
Number of Pumps	3
Design Pressure, Discharge, psig	2,800
Design Pressure, Suction, psig	200
Design Temperature, °F	300
Design Flow Rate, gpm	150
Maximum Flow Rate, gpm	580 Runout
Design Head, ft	5,800 (minimum)
Type	Horizontal centrifugal
Low Head Safety Injection Pumps	
Number of Pumps	2
Type	Vertical centrifugal
Design Pressure, Discharge, psig	300 (150 lb ASA discharge flange)
Design Temperature, °F	300
Design Flow, gpm	3,000
Design Head, ft	257
Maximum Flow Rate, gpm	4,000

Table 6.3-8

ECCS RELIEF VALVE DATA

Descriptions	Fluid Discharged	Fluid Inlet Temp. Normal °F	Temp. Relieving Fluid °F	Set Pressure Psig	Back Pressure Constant	Psig Build-Up	Capacity
N_2 supply to Accumulators	N_2	120	70	700	0	0	1,500 scfm
LHSI Pump Discharge	Water	120	120	235	3	50	20 gpm
Boron Injection Tank Relief	Boric Acid Solution	165	180	2,735	3	12	20 gpm
Accumulator to Containment	Water or N_2 Gas	120	120	700	0	0	1,500 scfm

Table 6.3-9

NOMINAL (DESIGN) EXTERNAL RECIRCULATION LOOP LEAKAGE (SAFETY INJECTION SYSTEM ONLY)

	Items	No. of Units	Type of Leakage Control and Unit Leakage Rate	Leakage to Atmosphere, cc per hr	Leakage to Vent and Drain System, cc per hr
1.	Low Head Safety Injection Pumps	2	Tandem mechanical seal with demineralized water interface between seals - leakage essentially zero[(1)]	0	0
2.	Safety Injection Charging (High Head) Pumps	3	Mechanical seal with leakoff - (10 cc/hr/seal) 2 seals per pump	0	60
3.	Flanges:				
	a. Pumps		Gasket - adjusted to zero leakage following any test - ten drops per minute, per flange (30 cc/hr)		
	(1) LHSI	2		60	0
	(2) HHSI	15		450	0
	b. Misc. large valves > 2 inches	57	Gasket and/or packed stems (20 cc/hr)	1,140	0
	c. Misc. flanges	10	Gasket - ten drops per minute, per flange (30 cc/hr)	300	0
4.	Misc. Small Valves ≤ 2 inches	144	Gasket and/or packed stems (3 cc/hr)	432	0
			TOTALS	2,382	60

(1) Seals are acceptance tested to essentially zero leakage.

Table 6.3-11

TABLE OF FAULT PROBABILITIES

Motor Starter Au. Closed contacts Short Out	5×10^{-7}/hr	P (failure during mission) = 5×10^{-7} hr^{-1} x 0.1 hr = 5×10^{-8}
Manual Switch Closing Contacts Short Out	7.5×10^{-8}/hr	P (failure during mission) = 7.5×10^{-8} hr^{-1} x 0.1 hr = 7.5×10^{-9}
Hot Short in Cable Run - Voltage to closing coil	1.0×10^{-9}/hr	P (failure during mission) = 1.0×10^{-9} hr^{-1} x 0.1 hr = 1×10^{-10}
Hot Short in MCC-bypasses Switch (closed) contact only	1.0×10^{-9}/hr	P (failure during mission) = 1.0×10^{-9} hr^{-1} x 0.1 hr = 1×10^{-10}
Operator Fails to Take Action On Light Indication	1×10^{-3}/hr	P (no action for 1 shift) = 10^{-3}/hr^{-1} x 8 hr = 8×10^{-3}
Limitorque Fails Indicating Open	7.5×10^{-8}/hr	P (limitorque failure/shift) = $.75 \times 10^{-7}$/hr^{-1} x 8 hr = 6×10^{-7}
Operator Fails to Open Valve After Maintenance	1×10^{-3}/main-tenance	P (error for 1 shift) = $\frac{10^{-3} \text{ x 1 maint} \quad \text{x 8 hr}}{\text{maint } 8.8 \times 10^{3} \text{ hr}}$ = 9×10^{-7}

Table 6.3-12

NORMAL/ACCIDENT FUNCTIONS EVALUATION

Component	Normal Operating Function	Normal Operating Arrangement	Accident Function	Accident Arrangement
Boron Injection Tank[(1)]	None	Lined up from the discharge of safety injection charging pumps to the closed BIT outlet valves.	Pathway to RCS	Lined up to discharge of safety injection charging pumps for cold leg injection
Refueling Water Storage Tank	Storage tank for refueling water	Lined up to suction of high head safety injection charging, low head safety injection and containment depressurization system quench pumps	Source of borated water for core and containment cooling	Lined up to suction of safety injection charging, low head safety injection and containment depressuri- zation system quench pumps.

(1) During normal operation the boron injection tank is isolated from the cold leg by redundant parallel BIT outlet valves, both of which receive a signal to open via the safety injection signal.

Component	Normal Operating Function	Normal Operating Arrangement	Accident Function	Accident Arrangement
Accumulators (3)	None	Lined up to discharge to cold legs of reactor coolant piping	Fast supply of borated water to core	Lined up to discharge to cold legs of reactor coolant piping
Safety Injection Charging Pumps (3)	Charging	Take suction from volume control tank and discharge to normal charging connections	Supply borated water to core	Lined up to take suction from RWST or low-head safety injection pumps and discharge to cold legs of reactor coolant piping
Low Head Safety Injection Pumps (2)	Refueling fill or refueling cavity	Lined up to take suction from refueling water storage tank and discharge to cold legs of reactor coolant piping	Supply borated water to core	Lined up to take suction from refueling water storage tank and discharge to reactor coolant system cold legs during injection phase or recirculation phase lined up to RCS hot or cold legs

Table 6.3-13

ACCUMULATOR INLEAKAGE[1]

Observed Leak Rate, cc per hr	Time Period Between Level Adjustments
543	1 month
272	2 months
30.3	18 months

(1) A total of 14 cubic feet, added to the initial amount, can be accepted in each accumulator before an alarm is sounded.

Table 6.4-1

QUENCH SPRAY SUBSYSTEM COMPONENT DESIGN DATA

Refueling Water Storage Tank

Number	1
Capacity, gallons	441,100
Design pressure	Atmospheric
Design temperature, °F	150
Operating pressure	Atmospheric
Operating temperature,°F	Per requirements of Technical Specification and Licensing Requirements Manual
Material[(2)]	ASTM A240 Type 304L

Refueling Water Storage Tank Coolers

Number		2
Design Duty, Btu/hr		375,000
	Shell	Tube
Fluid	Chilled Water	Refueling Water
Design pressure, psig	100	100
Design temperature, °F	250	250
Operating temperature, °F	35 to 70	45 to 140
Operating pressure, psig	75	85
Material[(2)]	ASME SA-53-B	ASTM A249 Type 304

Table 6.4-1 (CONT'D)

QUENCH SPRAY SUBSYSTEM COMPONENT DESIGN DATA

<u>Quench Spray Pumps</u>	
Number	2
Type	Horizontal, Centrifugal
Motor horsepower	200
Seals	Single Mechanical
Capacity, gpm	2,500
Head at rated capacity, ft	270
Design pressure, psig	250
Design temperature, °F	45
Materials(2)	
Pump casing	ASTM A351 CF8
Shaft	ASTM A182 F-316
Impeller	ASTM A351 CF8
<u>Refueling Water Circulation Pumps</u>	
Number	2
Type	Horizontal, Centrifugal
Motor horsepower	15/3.7(1)
Capacity, gpm	375/36(1)
Head at rated capacity, ft	70/31(1)
Design pressure, psig	225
Operating temperature, °F	45-100

Table 6.4-1 (CONT'D)

QUENCH SPRAY SUBSYSTEM COMPONENT DESIGN DATA

Refueling Water Circulation Pumps (Cont'd)

Materials[(2)]	
Pump casing	ASTM A-296 - CF8M
Shaft	ASTM A-276 - Type 316
Impeller	ASTM A-296 - CF8M

NOTES:

(1) Two speed pumps

(2) Materials listed in this table may be replaced with materials of equivalent design characteristics. The term "equivalent" is described in UFSAR Section 1.8.2, "Equivalent Materials."

Table 6.4-2

COMPONENT DESIGN DATA FOR THE RECIRCULATION SPRAY SUBSYSTEMS

Recirculation Spray Pumps

Number	4	
Type	Vertical-deep well type turbine	
Motor horsepower	300	
Seals	Tandem mechanical (outside)	Mechanical seal plus throttle bushing (inside)
Capacity, gpm	3,300	
TDH at rated capacity, ft	260	
Design Pressure, psig	265	
Design Temperature, °F	280	
Materials[4]		
Pump bowls	ASTM A351-CF8	
Shaft	17-4 pH	
Impeller	ASTM CF8	

Recirculation Spray Coolers

	RS-E-1A & 1C		RS-E-1B & 1D	
Number	2		2	
Design Duty, Btu per hr[3]	61×10^6		59.25×10^6	
	Shell	Tube	Shell	Tube
Fluid	Recirculated spray water	Ohio River water	Recirculated spray water	Ohio River water
Design Pressure, psig	150 int.	150 int. on tubes and channel	150 int.	150 int. on tubes and channel
Design Temperature, °F	280	280	280	280
Operating Temperature, °F	139 in 104 out[1]	85 in 115.6 out[1]	142.9 in 105.3 out[1]	90 in 119.5 out[1]

Table 6.4-2 (CONT'D)

COMPONENT DESIGN DATA FOR THE RECIRCULATION SPRAY SUBSYSTEMS

	RS-E-1A & 1C		RS-E-1B & 1D	
	Shell	Tube	Shell	Tube
Operating Pressure, psia	115 int. at inlet	65 int. at inlet and channel	115 int. at inlet	65 int. at inlet and channel
Material(4)	Stainless Steel - Type 304L	Stainless Steel - Type 304L	Stainless Steel - Type 304L	Stainless Steel - Type AL6XN(2)

NOTES:

(1) Temperatures vary as a function of time following a DBA.

(2) A low carbon, high purity, nitrogen bearing "super austenitic" stainless steel manufactured by Allegheny Ludlum.

(3) Original design for RS-E-1A & 1C did not include fouling factor. Rerated conditions for heat exchangers based on fouling factors, higher allowable river water temperature, and better defined containment conditions, provide a minimum rating for RS-E-1A and 1C used in containment analysis, that is other than the original design specification.

(4) Materials listed in this table may be replaced with materials of equivalent design characteristics. The term "equivalent" is described in UFSAR Section 1.8.2, "Equivalent Materials."

Table 6.4-3

CONSEQUENCES OF COMPONENT MALFUNCTIONS
IN THE CONTAINMENT DEPRESSURIZATION SYSTEM

	Components	Malfunction	Comments and Consequences
1.	Quench Pumps	Pump casing ruptures	The casing is designed for 450°F temperature; standard test pressure is 250 psig and maximum test pressure is 375 psig. These conditions exceed those which could occur during any operating conditions. The casings are made from cast steel (ASTM A351-CF8) or equivalent;(1) this metal has corrosion-erosion resistance and produces sound castings. The pumps conform to Seismic Category I design. Pumps are missile protected and may be inspected at any time. Rupture by missiles is not considered credible. Rupture of the pump casing is, therefore, not considered credible.
2.	Quench Pumps	Pump fails to start	The quench spray subsystem has two parallel 100 percent capacity pumps. Sufficient capacity is provided by one pump in case of failure of the other pump.
3.	Quench Pump Discharge Valve	Valve fails to open	Redundant parallel path is provided from second pump. Sufficient capacity is provided by one pump.
4.	Quench Pump Discharge Valve	Rupture of valve body	Valve body is designed for 150 lb. The castings are made from stainless steel; this material has corrosion-erosion resistance and produces sound castings. The valves are designed to be missile protected. Rupture of valve body is not considered credible.
5.	Quench Pumps	Weight loaded check valve in pump discharge line sticks closed	Valve is checked during cold shutdown. In addition, a parallel100 percent capacity quench spray subsystem is operable.

Table 6.4-3 (CONT'D)

CONSEQUENCES OF COMPONENT MALFUNCTIONS
IN THE CONTAINMENT DEPRESSURIZATION SYSTEM

	Components	Malfunction	Comments and Consequences
6.	Quench Piping	Pipe rupture	Piping is designed for 100°F temperature and 20 psig to 185 psig. These conditions exceed those which could occur during operation. The piping is fabricated of Type 304, or equivalent,[(1)] ASME Boiler and Pressure Vessel Code, stainless steel; this metal has corrosion-erosion resistance. Piping is designed for ASME Code, Class III or Class II (Section 6.2-2). Pipe rupture is not considered credible. However, in case of pipe rupture for pipe lines from the quench pumps, isolation valves are provided.
7.	Recirculation Spray Pump	Pump fails to start	Four 50 percent capacity recirculation spray pumps are provided. Only 2 out of 4 must operate.
8.	Recirculation Spray Cooler	Tube or shell rupture	Four 50 percent capacity recirculation spray coolers are provided. The recirculation spray coolers are designed to the ASME Code, Section III, Class C and Seismic Category I. A special jet force protective shield is installed for each cooler to ensure integrity of the coolers considering possible high energy sprays during a LOCA. Rupture of a cooler is considered unlikely and 150 percent backup exists. The river water discharged from the coolers is monitored for radiation. A high radiation level indicates the tube leakage. The river water lines to the affected cooler can be remote-manually isolated as necessary.
9.	Outside Recirculation Spray Pump	Rupture of pump casing	The casing is fabricated of Type 304 stainless steel; this metal is very corrosion resistant. The casings are missile protected and set in concrete. Rupture of the pump casing is not considered credible.

Table 6.4-3 (CONT'D)

CONSEQUENCES OF COMPONENT MALFUNCTIONS
IN THE CONTAINMENT DEPRESSURIZATION SYSTEM

	Components	Malfunction	Comments and Consequences
10.	Recirculation Piping	Rupture of piping	Piping is fabricated of Type 304 stainless steel and designed to safety ASME Code, Class 2. Piping is also missile protected. Rupture of piping is not considered credible. However, in case of pipe rupture for pipe lines outside the containment to and from recirculation pumps, isolation valves are provided.
11.	Outside Recirculation Spray Pump Discharge Valve	Rupture of valve body	Valve body is designed for 150 lb. The castings are made from stainless steel; this material has corrosion erosion resistance and produces sound castings. The valves are designed to be missile protected. Rupture of valve body is not considered credible.
12.	Motor Operated Valves (where opening is required)	Loss of power to one valve due to failure of electric bus	Electric power to valves in redundant line is supplied from separate bus.
13.	Automatic Electric and Control Instrumentation Trains to Actuate Containment Depressurization System	Failure of one train	Redundant train actuates redundant equipment.
14.	Spray Nozzles	Spray nozzles plugged	Strainers are provided in the discharge of the quench spray pumps. The containment sump strainer assembly is made up of strainer modules with 1/16" diameter perforations. The perforation size is small enough to prevent any material which could plug the spray nozzles from passing through. Sufficient margin is provided to accommodate plugging of 25 percent of the nozzles.
15.	Sump	Screens clogged	The clogging of all screens so as to prevent water reaching the pumps is not credible.

Table 6.4-3 (CONT'D)

CONSEQUENCES OF COMPONENT MALFUNCTIONS
IN THE CONTAINMENT DEPRESSURIZATION SYSTEM

Components	Malfunction	Comments and Consequences

NOTES:

(1) The term "equivalent" is described in UFSAR Section 1.8.2, "Equivalent Materials."

Table 6.4-4

RECIRCULATION SPRAY SUBSYSTEM LEAKAGE

Item	No. of Units	Type of Leakage Control and Unit Leakage Rate Used in the Analysis	Uncollected Leakage, cc per hr	Leakage to Vent and Drain System, cc per hr
Recirculation spray pumps	4	No leakage of spray water due to tandem seal arrangement (See text)	0	0
Flanges:		Adjusted to zero leakage following any test - assumed 10 drops per minute per flange		
a. Pump	8		240	0
b. Valves - body to bonnet (larger than 2 inches)	8		230	0
Valves - stem leakoffs	8	Backseated, double packing with leakoff - 1 cc per hr per inch stem diameter	0	8
Miscellaneous small valves	4	Flanged body, packed stems - 1 drop per minute	12	0
		Total	482	8

Table 6.6-1

SUPPLEMENTARY LEAK COLLECTION
AND RELEASE SYSTEM - DESIGN DATA

Service	Number of Units	Unit Capacity, Cfm	Static Pressure, in W.G.	Motor, H.P.
Leak Collection Exhaust Fans	2	50,000	15.0	150
Main Filter Bank	2	36,000	3.67	-

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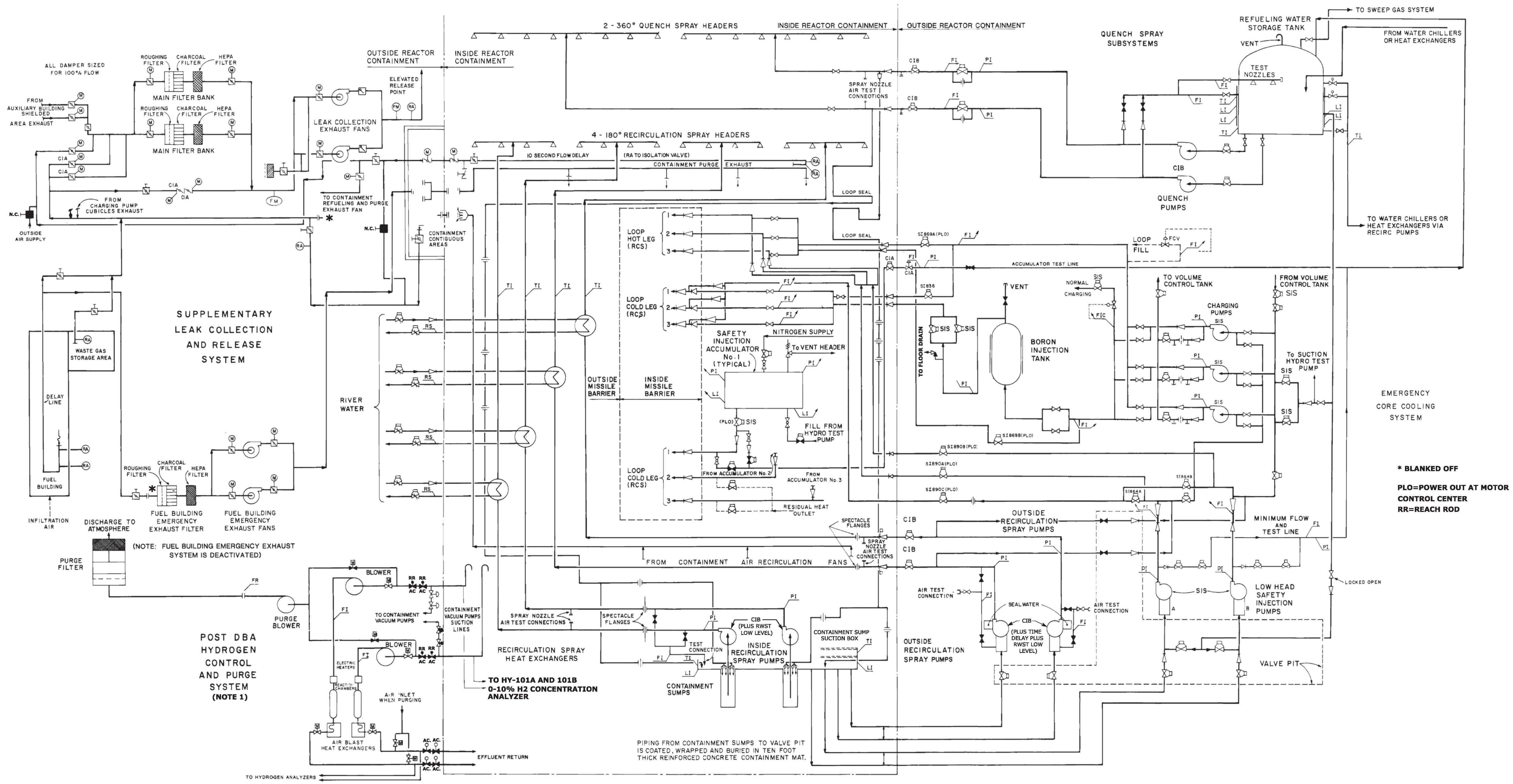

NOTE 1 - THE POST DBA HYDROGEN CONTROL AND PURGE SYSTEM (EXCEPT FOR HYDROGEN ANALYZERS) IS NO LONGER REQUIRED TO BE MAINTAINED. REFER TO UFSAR SECTION 6.5.1 .

FIGURE 6.1-1

ENGINEERED SAFETY FEATURES SYSTEMS

BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

M:\U1\USFAR\FIGURES_6.1_6.3_6.4\FIGURE 6.1-1.DGN

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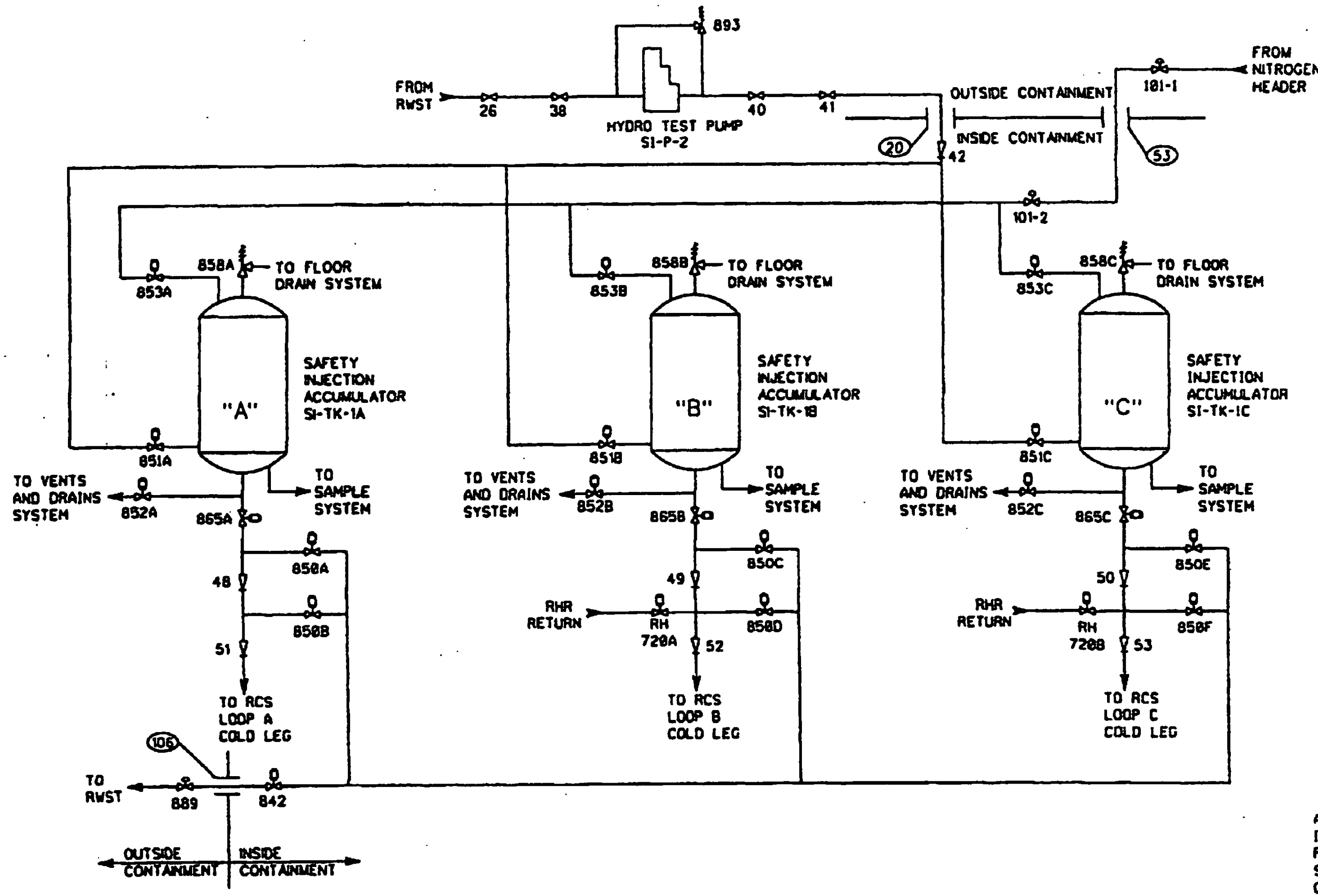

ALL VALVE AND EQUIPMENT IDENTIFICATION NUMBERS ON THIS FIGURE ARE PRECEDED BY THE SYSTEM DESIGNATOR "SI" UNLESS OTHERWISE INDICATED.

FIGURE 6.3-1
SAFETY INJECTION ACCUMULATORS

REFERENCE: STATION DRAWING RM-511
BEAVER VALLEY POWER STATION UNIT NO. 1
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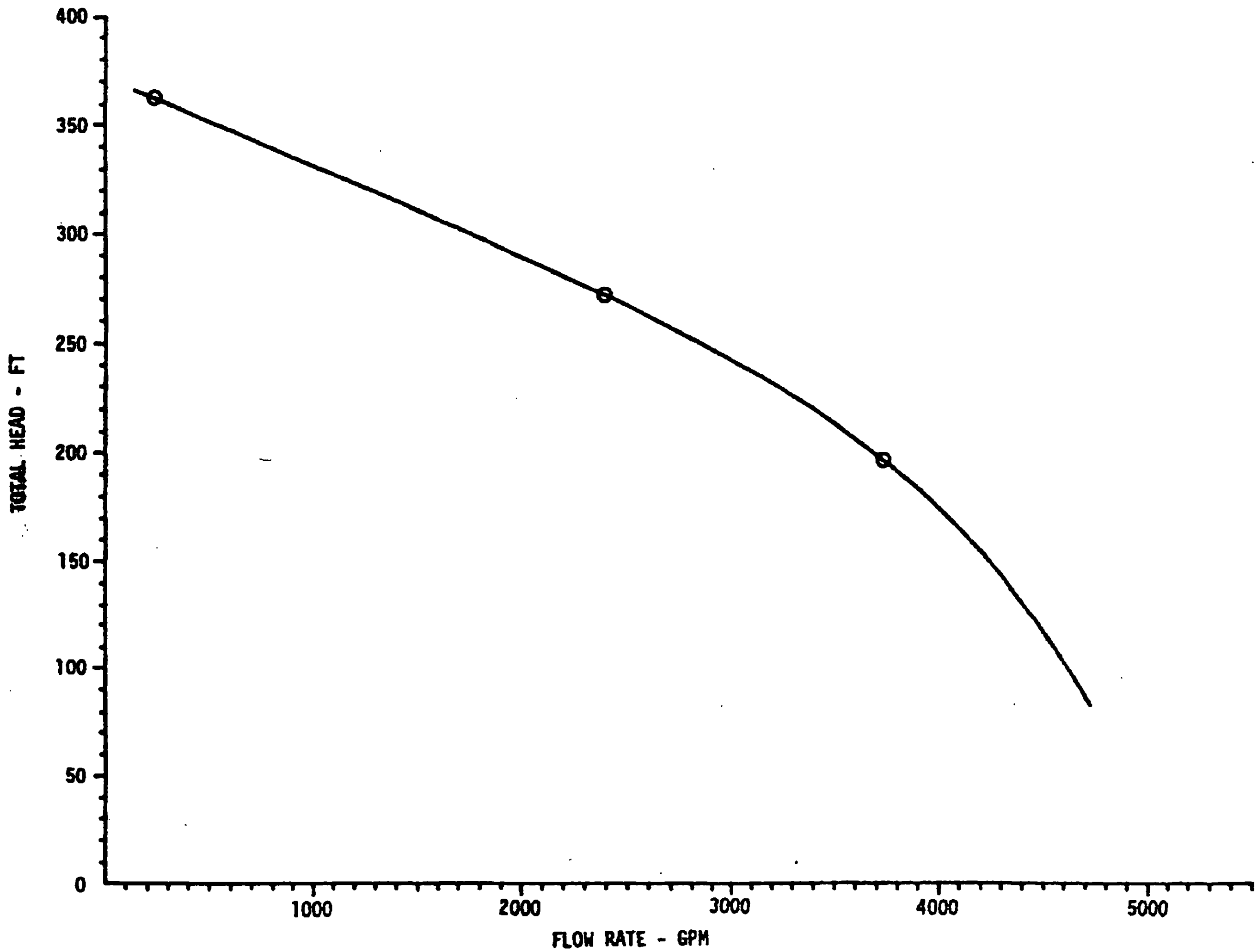
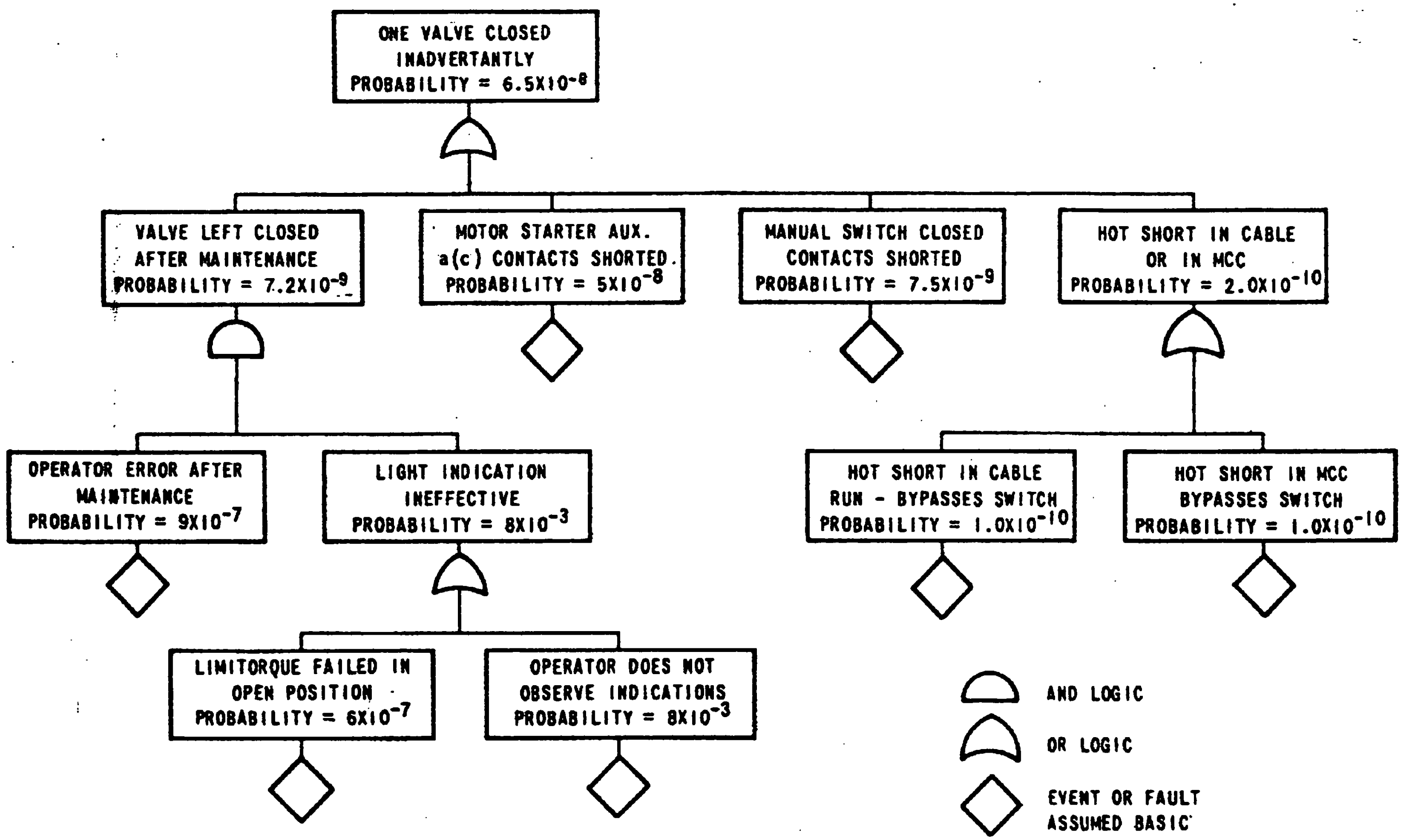


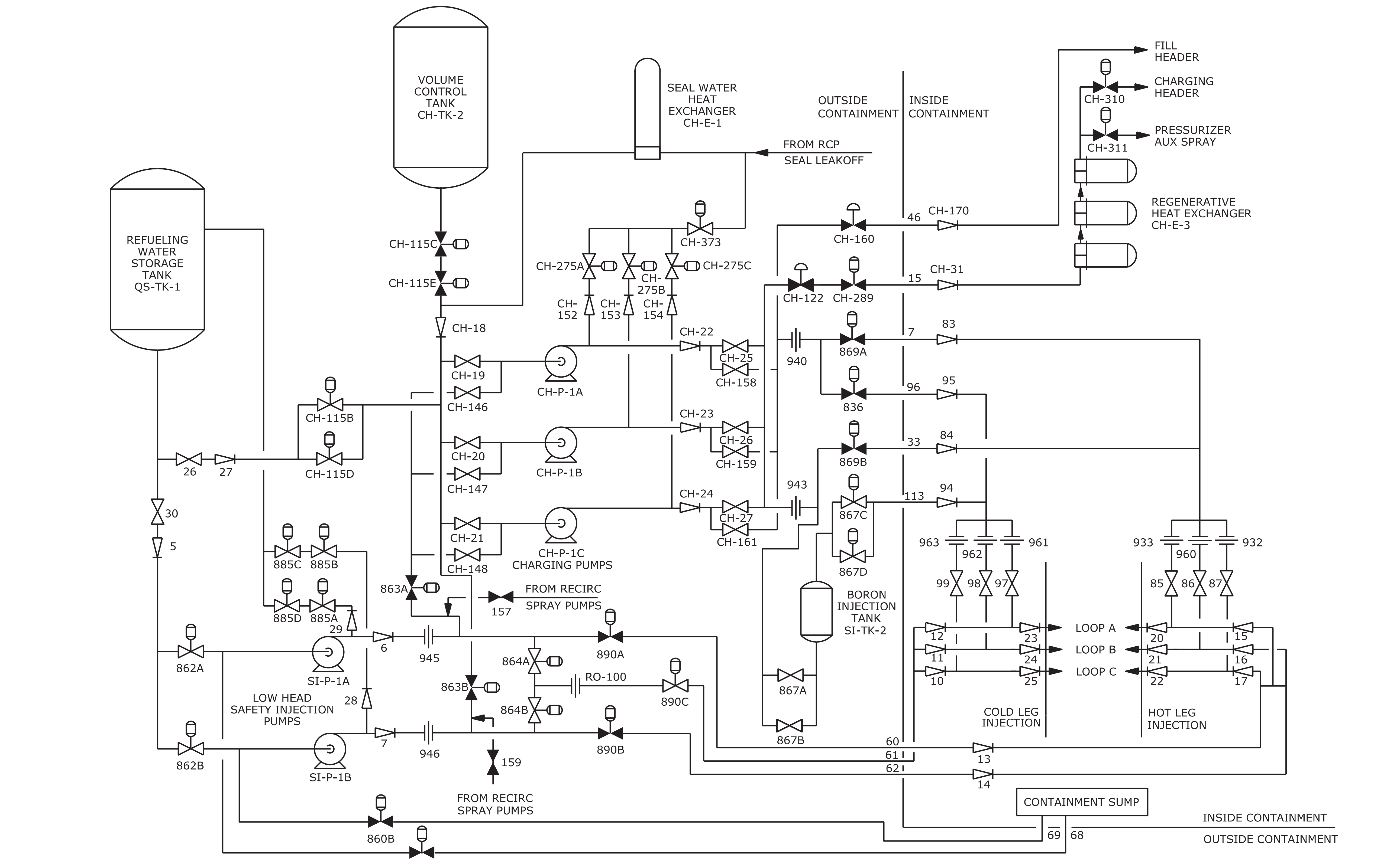
FIGURE 6.3-5
PERFORMANCE CURVE, LOW HEAD
SAFETY INJECTION PUMP
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT



