6.3 EMERGENCY CORE COOLING SYSTEMS

6.3.1 DESIGN BASES AND SUMMARY DESCRIPTION

Subsection 6.3.1 provides the design bases for the Emergency Core Cooling System (ECCS) and a summary description of the several systems as an introduction to the more detailed design descriptions provided in Subsection 6.3.2 and the performance analysis provided in Subsection 6.3.3.

6.3.1.1 Design Bases

6.3.1.1.1 Performance and Functional Requirements

The ECCS is designed to provide protection against postulated loss-of-coolant accidents (LOCA) caused by ruptures in primary system piping. The functional requirements, (for example, coolant delivery rates) specified in detail in Table 6.3-2B for Unit 1 and Table 6.3-2C for Unit 2, are such that the system performance under all LOCA conditions postulated in the design satisfies the requirements of 10CFR50.46. These requirements, the most important of which is that the post-LOCA peak cladding temperature be limited to 2200°F, are summarized in Subsection 6.3.3.2. In addition, the ECCS is designed to meet the following requirements:

1) Protection is provided for any primary line break up to and including the double-ended break of the largest line.

2) Two independent and diverse cooling methods (flooding and spraying) are provided to cool the core.

3) One high pressure cooling system is provided which is capable of maintaining water level above the top of the core and preventing ADS actuation for breaks of lines less than 1 inch nominal diameter.

4) With one exception, no operator action is required until 20 minutes after an accident to allow for operator assessment and decision. The only operator action assumed in the Section 6.3 ECCS analysis is that a RHR heat exchanger is placed in service within 20 minutes into the accident.

5) The ECCS is designed to satisfy all criteria specified in Section 6.3 for any normal mode of reactor operation.

6) A sufficient water source and the necessary piping, pumps and other hardware are provided so that the containment and reactor core can be flooded for possible core heat removal following a loss-of-coolant accident.

6.3.1.1.2 Reliability Requirements

The following reliability requirements apply:

1) The ECCS conforms to all licensing requirements, and good design practices of isolation, separation, and common mode failure considerations.

2) In order to meet the above requirements, the ECCS network has built-in redundancy so that adequate cooling can be provided, even in the event of specified failures. The following equipment makes up the ECCS:

   High Pressure Coolant Injection System (HPCI)
   Core Spray System (CS) (2 loops)
   Low Pressure Coolant Injection System (LPCI) (2 loops)
   Automatic Depressurization System (ADS)

3) The system is designed so that a single active or passive component failure, including power buses, electrical and mechanical parts, cabinets, and wiring will not disable the ADS.

4) In the event of a break in a pipe that is not a part of the ECCS, no single active component failure in the ECCS prevents automatic initiation and successful operation of less than the combination of ECCS equipment shown in Table 6.3-5 for a recirculation suction break (non-ECCS line break).

5) In the event of a break in a pipe that is a part of the ECCS, no single active component failure in the ECCS prevents automatic initiation and successful operation of less than the combination of ECCS equipment shown in Table 6.3-5 for a recirculation discharge break (ECCS line break).

   These are the minimum ECCS combinations which result after assuming the failures (from 4 above) and assuming that the ECCS line break disables a LPCI system loop.

6) Long-term cooling requires the removal of decay heat via the RHRSW system. In addition to the break which initiated the loss of coolant event, the system is able to sustain one failure, either active or passive, and still have at least one LPCI pump or one CS loop for makeup, and one RHR pump with a heat exchanger (including 100% RHRSW flow to the operating heat exchanger) for heat removal.

7) Off-site power is the preferred source of power for the ECCS network and every reasonable precaution is made to assure its high availability. However, on-site emergency power is provided with sufficient diversity and capacity so that all the above requirements are met even if off-site power is not available.

8) The on-site diesel fuel reserve is designed in accordance with IEEE-308 criteria as stated in Subsection 7.1.2.5.2.

9) Diesel-load configuration is 1 LPCI pump and 1 CS pump connected to a single diesel generator. (Typical for four aligned diesels in a one unit LOCA.)

10) Systems which interface with, but are not part of, the ECCS are designed and operated such that failure(s) in the interfacing systems do not propagate to and/or affect the performance of the ECCS.

11) Non-ECCS systems interfacing with the ECCS buses are automatically shed from and/or are initially inhibited from the ECCS buses of the affected unit when a LOCA signal exists and off-site AC power is not available.

12) No more than one storage battery is connectable to a DC power bus.

13) Each low pressure system of the ECCS including flow rate and sensing networks is capable of being tested during shutdown. All active components are capable of being tested during plant operation, including logic required to automatically initiate component action.

14) Provisions for testing the ECCS network components (electronic, mechanical, hydraulic and pneumatic, as applicable) are installed in such a manner that they are an integral and non-separable part of the design.

6.3.1.1.3 ECCS Requirements for Protection from Physical Damage

The emergency core cooling system piping and components are protected against damage from movement, from thermal stresses, from the effects of the LOCA and the Safe Shutdown Earthquake. The ECCS is protected against the effects of pipe whip, which might result from piping failures up to and including the LOCA. This protection is provided by separation, pipe whip restraints, or energy absorbing materials if required. Any of these three methods will be applied to provide protection against damage to piping and components of the ECCS which otherwise could result in a reduction of ECCS effectiveness to an unacceptable level.

Mechanical separation outside the drywell is achieved as follows:

1) The ECCS shall be separated into three functional groups:

   a. HPCI

   b. CS(A&C) + LPCI(A&C)

   c. CS(B&D) + LPCI(B&D)

2) The equipment in each group shall be separated from that in the other two groups. In addition, the HPCI and the RCIC (which is not part of the ECCS) shall be separated.

3) Separation barriers shall be constructed between the functional groups as required to assure that environmental disturbances such as fire, pipe rupture, falling objects, etc., affecting one functional group will not adversely affect the remaining groups. In addition, separation barriers shall be provided as required to assure that such disturbances do not affect both RCIC and HPCI. For additional discussion, refer to Section 3.12.

6.3.1.1.4 ECCS Environmental Design Basis

The low pressure systems of the ECCS have testable check valves in the drywell portions of their respective piping runs. These safety-related, injection/isolation valves are designed for abnormal environmental requirements.

The ECCS equipment (e.g., pumps, motors) is qualified for abnormal environmental requirements.

Abnormal environmental conditions to which these components are qualified or designed are described in Section 3.11.

For a listing of all safety-related valves located in the drywell subject to spray impingement from the containment Spray System, see Table 6.3-10. No safety-related valves become submerged because of spray from the Containment Spray System.

6.3.1.2 Summary Descriptions of ECCS

The ECCS injection network comprises a high pressure coolant injection (HPCI) system, a low pressure core spray (CS) system and the low pressure coolant injection (LPCI) mode of the Residual Heat Removal System. These systems are briefly described here as an introduction to the more detailed system design descriptions provided in Subsection 6.3.2. The Automatic Depressurization System (ADS) which assists the injection network under certain conditions is also briefly described. Boiling water reactors which employ the same ECCS design are listed in Table 1.3-3.

6.3.1.2.1 High Pressure Coolant Injection

The HPCI pumps water through the feedwater sparger. The primary purpose of HPCI is to maintain reactor vessel inventory after small breaks which do not depressurize the reactor vessel.

6.3.1.2.2 Core Spray

The two loops pump water into peripheral ring spray spargers mounted above the reactor core. The primary purpose of CS is to provide inventory makeup and spray cooling during large breaks in which the core is calculated to uncover. When assisted by ADS, CS also provides protection for small breaks.

6.3.1.2.3 Low Pressure Coolant Injection

LPCI is an operating mode of the Residual Heat Removal System. Four pumps deliver water from the suppression pool to the recirculation lines. The primary purpose of LPCI is to provide vessel inventory makeup following large pipe breaks. When assisted by ADS, LPCI also provides protection for small breaks.

6.3.1.2.4 Automatic Depressurization System

ADS utilizes a number of the reactor safety/relief valves to reduce reactor pressure during small breaks in the event of HPCI failure. When the vessel pressure is reduced to within the capacity of the low pressure systems (CS and LPCI), these systems provide inventory makeup so that acceptable post accident temperatures are maintained.

6.3.2 SYSTEM DESIGN

More detailed descriptions of the individual systems including individual design characteristics are provided in Subsections 6.3.2.1 through 6.3.2.4. The following discussion will provide details of the combined systems; and in particular, those design features and characteristics which are common to all systems.

6.3.2.1 Schematic Piping and Instrumentation Diagrams

The P&IDs and process diagrams for the ECCS are identified in Subsection 6.3.2.2.

6.3.2.2 Equipment and Component Descriptions

The starting signal for the ECCS comes from at least two independent and redundant sensors of high drywell pressure, low RPV pressure, and low reactor water level. The ECCS is actuated automatically and with one exception requires no operator action during the first 20 minutes following the accident. The only operator action assumed in the Section 6.3 ECCS analysis is that a RHR heat exchanger is placed in service within 20 minutes into the accident. A time sequence for a Design Basis LOCA analysis showing starting of the systems is provided in Table 6.3-1B-2 for Unit 1 and Table 6.3-1B-3 for Unit 2.

Electric power for operation of the ECCS (except the dc powered HPCI system) is from the preferred offsite ac power supply. Upon loss of the preferred source, operation is from the onsite standby diesel generators. Four diesel generators supplying individual ac buses have sufficient diversity and capacity so that failure of one diesel satisfies ECCS requirements. Section 8.3 contains a more detailed description of the power supplies for the ECCS.

6.3.2.2.1 High Pressure Coolant Injection (HPCI) System

The high pressure coolant injection (HPCI) system consists of a steam turbine driven constant-flow pump assembly, associated system piping, valves, controls, and instrumentation. The P&ID for HPCI, Dwgs. M-155, Sh. 1 and M-156, Sh. 1, shows the system components and their arrangement. The HPCI system Process Diagram, Dwg. M1-E41-4, Sh. 1, shows the design operating modes of the system.

The HPCI system equipment is installed in the reactor building. Suction piping comes from both the condensate storage tank and the suppression pool. Injection water is piped to the reactor feedwater line. Steam supply for the turbine is piped from a main steamline in the primary containment. This piping is provided with an isolation valve on each side of the primary containment. Remote controls for valve and turbine operation are provided in the main control room. The controls and instrumentation of the HPCI system are described, illustrated, and evaluated in Section 7.3.

The HPCI system is provided to ensure that the reactor core is adequately cooled to meet the design bases in the event of a small break in the reactor coolant pressure boundary (RCPB) and loss of coolant that does not result in rapid depressurization of the reactor vessel. This permits the plant to be shut down while maintaining sufficient reactor vessel water inventory until the reactor vessel is depressurized. The HPCI system continues to operate until the reactor vessel pressure is below the pressure at which LPCI operation or core spray system operation can maintain core cooling.

The HPCI system is designed to pump water into the reactor vessel for a wide range of pressures in the reactor vessel. Initially, demineralized water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor. Water from either source is pumped into the reactor vessel through the feedwater spargers.

The level instrumentation on the condensate storage tank is used to automatically transfer the HPCI suction from the condensate storage tank to the suppression pool at a level determined by conservative NPSH calculations. The calculations ensure adequate NSPH during the transfer process and ensure there is no unacceptable vortex formation in the suction lines during the transfer process. This suction transfer can be remotely overridden to realign suction to the CST. The portion of the suction piping exposed to outside air temperatures is protected from cold weather effects both by heat tracing and by insulation.

The temperature of the level instrumentation is monitored by temperature instrumentation which alarms in the control room if the temperature falls below 40^{o}F.

The pump assembly is located below the level of the condensate storage tank and below the water level in the suppression pool to ensure positive suction head to the pumps. Pump NPSH requirements are met by providing adequate suction head and adequate suction line size. Available NPSH is calculated using the assumptions of Regulatory Guide 1.1 (12/70). The NPSH calculations are shown in Sections 6.3.2.2.1.1 and 6.3.2.2.1.2. The required NPSH is shown in Table 6.3-8. See also Figure 6.3-3a.

The HPCI turbine-pump assembly and piping are protected from detrimental physical effects of the DBA, such as pipe whip, flooding, and high temperature. The equipment is located outside the primary containment.

The HPCI turbine is driven by steam from the reactor vessel which is generated by decay and residual heat. The steam is extracted from a main steamline upstream of the main steamline isolation valves. The inboard and outboard HPCI isolation valves in the steamline to the HPCI turbine are normally open. This keeps the piping to the turbine at an elevated temperature to permit rapid startup of the HPCI system. The inboard isolation valve has a bypass line containing a normally closed valve. This bypass line permits pressure equalization and drainage around the isolation valve and downstream line warmup prior to opening of the isolation valve. Signals from the HPCI control system open or close the supply valve adjacent to the turbine.

A condensate drain pot is provided upstream of the turbine stop valve to prevent the HPCI steam supply line from filling with water. The drain pot normally routes the condensate to the main condenser, but upon receipt of a HPCI initiation signal or a loss of control air pressure, isolation valves on the condensate line automatically close.

The turbine power is controlled by a flow controller, sensing pump discharge flow and providing a variable signal (1-5 volts DC) to the turbine governor, to maintain constant pump discharge flow over the pressure range of operation. The turbine control system is capable of limiting speed overshoot to 15 percent of maximum operating speed on a quick start while driving only the pump inertia load. Limit switches are provided on the turbine control valve to indicate fully open and closed positions. Both lights shall be "on" in midposition.

As reactor steam pressure decreases, the HPCI turbine control valve opens further to pass the steam flow required to provide the necessary pump flow. The capacity of the system is selected to provide sufficient core cooling to prevent clad temperatures in excess of the limits (10CFR50.46) while the pressure in the reactor vessel is above the pressure at which core spray and LPCI become effective.

Exhaust steam from the HPCI turbine is discharged to the suppression pool. A drain pot at the low point in the exhaust line collects moisture present in the steam. Collected moisture is discharged through an orifice to the barometric condenser.

The HPCI turbine gland seals are routed to the barometric condenser for cooling and containment of radioactive steam. Noncondensable gases from the barometric condenser are pumped to the Standby Gas Treatment System.

The check valves and two isolation valves are provided in the vacuum breaker line which connects the air space in the suppression chamber with the HPCI turbine exhaust line. This eliminates any possibility of water from the suppression pool being drawn into the HPCI turbine exhaust line. The isolation valve in this vacuum breaker line operates automatically via a combination of low reactor pressure and high drywell pressure. Test connections are provided on either side of the two check valves.

Startup of the HPCI system is completely independent of ac power. Only dc power from the station battery and steam extracted from the nuclear system are necessary.

The various operations of the HPCI components are summarized as follows:

The HPCI controls automatically start the system and bring it to design flowrate within 30 seconds from receipt of a reactor pressure vessel (RPV) low water level signal or a primary containment (drywell) high pressure signal. Refer to Chapter 15 for more analysis details.

The HPCI turbine is shut down automatically by any of the following signals:

1) Turbine overspeed - This prevents damage to the turbine.

2) RPV high water level - This indicates that core cooling requirements are satisfied.

3) HPCI pump low suction pressure - This prevents damage to the pump due to loss of flow.

4) HPCI turbine exhaust high pressure - This indicates a turbine or turbine control malfunction.

If an initiation signal is received after the turbine is shut down, the system will restart automatically if no shutdown signals exists.

Additionally, because the steam supply line to the HPCI turbine is part of the RCPB, certain signals automatically isolate this line, causing shutdown of the HPCI turbine. The auto isolation signal will not clear until manually reset after the clearance of all isolation signals. Automatic shutoff of the steam supply is described in Section 7.3. However, automatic depressurization and the low pressure systems of the ECCS act as backup, and automatic shutoff of the steam supply does not negate the ability of the ECCS to satisfy the safety objective.

In addition to the automatic operational features of the system, provisions are included for remote manual startup, operation, and shutdown (provided automatic initiation or shutdown signals do not exist).

HPCI operation automatically actuates the following valves:

1) HPCI pump discharge shutoff valves
2) HPCI steam admission valve
3) HPCI turbine stop valve
4) HPCI turbine control valve
5) HPCI steamline drain isolation valves
6) HPCI test return valve to CST if open
7) Minimum flow bypass valve

Prior to startup, the turbine control system will be held at the low speed design condition. Upon receipt of an initiating signal a speed ramp generator module will automatically run the control system toward its high speed design point, thereby controlling the transient acceleration of the turbine. The flow controller will automatically over-ride the speed ramp generator and when rated flow is established, the flow controller signal adjusts the setting of the turbine control so that rated flow is maintained as nuclear system pressure decreases.

Startup of the auxiliary oil pump and proper functioning of the hydraulic control system is required to open the turbine stop and control valves. Operation of the barometric condenser components is required to prevent outleakage from the turbine shaft seals. Startup of the condenser equipment is automatic, but its failure does not prevent the HPCI system from fulfilling its core cooling objective.

A minimum flow bypass is provided for pump protection. The bypass valve automatically opens on a low flow signal if the HPCI pump discharge pressure permissive is present, and automatically closes on a high flow signal. When the bypass is open, flow is directed to the suppression pool.

A line used for system testing leads from the HPCI pump discharge line to the condensate storage tank. The shutoff valves in this line are sequenced closed upon HPCI system initiation. To prevent pumping suppression pool water to the CST, these valves are also interlocked closed when the pump suction valve from the suppression pool is open. All automatically operated valves are equipped with a remote manual functional test feature.

The HPCI system initially injects water from the condensate storage tank. When the water level in the tank falls below some predetermined level, the pump suction is automatically transferred to the suppression pool. This level was determined by conservative calculations, which ensure that no unacceptable vortex formation would occur during the transfer process. In addition a vortex breaker is located at the suction nozzle of the CST to prevent vortex formation. Preoperational testing demonstrated that vortex formation did not occur. This test was performed with the condensate storage tank level at the transfer level, with the core spray pumps operating at 6000 gpm. This transfer may also be made from the main control room using remote controls. When the pump suction has been transferred to the suppression pool, a closed loop is established for recirculation of water escaping from a break. Suction can also be transferred to the CST if desired to access the remaining available volume.

To assure continuous core cooling, signals to isolate the containment do not operate any HPCI valves.

The HPCI system incorporates a relief valve in the pump suction line to protect the components and piping from inadvertent overpressure conditions.

The HPCI pump and piping are positioned to avoid damage from the physical effects of design basis accidents, such as pipe whip, missiles, high temperature, pressure, and humidity.

The HPCI equipment and support structures are designed in accordance with Seismic Category I criteria (see Chapter 3). The system is assumed to be filled with water for seismic analysis.

Provisions are included in the HPCI system which will permit the HPCI system to be tested. These provisions are:

1) A full flow test line is provided to route water to the condensate storage tank without entering the reactor pressure vessel.

2) A minimum flow bypass test line is provided to route water to the suppression pool without entering the reactor pressure vessel.

3) Instrumentation is provided to indicate system performance during normal test operations.

4) All motor-operated valves are capable of either local or remote manual operation for test purposes.

5) Drains are provided to leak test the major system valves.

The operating parameters for the components of the HPCI system, defined below, are shown on Dwg. M1-E41-4, Sh. 1.

1) One 100 percent capacity booster and main pump assembly and accessories

2) Piping, valves, and instrumentation for:

    a. Steam supply to the turbine

    b. Turbine exhaust to the suppression pool

    c. Supply from the condensate storage tank to the pump suction

    d. Supply from the suppression pool to the pump suction

    e. Pump discharge to the feedwater line spargers, including a test line to the condensate storage tank, a minimum flow bypass line to the suppression pool, and a cooling water supply to accessory equipment.

The basis for the design conditions was the ASME Section III, Nuclear Power Plant Components. The design parameters for the HPCI system components are shown in Table 6.3-8.

6.3.2.2.1.1 NPSH Available with Suction from the Condensate Storage Tank

The available NPSH is calculated in accordance with Regulatory Guide 1.1. The following data was used in the calculation:

a. Condensate storage tank water level is conservatively assumed to be two feet below the transfer level.

b. Condensate storage tank water is at 100°F.

c. Both HPCI and RCIC are in operation.

d. NPSHA = hs - hf + ha - hvpa

hs = static head
hf = friction head loss
ha = atmospheric pressure head
hvpa = vapor pressure

Unit I

hs = 673.75' - 650.25' = 23.5 ft.
hf = 7.1 ft.
ha = 33.16 ft.
hvpa = 2.2 ft.
NPSHA = 47.36 ft.

Unit II

hs = 673.75'-650.25' = 23.5 ft.
hf = 12.19 ft.
ha = 33.16 ft.
hvpa = 2.21 ft.
NPSHA = 42.26 ft.
NPSHR = 21 ft.

6.3.2.2.1.2 NPSH Available with Suction from the Suppression Pool

The available NPSH is calculated in accordance with Regulatory Guide 1.1. The following data was used in the calculation:

a. Suppression pool is at the minimum level of El. 668.5 feet. 668.5 El. is due to pool level drop for worst case passive failure in an ECCS pump room (Subsection 6.3-6).

b. Suppression pool water is at its maximum temperature for the given operating mode, 140□F.

c. Atmospheric pressure is assumed over the suppression pool.

d. NPSHA = hs - hf + ha - hvpa

hs = 668.5' - 650.25' = 18.25'
hf = 13.23 ft.
ha = 33.16 ft.
hvpa = 6.8 ft.
NPSHA = 31.4 ft. NPSHR = 21 ft.

6.3.2.2.2 Automatic Depressurization System (ADS)

If the RCIC and the HPCI cannot maintain the reactor water level, the automatic depressurization system, which is independent of any other system of the ECCS, reduces the reactor pressure so that flow from the LPCI and CS systems enters the reactor vessel in time to cool the core and limit any increase in fuel cladding temperature.

The automatic depressurization system employs nuclear system safety/relief valves to relieve high pressure steam to the suppression pool. The design, number, location, description, operational characteristics, and evaluation of the safety/relief valves are discussed in detail in Subsection 5.2.2. The operation of the ADS is discussed in Subsection 7.3.1.1a.1.4.

6.3.2.2.3 Core Spray (CS) System

Each of the two redundant core spray systems consists of: two 50% capacity centrifugal pumps that can be powered by normal auxiliary power or the standby ac power system; a spray sparger in the reactor vessel above the core (a separate sparger for each CS system); piping and valves to convey water from the suppression pool to the sparger; and associated controls and instrumentation. Dwg. M-152, Sh. 1, the CS system P&ID, presents the system components and their arrangement. The CS system Process Diagram, Dwg. M1-E21-15, Sh. 1, shows the design operating modes of the system. A simplified system flow diagram showing system injection into the reactor vessel is presented in Dwg. M1-E21-15, Sh. 1 for the CS system.

When low water level in the reactor vessel or high pressure in the drywell is sensed, and with reactor vessel pressure low enough, the core spray system automatically starts and sprays water into the top of the fuel assemblies to cool the core. The CS injection piping enters the vessel, divides, and enters the core shroud at two points near the top of the shroud. A sparger is attached to each outlet. Nozzles are spaced around the sparger to spray the water radially over the core and into the fuel assemblies.