NOTES:

1. OPERATOR ERRORS ASSUMED TO BE DETECTED AFTER ONE 8-HOUR SHIFT.
2. MISSION TIME ASSUMED TO BE 0.1 HOUR

FIGURE 6.3-7
FAULT TREE FOR INADVERTENT CLOSURE OF ONE NORMALLY OPEN ECCS MOTOR OPERATED VALVE



NOTES:

1. ALL VALVE AND EQUIPMENT IDENTIFICATION NUMBERS ON THIS FIGURE ARE PRECEDED BY THE SYSTEM DESIGNATOR "SI" UNLESS OTHERWISE INDICATED
2. VALVES SHOWN IN SAFETY INJECTION COLD LEG INJECTION PHASE ALIGNMENT

FIGURE 6.3-8
SAFETY INJECTION SYSTEM
INJECTION MODE

REFERENCE: STATION DRAWINGS RM-407 & 411
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

K:\U1\USFAR\FIGURE 6.1 THRU\FIG 6.3-8.dgn

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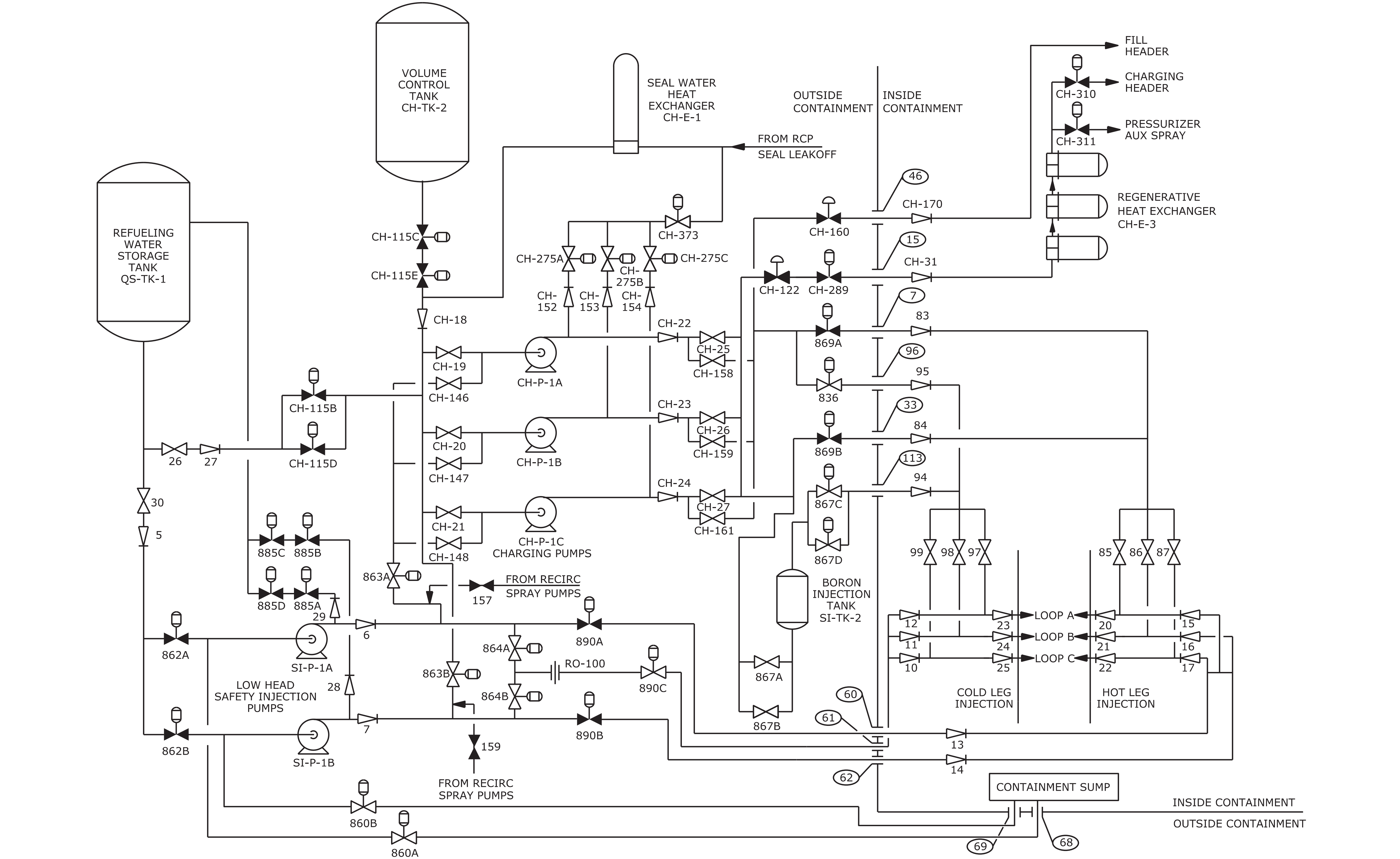

NOTES:

1. ALL VALVE AND EQUIPMENT IDENTIFICATION NUMBERS ON THIS FIGURE ARE PRECEDED BY THE SYSTEM DESIGNATOR "SI" UNLESS OTHERWISE INDICATED
2. VALVES SHOWN IN SAFETY INJECTION COLD LEG INJECTION PHASE ALIGNMENT

FIGURE 6.3-9
SAFETY INJECTION SYSTEM
RECIRCULATION MODE

REFERENCE: STATION DRAWINGS RM-407 & 411
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

K:\U1\USFAR\FIGURE 6.1 THRU\FIG 6.3-9.dgn

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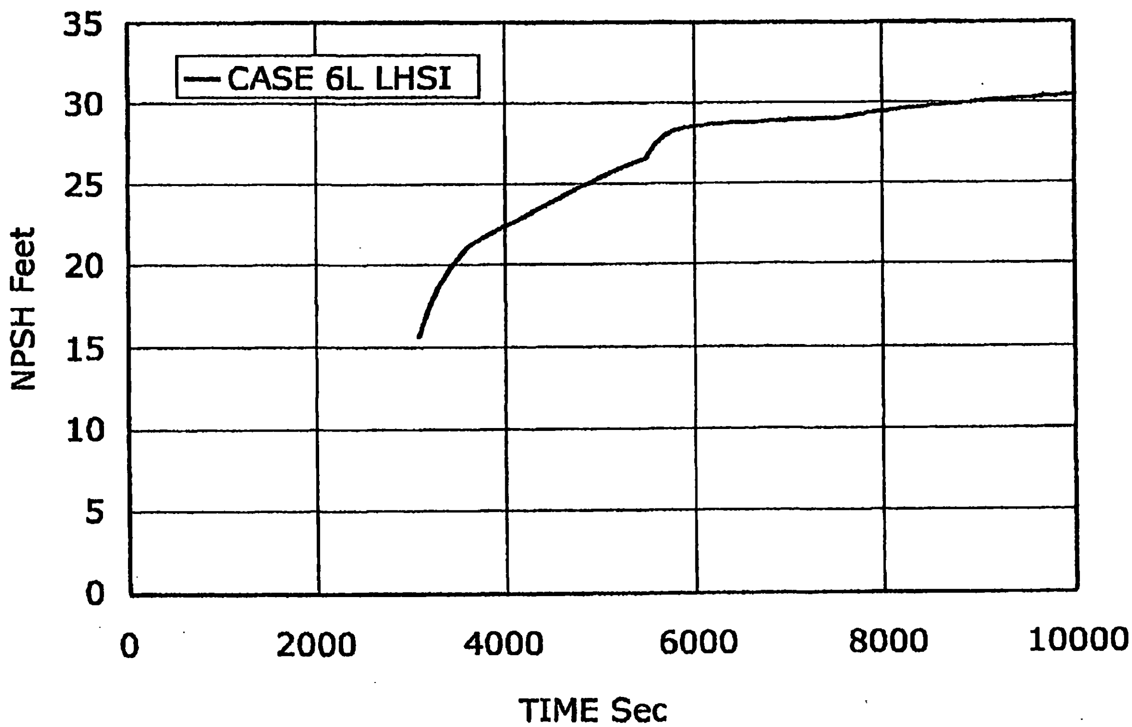

FIGURE 6.3-10

AVAILABLE NPSH FOR LHSI PUMP, PUMP SUCTION DER-Min. ESF

BEAVER VALLEY POWER STATION - UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT

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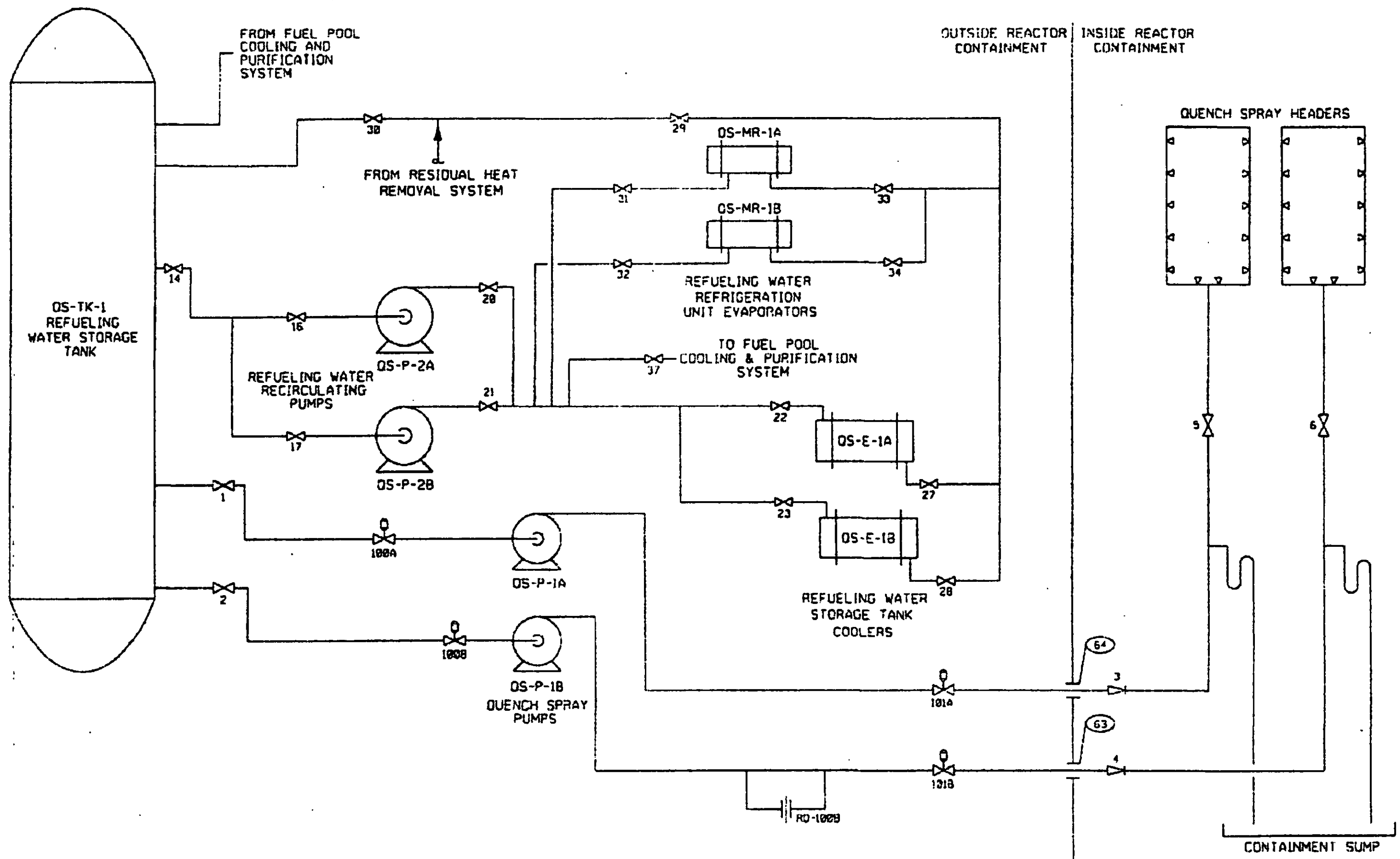
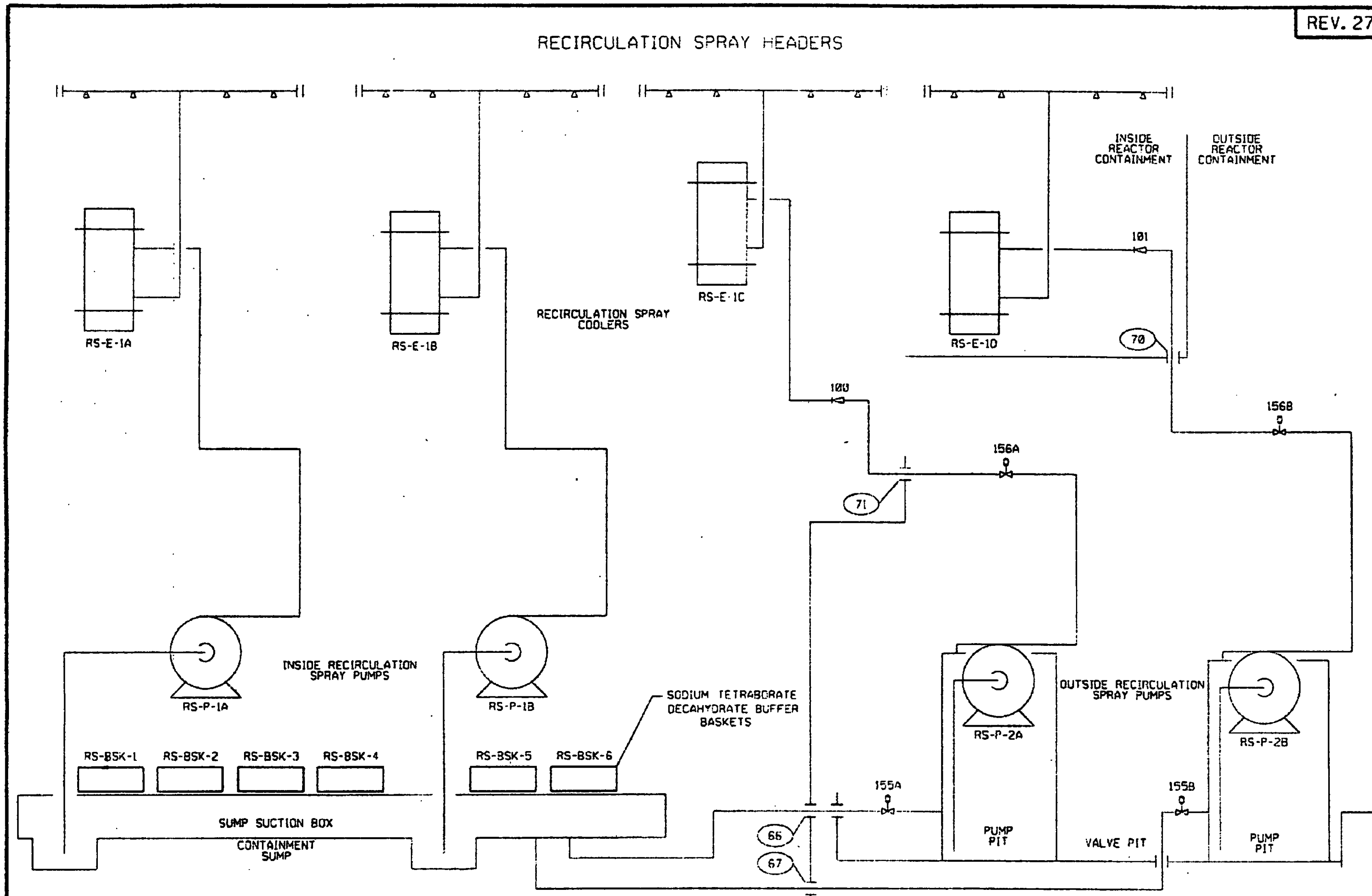


ALL VALVE AND EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR 'QS' UNLESS OTHERWISE INDICATED.

FIGURE 6.4-1A
CONTAINMENT DEPRESSURIZATION SYSTEM
QUENCH SPRAY SUBSYSTEM
REFERENCE: STATION DRAWING RM-513
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT



ALL VALVE AND EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR 'RS' UNLESS OTHERWISE INDICATED.

FIGURE 6.4-1B

CONTAINMENT DEPRESSURIZATION SYSTEM RECIRCULATION SPRAY SUBSYSTEM

REFERENCE: STATION DRAWING RM-513
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

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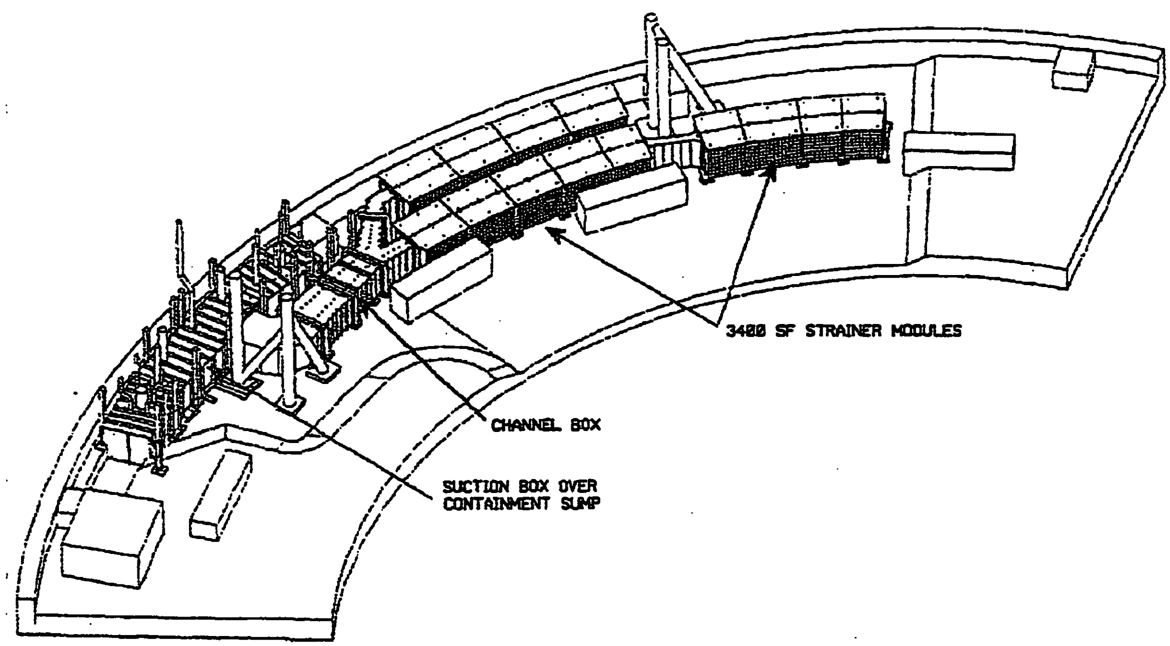

FIGURE 6.4-2

CONTAINMENT SUMP STRAINER ASSEMBLY

BEAVER VALLEY POWER STATION - UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT

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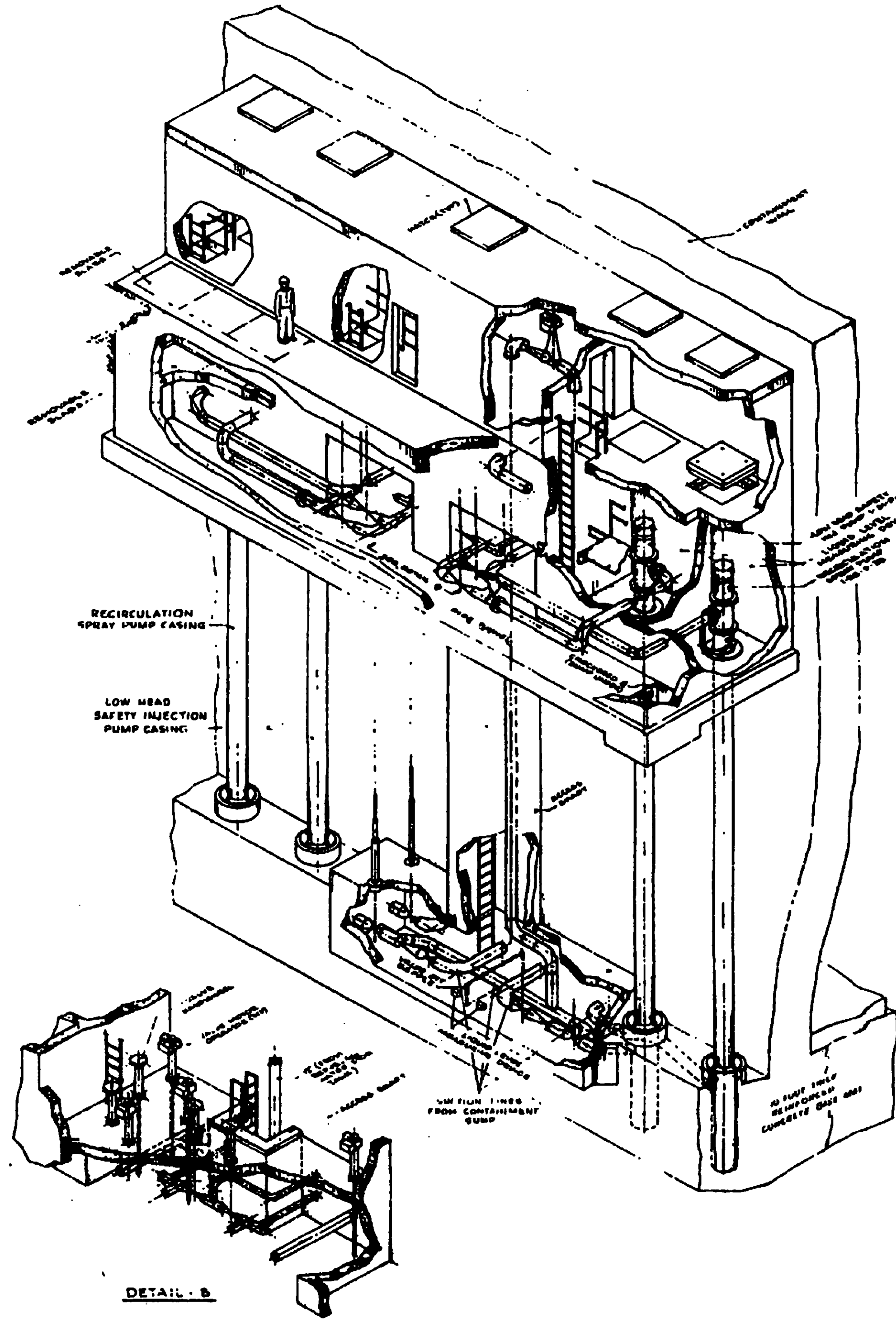


FIGURE 6.4-3
TYPICAL-GENERAL STRUCTURAL & PIPING ARRANGEMENT RECIRCULATION SPRAY & LOW HEAD SAFETY INJECTION SYSTEMS OUTSIDE THE REACTOR CONTAINMENT
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

NOTE: THIS FIGURE PROVIDES A TYPICAL REPRESENTATION OF THE GENERAL STRUCTURAL AND PIPING ARRANGEMENT AND SHALL NOT BE USED FOR DESIGN OR OPERATIONAL PURPOSES.

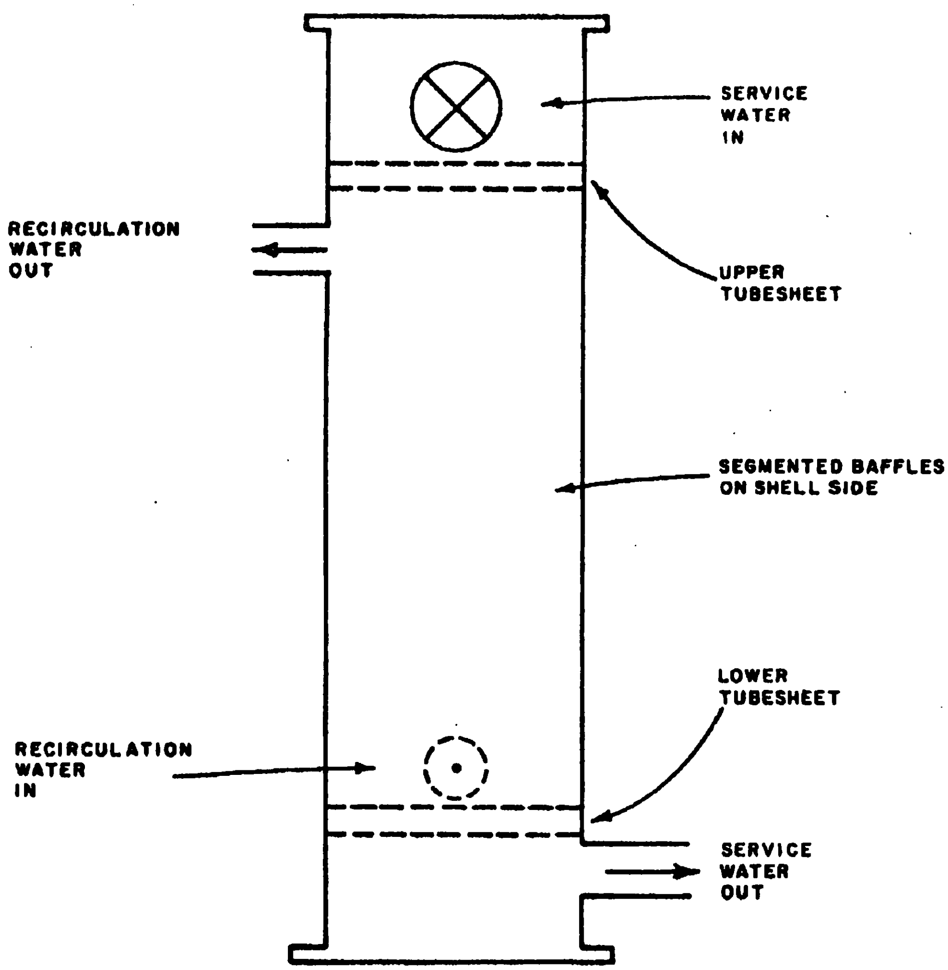


FIGURE 6.4-4
RECIRCULATION COOLER (TYPICAL) SHELL AND TUBE DESIGN - COUNTERCURRENT FLOW
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

REV. 0 (1/82)

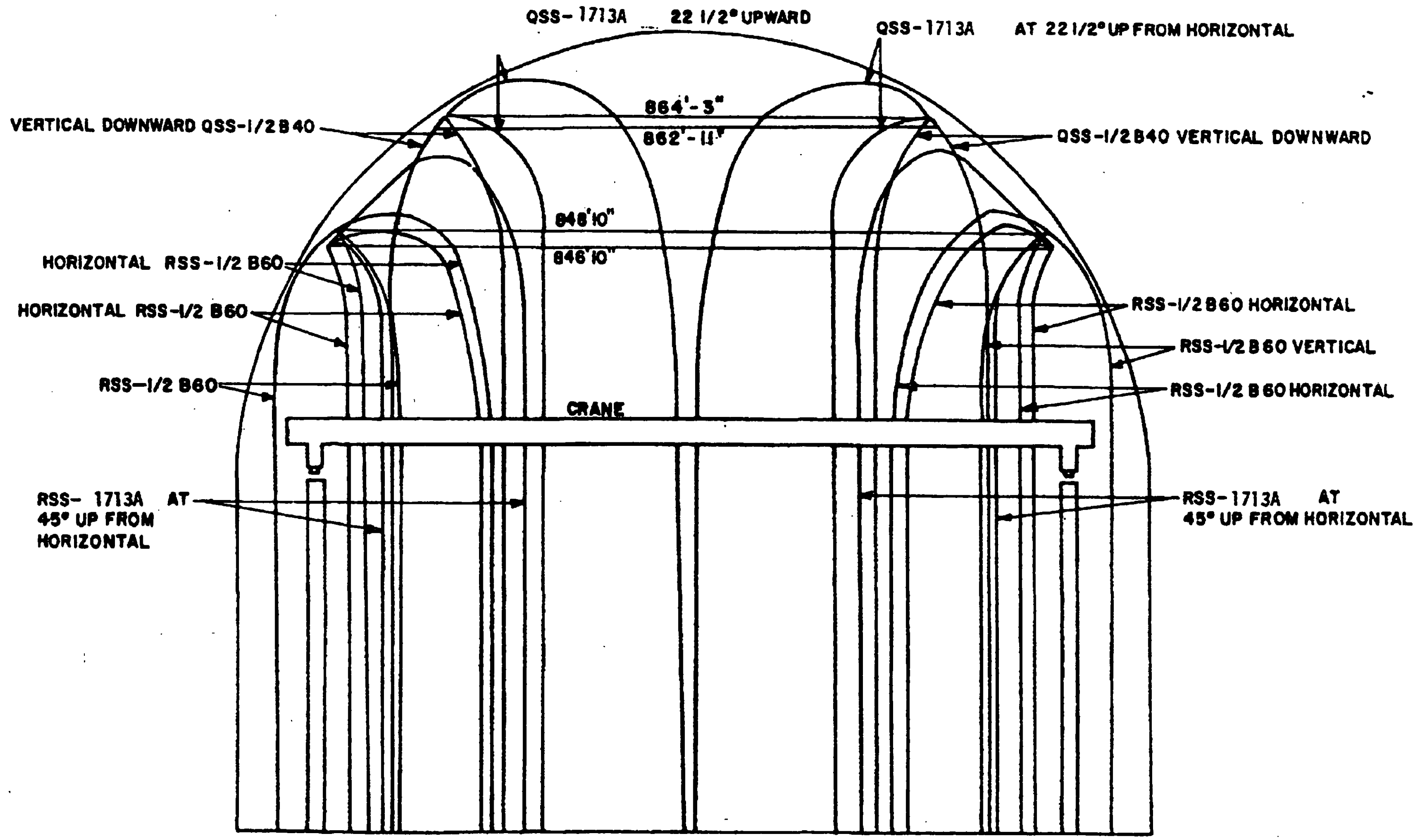


FIGURE 6.4-8
SPRAY PATTERNS IN CONTAINMENT
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

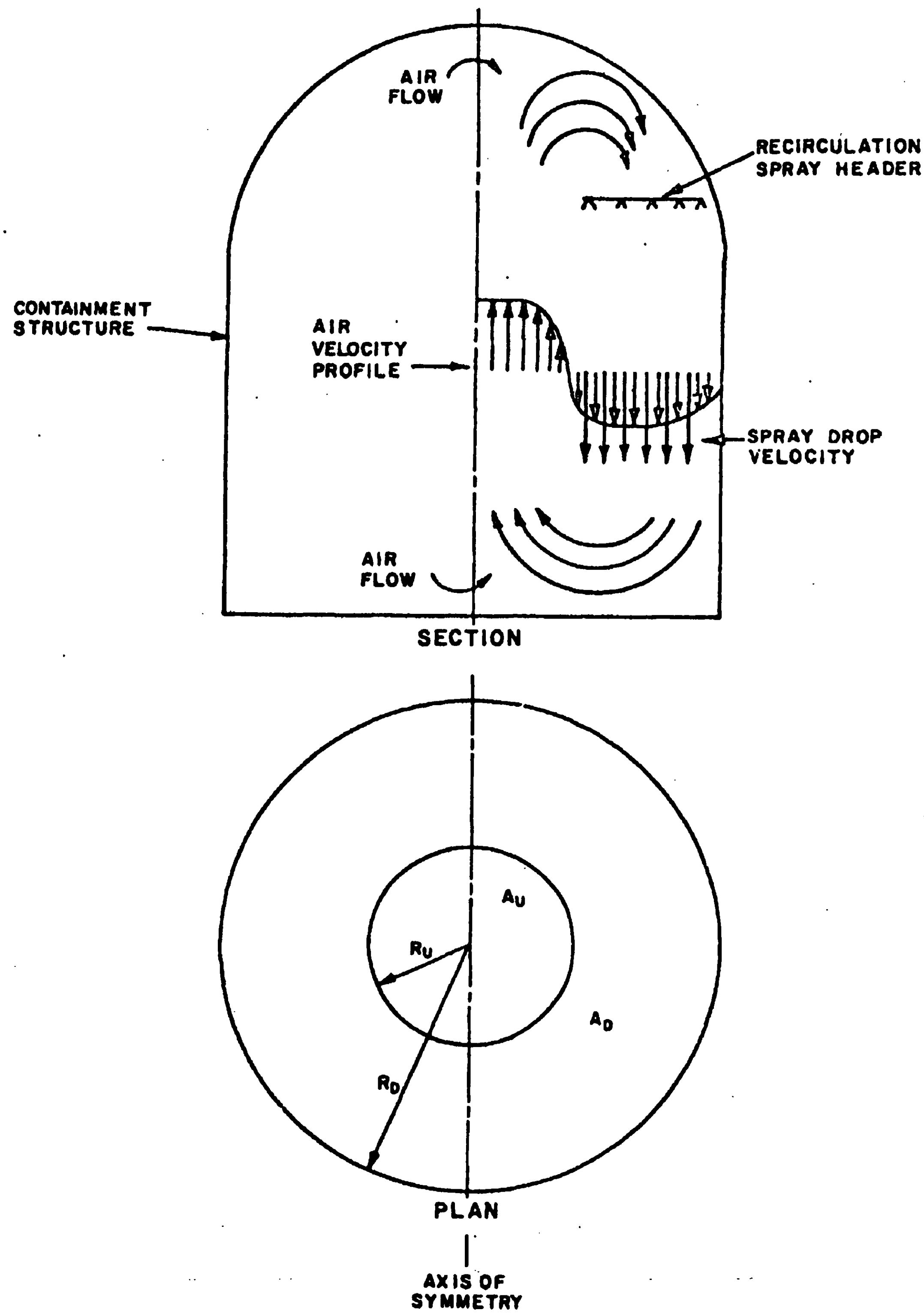
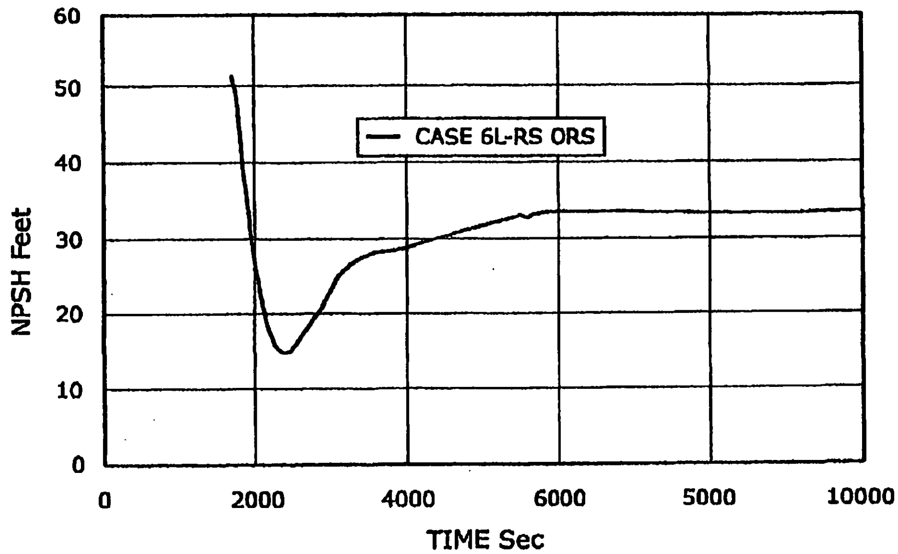


FIGURE 6.4-9
AIR MOTION INDUCED BY
RECIRCULATION SPRAYS
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

REV. 24



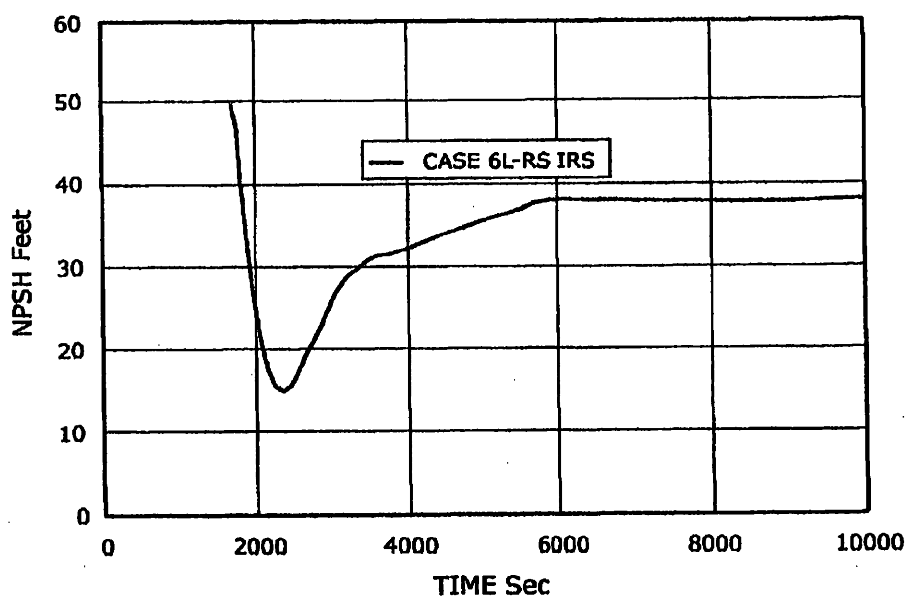


FIGURE 6.4-10

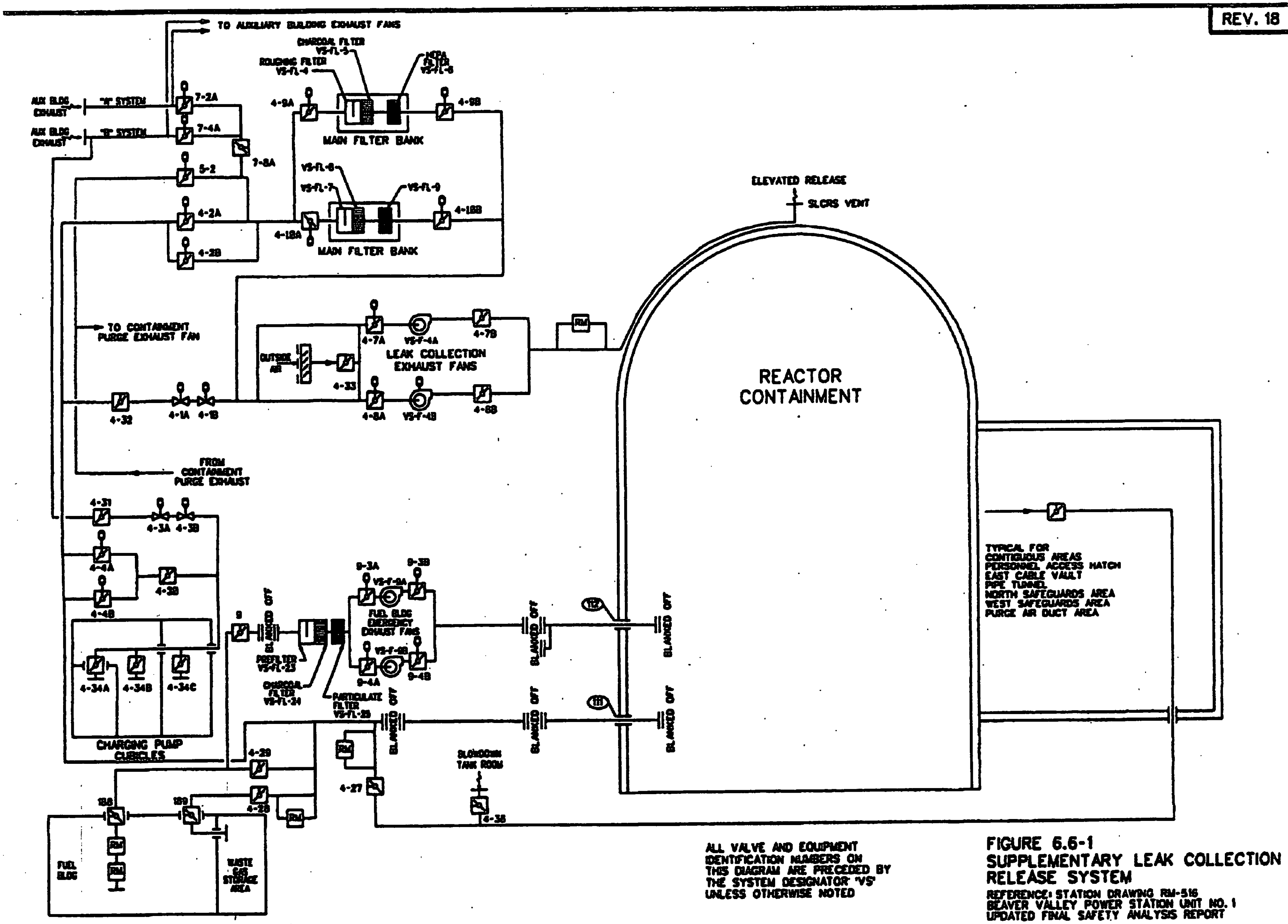
AVAILABLE NPSH FOR RECIRC. SPRAY PUMP, PUMP SUCTION DER-Min. ESF

BEAVER VALLEY POWER STATION - UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT

25-MAR-2008 12:40 M:\u1\UFSAR\g6040010.dgn

PREPARED ON THE CNSU CAEDDI SYSTEM

REV. 18

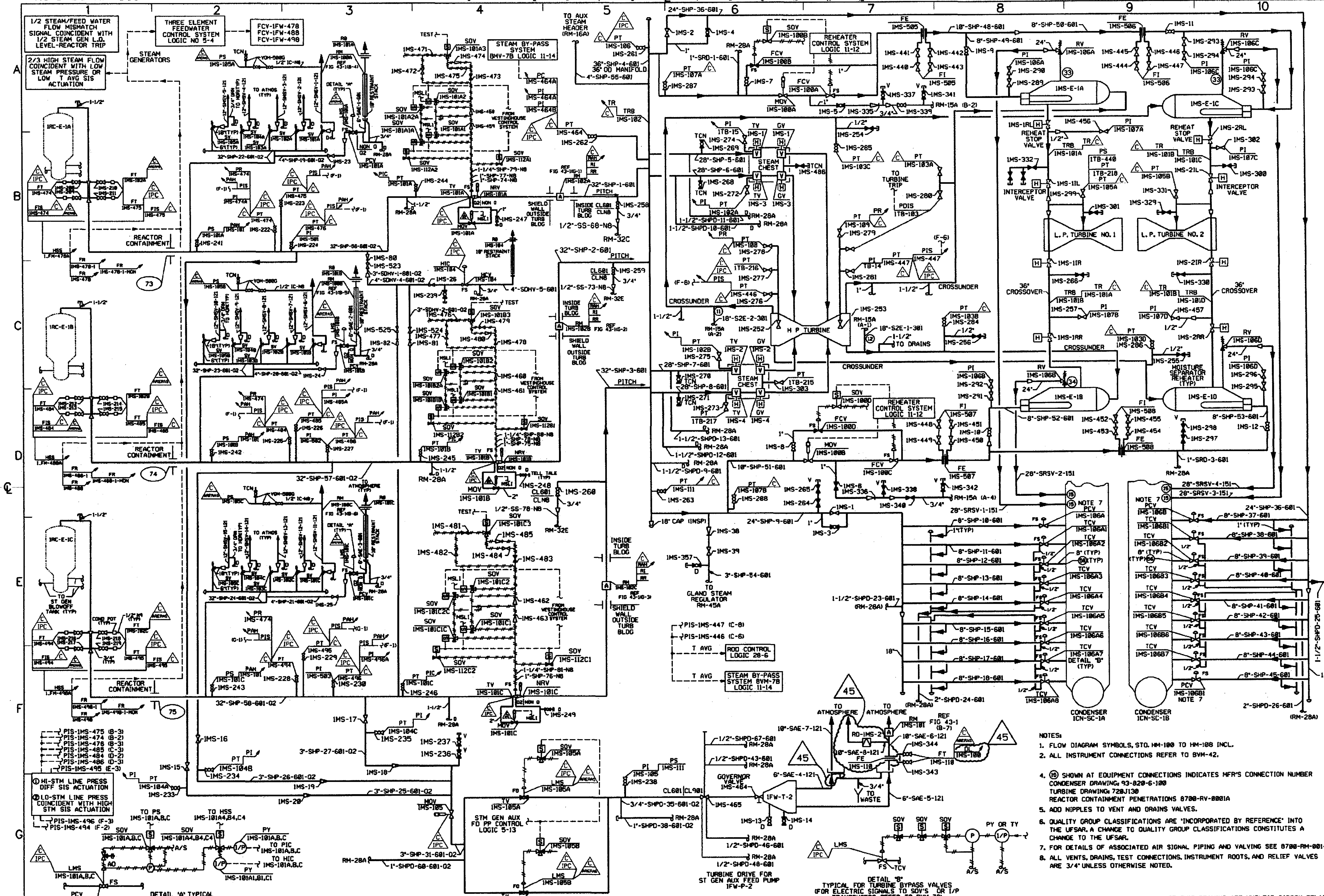


ALL VALVE AND EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR 'VS' UNLESS OTHERWISE NOTED

FIGURE 6.6-1
SUPPLEMENTARY LEAK COLLECTION RELEASE SYSTEM

REFERENCE: STATION DRAWING RM-516
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

No 8700-RM-0014A



NOTES:

1. FLOW DIAGRAM SYMBOLS, STD. HM-100 TO HM-108 INCL.
2. ALL INSTRUMENT CONNECTIONS REFER TO BVM-42.
4. (16) SHOWN AT EQUIPMENT CONNECTIONS INDICATES MFR'S CONNECTION NUMBER
   CONDENSER DRAWING: 93-820-6-100
   TURBINE DRAWING: 720J130
   REACTOR CONTAINMENT PENETRATIONS 8700-RV-0001A
5. ADD NIPPLES TO VENT AND DRAINS VALVES.
6. QUALITY GROUP CLASSIFICATIONS ARE 'INCORPORATED BY REFERENCE' INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.
7. FOR DETAILS OF ASSOCIATED AIR SIGNAL PIPING AND VALVING SEE 8700-RM-0014B.
8. ALL VENTS, DRAINS, TEST CONNECTIONS, INSTRUMENT ROOTS, AND RELIEF VALVES ARE 3/4" UNLESS OTHERWISE NOTED.

PORTIONS OF THIS DRAWING ARE NUCLEAR SAFETY RELATED

REVISION 45 INCORPORATED
DUN 13-0637-001-001
DUN 16-0515-001-001

FENOC — FIRST ENERGY NUCLEAR OPERATING COMPANY

SCALE NONE — DATE 11-2-16 — DRAWN BY LRH — FINAL APP FOR ISSUE 11/7/2016

BEAVER VALLEY POWER STATION UNIT NO 1

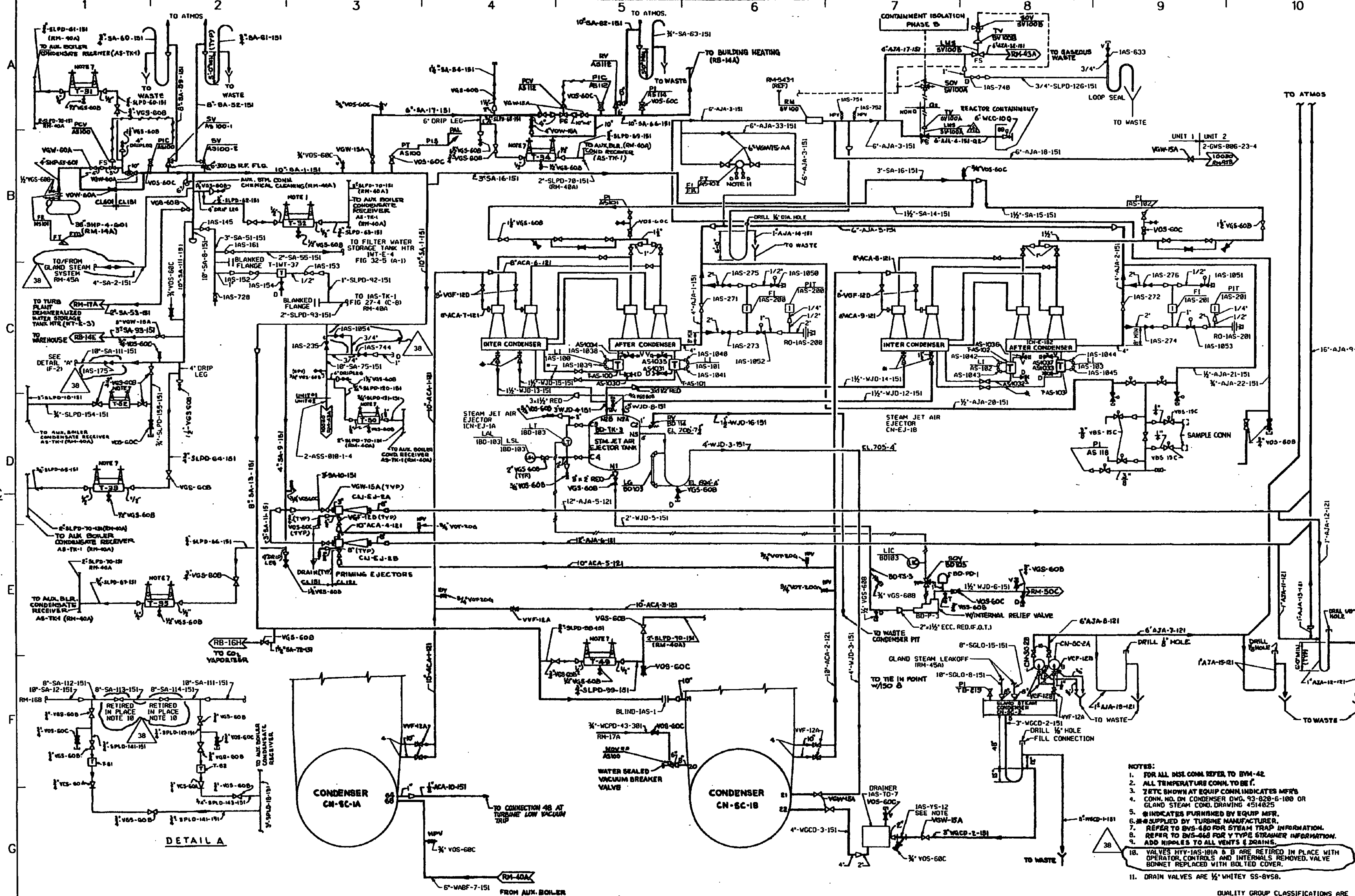
FLOW DIAGRAM
MAIN STEAM

AA — DWG NO 8700-RM-0014A — REV 45

PREPARED ON THE BVPS CAEDD SYSTEM

K:\U1\RM-0014A.DGN

No. 8700-RM-0016A



NOTES:

1. FOR ALL INST. CONN. REFER TO BVM-42.
2. ALL TEMPERATURE CONN. TO BE 1".
3. 2 ETC SHOWN AT EQUIP CONN. INDICATES MFR'S
4. CONN. NO. ON CONDENSER DWG. 93-820-6-100 OR GLAND STEAM COND. DRAWING 4514025
5. # INDICATES FURNISHED BY EQUIP MFR.
6. ## SUPPLIED BY TURBINE MANUFACTURER.
7. REFER TO BVS-680 FOR STEAM TRAP INFORMATION.
8. REFER TO BVS-468 FOR Y TYPE STRAINER INFORMATION.
9. ADD NIPPLES TO ALL VENTS & DRAINS.
10. VALVES HYV-1AS-101A & B ARE RETIRED IN PLACE WITH OPERATOR, CONTROLS AND INTERNALS REMOVED. VALVE BONNET REPLACED WITH BOLTED COVER. (38)
11. DRAIN VALVES ARE 1/2" WHITEY SS-8VS8.

QUALITY GROUP CLASSIFICATIONS ARE "INCORPORATED BY REFERENCE" INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.

PORTIONS OF THIS DWG ARE NUCLEAR SAFETY RELATED

REVISION 38 INCORPORATED
DUN 08-0157-001-002
DUN 10-0076-001-004
DUN 10-0404-003-002
DUN 10-0404-004-002

FENOC FirstEnergy Nuclear Operating Company

SCALE NONE | DATE 11/1/10 | DRAWN BY MVF

BEAVER VALLEY POWER STATION UNIT NO 1

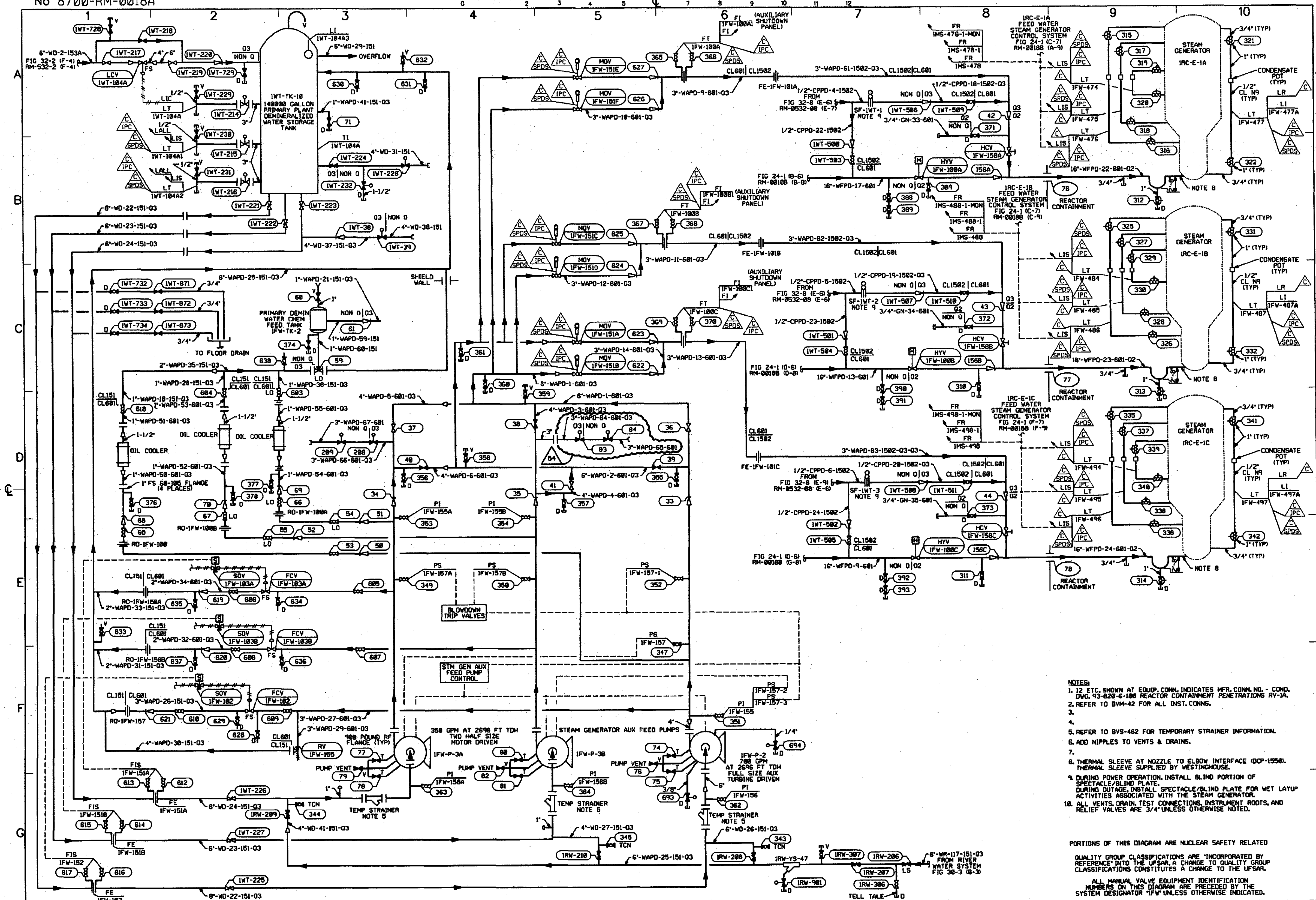
FLOW DIAGRAM
AUXILIARY STEAM & AIR REMOVAL-SH 1

AA | DWG. NO. 8700-RM-0016A | REV. 38

PREPARED ON THE BVPS CAEDD SYSTEM

03-DEC-2010 10:24 M:\u1\RM-0016A.DGN

No 8700-RM-0018A



NOTES:

1. 12 ETC. SHOWN AT EQUIP. CONN. INDICATES MFR. CONN. NO. - COND. DWG. 93-820-6-100 REACTOR CONTAINMENT PENETRATIONS RV-1A.
2. REFER TO BVM-42 FOR ALL INST. CONNS.
3.
4.
5. REFER TO BVS-462 FOR TEMPORARY STRAINER INFORMATION.
6. ADD NIPPLES TO VENTS & DRAINS.
7.
8. THERMAL SLEEVE AT NOZZLE TO ELBOW INTERFACE (DCP-1550). THERMAL SLEEVE SUPPLIED BY WESTINGHOUSE.
9. DURING POWER OPERATION, INSTALL BLIND PORTION OF SPECTACLE/BLIND PLATE. DURING OUTAGE, INSTALL SPECTACLE/BLIND PLATE FOR WET LAYUP ACTIVITIES ASSOCIATED WITH THE STEAM GENERATOR.
10. ALL VENTS, DRAIN, TEST CONNECTIONS, INSTRUMENT ROOTS, AND RELIEF VALVES ARE 3/4" UNLESS OTHERWISE NOTED.

PORTIONS OF THIS DIAGRAM ARE NUCLEAR SAFETY RELATED

QUALITY GROUP CLASSIFICATIONS ARE "INCORPORATED BY REFERENCE" INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.

ALL MANUAL VALVE EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR "1FW" UNLESS OTHERWISE INDICATED.

REVISIONS

REVISION 54 INCORPORATED DUN 13-0418-002-001

FENOC FIRST ENERGY NUCLEAR OPERATING COMPANY

BEAVER VALLEY POWER STATION UNIT NO 1

FLOW DIAGRAM
FEED WATER

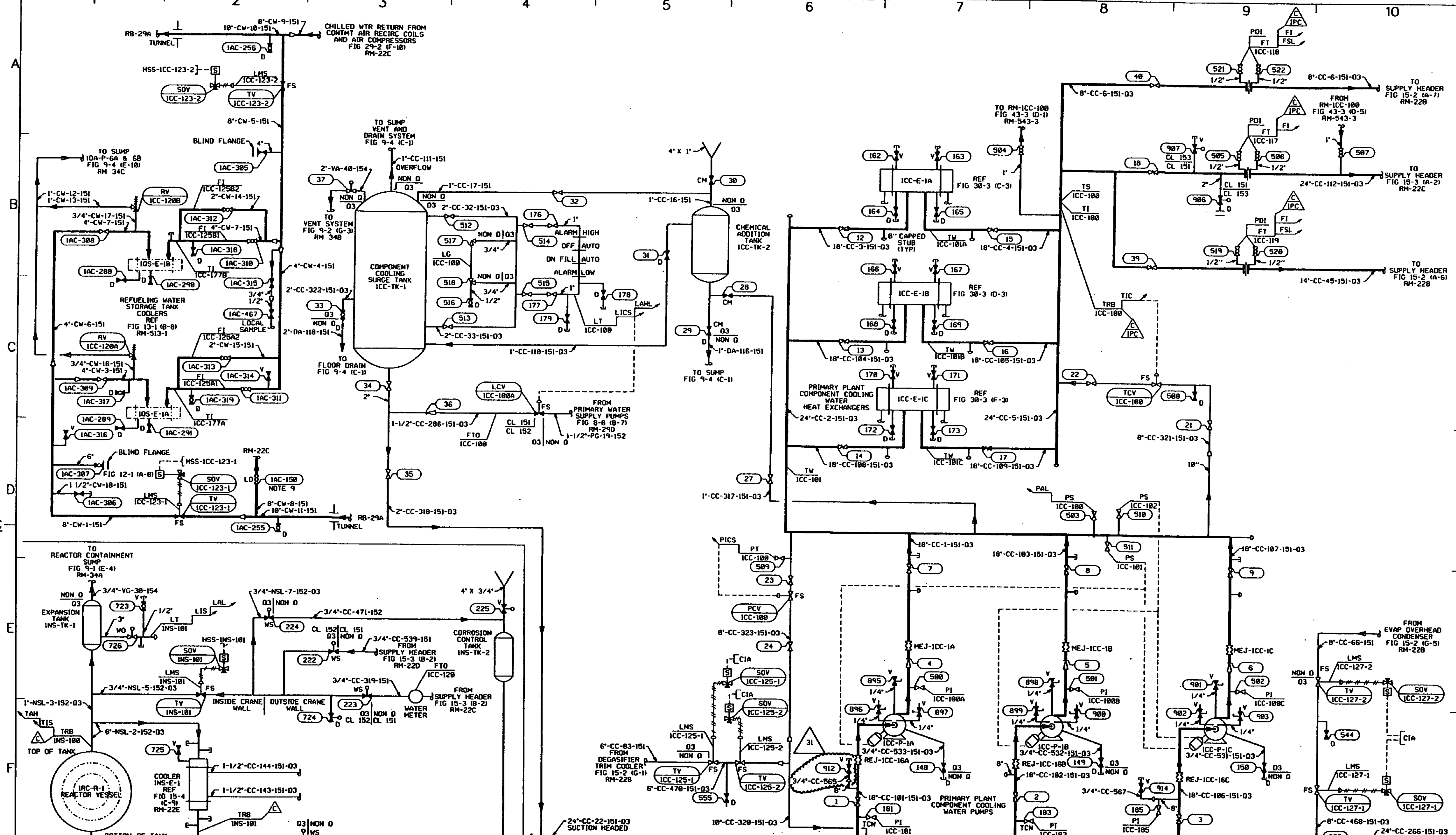
SCALE NONE | DATE 9-1-16 | DRAWN BY LRH | FINAL APP FOR ISSUE 9/1/2016

AA | DWG NO 8700-RM-0018A | REV 54

PREPARED ON THE BVPS CAEDD SYSTEM

K:\U1\RM-0018A.DGN

No. 8700-RM-0022A



NOTES:
1. ALL VENT, DRAIN, TEST CONNECTIONS, INSTRUMENT ROOTS AND RELIEF VALVES ARE 3/4" UNLESS OTHERWISE NOTED.
3. NUMBER SHOWN AT REACTOR CONTAINMENT LINE IS PIPING PENETRATION NUMBER ON RV-1A.
5. FOR INSTRUMENT CONNECTIONS REFER TO BVM-42.
6. VALVES IN PIPE SIZES: 10" OVER CLASS 121, 8" UNDER CLASS 151.
7. REFER TO BVS-462 FOR TEMPORARY STRAINER INFORMATION.
8. ADD NIPPLES TO VENTS & DRAINS.
9. VALVE REFERING TO THIS NOTE WAS A TV (GLOBE). OPERATOR WAS REMOVED AND A MANUAL OPERATOR PROVIVED.

REFERENCE DRAWINGS:
REACTOR CONTAINMENT PENETRATIONS RV-1A

FLOW DIAGRAM

AUX STEAM & AIR REMOVAL	RM-16A,B
COMPONENT COOLING WATER	RM-22B,C,D,E
BORON RECOVERY SYSTEM	RM-29A
LIQUID WASTE DISPOSAL SYSTEM	RM-30A,B
SAMPLE SYSTEM	RM-32A
VENT & DRAIN SYSTEM	RM-34A,B,C
REACTOR COOLANT SYSTEM	RM-37A,B
CHARGING & VOLUME CONTROL SYS.	RM-39B

FLOW DIAGRAM
CONDENSATE SYSTEM RM-17A
P&ID
RESIDUAL HEAT REMOVAL SYSTEM RM-410-1

PLUMBING SERVICE BUILDING - RB-18A,B,C,D,E,F,G
LOGIC DIAGRAMS LSK-12-7,8,11,12,13,14,18

PORTIONS OF THIS DIAGRAM ARE NUCLEAR SAFETY RELATED

QUALITY GROUP CLASSIFICATIONS ARE "INCORPORATED BY REFERENCE" INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR. ALL MANUAL VALVE EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR "1CCR" UNLESS OTHERWISE INDICATED.