The CS system is designed to provide cooling to the reactor core only when the reactor vessel pressure is low, as is the case for large LOCA break sizes. However, when the CS operates in conjunction with the ADS, the effective core cooling capability of the CS is extended to all break sizes because the ADS will rapidly reduce the reactor vessel pressure to the CS operating range.

The core spray pump and all motor operated valves can be operated individually by manual switches located in the control room. Operating indication is provided in the control room by a flowmeter and valve indicator lights.

To assure continuity of core cooling, signals to isolate the containment do not operate any core spray system valves.

The discharge line to the reactor is provided with two isolation valves. One of these valves is a testable (with an air-operated solenoid valve) check valve located inside the drywell as close as practical to the reactor vessel. The check valve will move to the close position on loss of air and/or power. CS injection flow causes this valve to open during LOCA conditions (i.e., no power is required for valve actuation during LOCA). If the CS line should break outside the containment, the check valve in the line inside the drywell will prevent loss of reactor water outside the containment.

The other isolation valve (which is also referred to as the CS injection valve) is a motor operated gate valve located outside the primary containment as close as practical to the CS discharge line penetration into the containment. This valve is capable of opening with the maximum differential across the valve expected for any system operating mode. The valve stroke time is less than or equal to 19 seconds. This valve is normally closed to back up the inside testable check valve for containment integrity purposes.

The CS system components and piping are arranged to avoid unacceptable damage from the physical effect of design-basis accidents, such as pipe whip, missiles, high temperature, pressure and humidity.

All principal active CS equipment is located outside the primary containment.

A check valve (one per CS pump), and in each loop one flow element and restricting orifice are provided in the CS discharge line from the pump to the injection valve. The check valve is located below the minimum suppression pool water level and is provided so the piping downstream of the valve can be maintained full of water by the discharge line fill system (see Subsection 6.3.2.2.5). The flow element is provided to measure system flow rate during LOCA and test conditions and for automatic control of the minimum low flow bypass gate valve. The measured flow is indicated in the main control room. The restricting orifice was sized during pre-operational test of the system to limit system flow to acceptable values as described on the CS system Process Diagram. (Dwg. M1-E21-15, Sh. 1)

The CS pump (pump performance test results) characteristics, head, flow, horsepower, and required NPSH are shown in Figure 6.3-118.

A low flow bypass line with a motor operated gate valve connects to the CS discharge line upstream of the check valve on the pump discharge line. The line bypasses water to the suppression pool to prevent pump damage due to overheating when other discharge line valves are closed or reactor pressure is greater than the CS system discharge pressure following system initiation. The valve automatically closes when flow in the main discharge line is sufficient to provide required pump cooling.

CS flow passes through a motor-operated pump suction valve that is normally open. This valve can be closed by a remote manual switch (located in the control room) to isolate the CS system from the suppression pool should a leak develop in the system. This valve is located in the core spray pump suction line as close to the suppression pool penetration as practical. Because the CS conveys water from the suppression pool, a closed loop is established for the spray water escaping from the break.

The design pressure and temperature for various portions of the system were established in accordance with the ASME Section III Boiler and Pressure Vessel code and the required core spray system design specification written for this system. The original Core Spray System Process Diagram was used as input in the development of the design specification.

Each of the two redundant core spray systems takes suction from the suppression pool through a suction line that has two high capacity stacked disk strainers. The strainers have sufficient capacity to filter their design debris source term under worst case conditions while maintaining strainer pressure drop below the maximum value required to provide adequate NPSH and system flow. The design debris source term consists of conservative amounts of insulation, paint chips and other drywell debris that could reach the strainers after being destroyed by LOCA jet forces and transported to the suppression pool through the downcomers. This debris is assumed to be filtered by the strainers along with corrosion products that would exist in the suppression pool prior to a LOCA. Correlations between the amount of debris filtered by the strainers and strainer pressure drop are based on testing performed on one of the Susquehanna RHR strainers (which have a design similar to the CS strainers) and NRC approved methodology outlined in NEDO-32686, "Utility Resolution Guide for ECCS Suction Strainer Blockage". The suppression pools are cleaned and inspected periodically to maintain corrosion product amounts at acceptable levels and to confirm the absence of miscellaneous debris that would be a strainer blockage threat.

The CS pump is located in the reactor building below the water level in the suppression pool to assure positive pump suction. Pump NPSH requirements are met with the containment at atmospheric pressure. A pressure gage is provided to indicate the suction head. The available NPSH has been calculated in accordance with NRC Regulatory Guide 1.1. The CS pump characteristics are shown in Figure 6.3-118.

The CS system incorporates relief valves to prevent the components and piping from inadvertent overpressure conditions. One relief valve, located on the pump discharge, is set at 500 psig with a capacity of 100 gpm at 10% accumulation. The second relief valve is located on the suction side of the pump and is set for 100 psig at a capacity of 10 gpm at 10% accumulation.

The CS system piping and support structures are designed in accordance with Seismic Category I criteria (see Chapter 3). The system is assumed to be filled with water for seismic analysis.

Provisions are included in the CS system which will permit the CS system to be tested. These provisions are:

1) All active CS components are testable during normal plant operation.

2) A full flow test line is provided to route water from and to the suppression pool without entering the reactor pressure vessel.

3) A suction test line supplying reactor grade water, is provided to test pump discharge into the reactor pressure vessel during normal plant shutdown.

4) Instrumentation is provided to indicate system performance during normal and test operations.

5) All motor-operated valves and check valves are capable of operation for test purposes. The Core Spray pump discharge check valves (152/252 F003A,B,C,&D) have local disc position indication on the valve hinge pins.

6.3.2.2.3.1 NPSH Available for CS

The available NPSH is calculated in accordance with Regulatory Guide 1.1. The following data was used in the calculation:

a. The suppression pool is at the minimum post-accident level of El. 667.3 feet. 667-3' El. is due to suppression pool draw down assuming the worst case break (Main Steam Line break inside containment).

b. The centerline of the pump suction is at El. 646'-10 5/8".

c. The suction strainers (total of two strainers for each suction line) are filtering their design debris source term. The maximum pressure loss across the strainers at maximum runout flow of 7900 gpm is 4.3 psi. The pump vendor required NPSH at runout flow is 4.0 feet.

d. Atmospheric pressure head is assumed to be equal to the vapor pressure, h_a - h_{vp}

e. The suppression pool water is assumed to be at 220°F.

f. $NPSH_A = h_s - h_f + h_a - h_{vp}$

h_s = static head
h_f = friction head loss
h_a = atmospheric pressure head
h_{vp}= vapor pressure

Based on Section 6.3.2.2.3.1.d, NPSHA = $h_s - h_f$

with h_s = 20.41 feet
h_f = 14.66 feet

$NPSH_A = 20.41 – 14.66$
$NPSH_A$ = 5.75 feet
$NPSH_r$ = 4.0 feet

6.3.2.2.4 Low Pressure Coolant Injection (LPCI) System

The low pressure coolant injection system is an operating mode of the RHR system. The LPCI system is automatically actuated by low water level in the reactor or high pressure in the drywell (high pressure in the drywell must be accompanied by a reactor vessel low pressure permissive signal) and uses four motor-driven RHR pumps to draw suction from the suppression pool and inject cooling water flow into the reactor core via the recirculation loop to accomplish cooling of the core by flooding.

The LPCI system, like the CS system, is designed to provide cooling to the reactor core only when the reactor vessel pressure is low, as is the case for large LOCA break sizes. However, when the LPCI operates in conjunction with the ADS, the effective core cooling capability of the LPCI is extended to all break sizes because the ADS will rapidly reduce the reactor vessel pressure to the LPCI operating range. NPSH for these flow conditions is shown in Figure 6.3-119.

Dwgs. M1-E11-3, Sh. 1 and M1-E11-3, Sh. 2, show a process diagram, (and process data) of the RHR system. The LPCI System P&ID is presented in Dwgs. M-151, Sh. 1, M-151, Sh. 2, M-151, Sh. 3, and M-151, Sh. 4.

LPCI operation includes using associated valves, control, instrumentation, and pump accessories. LPCI is normally powered from the preferred ac power source and from the standby ac power source upon a loss of preferred ac power.

In the event of a LOCA, the two halves of the LPCI system inject water into the discharge line in each recirculation loop. Since electrical power to each LPCI pump is separate, it is necessary to have a bus arrangement which permits the valves of a LPCI loop that has been disabled by a single failure of a divisional electrical supply to be energized by an alternate electrical supply. This feature preserves the ability of the LPCI to inject into the unbroken recirculation loop as well as the broken loop. See Table 6.3-5 for the LPCI pumps available during a LOCA and a limiting single failure.

To assure continuity of core cooling, signals to isolate the primary containment do not operate any RHR system valves which interfere with the LPCI mode of operation.

The process diagram, Dwgs. M1-E11-3, Sh. 1 and M1-E11-3, Sh. 2, and the P&ID, Dwgs. M-151, Sh. 1, M-151, Sh. 2, M-151, Sh. 3, and M-151, Sh. 4, indicate a great many available flow paths other than the LPCI injection line. However, the low water level or high drywell pressure signal and RPV low pressure signals which automatically initiate the LPCI mode are also used to realign containment cooling and spray modes of operation and revert other associated valves to the LPCI lineup. Inlet and outlet valves from the heat exchangers receive no automatic signals as the system is designed to provide rated flow to the vessel whether they are open or not.

A check valve in the pump discharge line is used together with a discharge line fill system (see Subsection 6.3.2.2.5) to prevent water hammer resulting from a pump start with an empty discharge line. A flow element in the pump discharge line is used to provide a measure of system flow and to originate automatic signals for control of the pump minimum flow valve. The minimum 1flow valve permits a small flow to the suppression pool in the event no discharge valve is open or in case vessel pressure is higher than pump shutoff head.

Using the suppression pool as the source of water for the LPCI establishes a closed loop for recirculation of LPCI water escaping from the break.

The design pressure and temperature for various portions of the system were established in accordance with the ASME Section III Boiler and Pressure Vessel code and the required RHR system design specification written for this system. The RHR System Process Diagram (Dwgs. M1-E11-3, Sh. 1, and M1-E11-3, Sh. 2) was used as input in the development of the design specification.

LPCI pumps and equipment are described in detail in Subsection 5.4.7, which also describes the other functions served by the same pumps if not needed for the LPCI function. The RHR heat exchangers are not associated with the emergency core cooling function. The heat exchangers are discussed in Subsection 6.2.2. The portions of the RHR required for accident protection including support structures are designed in accordance with Seismic Category I criteria (see Chapter 3). The LPCI pump characteristics are shown in Figure 6.3-119.

The LPCI system incorporates a relief valve on each pump suction line and the LPCI discharge header which protects the components and piping from inadvertent overpressure conditions. These valves are set to relieve pressure at 165 psig and 450 psig, respectively.

Provisions are included in the LPCI system to permit testing of the system. These provisions are:

1) All active LPCI components are designed to be testable during normal plant operation.

2) A discharge test line is provided for the four pumps to route suppression pool water back to the suppression pool without entering the reactor pressure vessel.

3) Instrumentation is provided to indicate system performance during normal and test operations.

4) All motor-operated valves, air-operated valves and check valves are capable of operation for test purposes. The RHR Pump discharge check valves (151/251 F031A,B,C&D) have local disc position indication on the valve hinge pins.

5) Shutdown lines taking suction from the recirculation system are provided to permit testing of the pump discharge into the reactor pressure vessel after normal plant shutdown and to provide for shutdown cooling.