REVISION 31 INCORPORATED
DUN 08-0530-001-002
ECP 08-0530

FENOC FIRSTENERGY NUCLEAR OPERATING COMPANY

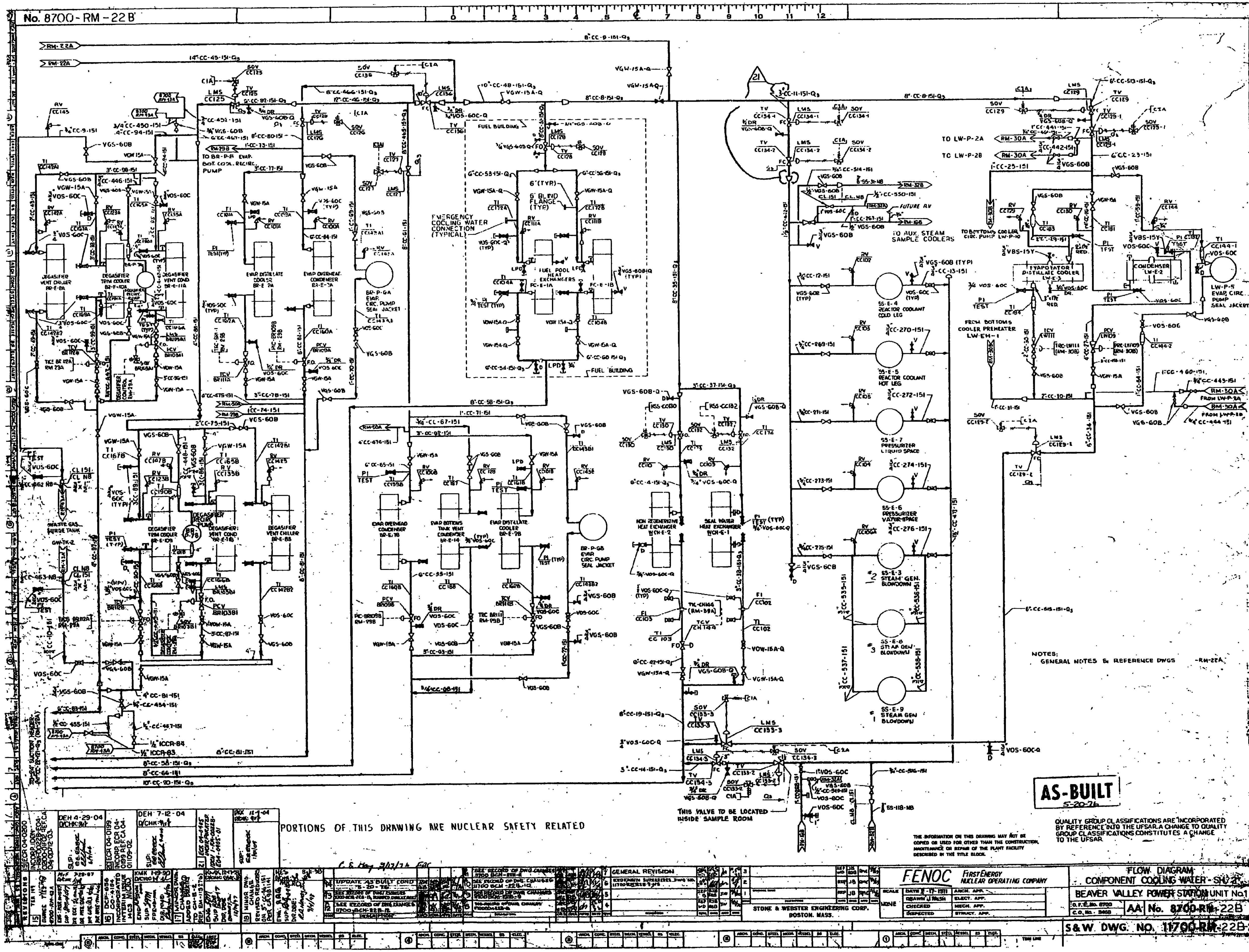
BEAVER VALLEY POWER STATION UNIT NO 1

FLOW DIAGRAM
COMPONENT COOLING - SH 1

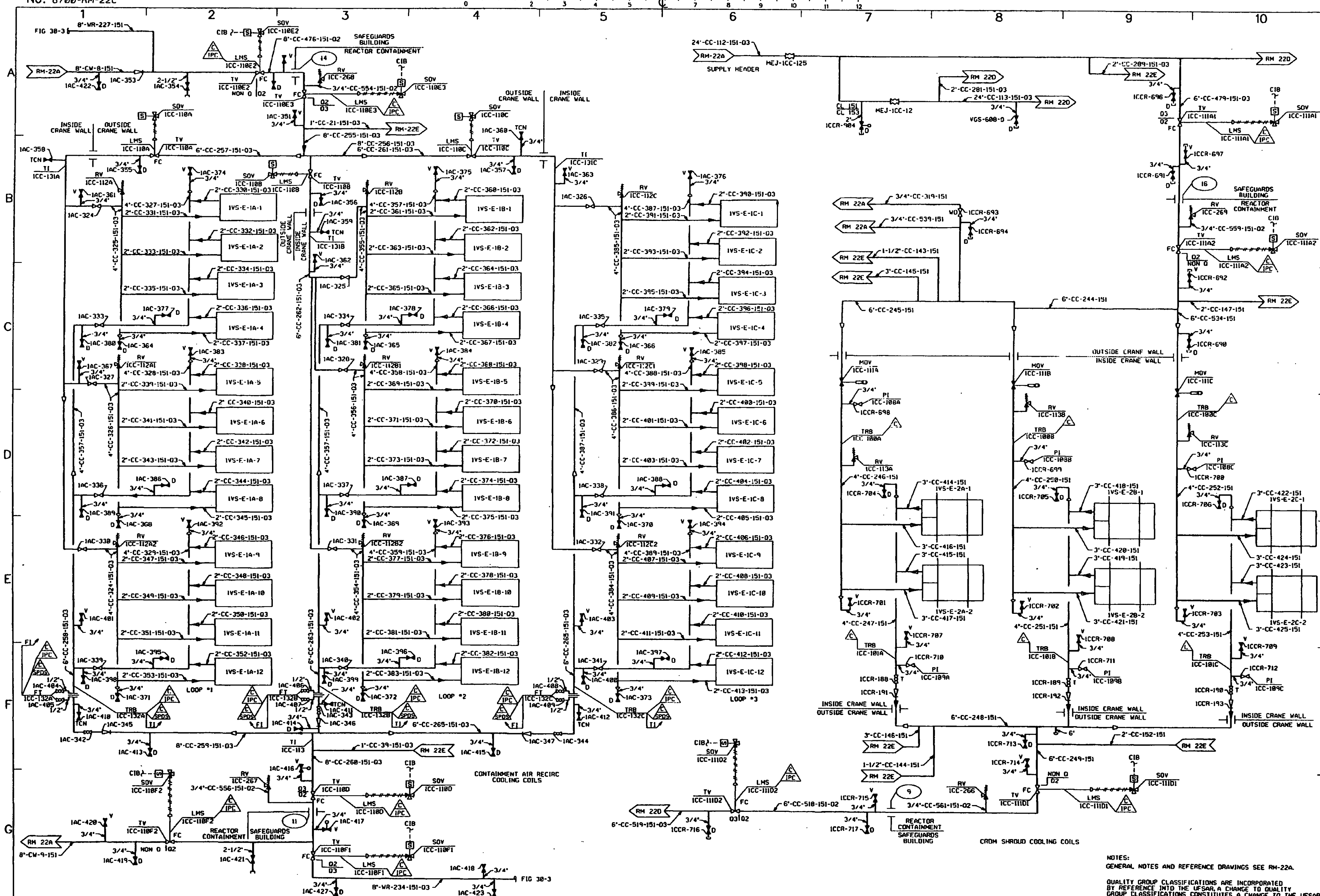
SCALE NONE | DATE 6/8/12 | DRAWN BY MVF | AA | DWG. NO. 8700- RM-0022A | REV. 31

08-JUN-2012 10:14 K:\u1\RM-0022A.DGN

PREPARED ON THE BVPS CAEDDY SYSTEM


No. 8700-RM-22B
EMERGENCY COOLING WATER CONNECTION (TYPICAL)
FUEL BUILDING
FUEL POOL HEAT EXCHANGERS
DEGASIFIER VENT CHILLER
DEGASIFIER TRIM COOLER
DEGASIFIER VENT COND
EVAP DISTILLATE COOLER
EVAP OVERHEAD CONDENSER
EVAP BOTTOMS TANK VENT CONDENSER
EVAP CIRC PUMP SEAL JACKET
WASTE GAS SURGE TANK
NON REGENERATIVE HEAT EXCHANGER
SEAL WATER HEAT EXCHANGER
REACTOR COOLANT COLD LEG
REACTOR COOLANT HOT LEG
PRESSURIZER LIQUID SPACE
PRESSURIZER VAPOR SPACE
STEAM GEN. BLOWDOWN
EVAPORATOR DISTILLATE COOLER
CONDENSER
TO LW-P-2A
TO LW-P-2B
TO AUX. STEAM SAMPLE COOLERS
FROM BOTTOMS COOLER PREHEATER
THIS VALVE TO BE LOCATED INSIDE SAMPLE ROOM
NOTES:
GENERAL NOTES & REFERENCE DWGS -RM-22A
AS-BUILT
5-20-76
QUALITY GROUP CLASSIFICATIONS ARE INCORPORATED BY REFERENCE INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.
THE INFORMATION ON THIS DRAWING MAY NOT BE COPIED OR USED FOR OTHER THAN THE CONSTRUCTION, MAINTENANCE OR REPAIR OF THE PLANT FACILITY DESCRIBED IN THE TITLE BLOCK.
PORTIONS OF THIS DRAWING ARE NUCLEAR SAFETY RELATED
FENOC FIRSTENERGY NUCLEAR OPERATING COMPANY
STONE & WEBSTER ENGINEERING CORP. BOSTON, MASS.
FLOW DIAGRAM
COMPONENT COOLING WATER - SH 2
BEAVER VALLEY POWER STATION UNIT No.1
SCALE NONE
AA No. 8700-RM-22B
S&W DWG. NO. 11700-RM-22B-21

No. 8700-RM-22C



NOTES:
GENERAL NOTES AND REFERENCE DRAWINGS SEE RM-22A.

QUALITY GROUP CLASSIFICATIONS ARE INCORPORATED BY REFERENCE INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.

REVISION 22
INCORPORATED
ECR-04-0310
IDCN 1-RM-0022C-E04-0310-01
IDCN 1-RM-0022C-E04-0310-02
DIAGRAM REDRAWN
ECR-04-0310 CHANGED SAFETY CLASS OF COMPONENTS
SEE IDCNS FOR LISTING OF COMPONENTS

FENOC FIRSTENERGY NUCLEAR OPERATING COMPANY

SCALE NONE | DATE 9-8-05 | DRAWN BY MVF

BEAVER VALLEY POWER STATION UNIT 1

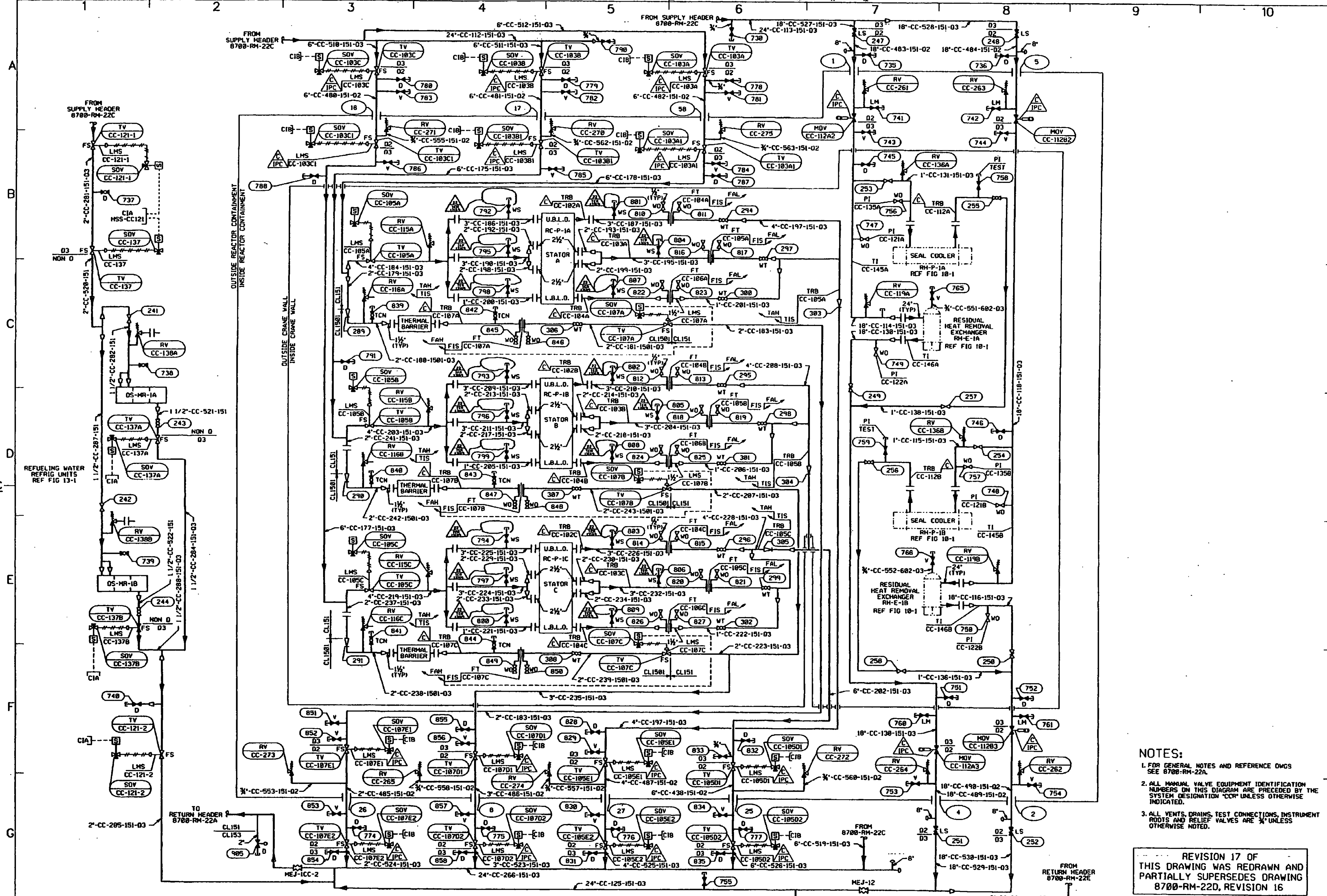
FLOW DIAGRAM
COMPONENT COOLING WATER SH 3

AA | DWG.NO. 8700-RM-22C | REV. 22

PREPARED ON THE BVPS CAEDDI SYSTEM

09-SEP-2005 14:26 M:\u1\RM0220C0.DGN

No. 8700-RM-22D



NOTES:

1. FOR GENERAL NOTES AND REFERENCE DWGS SEE 8700-RM-22A.
2. ALL MANUAL VALVE EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATION "CCR" UNLESS OTHERWISE INDICATED.
3. ALL VENTS, DRAINS, TEST CONNECTIONS, INSTRUMENT ROOTS AND RELIEF VALVES ARE ¾" UNLESS OTHERWISE NOTED.

REVISION 17 OF THIS DRAWING WAS REDRAWN AND PARTIALLY SUPERSEDES DRAWING 8700-RM-22D, REVISION 16

Duquesne Light Company NUCLEAR GROUP PITTSBURGH, PA

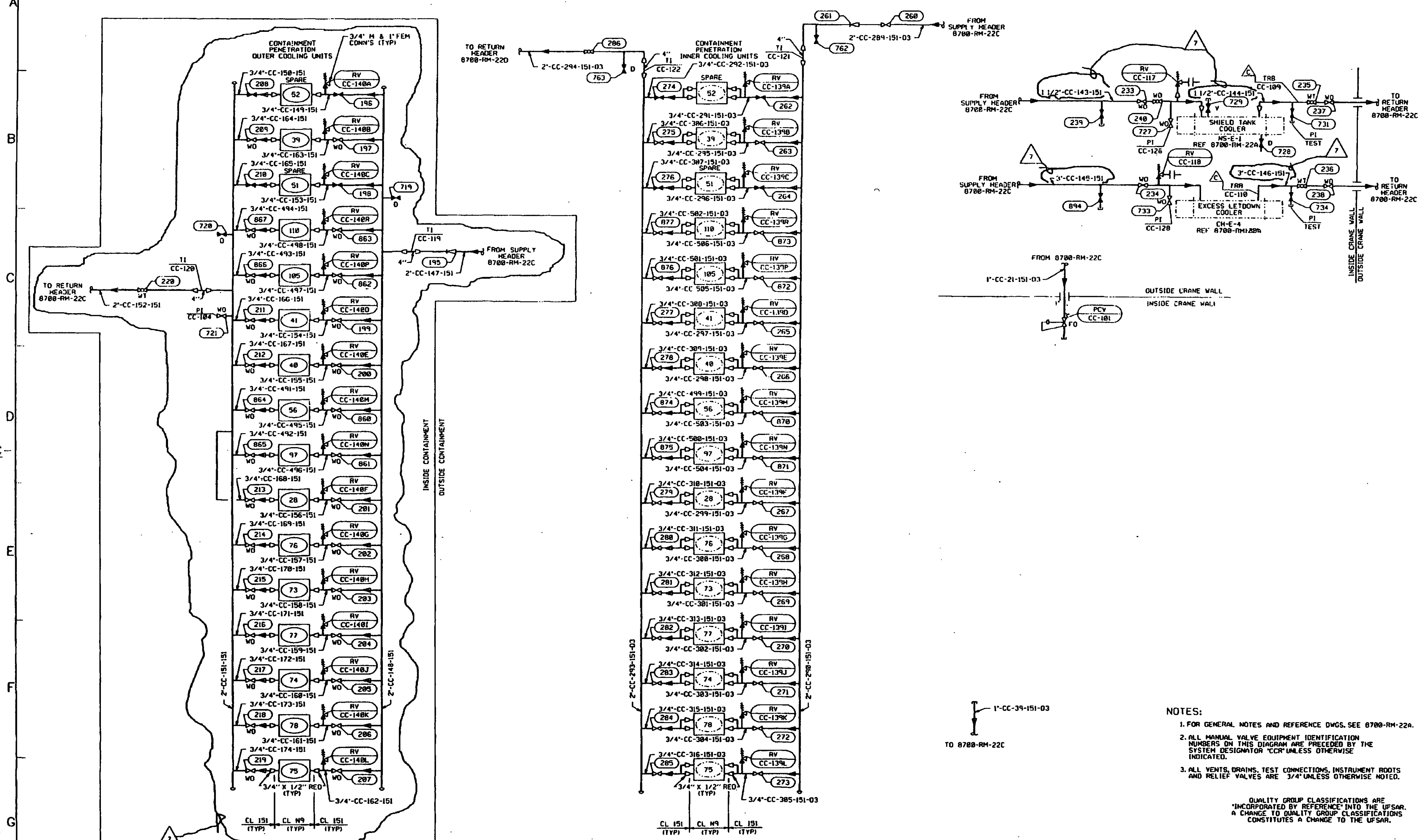
BEAVER VALLEY POWER STATION UNIT NO. 1

FLOW DIAGRAM COMPONENT COOLING WATER

SCALE	DATE	DRAWN BY	DWG. NO.	REV.
NONE	5-15-97	EBH/DKK	8700-RM-22D	22

PREPARED ON THE CADDS SYSTEM

No. 8700-RM-22E



NOTES:

1. FOR GENERAL NOTES AND REFERENCE DWGS. SEE 8700-RM-22A.
2. ALL MANUAL VALVE EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR "CCR" UNLESS OTHERWISE INDICATED.
3. ALL VENTS, DRAINS, TEST CONNECTIONS, INSTRUMENT ROOTS AND RELIEF VALVES ARE 3/4" UNLESS OTHERWISE NOTED.

QUALITY GROUP CLASSIFICATIONS ARE "INCORPORATED BY REFERENCE" INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.

REVISION 1 OF THIS DRAWING WAS REDRAWN FROM & PARTIALLY SUPERSEDES DRAWING 8700-RM-22D, REVISION 16

REVISION 7 INCORPORATED ECR 04-0318 IDCN-1-RM-0022E-E04-0318-01

FENOC FIRSTENERGY NUCLEAR OPERATING COMPANY

BEAVER VALLEY POWER STATION UNIT 1

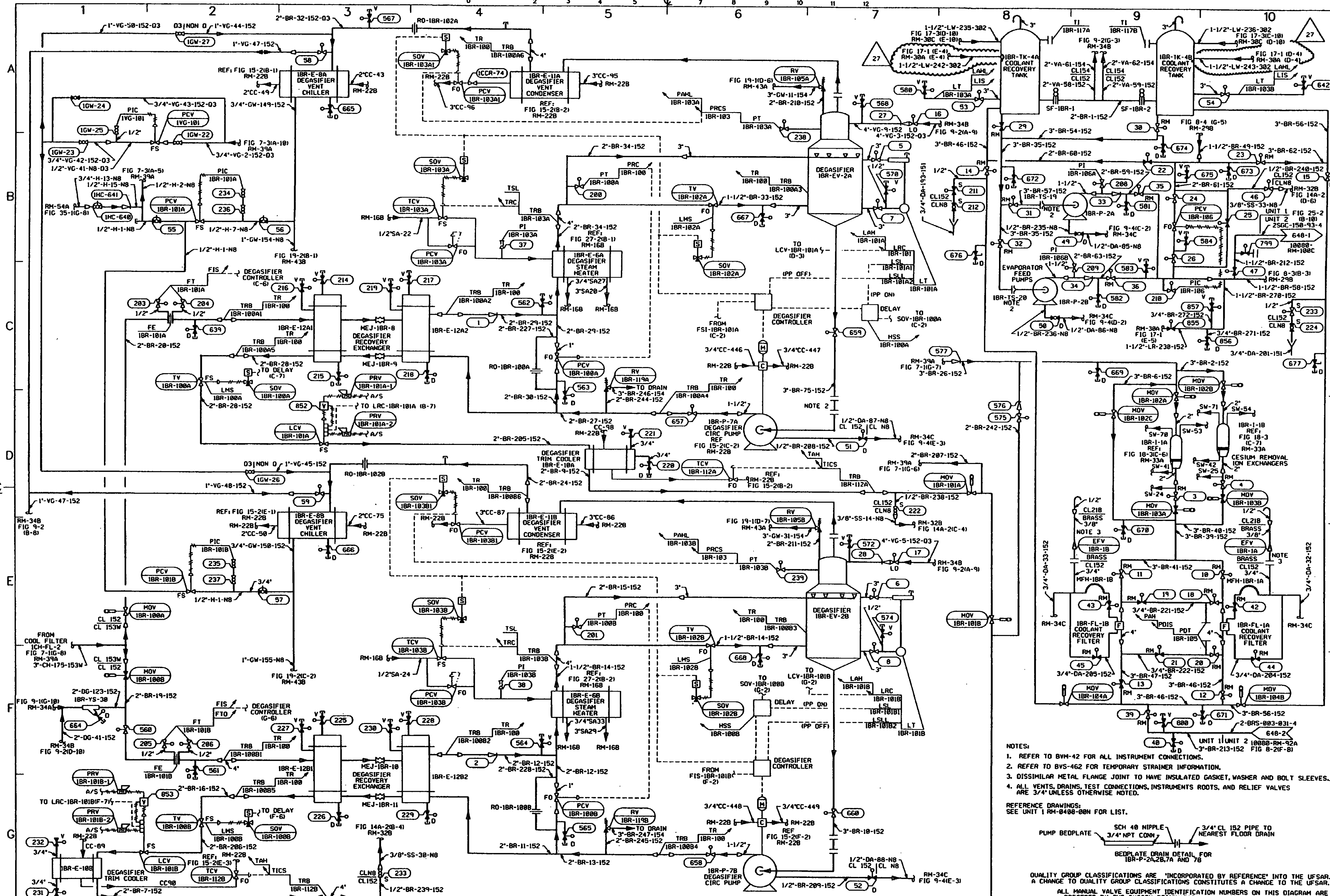
FLOW DIAGRAM COMPONENT COOLING WATER

NTS | 8-11-05 | MVF | AA | 8700-RM-22E | 7

PREPARED ON THE BVPS CAEDOC SYSTEM

07-SEP-2005 13:44

No. 8700-RM-0029A



NOTES:

1. REFER TO BVM-42 FOR ALL INSTRUMENT CONNECTIONS.
2. REFER TO BVS-462 FOR TEMPORARY STRAINER INFORMATION.
3. DISSIMILAR METAL FLANGE JOINT TO HAVE INSULATED GASKET, WASHER AND BOLT SLEEVES.
4. ALL VENTS, DRAINS, TEST CONNECTIONS, INSTRUMENTS ROOTS, AND RELIEF VALVES ARE 3/4" UNLESS OTHERWISE NOTED.

REFERENCE DRAWINGS:
SEE UNIT 1 RM-0408-00N FOR LIST.

BEDPLATE DRAIN DETAIL FOR 1BR-P-2A,2B,7A AND 7B

QUALITY GROUP CLASSIFICATIONS ARE 'INCORPORATED BY REFERENCE' INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.

ALL MANUAL VALVE EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR '1BR' UNLESS OTHERWISE INDICATED.

PORTIONS OF THIS DIAGRAM ARE NUCLEAR SAFETY RELATED

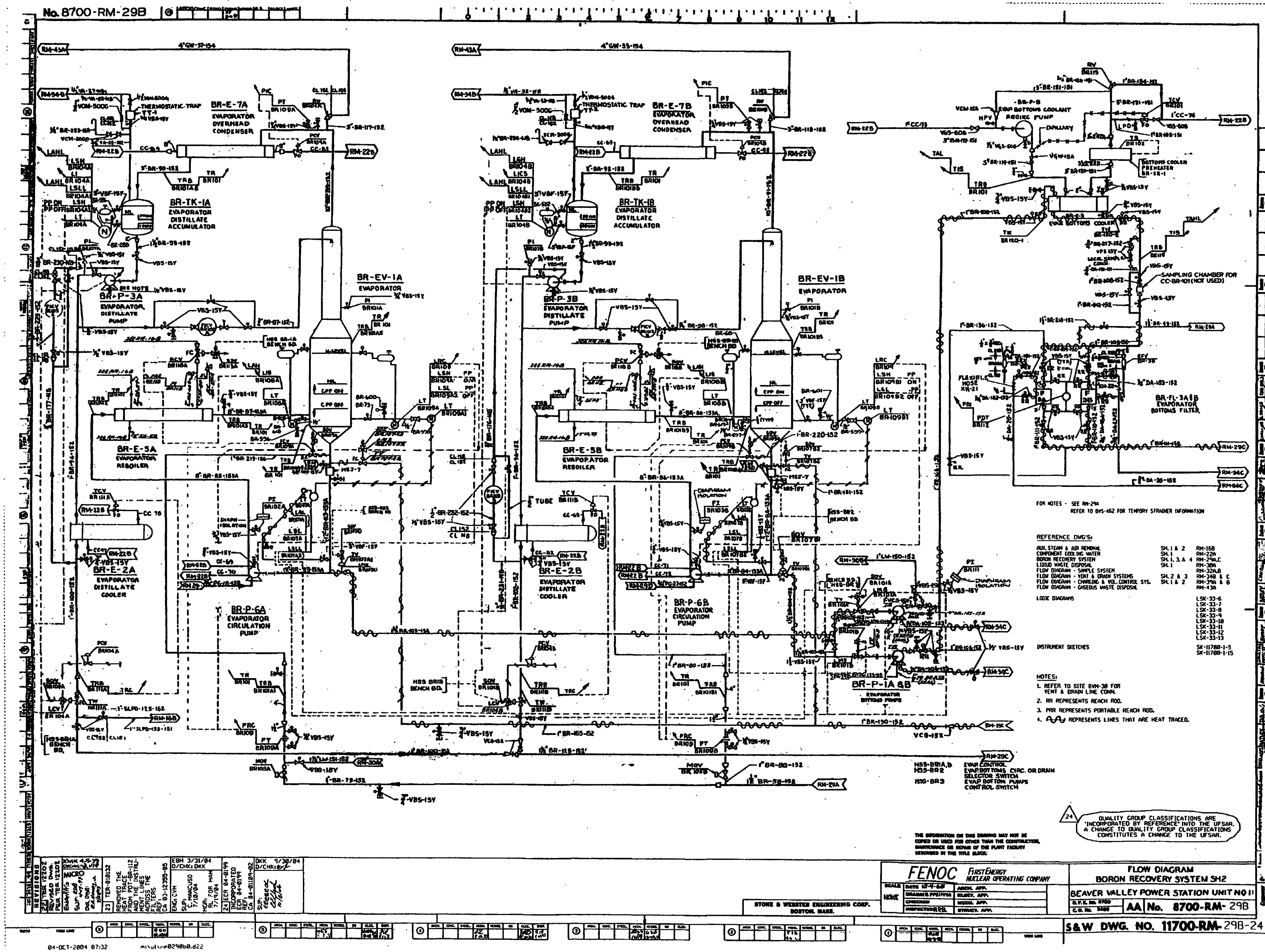
REVISION 27
INCORPORATED
DUN 11-0049-001-047
ECP 11-0049

FENOC FIRSTENERGY NUCLEAR OPERATING COMPANY

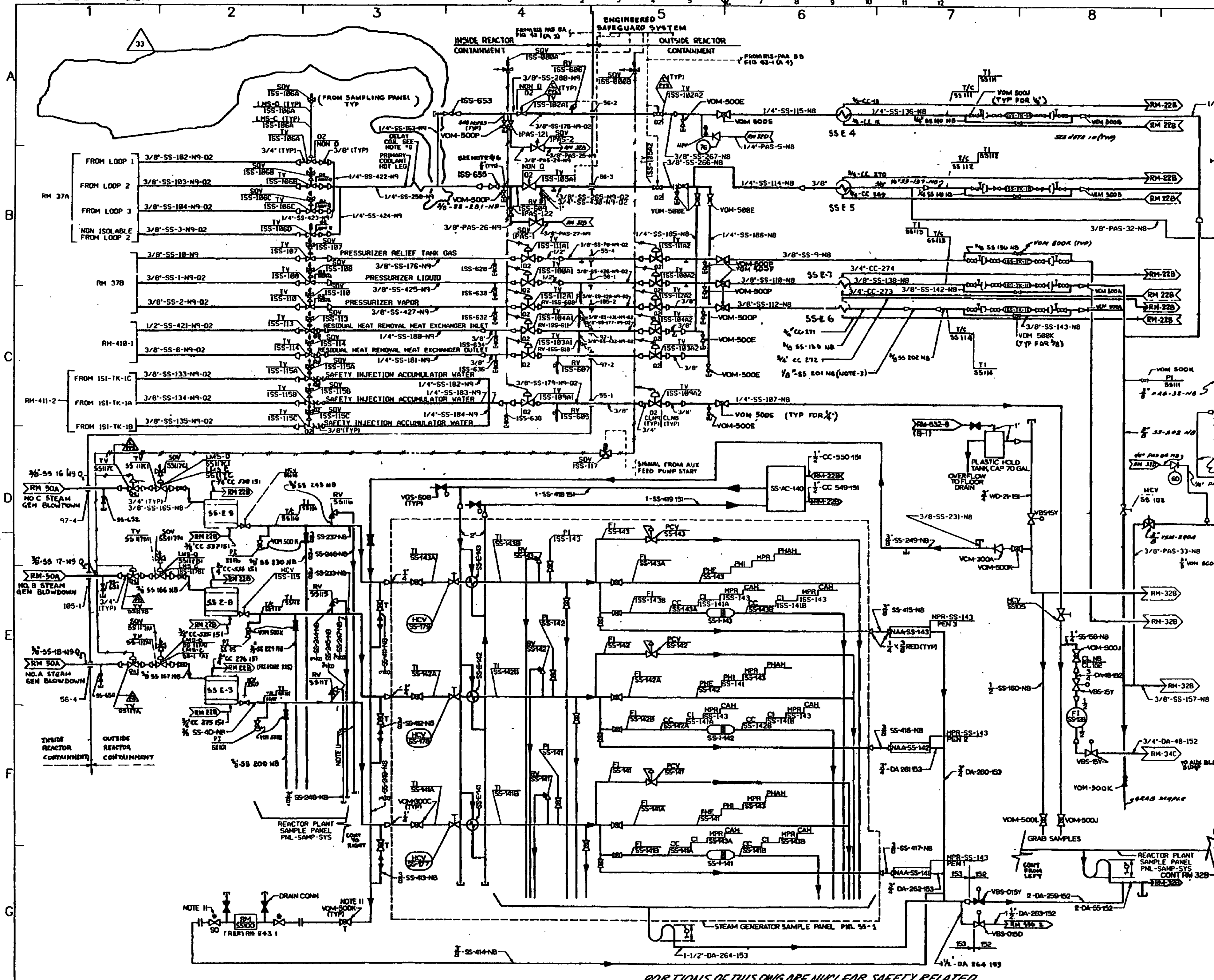
BEAVER VALLEY POWER STATION UNIT NO 1

FLOW DIAGRAM
BORON RECOVERY SYSTEM-SH 1

SCALE NONE | DATE 03/07/2012 | DRAWN BY MVF | AA | DWG.NO. 8700-RM-0029A | REV. 27


No. 8700-RM-29B
BR-E-7A
EVAPORATOR OVERHEAD CONDENSER
BR-TK-1A
EVAPORATOR DISTILLATE ACCUMULATOR
BR-P-3A
EVAPORATOR DISTILLATE PUMP
BR-EV-1A
EVAPORATOR
BR-E-5A
EVAPORATOR REBOILER
BR-E-2A
EVAPORATOR DISTILLATE COOLER
BR-P-6A
EVAPORATOR CIRCULATION PUMP
BR-E-7B
EVAPORATOR OVERHEAD CONDENSER
BR-TK-1B
EVAPORATOR DISTILLATE ACCUMULATOR
BR-P-3B
EVAPORATOR DISTILLATE PUMP
BR-EV-1B
EVAPORATOR
BR-E-5B
EVAPORATOR REBOILER
BR-E-2B
EVAPORATOR DISTILLATE COOLER
BR-P-6B
EVAPORATOR CIRCULATION PUMP
BR-FL-3A&B
EVAPORATOR BOTTOMS FILTER
BR-P-1A &B
EVAPORATOR BOTTOMS PUMPS
BOTTOMS COOLER PREHEATER
SAMPLING CHAMBER FOR CC-BR-101 (NOT USED)
FOR NOTES - SEE RM-29A
REFER TO BVS-462 FOR TEMPORY STRAINER INFORMATION
REFERENCE DWG'S:
LOGIC DIAGRAMS
LSK-33-6
LSK-33-7
LSK-33-8
LSK-33-9
LSK-33-10
LSK-33-11
LSK-33-12
LSK-33-13
INSTRUMENT SKETCHES
SK-11700-1-5
SK-11700-1-15
NOTES:
1. REFER TO SITE BVM-30 FOR VENT & DRAIN LINE CONN.
2. RR REPRESENTS REACH ROD.
3. PRR REPRESENTS PORTABLE REACH ROD.
4. REPRESENTS LINES THAT ARE HEAT TRACED.
QUALITY GROUP CLASSIFICATIONS ARE "INCORPORATED BY REFERENCE" INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.
FENOC FIRSTENERGY NUCLEAR OPERATING COMPANY
FLOW DIAGRAM
BORON RECOVERY SYSTEM SH2
BEAVER VALLEY POWER STATION UNIT NO II
AA No. 8700-RM-29B
STONE & WEBSTER ENGINEERING CORP. BOSTON, MASS.
S&W DWG. NO. 11700-RM-29B-24
04-OCT-2004 07:32

No. 8700-RM-32A



TYPICAL FOR ALL SAMPLE BOMBS (SEE TABLE BELOW FOR LINE NO'S)

BOMB NO	DEPRESSURING LINE NO
SS-TK 1A	1/4" SS 210-N8
SS TK-1A	1/4" SS 211-N8
SS-TK 1B	1/4" SS-212-N8
SS TK 1B	1/4" SS-213 N8
SS-TK-1C	1/4" SS 214-N8
SS TK-1C	1/4" SS 215 N8
1SS-TK-1D	1/4"-SS-216-N8
SS TK 1D	1/4" SS 217 N8
SS-TK-1E	1/4" SS-218-N8
SS TK 1E	1/4" SS-219 N8

NOTES:

1. LOCAL SAMPLE POINTS ARE SHOWN ON THE APPLICABLE DRAWING AND NOT ON THIS DRAWING.
2. FOR ALL INSTRUMENT CONN REFER TO BVM-42.
3. APPROX 90 FEET 1/8" OD X 0.035" WALL SS TUBING WOUND IN 4" DIAMETER COIL.
4. ALLOW WALL SLEEVES AND PANEL SPACE FOR FUTURE LINES.
5. PROVIDE FOR CAPPED LINE SIZE TEST CONNECTION ON ALL CONTAINMENT PENETRATIONS.
6. APPROX 50'-0" OF 1/4" OD CL N9 TUBING.
8. • SUPPLIED BY EQUIP MFG.
9. ADD NIPPLE TO VENTS AND DRAINS.
10. VALVE NUMBERS REMOVED. DID NOT AGREE WITH STATION VOND 8700-RM-179A-6.
11. SEE CR 99-1808 SINCE UPSTREAM RELIEF VALVES ARE CAPPED. RAD MONITOR RM-1SS-100 IS PROTECTED WITH THROTTLED INLET AND SEALED OPEN OUTLET.
12. QUALITY GROUP CLASSIFICATIONS ARE "INCORPORATED BY REFERENCE" INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO UFSAR.