6) All relief valves are removable for bench testing during plant shutdown.

6.3.2.2.4.1 NPSH AVAILABLE FOR RHR

The available NPSH is calculated in accordance with Regulatory Guide 1.1. The following data is used for a typical NPSH calculation:

a. The suppression pool is at the minimum post-accident level of El. 667.3 feet. 667.3' El. is due to suppression pool draw down assuming the worst case break (Main Steam Line break inside containment).

b. The centerline of the pump suction is at El. 648'-1/2".

c. The suction strainers (total of two strainers for each suction line) are filtering their design debris source term. The maximum pressure loss across the strainers at maximum runout flow of 13,800 gpm is 2.5 psi. The pump vendor required NPSH at runout flow is 5.0 feet.

d. Atmospheric pressure head is assumed to be equal to the vapor pressure, $h_a – h_{vp}$

e. The suppression pool water is assumed to be at 220°F.

f. $NPSH_A = h_s - h_f + h_a - h_{vp}$

h_s = static head
h_f = friction head loss
h_a = atmospheric pressure head
h_{vp} = vapor pressure

Based on Section 6.3.2.2.4.1.d, NPSHA = $h_S - h_f$

with h_s = 19.26 feet
h_f = 11.09 feet

$NPSH_A$ = 19.26 – 11.09
$NPSH_A$ = 8.17 feet
$NPSH_r$ = 5.0 feet

6.3.2.2.5 Discharge Line Fill System

The discharge line fill system, described in this section, serves the ECCS discharge lines (RHR, CS and HPCI) and the RCIC discharge line.

A requirement of the core cooling systems is that cooling water flow to the reactor vessel be initiated rapidly when the system is called on to perform its function. This quick-start system characteristic is provided by quick-opening valves, quick-start pumps, and a standby ac power source. The time lag between the signal to start the pump and the initiation of flow into the RPV is minimized by the ECCS and RCIC discharge line fill system which continuously keeps the core cooling pump discharge lines filled and simultaneously prevents water hammer during the rapid start transient of the ECCS and RCIC pumps.

The discharge line fill system consists of fill lines which provide a continuous supply of condensate from the condensate transfer system to the high points of the ECCS discharge piping. Following initial venting and system fill, a pressure above atmospheric pressure is maintained at the system's high points to prevent air accumulation. A minor, but continuous inflow into the discharge lines is required primarily to make up for leakage across the check or stop check valves provided near the ECCS and RCIC pumps. Past experience has shown that these valves will leak slightly, producing a small backflow. The estimated make-up for the pump discharge lines is less than 1 gpm. To ensure that the discharge lines are always filled, indication is provided in the Control Room as to whether the condensate transfer pumps are operating. An alarm will indicate low condensate transfer pump discharge pressure which can be verified on a pressure indicator in the control room.

A pressure switch is provided to initiate this low pressure alarm. Two pressure switches are provided to initiate auto start of the standby transfer pump. (Refer to FSAR Dwg. M-108, Sh. 1.) These pressure switches are primarily set to protect the condensate transfer pumps from operating at runout conditions. With one pump operation and approaching runout, tripping of the pressure switches will cause the second pump to start and thereby raise the pressure in the pump discharge header. The set point pressure for pump runout protection well exceeds the pump discharge pressure required for maintaining the injection lines pressurized. The fill lines for each system, therefore, are provided with pressure regulators to control the fill pressure to a few psi above atmospheric pressure at the systems high points so that entrapped air can be released through the high point vents during surveillance test. These pressure regulators have been permanently bypassed to allow the maximum available condensate transfer pressure to pass to the pump discharge headers. With the injection lines properly filled, vented, and pressurized, maintaining an adequate pump discharge header pressure will assure that the injection lines will remain filled with water.

A 2" fill line is provided for the discharge line of the HPCI train, the discharge line of the RCIC train, each of the two RHR trains, and each of the two core spray trains. The individual fill lines can be isolated to permit maintenance on the systems and on individual trains of a system without affecting the other train. Details are shown in the HPCI P&ID, Dwg. M-155, Sh. 1; the RHR P&ID, Dwgs. M-151, Sh. 1 and M-151, Sh. 3; the RCIC P&ID, Dwg. M-149, Sh. 1; and the CS P&ID,

Dwg. M-152, Sh. 1. The condensate transfer pumps with associated instrumentation, including the low pressure alarm, are shown on the condensate and refueling water P&ID, Dwg. M-108, Sh. 1.

No level transmitters are provided to detect air bubbles upstream of injection valves.

Air pockets will be prevented by proper venting and filling and by maintaining the discharge lines continuously pressurized such that the pressure at the high points always exceeds atmospheric pressure. This will require a minor but continuous feed flow into the discharge lines to make up for valve leakage.

The presence of small, local air bubbles upstream of the injection valves will not be detrimental to the ECCS during the start transient.

Each RHR train has its own fill line and can be isolated from the other train. If one pump in an RHR train needs to be isolated for maintenance, the discharge line for the other pump will remain filled and pressurized to the isolation valve of that pump, allowing the pump to perform its function.

The condensate transfer pump discharge low pressure alarm instrumentation is tested in accordance with the Technical Requirements Manual. A channel functional test and a channel calibration are required.

A backup keepfill function is provided by the Demineralized Water System via a gravity feed system from an atmospheric tank (Refer to FSAR Section 9.2.9). The passive, backup keepfill capability provided for the ECCS & RCIC Systems assures that the systems are available for non-design basis accident events such as an Appendix R Fire (General Reference: PLA-4945). The tank contains a minimum volume of 2000 gallons. The tank level is monitored on a periodic basis and is refilled as necessary. The minimum volume allows for reasonable operator response time in the event of a loss of the primary keepfill system from the Condensate Transfer System.

Surveillance tests to determine if the discharge lines for the RHR, HPCI, RCIC and CS systems are full are required by the plant Technical Specifications. The tests are performed by momentarily opening the vents at the system's high points to confirm the water fill and flow. No special fill and vent procedures are required prior to surveillance testing of these pumps.

6.3.2.3 Applicable Codes and Classification

The applicable codes and classification of the ECCS are specified in Section 3.2. All piping systems and components (pumps, valves, etc.) for the ECCS comply with applicable codes, addenda, code cases, and errata in effect at the time the equipment is procured. The piping and components of each system of the ECCS within the containment and out to and including the pressure retaining injection valve are Safety Class 1. The remaining piping and components are Safety Class 2, 3, or non-code as indicated in Section 3.2, and as indicated on the individual system P&ID. The equipment and piping of the ECCS are designed to Seismic Category I requirements. This seismic designation applies to all structures and equipments essential to the core cooling function. IEEE codes applicable to the controls and power supplies are specified in Section 7.1.

6.3.2.4 Materials Specifications and Compatibility

Materials specifications and compatibility for the ECCS are presented in Sections 6.1 and 3.2. Nonmetallic materials such as lubricants, seals, packings, paints and primers, insulation, as well as metallic materials, etc., are selected as a result of an engineering review and evaluation for compatibility with other materials in the system and the surroundings with concern for chemical, radiolytic, mechanical and nuclear effects. Materials used are reviewed and evaluated with regard to radiolytic and pyrolytic decomposition and attendant effects on safe operation of the ECCS.

6.3.2.5 System Reliability

A single failure analysis shows that no single failure prevents the starting of the ECCS when required, or the delivery of coolant to the reactor vessel. No individual system of the ECCS is single failure proof with the exception of the ADS, hence it is expected that single failures will disable individual systems of the ECCS. The most severe effects of single failures with respect to loss of equipment occur if the loss of coolant accident occurs in combination with an ECCS pipe break coincident with a loss of off-site power. The consequences of the most severe single failures are shown in Table 6.3-5.

6.3.2.6 Protection Provisions

Protection provisions are included in the design of the ECCS. Protection is afforded against missiles, pipe whip, and flooding. Also accounted for in the design are thermal stresses, loadings from a LOCA, and seismic effects.

The ECCS piping and components located outside the primary containment are protected from internally and externally generated missiles by the reinforced concrete structure of the ECCS pump rooms. The pump rooms layout and protection is covered in Subsection 6.2.3.

The ECCS is protected against the effects of pipe whip, which might result from piping failures up to and including the LOCA. This protection is provided by separation, pipe whip restraints, and energy absorbing materials. These three methods are applied to provide protection against damage to piping and components of the ECCS which otherwise could result in a reduction of ECCS effectiveness to an unacceptable level. See Section 3.6 for criteria on pipe whip.

The component supports which protect against damage from movement and from seismic events are discussed in Subsection 5.4.14. The methods used to provide assurance that thermal stresses do not cause damage to the ECCS are described in Subsection 3.9.3.

The discharge lines from RHR System relief valves PSV-15106 A&B (Unit 1) and PSV-25106 A&B (Unit 2) penetrate primary containment and discharge below the surface of the suppression pool. The corresponding line identification numbers for the discharge from these valves are 10"-HBB-120 (Unit 1) and 10"-HBB-220 (Unit 2). These lines have been designed and installed recognizing the full effect of the dynamic loads resulting from the normal water clearing of the submerged portion of the lines.

6.3.2.7 Provisions for Performance Testing

Periodic system and component testing provisions for the HPCI and Core Spray System are described in Subsection 6.3.2.2 as part of the individual system descriptions. These provisions for ADS are described in Subsection 6.3.4.2.2 and for LPCI in Subsection 6.3.4.2.4.

6.3.2.8 Manual Actions

With one exception, the ECCS is actuated automatically and requires no operator action during the first 20 minutes following the accident. The only operator action assumed in the Section 6.3 ECCS analysis is that a RHR heat exchanger is placed in service within 20 minutes into the accident.

The NPSH requirements of the CS pump and the LPCI (RHR) pump are shown in Figures 6.3-118 and 6.3-119, respectively.

During the long-term cooling period, the operator will take action as specified in Subsection 6.2.2.2 to place the containment cooling system into operation. Placing the containment cooling mode system into operation is the only manual action that the operator needs to accomplish during the course of the LOCA.

The operator has multiple instrumentation available in the control room to assist him in assessing the post-LOCA conditions. This instrumentation provides reactor vessel pressures and water levels, and containment pressure, temperature and radiation levels as well as indicating the operation of the ECCS. ECCS flow indication is the primary parameter available to assess proper operation of the system. Other indications such as position of valves, status of circuit breakers, and essential power bus voltage are also available to assist him in determining system operating status. The electrical and instrumentation complement to the ECCS is discussed in detail in Chapter 7.3. Other available instrumentation is listed in the P&IDs for the individual systems. Much of the monitoring instrumentation available to the operator is discussed in more detail in Chapter 5 and Section 6.2.

6.3.2.9 Position Verification for Manual Valves

Consideration has been given to the possibility that manual valves in the ECCS might be left in the wrong position when an accident occurs. Table 6.3-9 lists all the manually-operated valves in the ECCS (ADS, LPCI, Suppression Pool Cooling, Core Spray, and HPCI) and summarizes the methods for assuring correct valve position. The table lists only those manual valves which are related to the ECCS function of those systems. Thus, the only manual valves in the RHR system which were evaluated are those which comprise part of the LPCI and Suppression Pool Cooling modes. The boundaries of RHR for this purpose include sidestreams and connecting systems out to the first normally-closed remotely operated valve or to two check valves in series.

Many of the manual valves in these systems are vent, drain, or test connection valves which are normally closed and capped or have cam-locks where evaluated. These valves are identified in the "Function" column of Table 6.3-9. Such valves are not critical to the ECCS function; administrative controls, such as pre-startup valve lineup checks, should suffice to reasonably assure that such valves will not degrade ECCS performance.

Certain other valves are physically secured in their normal position to prevent inadvertent mispositioning. Valve manipulations are procedurally/administratively controlled in such a manner as to ensure accurate status control and proper restoration. In other cases, two isolation valves are provided in series to minimize the possibility of inter- or intra- system leakage. Again, such valves are identified in the "Function" column of Table 6.3-9.

Remote position indication of manual valves which are in the main flowpaths of the ECCS (except for makeup gas supply to the ADS valve accumulators) and which will be inaccessible during normal operation is provided in the control room. Proper administrative controls and/or surveillance testing are relied upon to assure the position of the remaining valves.

6.3.3 ECCS PERFORMANCE EVALUATION

ECCS performance is evaluated using approved 10CFR50 Appendix K models for demonstrating conformance to the acceptance criteria of 10CFR50.46. The ECCS performance is evaluated for the entire spectrum of break sizes for postulated loss of coolant accidents. The Section 15 accidents for which ECCS operation is required are:

- 15.2.8 Feedwater piping break
- 15.6.4 Spectrum of BWR steam system failures outside of containment
- 15.6.5 Loss-of-Coolant Accidents.

Section 15.6.5.5 provides radiological consequences of the DBA LOCA and the Main Steam Line Break events.

Evaluations for Susquehanna Units 1 and 2 with ATRIUM-10 and ATRIUM-11 fuel under uprated reactor power conditions have been performed. The results are used in analyses specific to the fuel neutronic design to demonstrate conformance to the acceptance criteria of 10CFR50.46. The analyses assumed an analytical power level of 4031 MWt which supports nominal rated powers as high as 3952 MWt. The analyses were performed using the NRC approved methodologies EXEM BWR LOCA given in Reference 6.3-23 for ATRIUM-10 and AURORA-B LOCA for ATRIUM-11 fuel. References 6.3-25 to 6.3-27 collectively comprise the AURORA-B LOCA Evaluation Model.

6.3.3.1 ECCS Bases for Technical Specifications

The maximum average planar linear heat generation rates calculated in this performance analysis provide the basis for Technical Specifications designed to ensure conformance with the acceptance criteria of 10CFR50.46. Minimum ECCS functional requirements are specified in Subsections 6.3.3.4 and 6.3.3.5, and testing requirements are discussed in Subsection 6.3.4. Limits on minimum suppression pool water level are discussed in Section 6.2.

6.3.3.2 Acceptance Criteria for ECCS Performance

The applicable acceptance criteria, extracted from 10 CFR 50.46, are listed and for each criterion, applicable parts of Subsection 6.3.3 where conformance is demonstrated are indicated.

Criterion 1, Peak Cladding Temperature - "The calculated maximum fuel element cladding temperature shall not exceed 2200°F." Conformance to Criterion 1 is shown in Table 6.3-3B-2 for Unit 1 and Unit 2.

Criterion 2, Maximum Cladding Oxidation - "The calculated total local oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation." Conformance to Criterion 2 is shown in Table 6.3-3B-2 for Unit 1 and Unit 2.

Criterion 3, Maximum Hydrogen Generation - "The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinder surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react." Conformance to Criterion 3 is shown in Table 6.3-3B-2 for Unit 1 and Unit 2.

Criterion 4, Coolable Geometry - "Calculated changes in core geometry shall be such that the core remains amenable to cooling." As described in Reference 6.3-14 conformance to Criterion 4 is demonstrated by conformance to Criteria 1, 2 and 3.

Criterion 5, Long-Term Cooling - "After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core." Conformance to Criterion 5 is demonstrated generically for General Electric BWRs in Reference 6.3-2, Section III.A as modified by Reference 6.3-20. Briefly summarized, the core remains covered to at least the jet pump suction elevation and the uncovered region is cooled by spray cooling and by steam generated in the covered part of the core.

6.3.3.3 Single Failure Considerations

The functional consequences of potential single failures in the ECCS are discussed in Subsection 6.3.2.5. There it is shown that all potential single failures are no more severe than one of the single failures identified in Table 6.3-5.

It is therefore only necessary to consider each of these single failures in the emergency core cooling system performance analyses.

Based on the EXEM BWR LOCA methodology (Reference 6.3-23) used for ATRIUM-10 fuel a large break in the recirculation suction piping with an LPCI single valve failure is the most severe failure.

A single failure in the ADS (one ADS valve) has no effect on large breaks. In calculations performed for the ATRIUM-10 fuel, a single failure in the ADS (one ADS valve fails to open) is explicitly evaluated as a distinct failure. For all other single failure scenarios that were analyzed, it was assumed that one ADS valve does not open.

Based on the AURORA-B LOCA methodology (References 6.3-25 to 6.3-27) used for ATRIUM-11 fuel, a 0.07 ft^2 break in the recirculation discharge piping with a single failure of the backup battery (DC) power is the most severe failure.

6.3.3.4 System Performance During the Accident

In general, the system response to an accident can be described as:

1) receiving an initiation signal,

2) a small lag time (to open all valves and have the pumps up to rated speed), and
3) finally the ECCS flow entering the vessel.

Key ECCS actuation set points and time delays for all the ECC systems are provided in Table 6.3-2B for Unit 1 and Unit 2. The minimization of the delay from the receipt of signal until the ECCS pumps have reached rated speed is limited by the physical constraints on acceleration of the diesel-generators and pumps.

Simplified piping and instrumentation and functional control diagrams for the ECCS are provided in Subsection 6.3.2. The operational sequence of ECCS for the limiting LOCA event is shown in Table 6.3-1B-2 for Unit 1 and Unit 2.

Operator action is not required, except as a monitoring function, during the short term cooling period following the LOCA. During the long-term cooling period, the operator will take action as specified in Subsection 6.2.2.2 to place the containment cooling system into operation. Long-term cooling capabilities are not impaired by a leak from the first isolation valve outside the suppression pool. Suppression pool water leakage can be made up by several methods. Leakage into the ECCS pump room will not flood high enough to communicate with other rooms. By maintaining water levels in the suppression pool, water level will eventually level out in the ECCS room stopping the leak. Making up suppression pool water can be done by either putting additional water directly into the suppression pool or into the vessel (assuming a LOCA).

6.3.3.5 Use of Dual Function Components for ECCS

With the exception of the LPCI system, the systems of the ECCS are designed to accomplish only one function: to cool the reactor core following a loss of reactor coolant. To this extent, components or portions of these systems (except for pressure relief) are not required for operation of other systems which have emergency core cooling functions, or vice versa. Because either the ADS initiating signal or the overpressure signal opens the safety relief valve, no conflict exists.

The LPCI system, however, uses the RHR pumps and some of the RHR valves and piping. When the reactor water level is low, the LPCI system has priority through the valve control logic over the other RHR subsystems for containment cooling or shutdown cooling. Immediately following a LOCA, the RHR system is directed to the LPCI mode. The LPCI system can be used to support long term containment cooling as well as its primary ECCS support function of cooling the reactor core following a loss of reactor coolant, as discussed in section 6.2.2.2 of the FSAR. The effects of this dual use for the LPCI system have been analyzed for ATRIUM-10 fuel (Reference 6.3-14) and ATRIUM-11 fuel (Reference 6.3-24).

6.3.3.6 Limits on ECCS System Parameters

The limits on the ECCS system parameters are discussed in Subsections 6.3.3.1 and 6.3.3.7.1.

Any number of components in any given system may be out of service, up to and including the entire system. The maximum allowable out of service time is a function of the level of redundance and the specified test intervals as discussed in Section15A.

6.3.3.7 ECCS Analyses for LOCA

6.3.3.7.1 LOCA Analysis Procedures and Input Variables

ATRIUM-10 Fuel

The procedures approved for LOCA analysis conformance calculations are described in detail in References 6.3-14 and 6.3-15. These procedures were used in the calculations discussed in this Subsection 6.3.3 for ATRIUM-10 fuel. The EXEM BWR LOCA application methodology (Reference 6.3-23) is used to demonstrate compliance with the first three 10CFR50.46 criteria.

The methodology defines the plant-specific break spectrum using inputs and models as required by 10CRF50 Appendix K.

Three major AREVA computer codes are used to determine the LOCA response for the Susquehanna LOCA analysis. These codes are RODEX2, RELAX and HUXY. Together, these codes evaluate the reactor vessel blowdown response to a pipe rupture, the subsequent core flooding by ECCS, and the fuel cladding heatup. Figure 4.1 of Reference 6.3-14 is a flow diagram of these computer codes, indicating the major code functions and the transfer of major parameters. The purpose of each code is described below.

FUEL PARAMETERS ANALYSIS (RODEX2)

A complete analysis for a given break starts with the specification of fuel rod parameters as determined by RODEX2. RODEX2 is first used to determine the initial stored energy for both the blowdown analysis (RELAX) system and hot channel) and the heatup analysis (HUXY). RODEX2 is also used to calculate fuel parameters such as fuel to cladding gap sizes and heat transfer coefficients for use in HUXY calculations.

BLOWDOWN ANALYSIS (RELAX)

Relax is used to calculate the system thermal hydraulic response during the blowdown phase of the LOCA. The RELAX analysis is performed from the time of the break initiation to the time the ECCS low pressure core spray flow reaches its rated value and the intact recirculation loop isolation valve is fully closed. The RELAX system blowdown calculation provides the system thermal-hydraulic conditions during this time as boundary conditions for the RELAX hot channel model and at the end of this time for use in initialing the refill/reflood analysis.

The RELAX hot channel analysis is performed to analyze the maximum power assembly of the core. The hot channel is assumed to be operating on fuel thermal limits prior to the occurrence of the postulated LOCA. The results from the RELAX hot channel calculation are heat transfer coefficients and fluid conditions that are used for input to the HUXY heatup analysis.

REFILL/REFLOOD ANALYSIS (RELAX)

The RELAX code is also used to analyze the LOCA beginning with the time of both rated core spray flow and intact recirculation loop isolation valve closure. The RELAX code computes the system hydraulic response during the refill/reflood phase of a LOCA. The refill stage is the period when the lower plenum is filling due to ECCS injection. The reflood period is when the core is being reflooded with ECCS water. The principal result of a RELAX calculation is the time when the two-phase fluid reaches the hot node in the core by entrainment of the reflooding fluid, termed as the “time of hot node (or core) reflood.” The RELAX calculations provide HUXY with the time of hot node reflood and the time when the liquid has risen in the bypass to the height of the axial plane of interest (time of bypass reflood)

HEATUP ANALYSIS (HUXY)

The HUXY code is used to perform heatup calculations for the entire LOCA transient and yields peak cladding temperature and local cladding oxidation. The heat generated by metal-water reaction is included in the HUXY analysis. HUXY is used to calculate the thermal response of each fuel rod in one axial plane of the hot rod. The HUXY code implements the clad swelling and rupture models from NUREG-0630 and complies with the NRC criteria stated in 10CFR50 Appendix K for LOCA Evaluation Models.

The significant input variables used by the LOCA codes for analysis of ATRIUM-10 fuel are listed in Table 6.3-2B.

ATRIUM-11 Fuel

The procedures approved for LOCA analysis conformance calculations are described in detail in Reference 6.3-24. These procedures were used in the calculations discussed in this Subsection 6.3.3 for ATRIUM-11 fuel. The AURORA-B LOCA application methodology (References 6.3-25 to 6.3-27) is used to demonstrate compliance with the first three 10CFR50.46 criteria. The methodology defines the plant-specific break spectrum using inputs and models as required by 10CRF50 Appendix K. The AURORA-B LOCA methodology employs two major computer codes to evaluate the system and fuel response during all phases of a LOCA. These are the S-RELAP5 and RODEX4 computer codes. A single S-RELAP5 model is used to calculate the response of the system, hot channels, and hot rods during all phases of the LOCA to determine the PCT and maximum local clad oxidation for each break spectrum or exposure analysis case.