REFERENCE DWGS

CONDENSATE SYSTEM	RM-17A
COMPONENT COOLING WATER	RM-22B
BORON RECOVERY SYSTEM	RM-29A & C
LIQUID WASTE DISPOSAL SYSTEM	RM-30A & B
VENT AND DRAIN SYSTEM	RM-34B & C
REACTOR COOLANT SYSTEM	RM-37A & B
RESIDUAL HEAT REMOVAL SYSTEM	RM-410-1
CHARGING AND VOLUME CONTROL SYSTEM	RM-39A & B
SAFETY INJECTION SYSTEM	RM-411-2
BLOWDOWN SYSTEM	RM-50A

REVISION 33 INCORPORATED ECP-05-0279 IDCN-1-RM-0032A-E05-0279-01

FENOC FirstEnergy Nuclear Operating Company

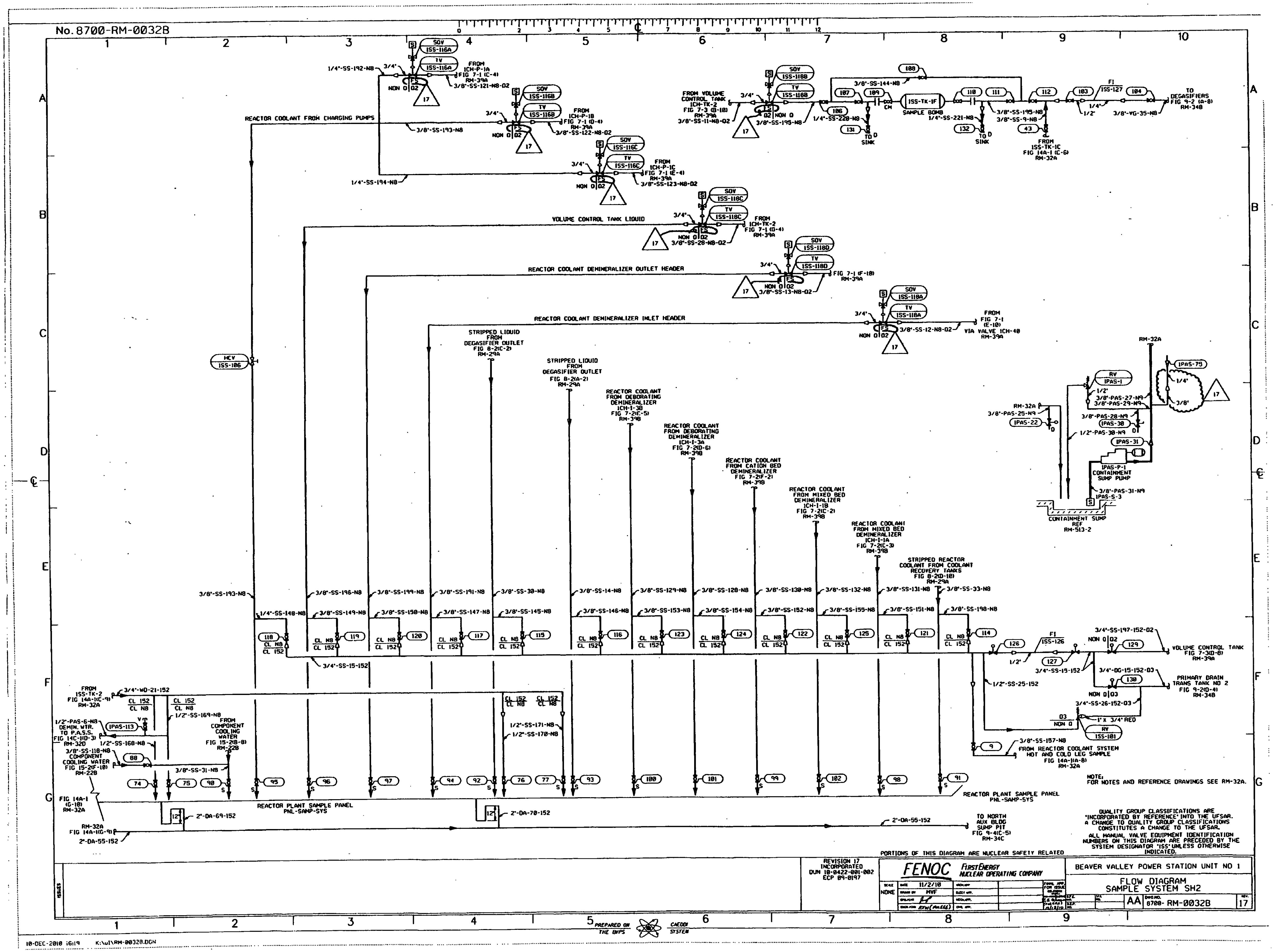
SCALE NONE | DATE 4-11-06 | DRAWN BY MVF | EBH EEE | EEE | FINAL APP. FOR ISSUE EEE 4/11/06

BEAVER VALLEY POWER STATION UNIT 1

FLOW DIAGRAM
SAMPLE SYSTEM SH 1

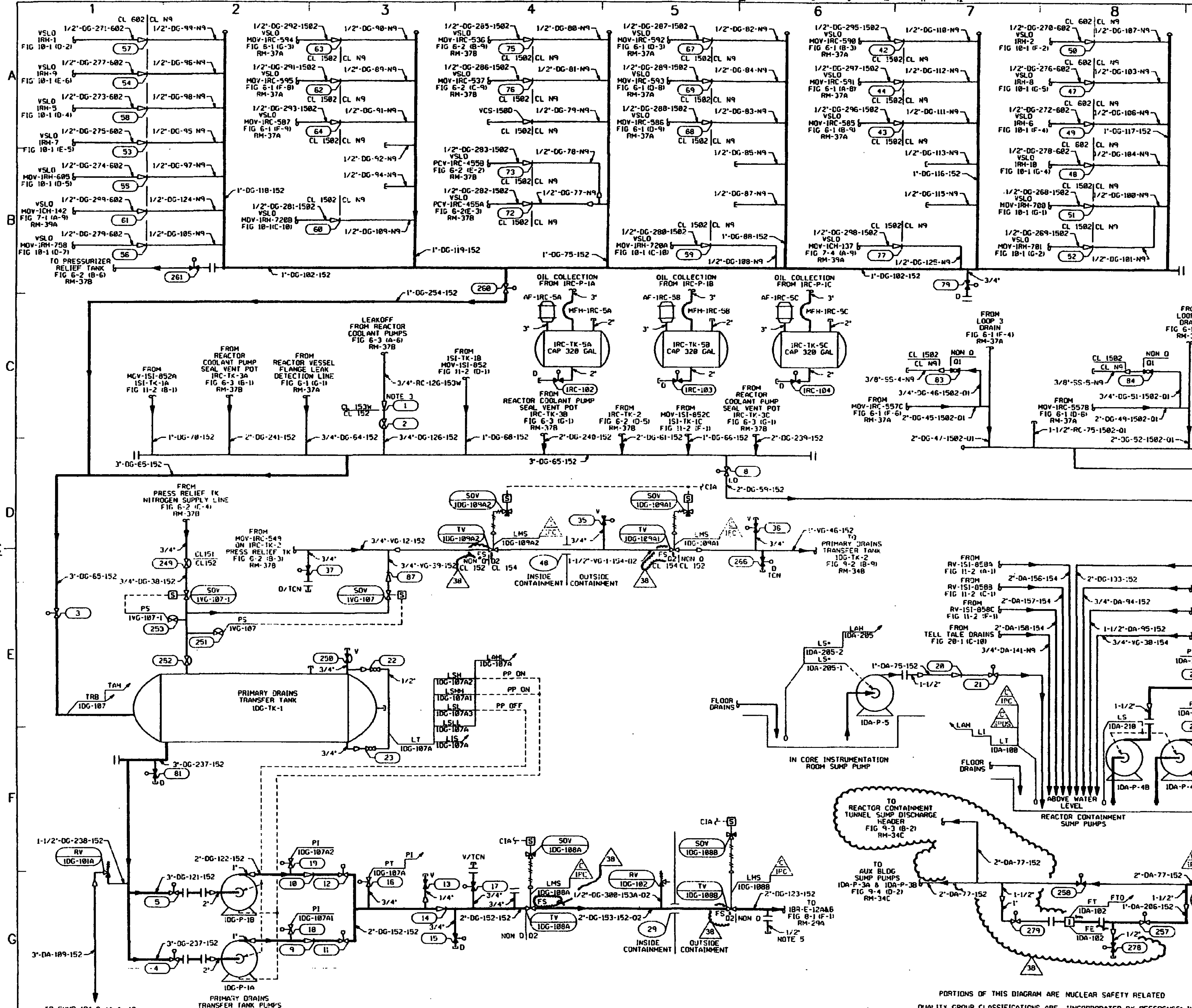
AA | DWG NO. 8700-RM-32A | REV. 33

20-APR-2006 16:59



No. 8700-RM-0032B
REACTOR COOLANT FROM CHARGING PUMPS
VOLUME CONTROL TANK LIQUID
REACTOR COOLANT DEMINERALIZER OUTLET HEADER
REACTOR COOLANT DEMINERALIZER INLET HEADER
FROM VOLUME CONTROL TANK 1CH-TK-2 FIG 7-3 (B-10) RM-39A
SAMPLE BOMB
TO DEGASIFIERS FIG 9-2 (A-8) RM-34B
STRIPPED LIQUID FROM DEGASIFIER OUTLET FIG 8-2(C-2) RM-29A
STRIPPED LIQUID FROM DEGASIFIER OUTLET FIG 8-2(A-2) RM-29A
REACTOR COOLANT FROM DEBORATING DEMINERALIZER 1CH-I-3B FIG 7-2(C-5) RM-39B
REACTOR COOLANT FROM DEBORATING DEMINERALIZER 1CH-I-3A FIG 7-2(D-6) RM-39B
REACTOR COOLANT FROM CATION BED DEMINERALIZER FIG 7-2(F-2) RM-39B
REACTOR COOLANT FROM MIXED BED DEMINERALIZER 1CH-I-1B FIG 7-2(C-2) RM-39B
REACTOR COOLANT FROM MIXED BED DEMINERALIZER 1CH-I-1A FIG 7-2(C-3) RM-39B
STRIPPED REACTOR COOLANT FROM COOLANT RECOVERY TANKS FIG 8-2(D-10) RM-29A
IPAS-P-1 CONTAINMENT SUMP PUMP
CONTAINMENT SUMP REF RM-5(3-2
VOLUME CONTROL TANK FIG 7-3(D-8) RM-39A
PRIMARY DRAIN TRANS TANK NO 2 FIG 9-2(D-4) RM-34B
FROM REACTOR COOLANT SYSTEM HOT AND COLD LEG SAMPLE FIG 14A-1(A-8) RM-32A
REACTOR PLANT SAMPLE PANEL PNL-SAMP-SYS
TO NORTH AUX BLDG SUMP PIT FIG 9-4(C-5) RM-34C
FROM COMPONENT COOLING WATER FIG 15-2B(-8) RM-22B
NOTE: FOR NOTES AND REFERENCE DRAWINGS SEE RM-32A.
QUALITY GROUP CLASSIFICATIONS ARE "INCORPORATED BY REFERENCE" INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.
ALL MANUAL VALVE EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR "1SS" UNLESS OTHERWISE INDICATED.
PORTIONS OF THIS DIAGRAM ARE NUCLEAR SAFETY RELATED
REVISION 17 INCORPORATED DUN 10-0422-001-002 ECP 09-0197
FENOC FIRSTENERGY NUCLEAR OPERATING COMPANY
BEAVER VALLEY POWER STATION UNIT NO 1
FLOW DIAGRAM SAMPLE SYSTEM SH2
11/2/10
AA 8700-RM-0032B 17
PREPARED ON THE BVPS CAEDDI SYSTEM
10-DEC-2010 16:19 K:\u1\RM-0032B.DGN

No. 8700-RM-0034A



NOTES:

1. NUMBER SHOWN AT CONTAINMENT LINE IS PIPING PENETRATION ON 1RV-1A.
2. REFER TO BVM-42 FOR ALL INSTRUMENT CONNECTIONS.
3. LOCATE A MINIMUM OF 2 FEET BELOW NUMBER 3 REACTOR COOLANT PUMP SEAL AND A MINIMUM OF 8 FEET ABOVE 1DG-TK-1.
4. REFER TO BVS-462 FOR TEMPORARY STRAINER INFORMATION.
5. TEST CONNECTION FLANGE WITH 1/2 INCH SWAGELOCK CONNECTION.
6. VSLO LINES ARE NON-Q.

REFERENCE DRAWINGS:

COMPONENT COOLING WATER SH 1, 2, & 3	RM-22A, B, & C
SAMPLE SYSTEM	RM-32A
VENT & DRAIN SYSTEM	RM-34A, B, & C
REACTOR COOLANT SYSTEM	RM-37A & B
BORON RECOVERY SYSTEM	RM-29A, B, C, & D
LIQUID WASTE DISPOSAL SYSTEM	RM-30A & B
RESIDUAL HEAT REMOVAL SYSTEM	RM-410-1
SAFETY INJECTION SYSTEM	RM-411-1 & 2
CHARGING AND VOLUME CONTROL SYS SH 1 & 2	RM-39A & B
STEAM GENERATOR BLOWDOWN	RM-50A
REACTOR CONTAINMENT - SCHEDULE OF PENETRATIONS	RV-1A
GASEOUS WASTE DISPOSAL SYSTEM	RM-43A
PLUMBING, ROOF DRAINS & FIRE PROTECTION	RB-18C
SPENT FUEL POOL SYSTEM	RM-420-1

SYMBOLS:

VSLO - VALVE STEM LEAKOFF
AC - ADMINISTRATIVELY CONTROLLED
TS - TEMPORARY STRAINER
• - DESIGNATES SUPPLIED WITH EQUIPMENT
LO - LOCKED OPEN

PORTIONS OF THIS DIAGRAM ARE NUCLEAR SAFETY RELATED

QUALITY GROUP CLASSIFICATIONS ARE "INCORPORATED BY REFERENCE" INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.

ALL MANUAL VALVE EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR "1DV" UNLESS OTHERWISE INDICATED.

REVISION 38 INCORPORATED
ECP 05-0423
ECP 09-0197
1DCN-1-RM-0034A-E05-0423-01

FENOC FirstEnergy Nuclear Operating Company

SCALE NONE | DATE 6/30/11 | DRAWN BY MVF

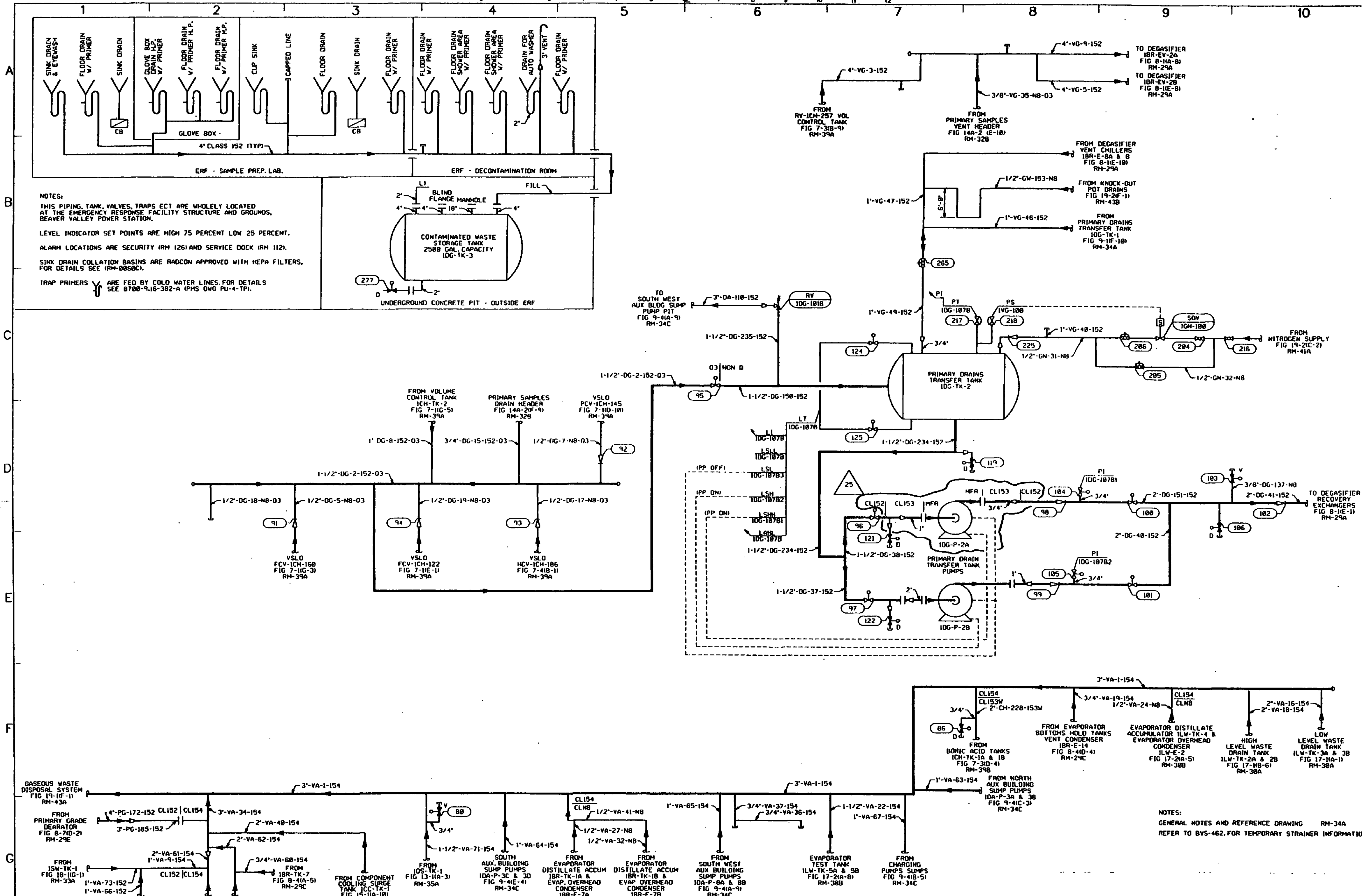
BEAVER VALLEY POWER STATION UNIT NO 1

FLOW DIAGRAM
VENT & DRAIN SYSTEM SH 1

AA | DWG NO. 8700-RM-0034A | REV 38

PREPARED ON THE BVPS CAEDD SYSTEM

No. 8700-RM-0034B



QUALITY GROUP CLASSIFICATIONS ARE INCORPORATED BY REFERENCE INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.

PORTIONS OF THIS DIAGRAM ARE NUCLEAR SAFETY RELATED

ALL MANUAL VALVE EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR "1DV" UNLESS OTHERWISE INDICATED.

REVISION 25 INCORPORATED ECP-05-0315 IDCN 1-RM-0034B-E05-0315-01

FENOC FirstEnergy Nuclear Operating Company

Scale: NONE | Date: 8-20-07 | Drawn by: MVF

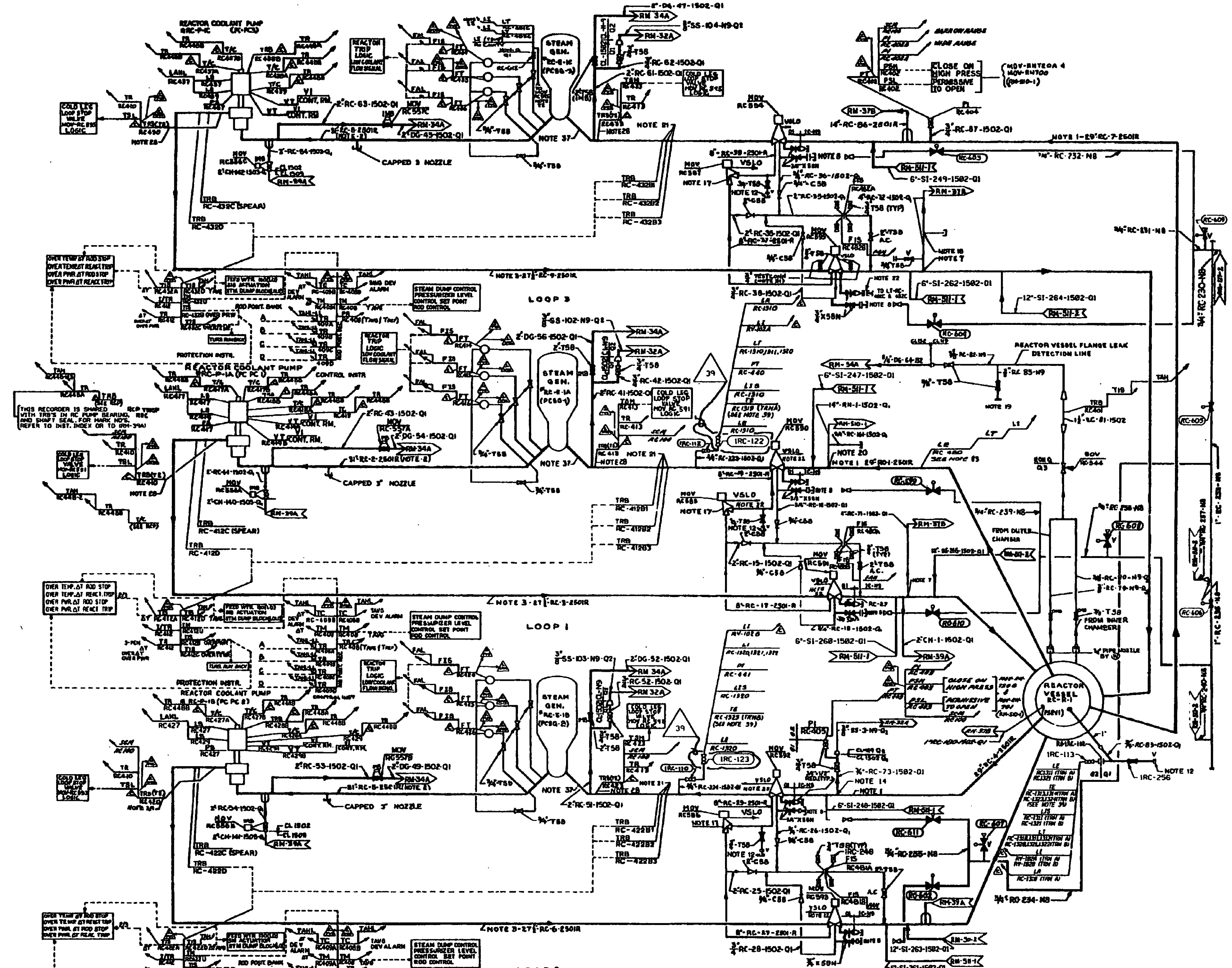
BEAVER VALLEY POWER STATION UNIT NO 1

FLOW DIAGRAM
VENT & DRAIN SYSTEMS SH 2

AA | DWG NO. 8700-RM-0034B | REV 25

PREPARED ON THE BVPS CAEDD SYSTEM

No. 8700-RM-0037A



PORTIONS OF THIS DWG ARE NUCLEAR SAFETY RELATED

NOTES:

1. 29" INSIDE DIAMETER
2. 31" INSIDE DIAMETER
3. 27.5" INSIDE DIAMETER
4. SLOPE PIPING DOWNWARD TOWARD SPRAY VALVES TO PROVIDE WATER SEAL BETWEEN PRESSURIZER AND SPRAY VALVES.
5. GLOBE VALVES NORMALLY INSTALLED WITH FLOW UNDER SEAT. EXCEPTION IS 554A, B, & C.
6. VENT HOLE PROVIDED.
7. SPRAY LINE SCOOP.
8. 1/2" SWAGELOK TEST CONNECTION FOR DISC. PRESSURIZATION SYSTEM.
9. ½" SWAGELOK TEST CONNECTION FOR TEMPORARY LEVEL INDICATION.
10. ACCESS OR REACH ROD REQUIRED FOR MANUAL ADJUSTMENT DURING NORMAL PLANT OPERATION.
11. ACCESS FOR INSPECTION OF REACTOR COOLANT PIPE TO REACTOR VESSEL NOZZLE WELDS TO BE PROVIDED.
12. ISOLATION FLANGE IS TO BE 2500#, Q1, TYPE 316. FLANGE MAY/MAY NOT HAVE TEST CONNECTIONS.
13. PLACE DETECTORS AT BOTTOM OF PIPE.
14. ¾" HALF COUPLING WITH SCOOP, UTILIZING A .234" I. D. ORIFICE.
15. SLOPE LINES TO PROVIDE WATER SEAL FOR RELIEF VALVES.
16.
17. 0.25" HOLE THROUGH VALVE PLUG.
18. LOCATE BELOW ELEVATION OF REACTOR VESSEL FLANGE.
19. LOCAL COLLECTION POINT FOR DETERMINING LEAKAGE RATE.
20. LOCATE CONNECTION 45° BELOW HORIZONTAL CENTERLINE.
21. THE BOTTOM TWO HOT LEG RTD THERMOWELLS (120° & 240°) ARE LOCATED IN THE SAME CROSSECTIONAL PLANE, THE TOP HOT LEG RTD THERMOWELL (0°) IS LOCATED 6½" TOWARD THE REACTOR FROM THAT CROSSECTIONAL PLANE.
22. LT-RC-481C & LT-RC-482C PROVIDE RCS LEVEL INDICATION DURING REFUELING AND MID-LOOP OPERATION VIA TEMPORARY CONNECTIONS TO L-FW494 & L-FW-476. A TEMPORARY SPOOL IS INSTALLED AT MOV-RC-595 TO PROVIDE AN ALTERNATE FLOW PATH TO LT-RC-481C & LT-RC-482C.
23. LE-RC-480 & LT-RC-480 ARE INSTALLED ONLY WHEN NEEDED FOR MID-LOOP LEVEL. THEY PROVIDE RCS LEVEL INDICATION VIA TEMP. CONNECTION TO L-FW-485.
24.
25.
26.
27.
28. RTD INSTALLED IN WELL.
29. PARALLEL PIPE PATHS SHOULD BE OF APPROX. EQUIVALENT LENGTHS. FLOW PATH LENGTH OF EACH 1" PIPING SECTION SHOULD BE APPROXIMATELY 6 FEET (12'-0" MAXIMUM).
30.
31. TEST CONN. FOR LOOP STOP VALVE ANALYSIS.
32. ALL INSTRUMENTS ARE SUPPLIED BY W.
33. LOCATE AS CLOSE TO CONTAINMENT AS POSSIBLE, DEAD WGT. TESTER PENETRATION DET. REFER TO SECT. 4.0 PAGE 5, I & C STDS.
34. REFER TO WESTINGHOUSE MFR. PRINTS FOR ALL INSTRUMENT CONNECTIONS..
    * ITEMS MARKED BY ASTERISK ARE SUPPLIED BY W. THE W EQUIPMENT NUMBER IS SHOWN IN PARENTHESIS.
35. TYGON HOSE TO BE CONNECTED BETWEEN THESE POINTS FOR FILLING & DRAINING OF PRESSURIZER.
36. ¾" NPT HOSE CONNECTIONS
37. 31" x 29" RED. ELL. CONNECTS HOT LEG STOP VALVE WITH STEAM GENERATOR.
38.
39. TEMPERATURE ELEMENTS LOCATED ON SENSING LINES.

FLOW DIAGRAM-SAMPLE SYSTEMS	RM-32A
FLOW DIAGRAM-COMPONENT COOLING WATER-SH.3&4	RM-15C&D
FLOW DIAGRAM-VENT & DRAIN SYSTEMS-SH-1	RM-34A
FLOW DIAGRAM-CHARGING & VOL.CONTROL SYS. SH.1	RM-39A
FLOW DIAGRAM-SAFETY INJECTION SYS. SH.1 & 2	RM-511-1 & 2
BLOCK DIAGRAM-ΔT/TAVG CONTROL	5656D70 SHEET 1
BLOCK DIAGRAM-ΔT/TAVG PROTECTION	5656D70 SHEET 3
BLOCK DIAGRAM PRESSURIZER LVL. CONT. & PROT.	5656D70 SHEET 4
BLOCK DIAGRAM PRESSURIZER PRSS. CONT. & PROT	5656D70 SHEET 5
FLOW DIAGRAM-RESIDUAL HEAT REMOVAL SYSTEM	RM-510-1

SYMBOLS:

D - BLANKED DRAIN CONNECTION
V - BLANKED VENT CONNECTION
FO - FAIL OPEN
FC - FAIL CLOSED
LMC - LEAK MONITORING CONN.
SIS - ACTUATED BY SAFETY INJECTION SIGNAL
IMB - INSIDE MISSILE BARRIER
OMB - OUTSIDE MISSILE BARRIER
VSLO - VALVE STEM LEAK OFF FOR CONTINUATION AND DETAIL SEE DRAWING RM-34A
NEEDLE VALVE
AC - ADMINISTRATIVELY CONTROLLED
CIA - ACTUATED BY CONTAINMENT ISOLATION PHASE A SIGNAL
FAI - FAIL AS IS.
HC - HOSE CONNECTION

QUALITY GROUP CLASSIFICATIONS ARE "INCORPORATED BY REFERENCE" INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.

REVISION 39
SATISFIED
ECP 09-0197
INCORPORATED
ECP 08-0004
DUN 08-0004-001-004
DUN 08-0004-002-004

FENOC FIRSTENERGY NUCLEAR OPERATING COMPANY

SCALE NONE | DATE 5/8/09 | DRAWN BY MVF | FINAL APP. FOR ISSUE 6/3/09

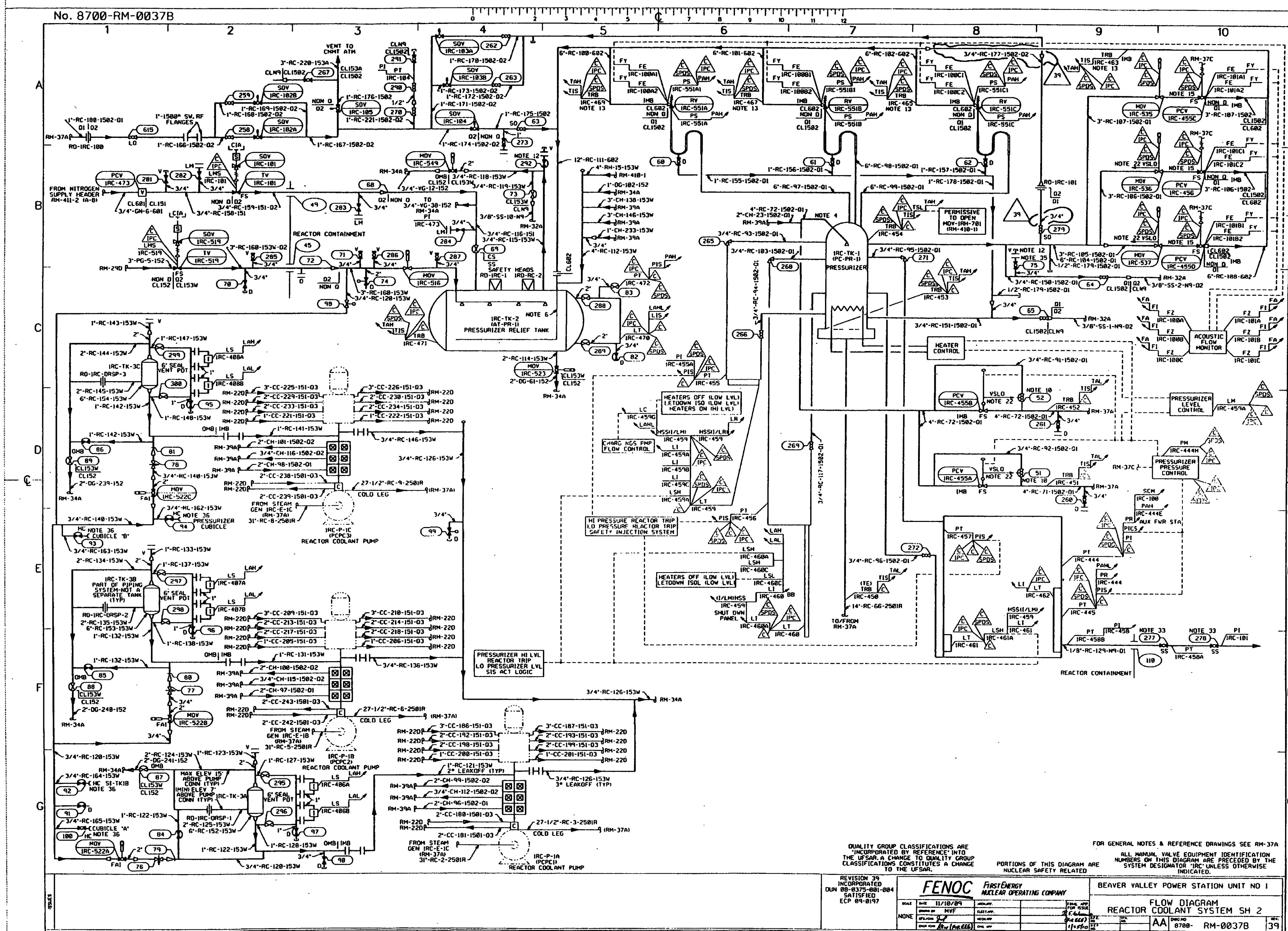
BEAVER VALLEY POWER STATION UNIT NO 1

FLOW DIAGRAM
REACTOR COOLANT SYSTEM-SH 1

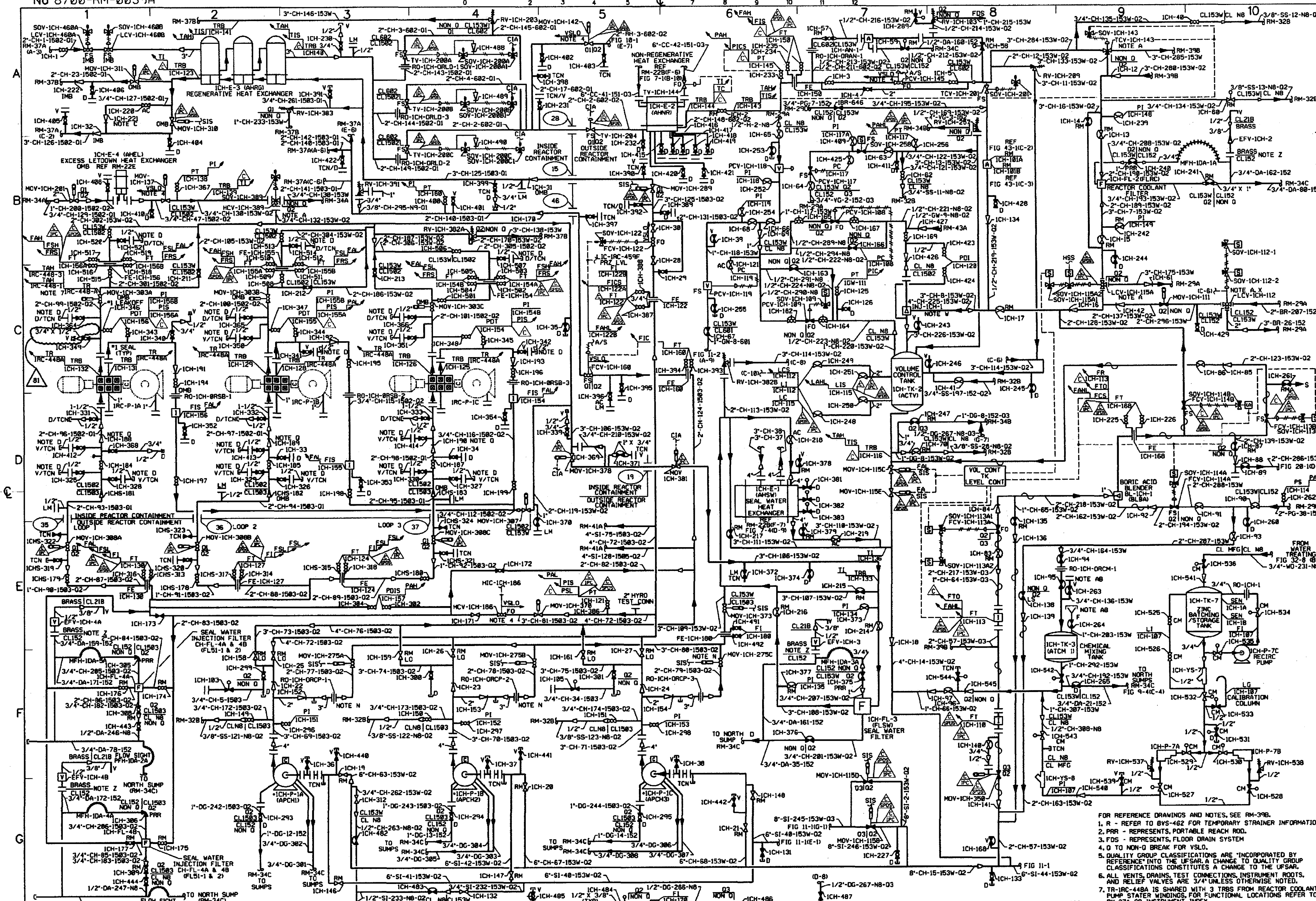
AA | DWG NO. 8700-RM-0037A | REV. 39

5 PREPARED ON THE BVPS CAEDD SYSTEM

28-MAY-2009 13:21


No. 8700-RM-0037B
REACTOR CONTAINMENT
IRC-TK-1 (PC-PR-1) PRESSURIZER
IRC-TK-2 (AT-PR-1) PRESSURIZER RELIEF TANK
IRC-P-1C (PCPC3) REACTOR COOLANT PUMP
IRC-P-1B (PCPC2) REACTOR COOLANT PUMP
IRC-P-1A (PCPC1) REACTOR COOLANT PUMP
ACOUSTIC FLOW MONITOR
PRESSURIZER LEVEL CONTROL
PRESSURIZER PRESSURE CONTROL
HEATER CONTROL
PRESSURIZER HI LVL REACTOR TRIP LO PRESSURIZER LVL SIS ACT LOGIC
HI PRESSURE REACTOR TRIP LO PRESSURE REACTOR TRIP SAFETY INJECTION SYSTEM
QUALITY GROUP CLASSIFICATIONS ARE 'INCORPORATED BY REFERENCE' INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.
PORTIONS OF THIS DIAGRAM ARE NUCLEAR SAFETY RELATED
FOR GENERAL NOTES & REFERENCE DRAWINGS SEE RM-37A
ALL MANUAL VALVE EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR 'IRC' UNLESS OTHERWISE INDICATED.
REVISION 39 INCORPORATED DUN 08-0375-001-004 SATISFIED ECP 09-0197
FENOC FIRSTENERGY NUCLEAR OPERATING COMPANY
BEAVER VALLEY POWER STATION UNIT NO 1
FLOW DIAGRAM
REACTOR COOLANT SYSTEM SH 2
SCALE NONE
DATE 11/10/09
MVF
AA
8700- RM-0037B
39

No 8700-RM-0039A



FOR REFERENCE DRAWINGS AND NOTES, SEE RM-39B.

1. R - REFER TO BVS-462 FOR TEMPORARY STRAINER INFORMATION.
2. PRR - REPRESENTS, PORTABLE REACH ROD.
3. FDS - REPRESENTS, FLOOR DRAIN SYSTEM
4. Q TO NON-Q BREAK FOR VSLO.
5. QUALITY GROUP CLASSIFICATIONS ARE "INCORPORATED BY REFERENCE" INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.
6. ALL VENTS, DRAINS, TEST CONNECTIONS, INSTRUMENT ROOTS, AND RELIEF VALVES ARE 3/4" UNLESS OTHERWISE NOTED.
7. TR-1RC-448A IS SHARED WITH 3 TRBS FROM REACTOR COOLANT PUMP STATER WINDINGS, FOR FUNCTIONAL LOCATIONS REFER TO RV-37A OR INSTRUMENT INDEX.

PORTIONS OF THIS DRAWING ARE NUCLEAR SAFETY RELATED

REVISIONS

REVISION 81 INCORPORATED
DUN 15-0426-002-004
ECP 15-0426
Ref: 15-0426-002-003 DUN

FENOC FIRST ENERGY NUCLEAR OPERATING COMPANY

SCALE NONE | DATE 10-24-16 | DRAWN BY LRH | RAH

FINAL APP FOR ISSUE 10/27/2016

BEAVER VALLEY POWER STATION UNIT NO 1

FLOW DIAGRAM
CHARGING & VOLUME CONTROL SYS SH 1

AA | DWG NO 8700-RM-0039A | REV 81

5 PREPARED ON THE BVPS CAEDD SYSTEM

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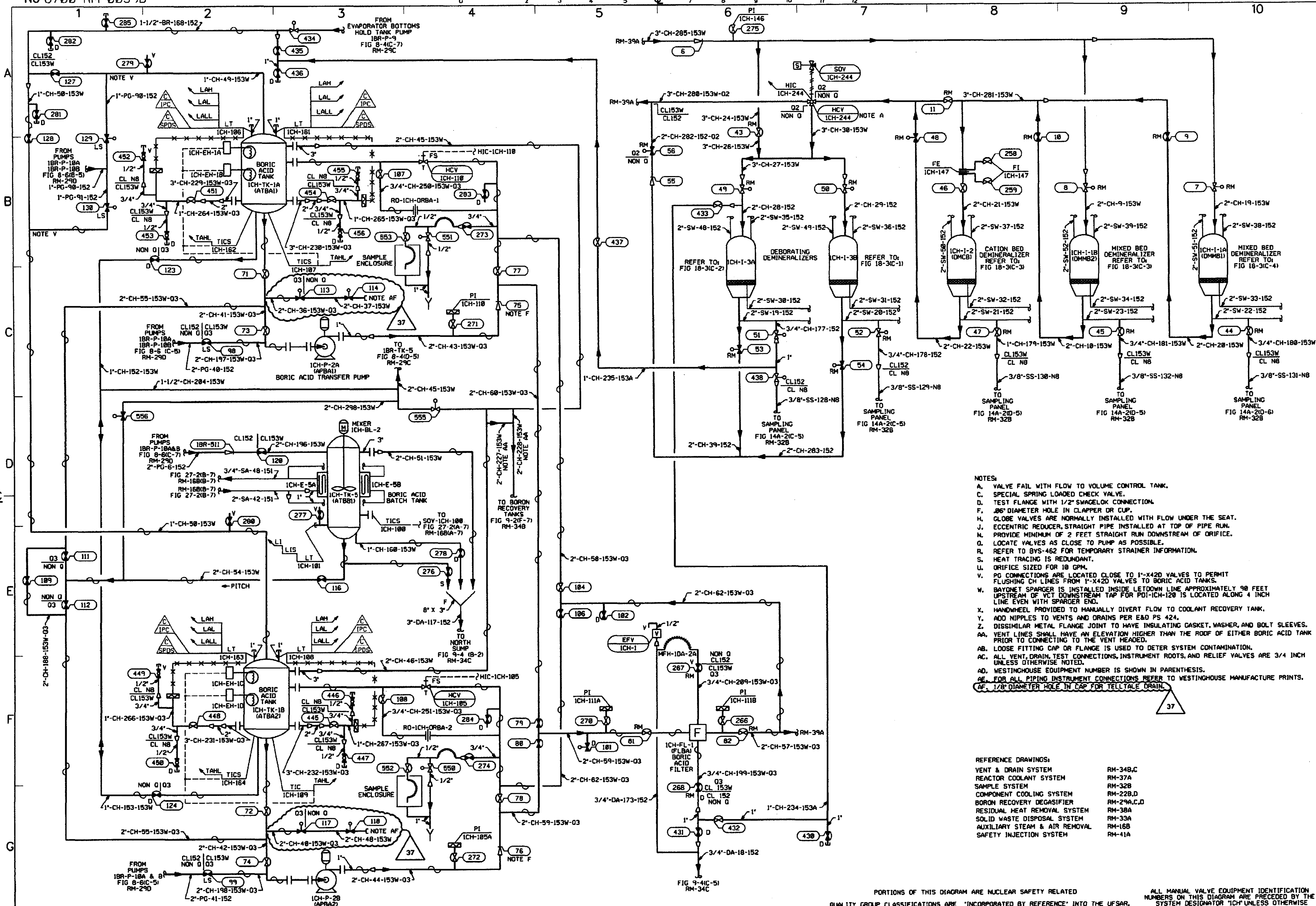
No 8700-RM-0039B



NOTES:

A. VALVE FAIL WITH FLOW TO VOLUME CONTROL TANK.
C. SPECIAL SPRING LOADED CHECK VALVE.
D. TEST FLANGE WITH 1/2" SWAGELOK CONNECTION.
F. .06" DIAMETER HOLE IN CLAPPER OR CUP.
H. GLOBE VALVES ARE NORMALLY INSTALLED WITH FLOW UNDER THE SEAT.
J. ECCENTRIC REDUCER. STRAIGHT PIPE INSTALLED AT TOP OF PIPE RUN.
N. PROVIDE MINIMUM OF 2 FEET STRAIGHT RUN DOWNSTREAM OF ORIFICE.
Q. LOCATE VALVES AS CLOSE TO PUMP AS POSSIBLE.
R. REFER TO BVS-462 FOR TEMPORARY STRAINER INFORMATION.
S. HEAT TRACING IS REDUNDANT.
U. ORIFICE SIZED FOR 10 GPM.
V. PG CONNECTIONS ARE LOCATED CLOSE TO 1"-X42D VALVES TO PERMIT FLUSHING CH LINES FROM 1"-X42D VALVES TO BORIC ACID TANKS.
W. BAYONET SPARGER IS INSTALLED INSIDE LETDOWN LINE APPROXIMATELY 90 FEET UPSTREAM OF VCT DOWNSTREAM TAP FOR PDI-1CH-120 IS LOCATED ALONG 4 INCH LINE EVEN WITH SPARGER END.
X. HANDWHEEL PROVIDED TO MANUALLY DIVERT FLOW TO COOLANT RECOVERY TANK.
Y. ADD NIPPLES TO VENTS AND DRAINS PER E&D PS 424.
Z. DISSIMILAR METAL FLANGE JOINT TO HAVE INSULATING GASKET, WASHER, AND BOLT SLEEVES.
AA. VENT LINES SHALL HAVE AN ELEVATION HIGHER THAN THE ROOF OF EITHER BORIC ACID TANK PRIOR TO CONNECTING TO THE VENT HEADED.
AB. LOOSE FITTING CAP OR FLANGE IS USED TO DETER SYSTEM CONTAMINATION.
AC. ALL VENT, DRAIN, TEST CONNECTIONS, INSTRUMENT ROOTS, AND RELIEF VALVES ARE 3/4 INCH UNLESS OTHERWISE NOTED.
AD. WESTINGHOUSE EQUIPMENT NUMBER IS SHOWN IN PARENTHESIS.
AE. FOR ALL PIPING INSTRUMENT CONNECTIONS REFER TO WESTINGHOUSE MANUFACTURE PRINTS.
AF. 1/8" DIAMETER HOLE IN CAP FOR TELLTALE DRAIN.

37

REFERENCE DRAWINGS:

VENT & DRAIN SYSTEM	RM-34B,C
REACTOR COOLANT SYSTEM	RM-37A
SAMPLE SYSTEM	RM-32B
COMPONENT COOLING SYSTEM	RM-22B,D
BORON RECOVERY DEGASIFIER	RM-29A,C,D
RESIDUAL HEAT REMOVAL SYSTEM	RM-38A
SOLID WASTE DISPOSAL SYSTEM	RM-33A
AUXILIARY STEAM & AIR REMOVAL	RM-16B
SAFETY INJECTION SYSTEM	RM-41A

PORTIONS OF THIS DIAGRAM ARE NUCLEAR SAFETY RELATED

QUALITY GROUP CLASSIFICATIONS ARE "INCORPORATED BY REFERENCE" INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.

ALL MANUAL VALVE EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR "1CH" UNLESS OTHERWISE INDICATED.

REVISIONS

REVISION 37 INCORPORATED
DUN 13-0634-003-001
DUN 13-0634-004-001

FENOC FIRST ENERGY NUCLEAR OPERATING COMPANY

SCALE: NONE | DATE: 7-8-16 | DRAWN BY: LRH | FINAL APP FOR ISSUE: 7/12/2016

BEAVER VALLEY POWER STATION UNIT NO 1

FLOW DIAGRAM - CHARGING & VOLUME CONTROL SYSTEM SH 2

AA | DWG NO 8700- RM-0039B | REV 37

PREPARED ON THE BVPS CAEDD SYSTEM

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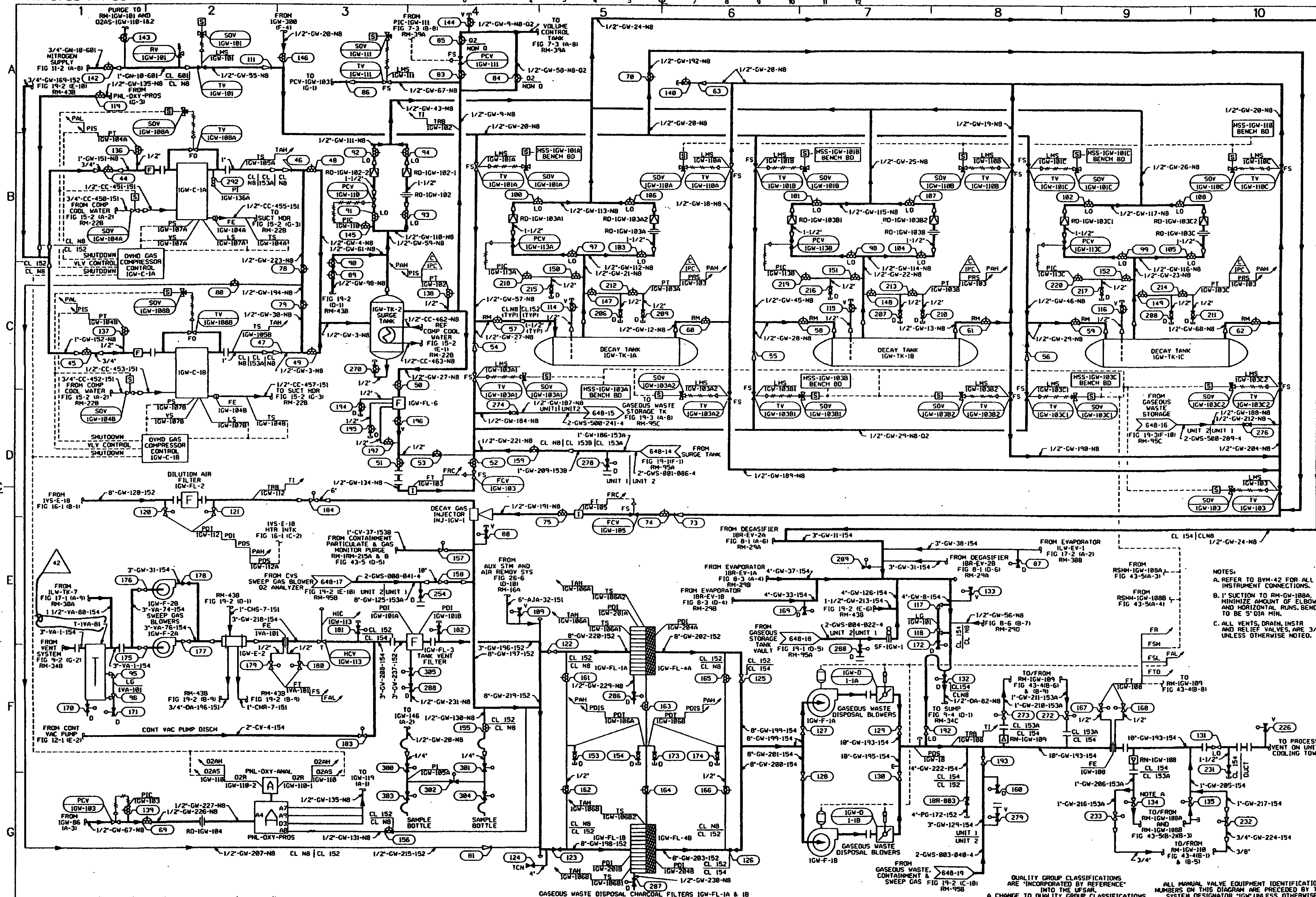
No. 8700-RM-0043A



REVISION 42 INCORPORATED DUN 08-0612-001-002

FENOC FirstEnergy Nuclear Operating Company

BEAVER VALLEY POWER STATION UNIT NO 1

FLOW DIAGRAM
GASEOUS WASTE DISPOSAL SYSTEM

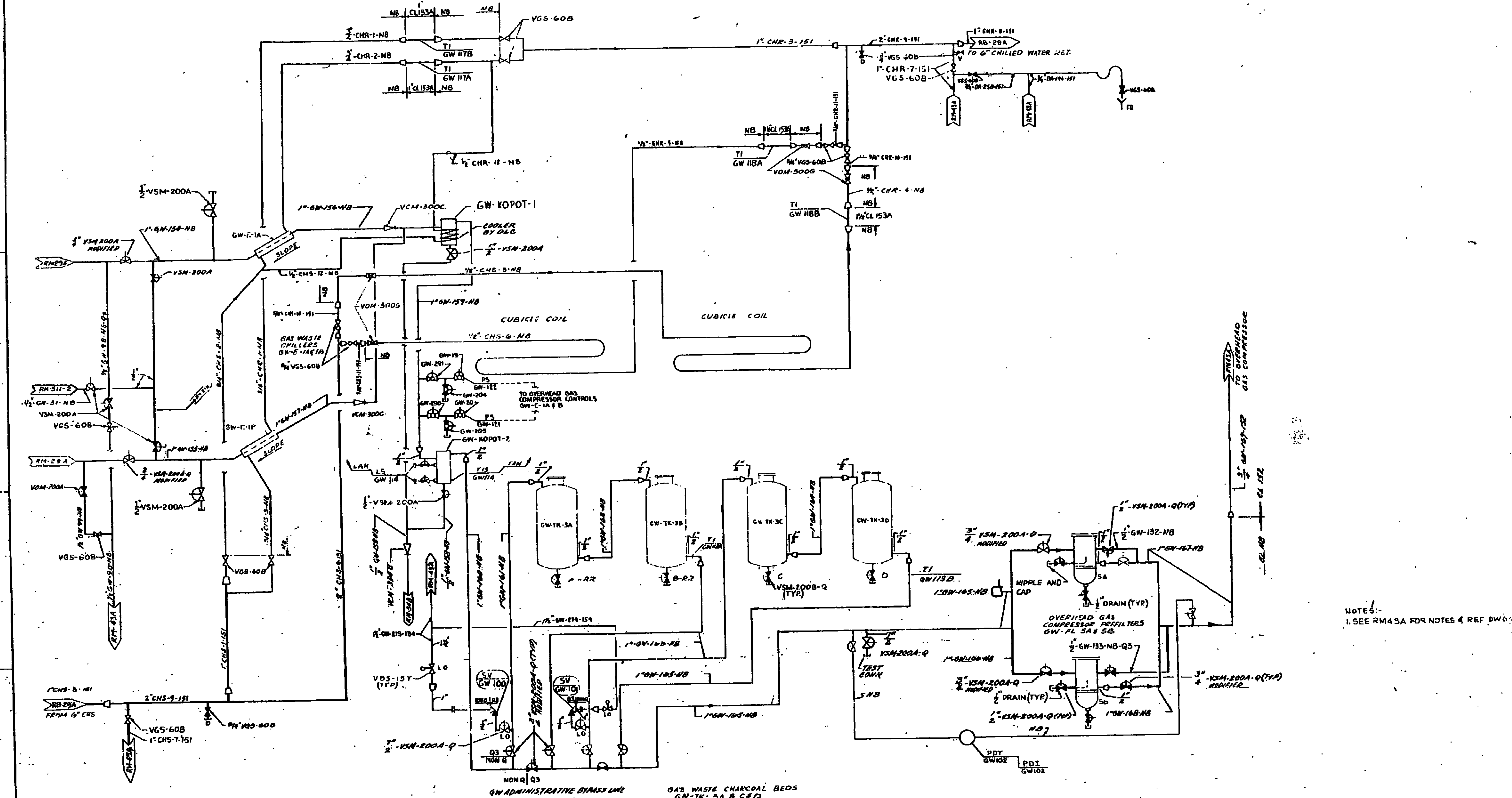
SCALE NONE — DATE 11/18/08

AA — DWG. NO. 8700-RM-0043A — REV. 42

PREPARED ON THE BVPS CAEDDS SYSTEM

21-NOV-2008 08:07 Ks:\ul\RM-0043A.DGN

No. 8700-RM-43B



NOTES:-
1. SEE RM43A FOR NOTES & REF DWG'S.

PORTIONS OF THIS DWG ARE NUCLEAR SAFETY RELATED

THE INFORMATION ON THIS DRAWING MAY NOT BE COPIED OR USED FOR OTHER THAN THE CONSTRUCTION, MAINTENANCE OR REPAIR OF THE PLANT FACILITY DESCRIBED IN THE TITLE BLOCK.

(18) QUALITY GROUP CLASSIFICATIONS ARE INCORPORATED BY REFERENCE INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.

FENOC FIRSTENERGY NUCLEAR OPERATING COMPANY

STONE & WEBSTER ENGINEERING CORP. BOSTON, MASS.

SCALE	
DATE 4-3-72	ARCH. APP.
DRAWN E. GITNIK	ELECT. APP.
CHECKED	MECH. APP.
INSPECTED	STRUCT. APP.

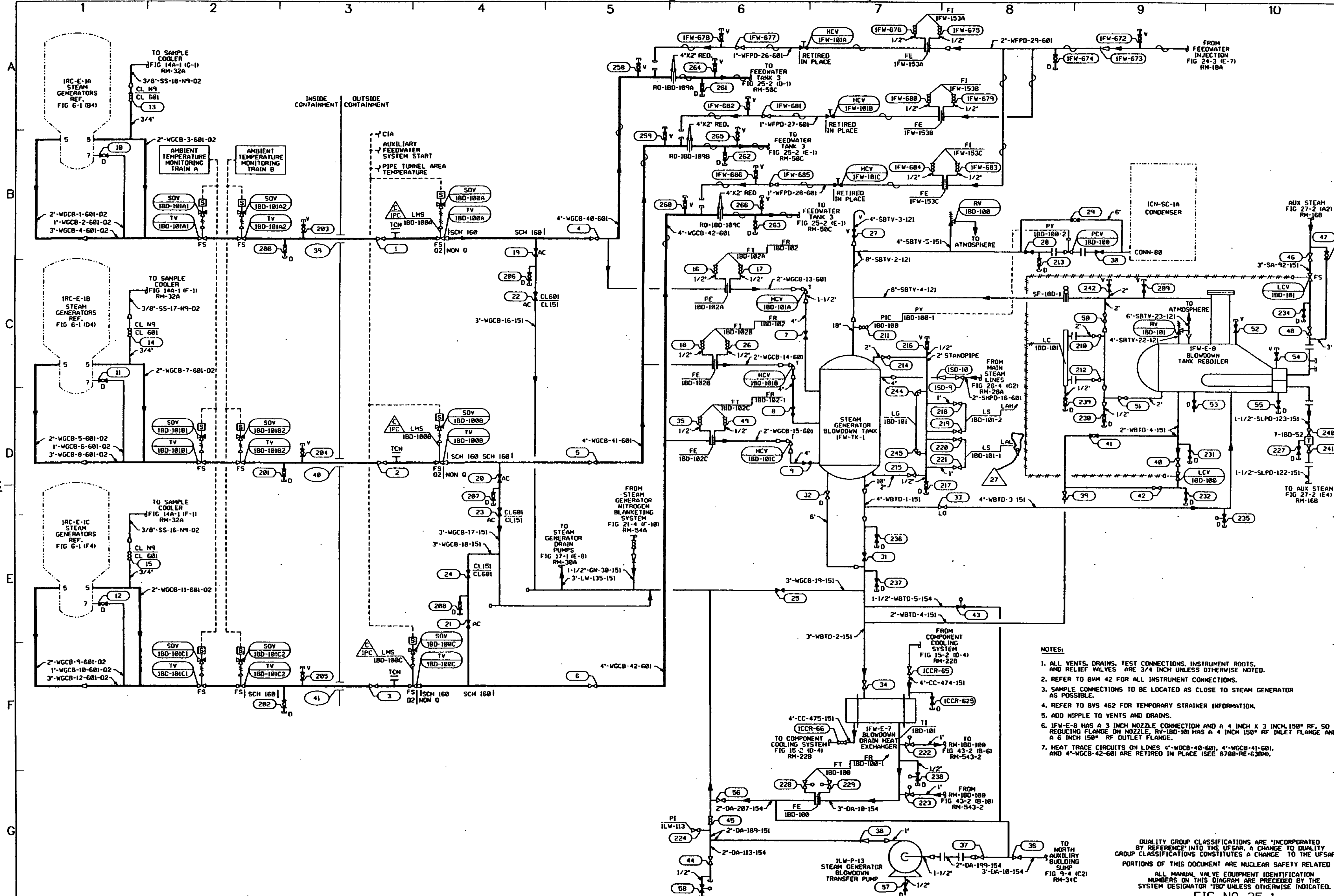
FLOW DIAGRAM
GASEOUS WASTE DISPOSAL SYSTEM-SH. 2
BEAVER VALLEY POWER STATION UNIT NO. 1

O.F.E. No. 8700
C.O. No. 5465

AA No. 8700-RM-43B

S&W DWG. NO. 11700-RM-43B-18

No. 8700-RM-50A



NOTES:

1. ALL VENTS, DRAINS, TEST CONNECTIONS, INSTRUMENT ROOTS, AND RELIEF VALVES ARE 3/4 INCH UNLESS OTHERWISE NOTED.
2. REFER TO BVM 42 FOR ALL INSTRUMENT CONNECTIONS.
3. SAMPLE CONNECTIONS TO BE LOCATED AS CLOSE TO STEAM GENERATOR AS POSSIBLE.
4. REFER TO BVS 462 FOR TEMPORARY STRAINER INFORMATION.
5. ADD NIPPLE TO VENTS AND DRAINS.
6. 1FW-E-8 HAS A 3 INCH NOZZLE CONNECTION AND A 4 INCH X 3 INCH, 150# RF, SO REDUCING FLANGE ON NOZZLE, RV-1BD-101 HAS A 4 INCH 150# RF INLET FLANGE AND A 6 INCH 150# RF OUTLET FLANGE.
7. HEAT TRACE CIRCUITS ON LINES 4"-WGCB-40-601, 4"-WGCB-41-601, AND 4"-WGCB-42-601 ARE RETIRED IN PLACE (SEE 8700-RE-638M).

QUALITY GROUP CLASSIFICATIONS ARE "INCORPORATED BY REFERENCE" INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.

PORTIONS OF THIS DOCUMENT ARE NUCLEAR SAFETY RELATED

ALL MANUAL VALVE EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR "1BD" UNLESS OTHERWISE INDICATED.

FIG NO 25-1

REVISION 27 INCORPORATED ECP-05-0028 IDCN-1-RM-0050A-E05-0028-01

FENOC FirstEnergy Nuclear Operating Company

SCALE NONE | DATE 1-11-06 | DRAWN BY MVF

BEAVER VALLEY POWER STATION UNIT NO 1

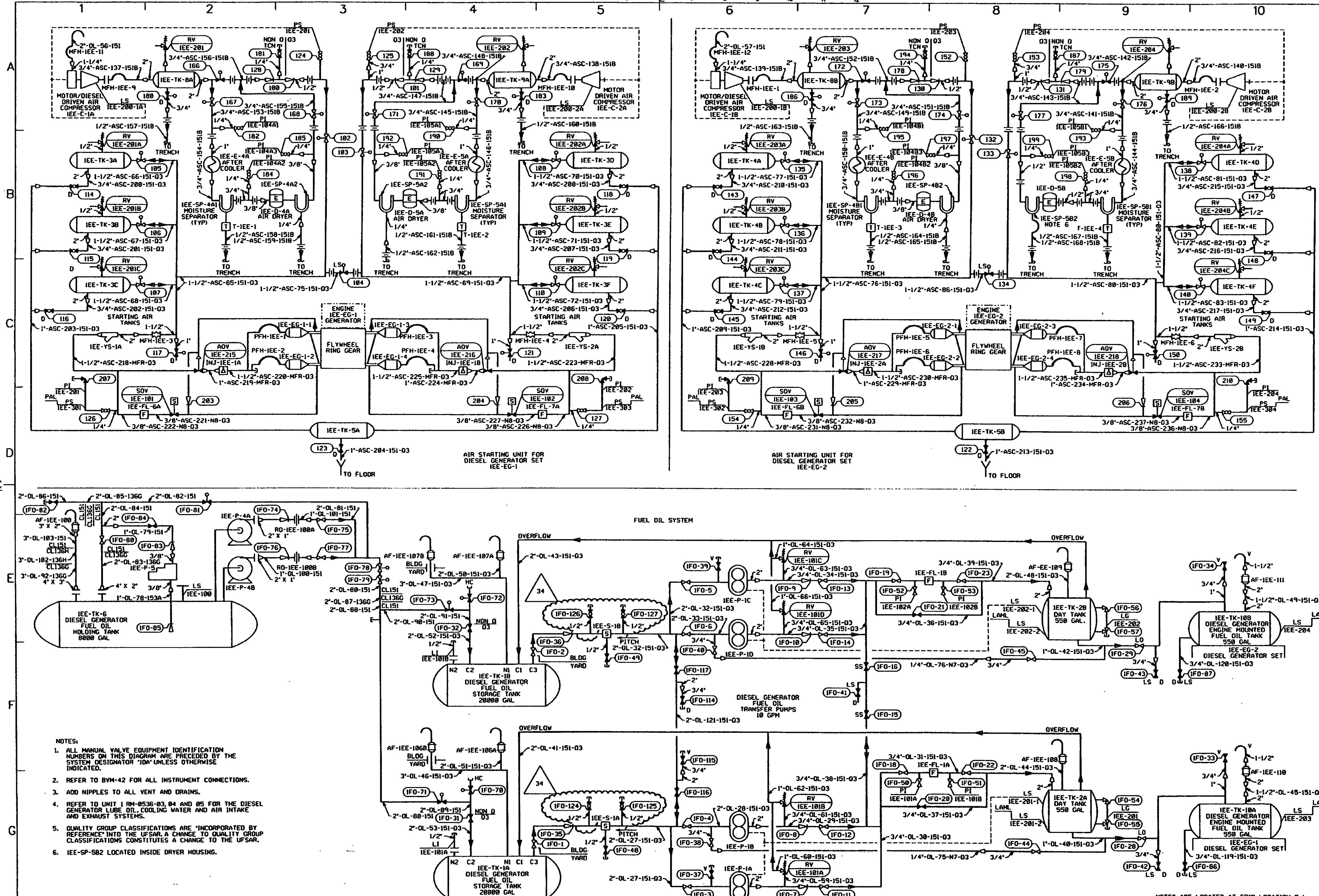
FLOW DIAGRAM STEAM GENERATOR BLOWDOWN

AA | DWG NO 8700-RM-50A | REV 27

PREPARED ON THE BVPS CAEDDI SYSTEM

17-JAN-2006 14:34 K:\u1\rm0050a.dgn

No. 8700-RM-0053A



NOTES:

1. ALL MANUAL VALVE EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR '1DA' UNLESS OTHERWISE INDICATED.
2. REFER TO BVM-42 FOR ALL INSTRUMENT CONNECTIONS.
3. ADD NIPPLES TO ALL VENT AND DRAINS.
4. REFER TO UNIT 1 RM-0536-03, 04 AND 05 FOR THE DIESEL GENERATOR LUBE OIL, COOLING WATER AND AIR INTAKE AND EXHAUST SYSTEMS.
5. QUALITY GROUP CLASSIFICATIONS ARE "INCORPORATED BY REFERENCE" INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.
6. 1EE-SP-5B2 LOCATED INSIDE DRYER HOUSING.