A complete analysis starts with the specification of fuel parameters using RODEX4 (Reference 6.3-27). RODEX4 is used to determine the maximum initial stored energy of the fuel over the exposure range and to provide fuel properties to S-RELAP5 through a data file at specific exposures. The initial stored energy used in S-RELAP5 is increased to be higher than that calculated by RODEX4 for the power, exposure, and fuel design being considered.

6.3.3.7.2 Accident Description

A detailed description of the LOCA calculation is provided in Reference 6.3-14 for the ATRIUM-10 fuel and Reference 6.3-24 for ATRIUM-11 fuel.

The LOCA analyses covered a spectrum of break sizes in the suction and discharge piping of one of the recirculation loops. For the double-ended guillotine (DEG) break, the discharge coefficients, that characterize the rate at which coolant can escape from the break, were varied to span the condition at which the maximum PCT may occur. A discharge coefficient of 1.0 corresponds to full double-ended guillotine break with an area of 7.0 ft^2. For smaller pipe breaks, (longitudinal splits in the piping), areas of the opening were varied from 3.5 ft^2 to 0.2 ft^2. Two average axial power shapes (top peaked and mid peaked) were analyzed. Two single failure conditions were analyzed, battery failure (disabling the HPCI system) and a failure of a single injection valve in the LPCI.

A double ended guillotine break (1.0 DEG) of the recirculation suction line with a single LPCI valve failure using ECCS nominal delay times is the limiting break (highest PCT) for ATRIUM-10. System hot channel and hot node response curves for the ATRIUM-10 analysis are shown in Figures 6.3-201 through 6.3-221.

For this analysis, the discharge valve in the broken recirculation loop is assumed to close when a suction side break occurs. This assumption allows credit to be taken for the LPCI flow in the failed recirculation loop.

Detailed discussion of the limiting LOCA for and system response curves for the ATRIUM-11 analysis are provided in Reference 6.3-24.

A short description of the major events during the limiting design basis accident (DBA) is included below.

At the beginning of the event the pipe break occurs, and offsite power is assumed to be unavailable. The core flow drops rapidly during the first 1 second, then pauses for the next 5 seconds before dropping to essentially zero between 10 and 20 seconds. The pause in the core flow reduction results from pressurization due to closure of MSIV. Loss of offsite power initiates the MSIV closure, which is delayed by 2 seconds. In response to the MSIV position at 85% open, a control rod scram signal is generated. The scram serves to shut down the reactor from full power operation.

During the early portion of the event, the recirculation drive flow in the broken loop ceases almost immediately. The flow in the drive line of the broken loop then reverses causing vessel inventory depletion through the jet pump nozzles and out the break. Also, at the beginning of the event, the intact loop recirculation pump is tripped due to the assumed loss of offsite power and begins to coast down. The remainder of the coastdown (after about 5 seconds) is then governed by natural circulation as the intact loop coastdown progresses, until the pressure in the lower plenum reaches the saturation pressure of the water in the lower plenum.

At around 8 seconds, the water level in the downcomer region outside the core shroud reaches the top of the jet pumps. Once the top of the jet pumps are uncovered, the core flow drops almost immediately to zero. Prior to this time, the core flow had been sustained primarily by natural circulation effects. However, once the top of the jet pumps uncover, the natural circulation flow from the downcomer region drop to zero, and the jet pumps are no longer capable of functioning in their intended manner. The result is an almost complete stagnation of core flow.

Liquid continues to be lost from the downcomer region until the break at the recirculation suction nozzle is uncovered (around 11 seconds). At this time the vessel depressurizes more rapidly as the break flow changes form primarily liquid flow to a liquid steam mixture. Shortly thereafter, the vessel pressure reaches the saturation point of the previously subcooled liquid in the lower plenum. At this time, around 14 seconds, a significant portion of the fluid in the lower plenum flashes (vaporizes) to steam. The lower plenum flashing causes a brief but significant increase in the core flow as the vaporization displaces the liquid, and the volume expansion pushes steam and liquid into the core.

The core flow decrease due to jet pump uncovery causes the liquid mass in the bundles to drop rapidly due to the combination of the lack of core flow and the ensuing vaporization of fluid in the bundles. The lower plenum flashing provides a source of liquid to the core which is vaporized to steam within a few seconds, causing the liquid mass in the bundles to once again drop rapidly. This begins the period of core uncovery, which persists until the ECC systems begin injecting coolant into the reactor vessel.

The following discussion explains the interrelationship between the thermal hydraulic phenomena and the fuel response (primarily peak cladding temperature).

Figure 6.3-221 represents the temperature versus time for the fuel rod where the maximum PCT value is observed.

The first notable occurrence affecting the PCT response occurs near the time of jet pump uncovery. At this time, the heat transfer rates in the hot channel drop dramatically. This causes a heat up in the upper nodes in the bundle as the fuel rods are exposed to a predominantly vapor mixture. Shortly after the jet pump uncovery, the lower plenum flashed. This flashing provides a surge of core flow which replenishes the bundle inventory with a predominately liquid mixture. This causes a short term improvement in heat transfer. As a result, the cladding temperature decreases for the corresponding time.

Once the flashing in the lower plenum subsides and the steam-liquid mixture in the core begins to recede, all the fuel nodes uncover and begin to heat up. This condition continues until the ECC systems begin to inject coolant into the core.

At approximately 50 seconds, core spray injection begins and coolant is sprayed over the top of the core. Some of the core spray flow will penetrate down into the fuel channels, providing spray cooling to the bundles. In some locations, the fluid will be unable to penetrate into the fuel channels due to the high steam upflow. This phenomena, termed counter current flow limitation (CCFL), may cause a build up of accumulated core spray flow in the upper plenum if the injected flow exceeds that which can drain into the core region. A portion of the fluid which accumulates in the upper plenum will drain down into the core bypass region (inside the core but outside the fuel channels). This fluid may also drain into the fuel bundles located on the core periphery, or may drain into the CCFL limited bundles as the rate of steam upflow decreases through these bundles. The fluid which enters the core bypass region will either pass down into the lower plenum region through the leakage paths in the lower fuel support casting or lower tie plate, or may pass through the leakage paths in the channel directly into the lower portion of the fuel bundles.

The availability of coolant from the core spray system provides a direct heat transfer benefit after the end of system blowdown at about 66 seconds. Once rated core spray flow is attained credit for cooling due to core spray is taken in the ATRIUM-10 and ATRIUM-11 analyses. This approach is used since testing has demonstrated the applicability of the spray cooling coefficients provided in Appendix K of 10CFR50 to the ATRIUM-10 and ATRIUM-11 fuel designs.

The LPCI system has a failed valve and is unavailable in the intact recirculation loop. However, the full LPCI flow is available in the broken loop when the discharge valve in that loop is fully closed at 84 seconds, (Figure 6.3-210). Water introduced into the upper plenum from the CS starts passing through the core by way of the passages between the fuel channels and through lower power fuel assemblies to the lower plenum at about 66 seconds. The lower plenum continues to fill providing a water-steam mixture to the core. At about 120 seconds, the cladding temperature reaches a peak when the flow of liquid to the hot node of the core is sufficient to maintain cooling.

6.3.3.7.3 Break Spectrum Calculations

A complete spectrum of postulated break sizes and locations is considered in the evaluation of ECCS performance.

A summary of the results of the break spectrum calculations is shown in Table 6.3-3C. These results are from References 6.3-14 and 6.3-24. Conformance to the acceptance criteria (PCT ≤ 2200°F, local oxidation ≤ 17% and core wide metal-water reaction ≤ 1%) is provided in Table 6.3-3B-2.

The peak clad temperature for the limiting break shown in Table 6.3-3B-2 differs slightly from those shown in Table 6.3-3C. Table 6.3-3B-2 results are from Reference 6.3-15, which is the analysis that establishes the MAPLHGR for the SSES Units. The results in Table 6.3-3C provide a relative comparison for the various conditions assumed for the break spectrum analyses.

For convenience in describing the LOCA phenomena, the break spectrum has been separated into two regions: small breaks and large breaks. The large breaks are those in the area range of 7.0 to 3.5 ft^2, while small breaks are those smaller than 3.5 ft^2.

The small break region provides a slower depressurization of the reactor vessel, delaying the time for low pressure ECC systems to become effective. As a result, failure of high pressure coolant injection (HPCI) is expected to be involved in the most limiting single failure scenario.

The large break region provides a rapid depressurization of the reactor vessel, making the HPCI and ADS systems relatively unimportant in determining the consequences of a LOCA. As a result, a failure involving the low pressure ECC systems is expected to be involved in the most limiting single failure scenario. As discussed in Section 6.3.3.7.4, the most limiting single failure in this region is the LPCI injection valve, disabling LPCI injection to the intact loop.

As demonstrated in Table 6.3-5, plants which incorporate the LPCI modification have a different complement of ECCS components available depending on break location (recirculation discharge or suction piping) and single failure assumptions. Analyses are performed for both locations to determine at which location the limiting DBA occurs.

6.3.3.7.4 Large Recirculation Line Break Calculations

In this region, the vessel depressurizes rapidly and the HPCI has an insignificant effect on the event. Consequently, failure of the core spray or LPCI is more severe.

The highest calculated PCT for large breaks corresponds to a 1.0 DEG guillotine break with the break flow at each of the two ends of the recirculation suction piping unimpeded by the other (Tables 6.3-3B-2 and 6.3-3C). The ATRIUM-10 limiting large break results are shown in Figures 6.3-201 through 6.3-221. The ATRIUM-11 limiting break results are provided in Reference 6.3-24.

6.3.3.7.5 Transition Recirculation Line Break Calculations

The break sizes analyzed for the ATRIUM-10 and ATRIUM-11 fuel are characterized as large and small only.

6.3.3.7.6 Small Recirculation Line Break Calculations

As described in Section 6.3.3.7.2, analyses were performed for two initial flow conditions, two average axial power shapes (top peaked and mid-peaked), and two single failure conditions (single LPCI valve failure and battery failure – disabling the HPCI system). Failures were assumed to occur in either the suction side or the discharge side of the recirculation loop piping.

The ATRIUM-10 analyses showed that the most limiting break for all of these conditions was a large double-ended guillotine break and that small breaks were not limiting regardless of the single failure condition, (Reference 6.3-14). The ATRIUM-11 analyses demonstrate that a small break is the most limiting break (Reference 6.3-24). See Table 6.3-3C for results with different break sizes.

6.3.3.7.7 Calculations for Other Break Locations

General Electric performed analyses for non-recirculation line breaks as part of the original power uprate analysis (References 6.3-8 and 6.3-10). Events evaluated included: core spray line break, feedwater line break, steamline break inside containment, and steamline break outside containment. These analyses clearly demonstrated that these postulated events are significantly less limiting than the postulated recirculation line breaks.

Framatome analyzed the core spray line break at 4031 MWt in References 6.3-14 and 6.3-24 and confirmed that this break remains less limiting than the recirculation line breaks.

Thus, non-recirculation line breaks are not considered to be potentially limiting and are not specifically analyzed.

6.3.3.8 LOCA Analysis Conclusions

Having shown compliance with the applicable acceptance criteria of Subsection 6.3.3.2, it is concluded that the ECCS will perform its function in an acceptable manner and meet all of the 10CFR50.46 acceptance criteria, given operation at or below the maximum average planar linear heat generation rates as specified in the current cycle Core Operating Limit Report (COLR) for each unit (see FSAR Section 16.3, Technical Requirements Manuals).

6.3.4 TESTS AND INSPECTIONS

6.3.4.1 ECCS Performance Tests

All systems of the ECCS were tested for their operational ECCS function during the pre-operational and/or startup test program. Each component was tested for power source, range, direction of rotation, set point, limit switch setting, torque switch setting, etc. Each pump was tested for flow capacity for comparison with vendor data. (This test was also used to verify flow measuring capability.) The flow tests involved the same suction and discharge source; i.e., suppression pool or condensate storage tank.

All logic elements except sensors and relays were tested individually and then as a system to verify complete system response to emergency signals including the ability of valves to revert to the ECCS alignment from other positions.

Finally the entire system was tested for response time and flow capacity taking suction from its normal source and delivering flow into the reactor vessel. This last series of tests was performed with power supplied from both offsite power and onsite emergency power.

See Chapter 14 for a thorough discussion of pre-operational testing for these systems.

6.3.4.2 Reliability Tests and Inspections

The average reliability of a standby (non-operating) safety system is a function of the duration of the interval between periodic functional tests. The factors considered in determining the periodic test interval of the ECCS are: the desired system availability (average reliability), the number of redundant functional system success paths, the failure rates of the individual components in the system, and the schedule of periodic tests (simultaneous versus uniformly staggered versus randomly staggered). For the ECCS the above factors were used to determine safe test intervals utilizing the methods described in Reference 6.3-1.

All of the active components of the HPCI, CS, and LPCI systems are designed so that they may be tested during normal plant operation. Full flow test capability is provided by a test line back to the suction source. The full flow test is used to verify the capacity of each ECCS pump loop while the plant remains undisturbed (except for a slight disturbance during HPCI testing) in the power generation mode. In addition, each individual valve may be tested during normal plant operation. Input jacks are provided such that by opening the injection valve breaker, each ECCS loop can be tested for response time.

All of the active components of the ADS system are also designed so that they may be tested during normal plant operation. Tests performed every 24 months include a logic system functional test of the ADS system and manual operation of the ADS valves. ADS valves and their associated solenoid valves which have been overhauled during a plant outage are tested during the startup following that outage.

Testing of the initiating instrumentation and controls portion of the ECCS is discussed in Subsection 7.3.1. The emergency power system, which supplies electrical power to the ECCS in the event that offsite power is unavailable, is tested as described in Subsection 8.3.1. The frequency of testing is specified in the Technical Specifications. Visual inspections of all the ECCS components located outside the drywell can be made at any time during power operation subject to ALARA concerns. Components inside the drywell can be visually inspected only during periods of access to the drywell. When the reactor vessel is open, the spargers and other internals can be inspected.

6.3.4.2.1 HPCI Testing

The HPCI system can be tested at full flow with condensate storage tank water at any time during plant operation except when the reactor vessel water level is low, or when the condensate level in the condensate storage tank is below the reserve level, or when the valves from the suppression pool to the pump are open. If an initiation signal occurs while the HPCI system is being tested, the system returns automatically to the operating mode except as noted below. The following actions, which prevent the system from automatically returning to the operating mode, can be taken during HPCI system testing:

a) The F006 (injection valve) breaker may be opened to prevent inadvertent RPV injection.

b) The F008 and F011 test line valves may be prevented from automatic closure on a HPCI initiation. This is done to prevent possible system damage that could unnecessarily occur should the valves close during the test.

c) Additionally, some HPCI testing is required to be performed with the flow controller in manual.

For all configurations, manual actions can be taken to realign the system for vessel injection.

A design flow functional test of the HPCI system over the operating pressure and flow range is performed by pumping water from the condensate storage tank and back through the full flow test return line to the condensate storage tank. The HPCI system turbine pump is driven at its rated output by steam from the reactor. The suction valves from the suppression pool and the discharge valves to the feedwater lines remain closed. These two valves are tested separately to ensure their operability.

Inservice testing of pumps and valves in the HPCI system is discussed in Section 3.9.6. The HPCI pump discharge line, as well as its tributary filling line from the condensate transfer system, is normally protected from full reactor pressure by containment isolation valves which are designated as ASME Section XI-IWV category A. These valves will be leak-rate tested in accordance with the Code. Valves HV-E41-1(2)F006, 1(2)-55-038, and B21-1(2)F010B comprise the containment isolation arrangement for the HPCI discharge line. Valve F006 is a normally-closed, motor-operated gate valve which will open automatically only upon a HPCI initiation signal (coincidence of RPV low level and/or high drywell pressure). Valve 1(2)-55-038 is a normally locked-closed manual valve. Valve F010B is a quick-closing, tilting-disk check valve.

In addition, further isolation of the HPCI discharge line from the reactor coolant pressure boundary is provided by check valve 1(2)41818B, which is designated as ASME Category A. Therefore, adequate protection of the low pressure portions of the HPCI discharge line is provided by the aforementioned valves, and leakrate testing of check valves FOO5, 1(2)-55-012, and 1(2)-55-013 is unwarranted. The HPCI test conditions are tabulated on the HPCI process flow diagram, Dwg. M1-E41-4, Sh. 1.

6.3.4.2.2 ADS Testing

The ADS valves are tested every 24 months. This testing includes simulated automatic actuation of the system throughout its emergency operating sequence, but excludes actual valve actuation. Each individual ADS valve is manually actuated.

During plant operation the ADS system can be checked as discussed in Subsection 7.3.1.1a.1.4.

6.3.4.2.3 CS Testing

The CS pump and valves are tested periodically during reactor operation. With the injection valve closed and the return line open to the suppression pool, full flow pump capability is demonstrated. The injection valve and the check valve are tested in a manner similar to that used for the LPCI valves. The system test conditions during reactor shutdown are shown on the CS system process diagram, Dwg. M1-E21-15, Sh. 1. The portion of the CS outside the drywell may be inspected for leaks during tests.

Inservice testing of pumps and valves in the Core Spray system is discussed in Section 3.9.6. The core spray pump discharge lines are protected from full reactor pressure by containment isolation valves which are designated as ASME Category A. These valves will be leak-rate tested in accordance with the Code. Valves HV-E21-1(2) F005 A and B, 1(2) F006 A and B, and 1(2) F037A and B comprise the containment isolation arrangement for the core spray injection lines. Normally-closed, motor-operated valve F005A(B) is interlocked such that it will open automatically upon receipt of a LOCA signal only if reactor pressure is below core spray system design pressure. This design configuration is considered to conform with the criteria set forth in NRC Standard Review plan (NUREG 75-087) Section 6.3, Paragraph III.11.a.

6.3.4.2.4 LPCI Testing

Each LPCI loop can be tested during reactor operation. The test conditions are tabulated in Dwgs. M1-E11-3, Sh. 1 and M1-E11-3, Sh. 2. During plant operation, this test does not inject cold water into the reactor because the injection line check valve is held closed by the recirculation loop pressure, which is higher than the pump pressure, and because of the normally closed injection valve (F015). The injection line portion is verified not to be obstructed whenever the Shutdown Cooling (SDC) Mode of the RHR System is placed in service. Verification during operation in the SDC Mode in lieu of an additional test minimizes thermal stresses.

To test a LPCI pump at rated flow, the test line valve to the suppression pool is required to be open, the pump suction valve from the suppression pool is required to be open (this valve is normally open), and the pumps are started using the remote/manual switches in the control room. Correct operation is determined by observing the instruments in the control room.

If an initiation signal occurs during the test, the RHR system returns to the LPCI mode. The valves in the test bypass lines are closed automatically to assure that the LPCI pump discharge is correctly routed to the recirculation loop.

6.3.5 INSTRUMENTATION REQUIREMENTS

Design details including redundancy and logic of the ECCS instrumentation are discussed in Section 7.3.

All instrumentation required for automatic and manual initiation of the HPCI, CS, LPCI and ADS is discussed in Subsection 7.3.2 and is designed to meet the requirements of IEEE 279 and other applicable regulatory requirements. The HPCI, CS, LPCI and ADS can be manually initiated from the control room.

The HPCI System is automatically initiated on low reactor water level or high drywell pressure. CS and LPCI are automatically initiated on a low reactor water level or high drywell pressure initiation signal and the high drywell pressure initiation must be accomplished by a reactor vessel low pressure permissive signal. (See Table 6.3-2 for specific initiation levels for each system.) The ADS is automatically actuated by sensed variables for reactor vessel low water level and drywell high pressure plus the indication that at least one LPCI pump or both CS pumps in the same loop are operating. The CS and LPCI automatically return from system flow test modes to the emergency core cooling mode of operation following receipt of an automatic initiation signal. HPCI will realign except for the conditions discussed in Section 6.3.4.2.1. The CS and LPCI system injection into the RPV begin when reactor pressure decreases to system discharge shutoff pressure.

HPCI injection begins as soon as the HPCI turbine pump is up to speed and the injection valve is opened since the HPCI is capable of injecting water into the RPV over a pressure range from 150 psig to 1210 psig.

6.3.6 NPSH MARGIN AND VORTEX FORMATION AFTER A PASSIVE FAILURE IN A WATER TIGHT ECCS PUMP ROOM

NPSH calculations for ECCS pumps have shown adequate margin to assure capability of proper pump operation after a pool level drop due to a worst case passive failure in an ECCS water tight pump room. This capability was initially verified during preoperational testing in SSES Units 1 and 2. The tests assumed a passive failure in the ECCS pump room resulting in the lowest pool level with subsequent operation of the ECCS pump with the smallest NPSH margin above NPSH required. Subsequent capability, due to regulatory driven changeout of strainers, was verified by full scale tests of the Unit 1 strainers at the EPRI Evaluation Center. These strainers were tested for vortexing potential and NPSH impact on existing margins and were shown to satisfy all system requirements, including a passive leak in an ECCS pump room. Even though the Unit 2 strainers were not tested, the effect of the changeout on Unit 2 margins is verifiable by hydraulic similitude through system and structure similarity between the two units. ECCS pump data is presented in

Figures 6.3–118 and 6.3–119. Figures 6.3–120 and 6.3–121 are provided for reference information, actual performance is defined by Test Procedures.

The pool level drop has been determined assuming a passive failure in a ECCS water tight pump room with operator action 10 minutes after an alarm in the room indicating high water level. Vortex tests of ECCS pumps taking suction from the suppression pool (or condensate storage tank) were performed at the design basis minimum suppression pool (or condensate storage tank) water level during the Unit 1 preoperational testing. These tests showed no vortex formation, abnormal noise levels or signs of abnormal suction flow. However, as discussed above, new strainers replaced the original and these were tested for vortexing effect on the system by full scale tests at the EPRI Evaluation Center. The results of the tests indicated no visible vortexing. Since these tests were performed in a full scale test environment, only modified to observe vortex formation/air entrainment, similar performance results are expected with the equipment installed in the suppression pool. Also, since the tests were done only for Unit 1 strainers, differences are not expected in Unit 2 system margins due to hydraulic similitude through system and structure similarity between the two units as was the situation for NPSH.

6.3.7 REFERENCES

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6.3-2 "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K," NEDO-20566A, September 1986.

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6.3-10 "SAFER/GESTR-LOCA Analysis Basis Documentation for Susquehanna Steam Electric Station Units 1 and 2," NEDC-32281P, September 1993.