PORTIONS OF THIS DIAGRAM ARE NUCLEAR SAFETY RELATED

NOTES ARE LOCATED AT GRID LOCATION F-1

REVISION 34 INCORPORATED
DUN 11-0691-001-002
DUN 11-0691-002-002
ECP 11-0691

FENOC FirstEnergy Nuclear Operating Company

SCALE: NONE | DATE: 5/4/12 | DRAWN BY: MVF | FINAL APP. FOR ISSUE: 8/6/12

BEAVER VALLEY POWER STATION UNIT NO 1

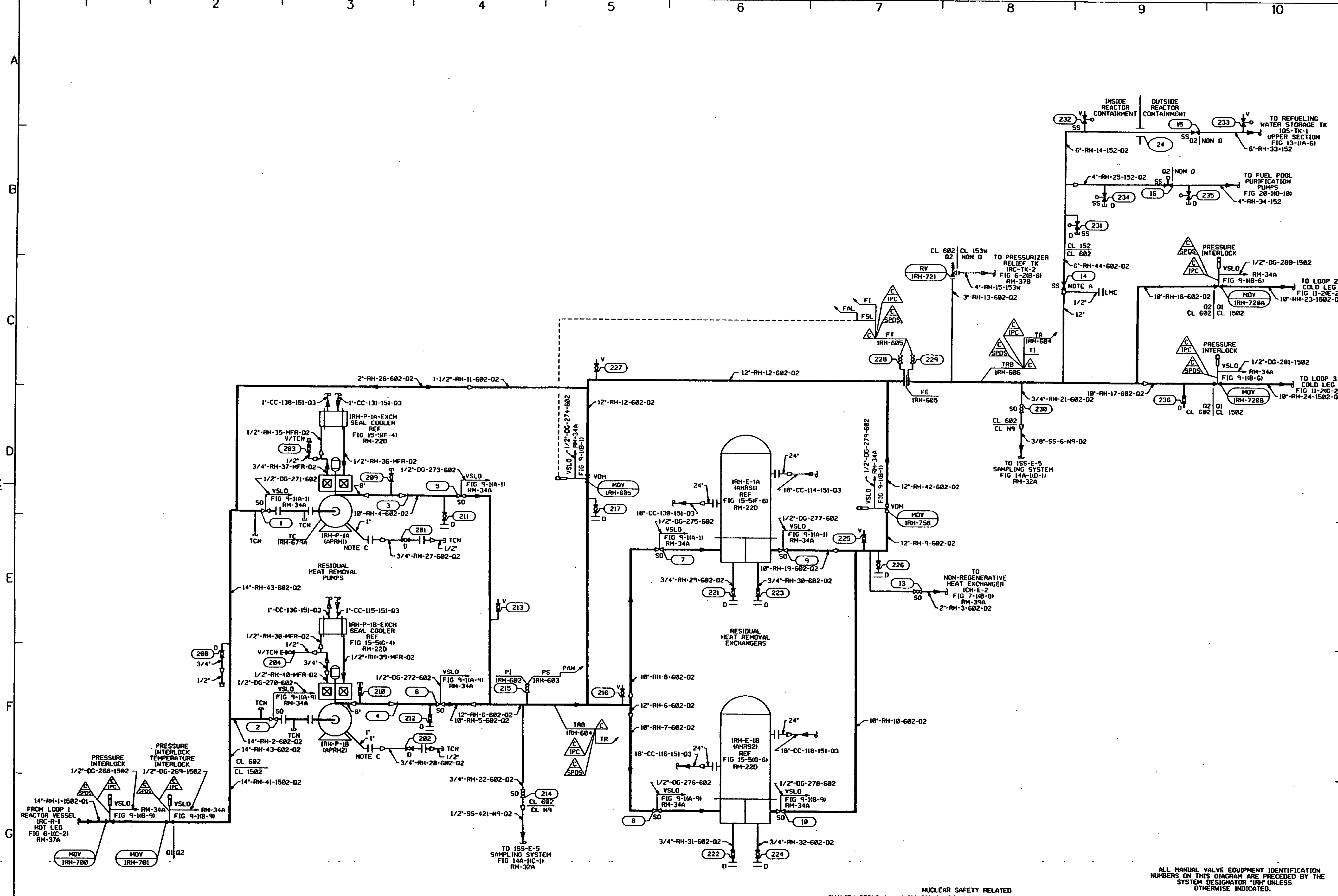
FLOW DIAGRAM
EMERGENCY DIESEL GENERATOR
FUEL AND AIR SYSTEMS

AA | DWG. NO. 8700-RM-0053A | REV. 34

PREPARED ON THE BVPS CAEDD SYSTEM

17-MAY-2012 10:05 K:\u1\RM-0053A.DGN

No. 8700-RM-0410-001



ALL MANUAL VALVE EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR "1RH" UNLESS OTHERWISE INDICATED.

NUCLEAR SAFETY RELATED
QUALITY GROUP CLASSIFICATIONS ARE "INCORPORATED BY REFERENCE" INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.

OP MANUAL FIG NO 10-1

REVISION 14 INCORPORATED
IDCN 1-RM-0410-001-E02-0258-01

FENOC FirstEnergy Nuclear Operating Company

DATE 5/7/12 DRAWN BY MVF
SCALE NONE

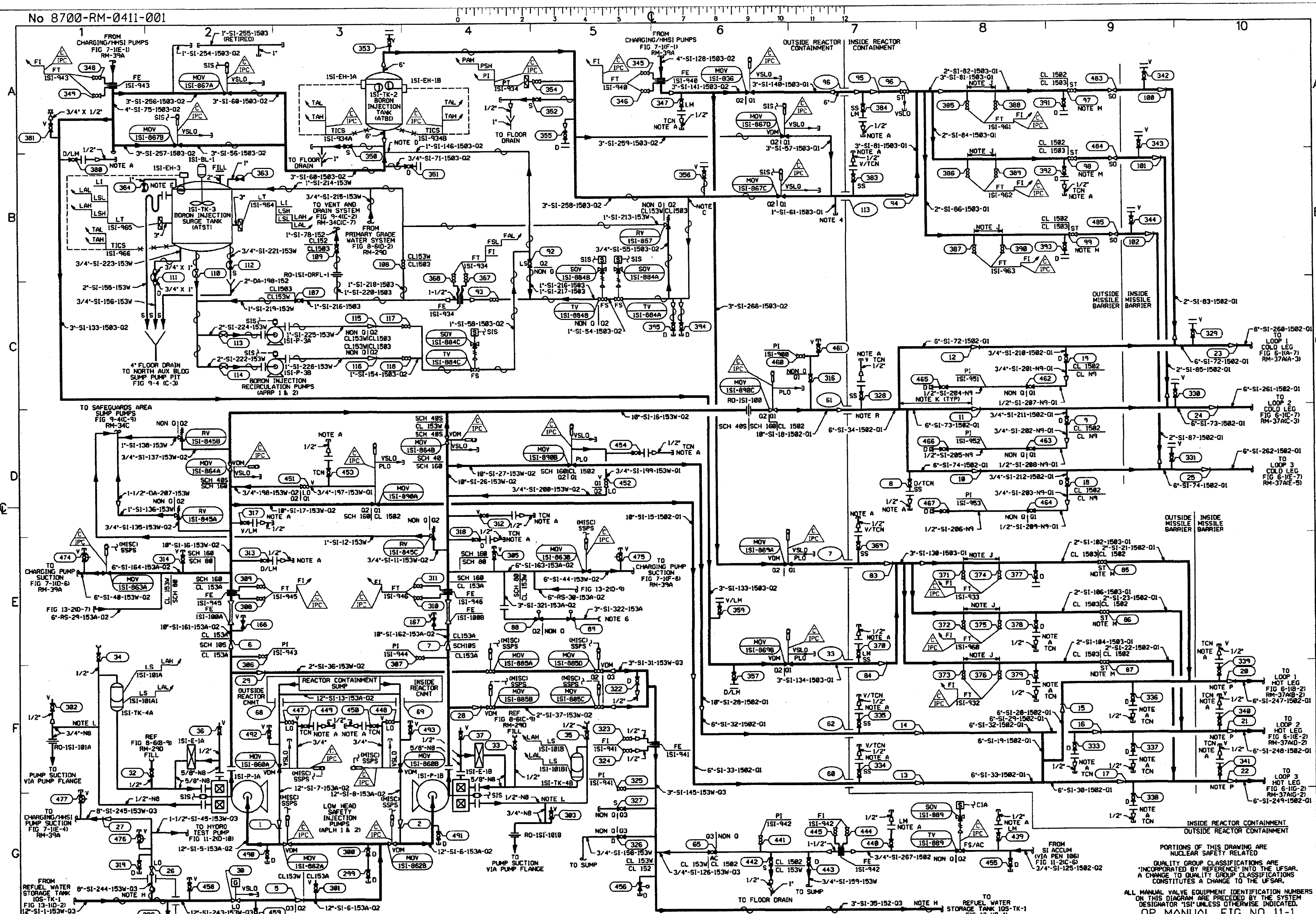
PIPING & INSTRUMENTATION DIAGRAM
RESIDUAL HEAT REMOVAL SYSTEM

AA 8700-RM-0410-001 REV. 14

17-MAY-2012 10:10 K:\u1\RM-0410-001.DGN

PREPARED ON THE BVPS CAEDDI SYSTEM

No 8700-RM-0411-001



PORTIONS OF THIS DRAWING ARE NUCLEAR SAFETY RELATED

QUALITY GROUP CLASSIFICATIONS ARE "INCORPORATED BY REFERENCE" INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.

ALL MANUAL VALVE EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR "1SI" UNLESS OTHERWISE INDICATED.

OP MANUAL FIG NO 11-1

REVISION 28 INCORPORATED DUN 13-0418-006-001 DUN 15-0426-004-001 DUN 16-0500-001-001

FENOC — FirstEnergy Nuclear Operating Company

SCALE NONE — DATE 10-31-16 — DRAWN BY LRH — FINAL APP FOR ISSUE 11/1/2016

BEAVER VALLEY POWER STATION UNIT NO 1

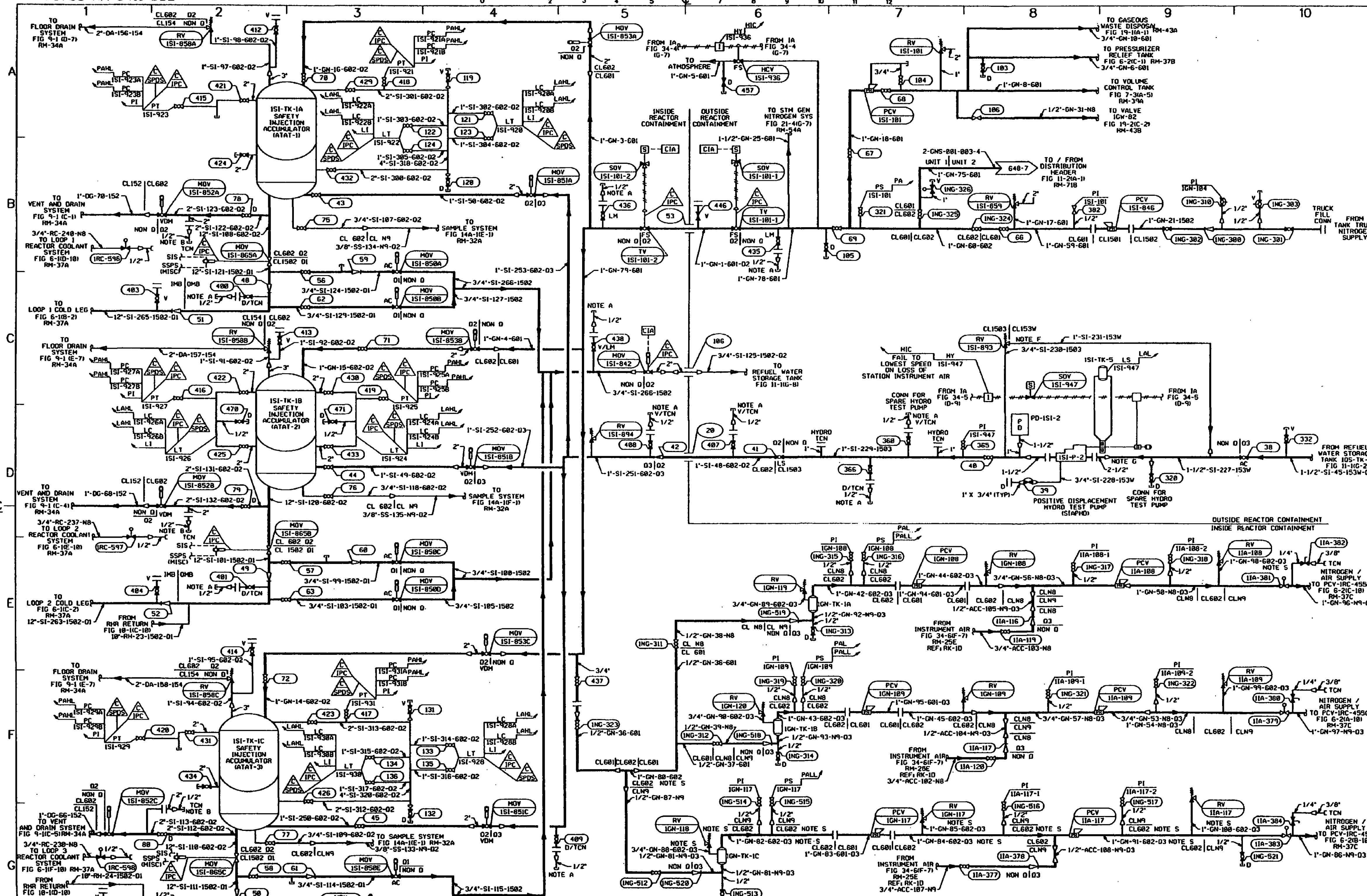
PIPING & INSTRUMENTATION DIAGRAM SAFETY INJECTION SYSTEM

AA 8700-RM-0411-001 REV 28

PREPARED ON THE BVPS CAEDD SYSTEM

K:\L1\RM-0411-001.DGN

No. 8700-RM-0411-002



OP FIG NO 11-2

PORTIONS OF THIS DRAWING ARE NUCLEAR SAFETY RELATED

QUALITY GROUP CLASSIFICATIONS ARE "INCORPORATED BY REFERENCE" INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.

ALL MANUAL VALVE EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR "1SI" UNLESS OTHERWISE INDICATED.

REVISION 16 INCORPORATED DUN 11-0488-001-001 ECP 09-0197

FENOC FirstEnergy Nuclear Operating Company

BEAVER VALLEY POWER STATION UNIT NO 1

PIPING & INSTRUMENTATION DIAGRAM SAFETY INJECTION SYSTEM

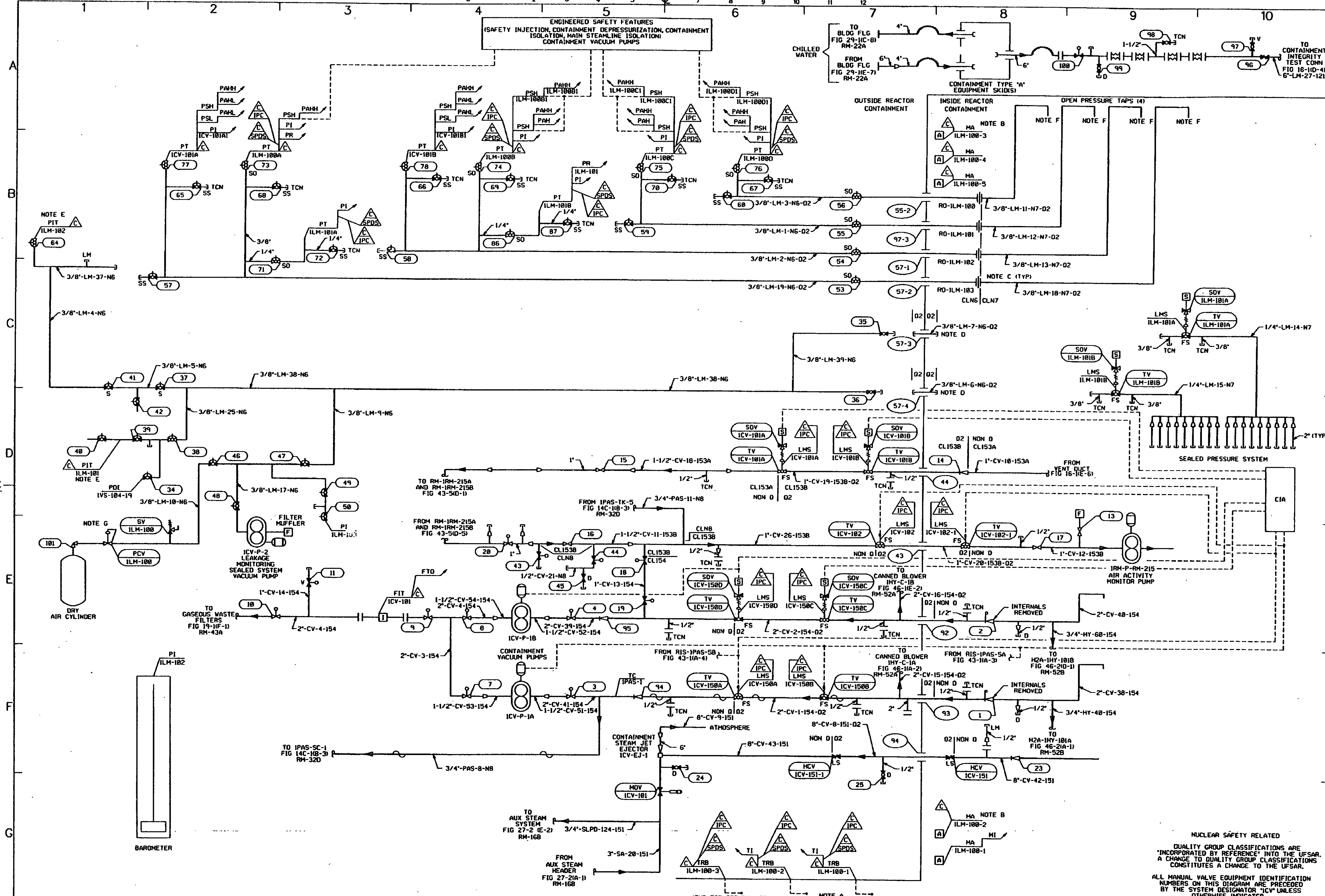
SCALE NONE | DATE 11/23/11 | DRAWN BY MVF

AA | 8700-RM-0411-002 | 16

PREPARED ON THE BVPS CAEDDI SYSTEM

23-NOV-2011 09:36 K:\U1\RM-0411-002.DGN

No. 8700-RM-0412-001



NUCLEAR SAFETY RELATED

QUALITY GROUP CLASSIFICATIONS ARE "INCORPORATED BY REFERENCE" INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.

ALL MANUAL VALVE EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR "ICV" UNLESS OTHERWISE INDICATED.

OP MANUAL FIG NO 12-1

REVISION 11
DUN 11-0628-001-001
SATISFIED
ECP 09-0197

FENOC FirstEnergy Nuclear Operating Company

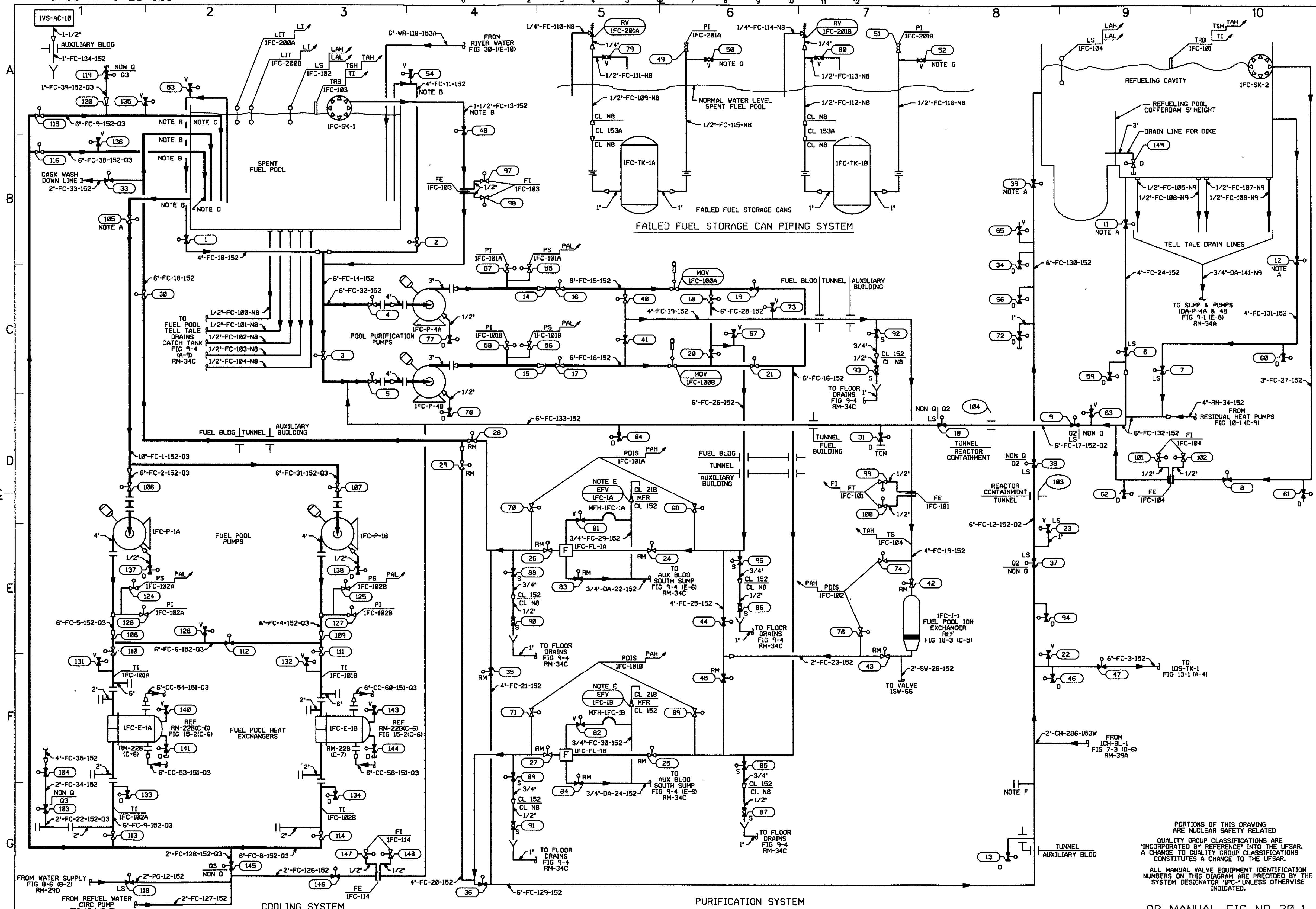
BEAVER VALLEY POWER STATION UNIT NO 1

PIPING & INSTRUMENTATION DIAGRAM
CONT VACUUM & LEAKAGE MONITORING

SCALE NONE | DATE 11/17/11 | DRAWN BY MVF

AA | DWG NO. 8700-RM-0412-001 | REV. 11

No. 8700-RM-0420-001



PORTIONS OF THIS DRAWING ARE NUCLEAR SAFETY RELATED

QUALITY GROUP CLASSIFICATIONS ARE "INCORPORATED BY REFERENCE" INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.

ALL MANUAL VALVE EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR "1FC-" UNLESS OTHERWISE INDICATED.

OP MANUAL FIG NO 20-1

REVISIONS

REVISION 12 INCORPORATED DUN 13-0561-003-012 DUN 13-0561-006-008

FENOC — FirstEnergy Nuclear Operating Company

SCALE NONE — DATE 6/9/15 — DRAWN BY MVF

BEAVER VALLEY POWER STATION UNIT NO 1

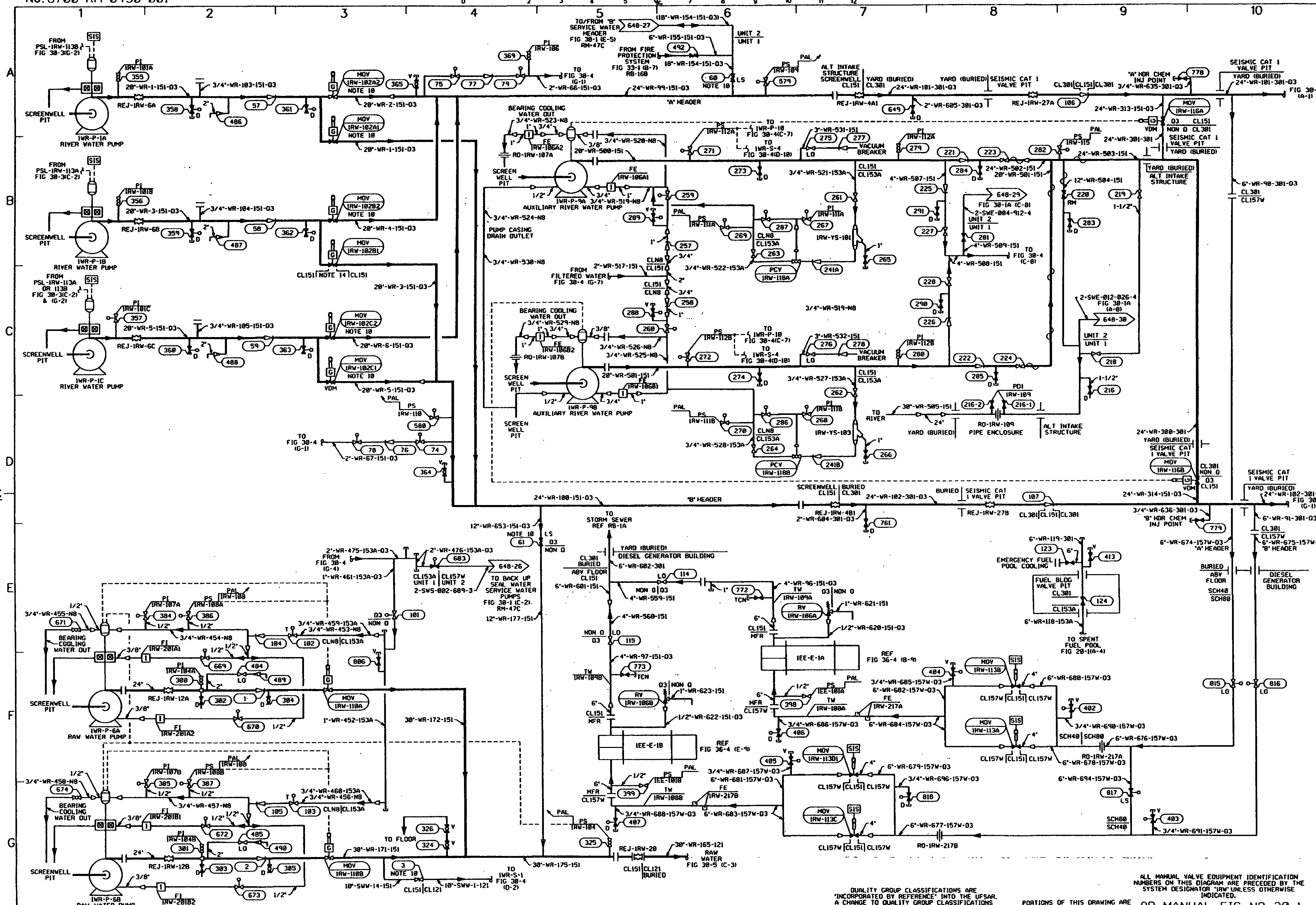
PIPING & INSTRUMENTATION DIAGRAM
FUEL POOL COOLING & PURIFICATION SYS

AA — DWG NO 8700-RM-0420-001 — REV 12

PREPARED ON THE BVPS CAEDD SYSTEM

k:\ui\RM-0420-001.DGN

No. 8700-RM-0430-001



QUALITY GROUP CLASSIFICATIONS ARE "INCORPORATED BY REFERENCE" INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.

ALL MANUAL VALVE EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR "1RW" UNLESS OTHERWISE INDICATED.

PORTIONS OF THIS DRAWING ARE NUCLEAR SAFETY RELATED

OP MANUAL FIG NO 30-1

REVISION 32 INCORPORATED DUN 11-0296-002-004

FENOC FirstEnergy Nuclear Operating Company

BEAVER VALLEY POWER STATION UNIT NO 1

PIPING & INSTRUMENTATION DIAGRAM RIVER WATER SYSTEM

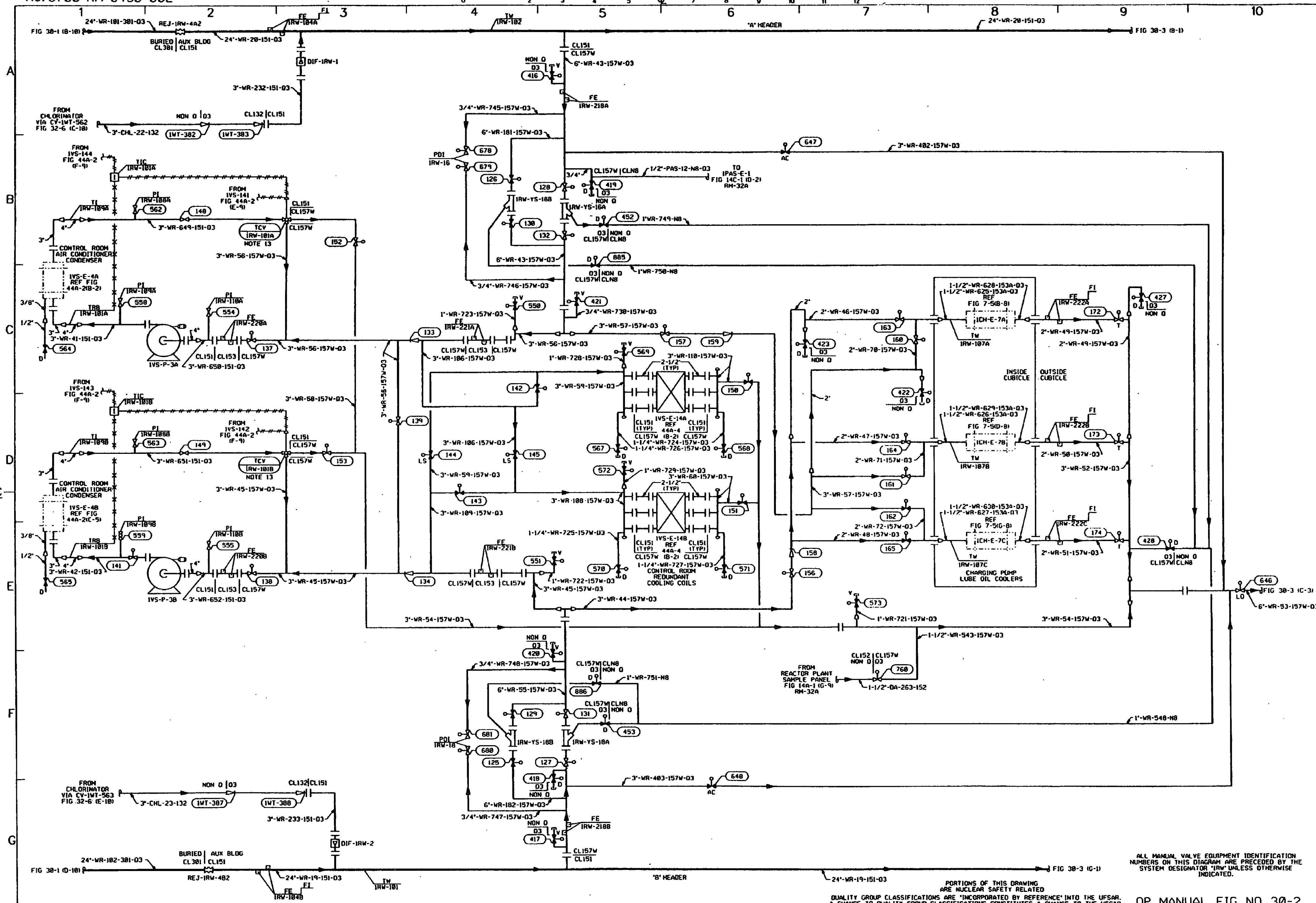
DATE 4/25/12 | SCALE NONE | DRAWN BY HVF

AA | DWG.NO. 8700-RM-0430-001 | REV. 32

PREPARED ON THE BVPS CAEDDI SYSTEM

21-AUG-2012 08:40 K:\u1\RM-0430-001.DGN

No. 8700-RM-0430-002



ALL MANUAL VALVE EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR 'IRW' UNLESS OTHERWISE INDICATED.

PORTIONS OF THIS DRAWING ARE NUCLEAR SAFETY RELATED

QUALITY GROUP CLASSIFICATIONS ARE 'INCORPORATED BY REFERENCE' INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.

OP MANUAL FIG NO 30-2

REVISION 20 INCORPORATED DCN 1-RM-0430-002-E03-0250-01

FENOC FirstEnergy Nuclear Operating Company

SCALE NONE | DATE 11/04/10 | DRAWN BY MVF

BEAVER VALLEY POWER STATION UNIT NO 1

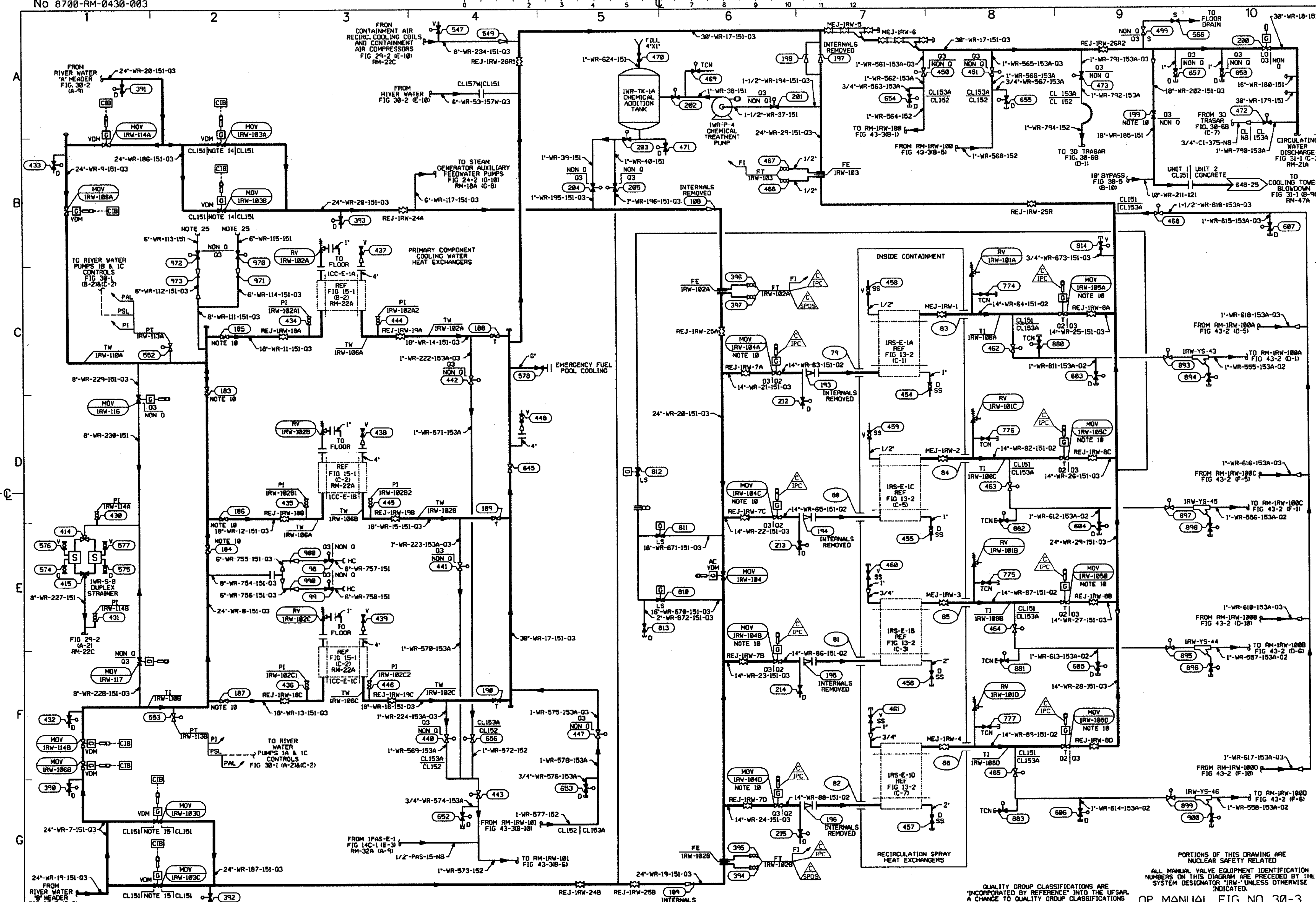
PIPING & INSTRUMENTATION DIAGRAM
RIVER WATER SYSTEM

AA | DWG NO. 8700-RM-0430-002 | REV 20

PREPARED ON THE BVPS CAEDD SYSTEM

04-NOV-2010 08:08

No 8700-RM-0430-003



OP MANUAL FIG NO 30-3

REVISIONS

REVISION 31 INCORPORATED ECP 13-0633 & DUN 13-0633-001-001

FENOC FIRST ENERGY NUCLEAR OPERATING COMPANY

SCALE NONE	DATE 10-4-16	FINAL APP FOR ISSUE	
	DRAWN BY LRH	10/27/2016	
	SWG		

BEAVER VALLEY POWER STATION UNIT NO 1

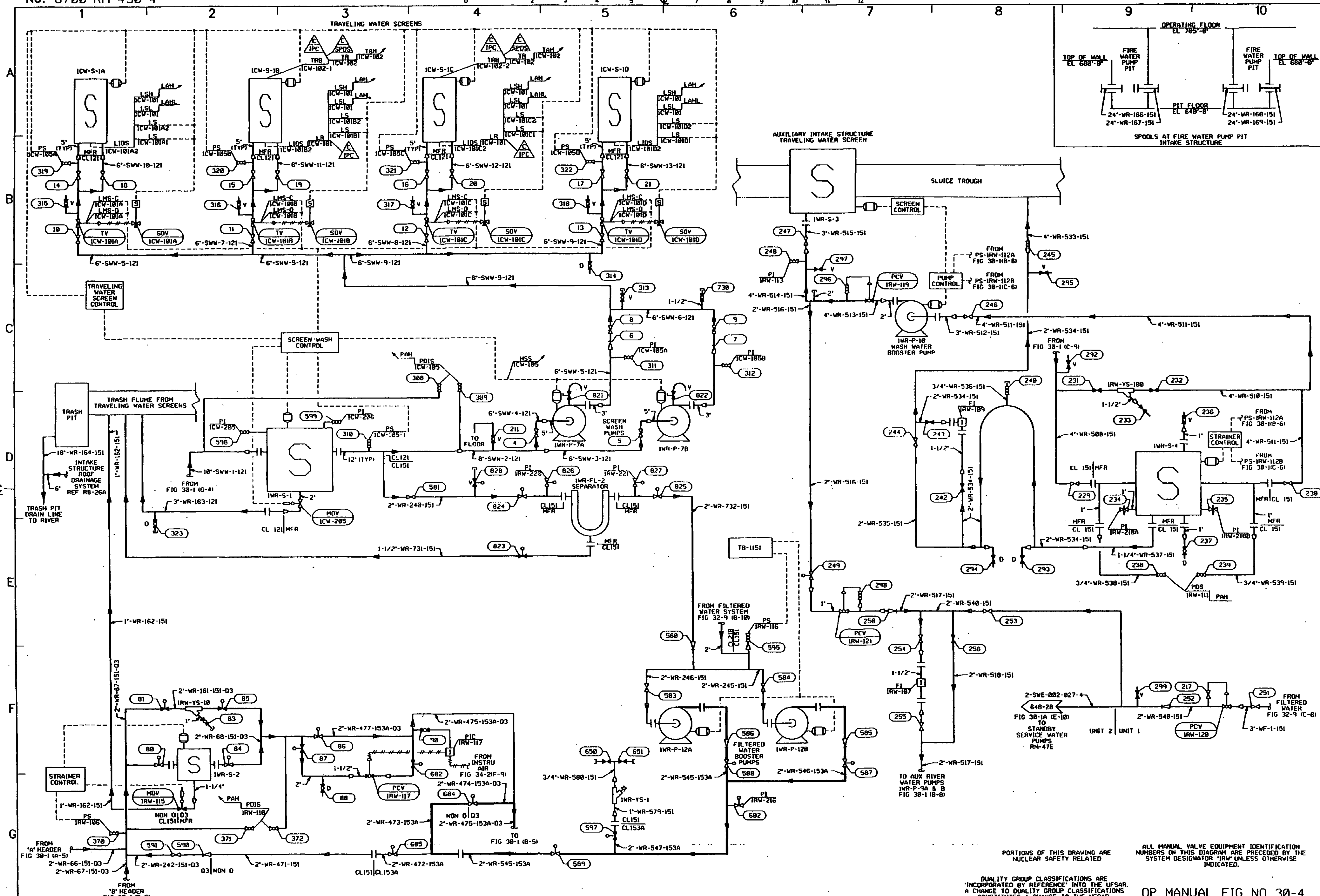
PIPING & INSTRUMENTATION DIAGRAM RIVER WATER SYSTEM

AA | DWG NO 8700-RM-0430-003 | REV 31

PREPARED ON THE BVPS CADD SYSTEM

K:\U1\RM-0430-003.DGN

No. 8700-RM-430-4



OP MANUAL FIG NO 30-4

REVISION 17
INCORPORATED
ECR-05-0367
IDCN-1-RM-0430-004-E05-0367-01

FENOC FirstEnergy Nuclear Operating Company

11-15-05

BEAVER VALLEY POWER STATION UNIT NO 1

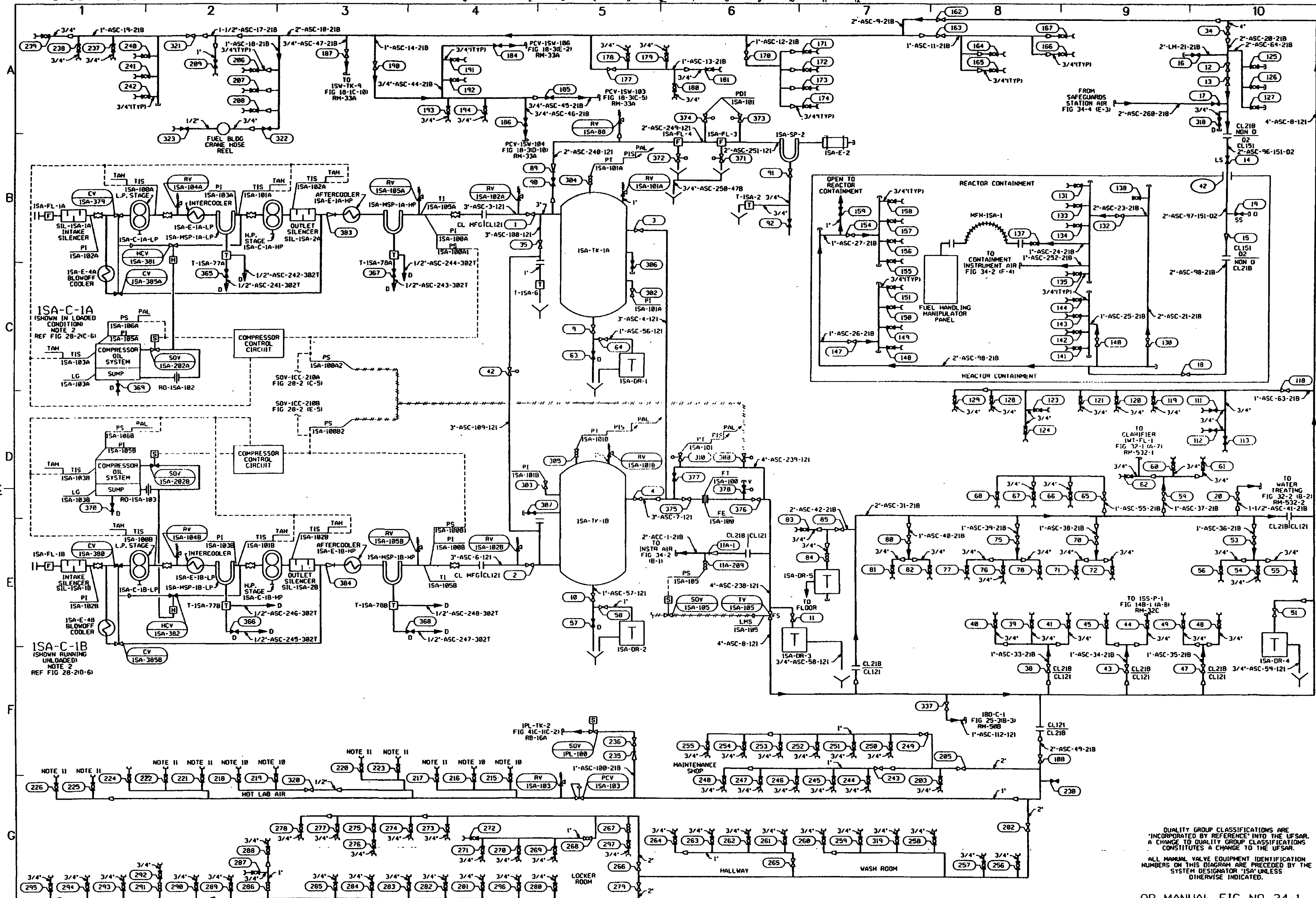
PIPING & INSTRUMENTATION DIAGRAM
RIVER WATER SYSTEM

AA 8700-RM-430-4 17

PREPARED ON THE BVPS CAEDDI SYSTEM

15-NOV-2005 16:21

No. 8700-RM-0434-001



OP MANUAL FIG NO 34-1

PORTIONS OF THIS DIAGRAM ARE NUCLEAR SAFETY RELATED

REVISION 19
INCORPORATED
ECP-06-0214
(DCN 1-RM-0434-001-E06-0214-01)

FENOC FirstEnergy Nuclear Operating Company

BEAVER VALLEY POWER STATION UNIT NO 1

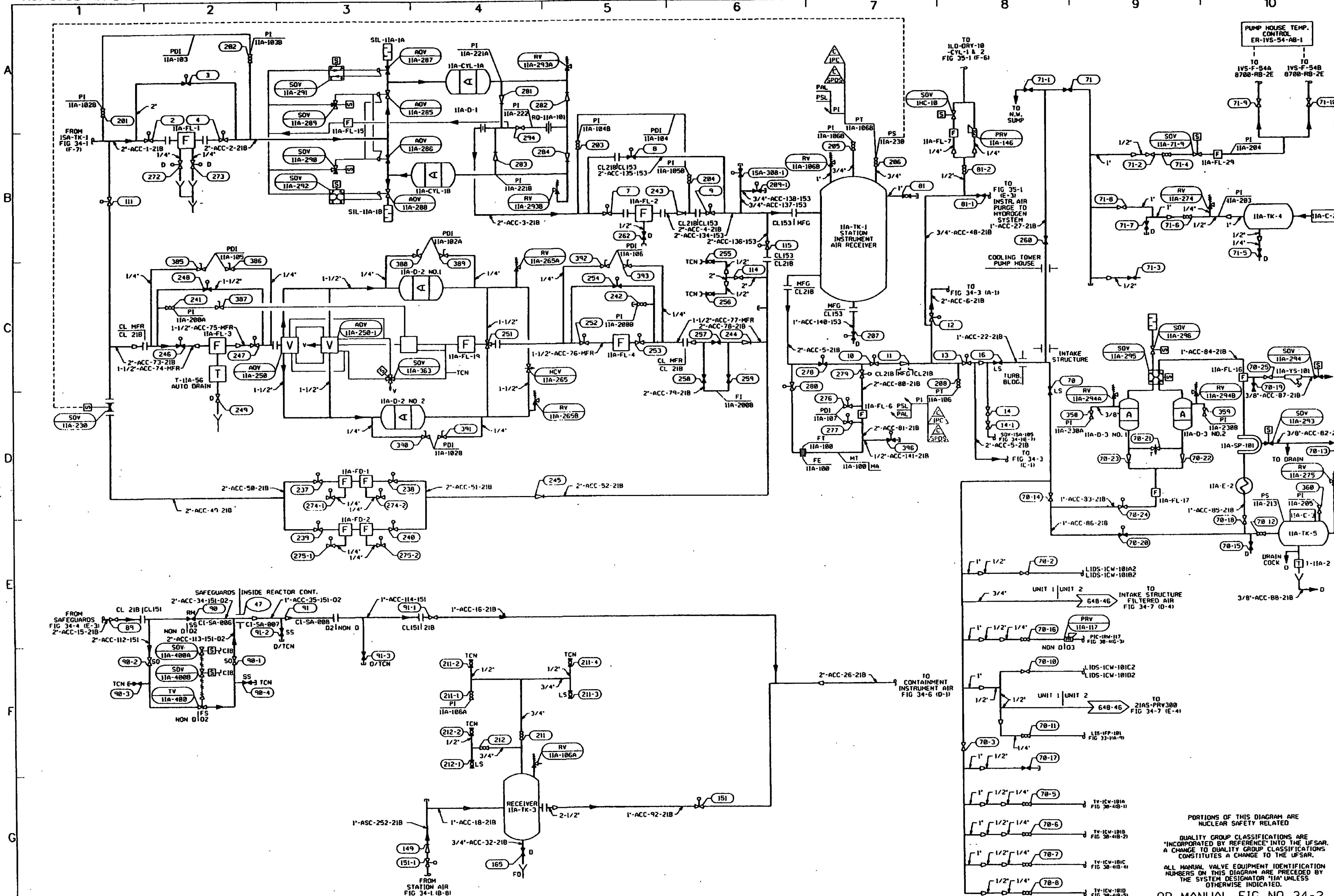
PIPING & INSTRUMENTATION DIAGRAM
STATION COMPRESSED AIR

5-25-07 | MVF | NONE | AA | 8700-RM-0434-001 | 19

PREPARED ON THE BVPS CAEDD SYSTEM

18-MAY-2007 13:43

No. 8700-RM-0434-002



PORTIONS OF THIS DIAGRAM ARE NUCLEAR SAFETY RELATED

QUALITY GROUP CLASSIFICATIONS ARE "INCORPORATED BY REFERENCE" INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.

ALL MANUAL VALVE EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR "IIA" UNLESS OTHERWISE INDICATED.

OP MANUAL FIG NO 34-2

REVISION 26 INCORPORATED DUN 09-0421-001-002 SATISFIED ECP 09-0197

FENOC FirstEnergy Nuclear Operating Company

BEAVER VALLEY POWER STATION UNIT NO 1

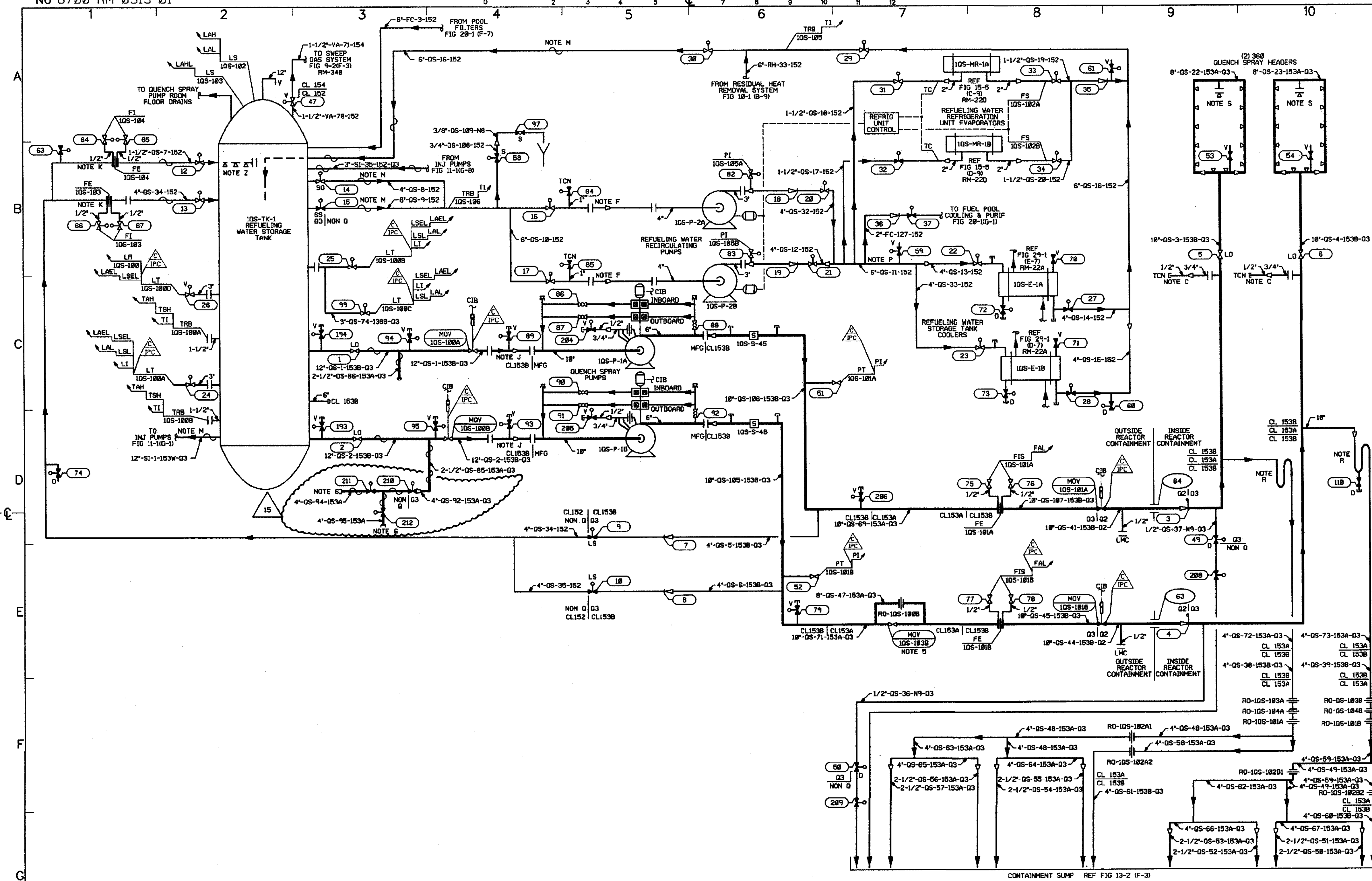
PIPING & INSTRUMENTATION DIAGRAM
INSTRUMENT AIR & CONT INST AIR

AA 8700-RM-0434-002 26

PREPARED ON THE BVPS CAEDDI SYSTEM

05-NOV-2009 13:29

No 8700-RM-0513-01



QUALITY GROUP CLASSIFICATIONS ARE 'INCORPORATED BY REFERENCE' INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.

PORTIONS OF THIS DIAGRAM ARE NUCLEAR SAFETY RELATED

ALL MANUAL VALVE EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR '1QS-' UNLESS OTHERWISE INDICATED.

REVISIONS

REVISION 15 INCORPORATED
DUN 13-0632-001-001
DUN 16-0518-001-003

FENOC — FIRST ENERGY NUCLEAR OPERATING COMPANY

SCALE NONE | DATE 11-7-16 | DRAWN BY J.M.POPE | SWG

FINAL APP FOR ISSUE 11/8/2016

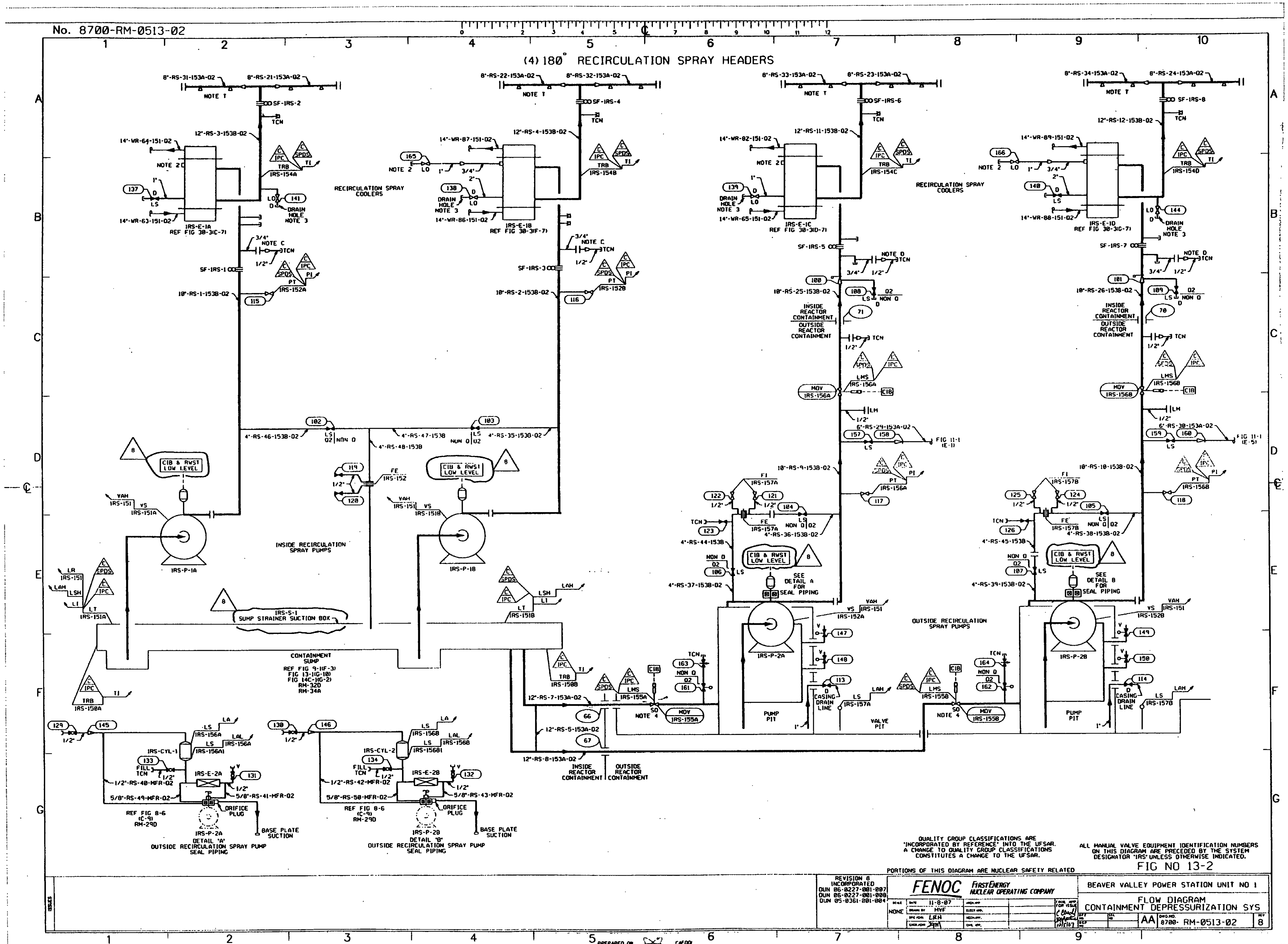
BEAVER VALLEY POWER STATION UNIT NO 1

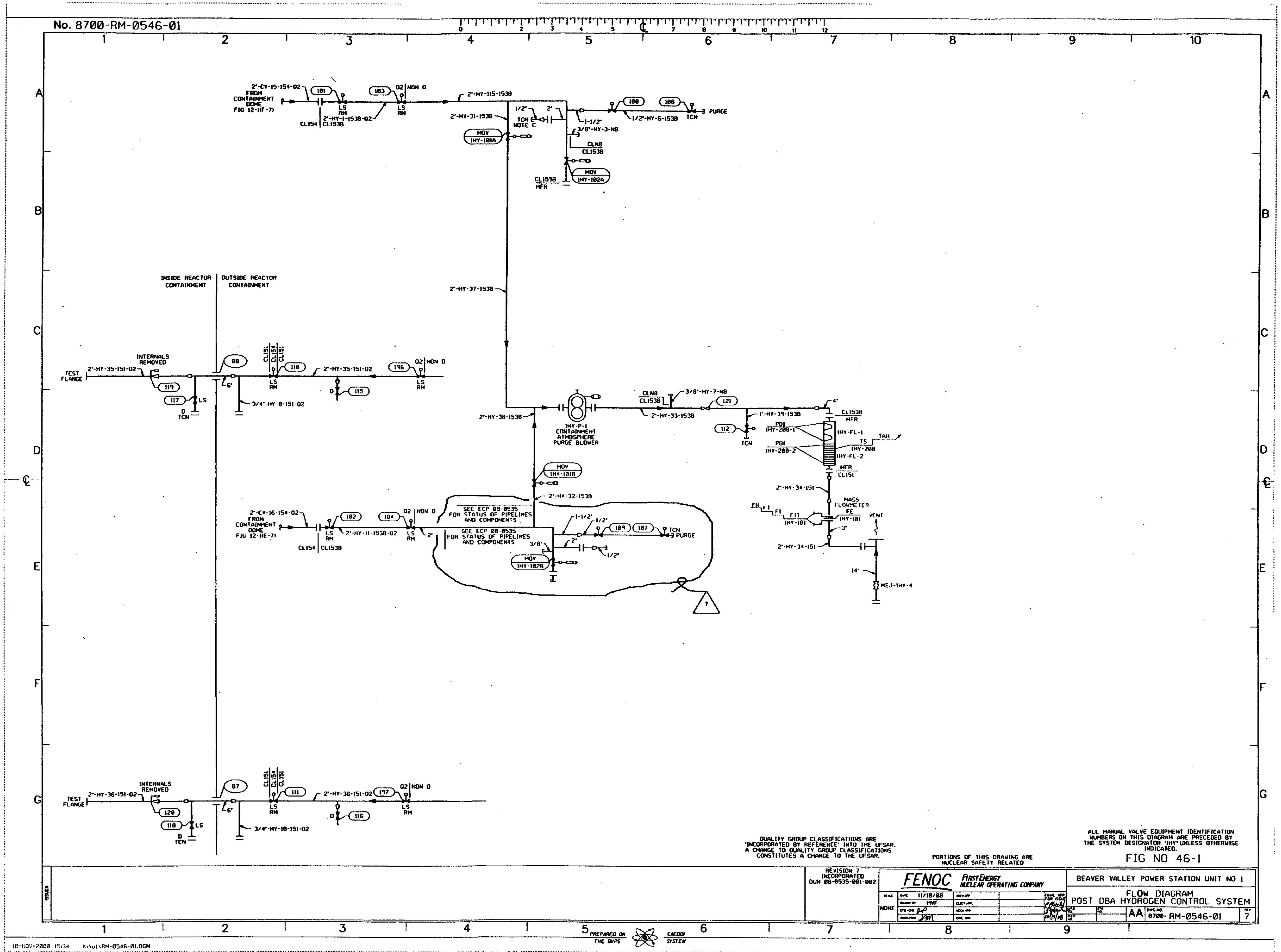
FLOW DIAGRAM
CONTAINMENT DEPRESSURIZATION SYS

AA | DWG NO 8700- RM-0513-01 | REV 15

PREPARED ON THE BVPS CAEDDI SYSTEM

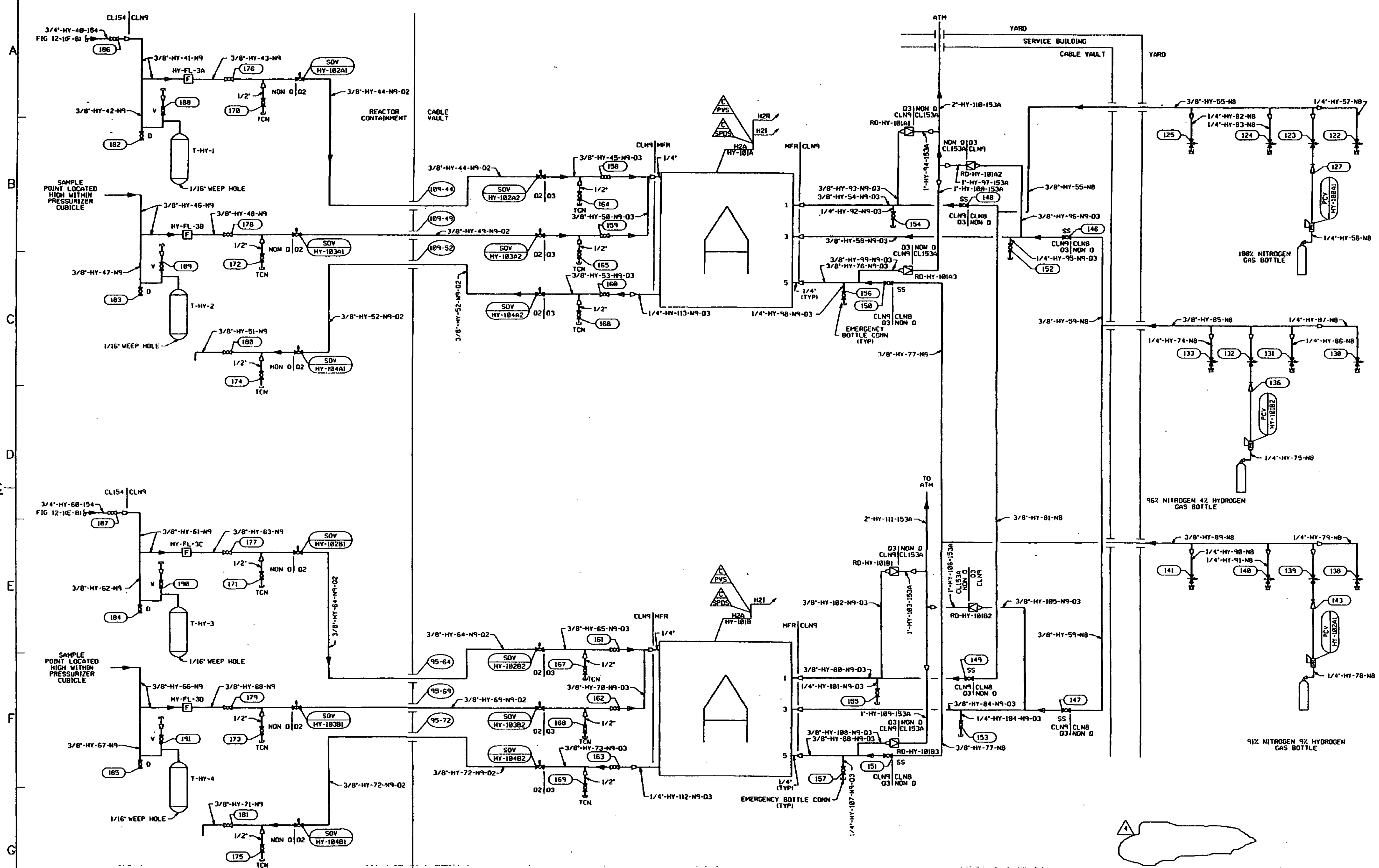
K:\U1\RM-0513-01.dgn


No. 8700-RM-0513-02
(4) 180° RECIRCULATION SPRAY HEADERS
RECIRCULATION SPRAY COOLERS
INSIDE RECIRCULATION SPRAY PUMPS
OUTSIDE RECIRCULATION SPRAY PUMPS
SUMP STRAINER SUCTION BOX
CONTAINMENT SUMP
REF FIG 9-1(F-3)
FIG 13-1(G-10)
FIG 14C-1(G-2)
RM-32D
RM-34A
INSIDE REACTOR CONTAINMENT
OUTSIDE REACTOR CONTAINMENT
PUMP PIT
VALVE PIT
CIB & RWST LOW LEVEL
DETAIL 'A' OUTSIDE RECIRCULATION SPRAY PUMP SEAL PIPING
DETAIL 'B' OUTSIDE RECIRCULATION SPRAY PUMP SEAL PIPING
BASE PLATE SUCTION
ORIFICE PLUG
QUALITY GROUP CLASSIFICATIONS ARE 'INCORPORATED BY REFERENCE' INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.
ALL MANUAL VALVE EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR '1RS' UNLESS OTHERWISE INDICATED.
FIG NO 13-2
PORTIONS OF THIS DIAGRAM ARE NUCLEAR SAFETY RELATED
REVISION 8 INCORPORATED
DUN 06-0227-001-007
DUN 06-0227-001-008
DUN 05-0361-001-004
FENOC FirstEnergy Nuclear Operating Company
BEAVER VALLEY POWER STATION UNIT NO 1
FLOW DIAGRAM
CONTAINMENT DEPRESSURIZATION SYS
AA 8700-RM-0513-02 8
PREPARED ON THE BVPS CADD SYSTEM



No. 8700-RM-0546-01
INSIDE REACTOR CONTAINMENT
OUTSIDE REACTOR CONTAINMENT
TEST FLANGE
INTERNALS REMOVED
FROM CONTAINMENT DOME FIG 12-1(F-7)
FROM CONTAINMENT DOME FIG 12-1(E-7)
PURGE
1HY-P-1 CONTAINMENT ATMOSPHERE PURGE BLOWER
MOV 1HY-101A
MOV 1HY-102A
MOV 1HY-101B
MOV 1HY-102B
SEE ECP 08-0535 FOR STATUS OF PIPELINES AND COMPONENTS
MASS FLOWMETER
VENT
MEJ-1HY-4
QUALITY GROUP CLASSIFICATIONS ARE "INCORPORATED BY REFERENCE" INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.
PORTIONS OF THIS DRAWING ARE NUCLEAR SAFETY RELATED
ALL MANUAL VALVE EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR '1HY' UNLESS OTHERWISE INDICATED.
FIG NO 46-1
REVISION 7 INCORPORATED DUN 08-0535-001-002
FENOC FIRSTENERGY NUCLEAR OPERATING COMPANY
BEAVER VALLEY POWER STATION UNIT NO 1
FLOW DIAGRAM
POST DBA HYDROGEN CONTROL SYSTEM
8700-RM-0546-01
7
PREPARED ON THE BVPS CADD SYSTEM

No. 8700-RM-546-2



PORTIONS OF THIS DRAWING ARE NUCLEAR SAFETY RELATED

REVISION 1 OF THIS DRAWING SUPERSEDES DRAWING 8700-RM-528 REV 4

QUALITY GROUP CLASSIFICATIONS ARE 'INCORPORATED BY REFERENCE' INTO THE UFSAR. A CHANGE TO QUALITY GROUP CLASSIFICATIONS CONSTITUTES A CHANGE TO THE UFSAR.

ALL MANUAL VALVE EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR 'HY' UNLESS OTHERWISE INDICATED.

FIG NO 46-2

REVISION 4 INCORPORATED ECR-04-0199 CA 04-01109-2

FENOC FirstEnergy Nuclear Operating Company

BEAVER VALLEY POWER STATION UNIT NO 1

FLOW DIAGRAM
POST DBA HYDROGEN ANALYZER SYSTEM

SCALE	NONE
DATE	8-4-04
DRAWN BY	MVF
DWG. NO.	AA 8700-RM-546-2
REV	4

04-AUG-2004 10:07 K:\ul\M5460020.DGN

PREPARED ON THE B/PS CADD SYSTEM