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6.3-14 ”Susquehanna LOCA Break Spectrum Analysis for ATRIUM-10 Fuel and Extended Power Uprate” Framatome ANP EMF-3242(P), Rev. 1, March 2006.

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6.3-17 PL-NF-98-010, Rev. 0, Susquehanna SES Unit 1 Cycle 11 Core Operating Limits Report,” July 1998.

6.3-18 SSES Unit 2 Cycle 9 Core Operating Limits Report, PL-NF-97-005, Rev. 0, April 1997.

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6.3-24 “Susquehanna Units 1 and 2 LOCA Analysis for ATRIUM-11 Fuel,” Framatome ANP-3784P Rev. 0, June 2019.

6.3-25 “AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios,” Framatome ANP-10332P-A Rev. 0, March 2019.

6.3-26 “Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model,” Exxon Nuclear Company XN-NF-82-07(P)(A) Rev. 1, November 1982.

6.3-27 “Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors,” Framatome BAW-10247PA Rev. 0, February 2008.

TABLE 6.3-1B-2
Unit 1 and Unit 2

EVENT TIMES FOR THE HIGHEST PCT CASE FROM THE TLO RECIRCULATION LINE BREAK SPECTRUM ANALYSIS

ATRIUM-10 Fuel		ATRIUM-11 Fuel	
Limiting Event: Large break in the recirculation suction piping with an LPCI single valve failure	Time (sec)	Limiting Event: 0.07 ft^2 break in the recirculation discharge piping with a single failure of the backup battery (DC) power	Time (sec)
Initiate Break	0.0	Initiate Break	0.0
Initiate Scram	2.5	Initiate MSIV closure	2.0
MSIV closed	5.0	Initiate scram (MSIV < 85% open)	2.5
L2 low water level, HPCI signaled	5.7	MSIV fully closed	5.0
L1 low water level, DG signaled	7.7	L2 Low-low liquid level, HPCI signaled	34.5
Jet pump suction uncovers	8.4	L1 Low-low-low liquid level, DG signaled	129.9
Recirc suction uncovers	11.5	DG power to valves	164.3
Lower plenum flashes	14.5	HPCI flow starts	N/A
DG power at ESS bus	32.8	LPCI pumps to rated speed	166.8
LPCI pump starts	36.8	LPCI valve starts to open	372.1
HPCI flow starts	39.8	LPCI flow starts	408.5
CS pump starts	44.3	LPCS pumps to rated speed	170.3
CS valve opens	44.4	LPCS valve starts to open	372.1
Intact loop LPCI valve opens	NA	LPCS flow starts	397.4
Intact loop LPCI flow starts	NA	ADS valve starts to open	250.3
Broken Loop valve opens	44.4	RDIV closure starts	438.3
Broken Loop flow starts	44.5	RDIV	471.3
CS flow starts	47.8	End of blowdown	565.2
Recirc Discharge valve closure starts	51.3	Bypass reflood	None
End of Blowdown	65.5		
Begin rated spray	65.5		
Recirc Discharge valve closure complete	84.3		
Core reflood	118.0		
PCT	118.0		
ADS valve opens	127.7		
Bypass reflood	129.2		

TABLE 6.3-2B
Unit 1 and Unit 2

SIGNIFICANT INPUT PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS

PLANT PARAMETERS:	
Core Thermal Power	4031 MWt (ATRIUM-10 / ATRIUM-11 Fuel)
Vessel Steam Dome Pressure	1054 psia (ATRIUM-10 / ATRIUM-11 Fuel)
Maximum Recirculation Line Break Area (ft^2)	7.0[(1)] (ATRIUM-10 / ATRIUM-11 Fuel)
Recirculation Line Break Area for Small and Intermediate Breaks (ft^2)	3.5 to 0.05 (ATRIUM-10 / ATRIUM-11 Fuel)
FUEL PARAMETERS:	
Fuel Types	ATRIUM-10 / ATRIUM-11
Number of fuel rods	91 (8 are part length) (ATRIUM-10 Fuel) 112 (20 are part length) (ATRIUM-11 Fuel)
Peak Technical Specification Linear Heat Generation Rate (kw/ft)	13.4 (ATRIUM-10 Fuel) 13.6 (ATRIUM-11 Fuel)
A more detailed list of inputs to the model and its source is presented for ATRIUM-10 fuel in References 6.3-14 and 6.3-15, and for ATRIUM-11 in Reference 6.3-24.	

TABLE 6.3-2B
Unit 1 and Unit 2

SIGNIFICANT INPUT PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS

EMERGENCY CORE COOLING SYSTEM PARAMETERS:		
Low Pressure Coolant Injection System		
Vessel Pressure at which flow may commence	psid (vessel to drywell)	270
Minimum Rated Flow at Vessel Pressure	GPM/psid (vessel to drywell)	Fig. 6.3-80C (ATRIUM-10 / ATRIUM-11 Fuel)
Initiating signals: low water level **or** high drywell pressure plus low reactor pressure permissive(6)	 ft. above top of active fuel psig psig	 ≤ 0.06 (ATRIUM-10 / ATRIUM-11 Fuel) ≥ 2.0 400 (ATRIUM-10 Fuel) 380 (ATRIUM-11 Fuel)
Maximum allowable time delay from initiating signal to pumps at rated speed	sec	36.6(2)
Pressure at which injection valve may open	psig (vessel pressure)	400 (ATRIUM-10 Fuel) 380 (ATRIUM-11 Fuel)
Pressure at which recirculation discharge valve signaled to close	psig	200
Maximum allowed recirculation discharge valve closing time	sec	33
Core Spray System		
Vessel pressure at which flow may commence	psid (vessel to drywell)	303 (ATRIUM-10 Fuel) 298 (ATRIUM-11 Fuel)
Minimum rated flow at Vessel Pressure	GPM/Core Spray Loop psid (vessel to drywell)	Fig. 6.3-79C(3)

TABLE 6.3-2B
Unit 1 and Unit 2

SIGNIFICANT INPUT PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS

Initiating signals: low water level **or** high Drywell Pressure plus low reactor pressure permissive[6]	ft. above top of active fuel psig psig	≤ 0.06 (ATRIUM-10 / ATRIUM-11 Fuel) ≥2.0 400 (ATRIUM-10 Fuel) 380 (ATRIUM 11 Fuel)
Maximum allowed (runout) flow	GPM/ Core Spray Loop	6885 (ATRIUM-10 Fuel) 6785 (ATRIUM-11 Fuel) Fig. 6.3-79C
Maximum allowed delay time from initiating signal to pump at rated speed	sec	40.1[4]
Pressure at which injection valve may open	psig (vessel pressure)	400 (ATRIUM-10 Fuel) 380 (ATRIUM-11 Fuel)
High Pressure Coolant Injection		
Vessel pressure at which flow may commence	psia	1225 to 165
Minimum rated flow available at vessel pressure	GPM psid (vessel to pump suction)	4500 1210 to 165 psid (ATRIUM-10 / ATRIUM-11 Fuel)
Initiating Signals: low water level **or** high Drywell Pressure[6]	ft above top of active fuel psig	≤7.65 (ATRIUM-10 / ATRIUM-11 Fuel) ≥ 2.0
Maximum allowed delay time from initiating signal to rated flow available and injection valve wide open	sec	35

TABLE 6.3-2B
Unit 1 and Unit 2

SIGNIFICANT INPUT PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS

Automatic Depressurization System		
Total number of valves installed		6
Number of valves used in analysis		6 (ATRIUM-10 / ATRIUM-11 Fuel)(5)
Minimum Flow Capacity of any 5 valves at vessel pressure	lbm/hr psig (vessel to suppression pool)	4.0×10^6 1125
Initiating Signals: low water level **and** high Drywell Pressure **and** Signal that at least 1 LPCI pump or 1 CS loop (2 pumps per loop) are running (pump discharge pressure)	ft above top of active fuel psig psig (CS) psig (LPCI)	≤ 0.06 (ATRIUM-10 / ATRIUM-11 Fuel) ≥ 2.0 115 to 175 100 to 150
Delay time from all initiating signals completed to the time valves are open	sec	120 (ATRIUM-10 / ATRIUM-11 Fuel)

(1) Calculation of maximum line break area is based on maximum area at the break location. Break flow rate will be limited by the minimum flow area encountered in break path.
(2) Analysis assumes a 24 second LPCI valve opening time.
(3) Accounts for 100 gpm leakage in the piping connection between the vessel nozzle and the shroud.
(4) Analysis assumes a 19 second CS valve opening time.
(5) For calculations in which the single failure of interest is the ADS, only 5 valves are operable.
(6) Calculations for ATRIUM-10 fuel also performed to justify an additional 5 second of delay time. No credit is assumed for the start of HPCI, CS, or LPCI due to high drywell pressure.

TABLE 6.3-3B-2

Unit 1 and UNIT 2

**RESULTS FOR LIMITING TWO LOOP OPERATION
RECIRCULATION LINE BREAK
1.0 DEG PUMP SUCTION SF-LPCI
TOP-PEAKED AXIAL 102% POWER (4031 MWt)
80 Mlbs/HR FLOW**

Peak Cladding Temperature, ºF	**1844ºF**
Local Cladding Oxidation (Max%)	**0.80%**
Total Hydrogen Generated (% of Total Hydrogen Possible)	**<0.2%**

TABLE 6.3-3C

Unit 1 and Unit 2

RECIRCULATION LINE BREAK RESULTS
HIGHEST PCT CASES

Single Failure	Axial Power Shape			
	Mid-Peaked		Top-Peaked	
	Break Size and Location	PCT (ºF)	Break Size and Location	PCT (ºF)
ATRIUM-10 108 Mlb/hr Flow 102% Power (4031 MWt)				
SF-BATT	0.6 DEG pump suction	1648	0.7 ft^2 pump discharge	1706
SF-LPCI	1.0 DEG pump suction	1698	0.8 DEG pump suction	1730
ATRIUM-10 80 Mlb/hr Flow 102% Power (4031 MWt)				
SF-BATT	0.6 DEG pump suction	1671	1.0 DEG pump discharge	1684
SF-LPCI	1.5 $ft.^2$ pump discharge	1728	1.0 DEG pump discharge	1803
ATRIUM-11 108 Mlb/hr Flow 102% Power (4031 MWt)				
SF-BATT	0.06 ft^2 pump suction	1734	0.07 ft^2 pump discharge	1771
SF-LOCA	1.0 DEG pump suction	1588	1.0 DEG pump suction	1639
ATRIUM-11 99 Mlb/hr Flow 102% Power (4031 MWt)				
SF-BATT	0.08 ft^2 pump discharge	1732	0.06 ft^2 pump discharge	1759
SF-LOCA	1.0 DEG pump suction	1578	0.3 ft^2 pump discharge	1631

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TABLE 6.3-5

SINGLE FAILURES AND AVAILABLE SYSTEMS FOR SUSQUEHANNA

Assumed Failure")	Recirculation Suction Break Systems Remaining'''	Recirculation Discharge Break Systems Remaining
Opposite Unit False LOCA Signal	ADS[3], HPCI, 1 CS system (2 pumps), 2 LPCI pumps (one in each loop)	ADS, HPCI, 1 CS system (2 pumps), 1 LPCI pump
Battery[4]	ADS, 1 CS system (2 pumps), 3 LPCI pumps	ADS, 1 CS system (2 pumps), 1 LPCI pump
LPCI Injection Valve	ADS, HPCI, 2 CS systems (4 pumps), 2 LPCI pumps (in one loop)	ADS, HPCI, 2 CS systems (4 pumps)
Diesel-Generator[4]	ADS, HPCI, 1 CS system (2 pumps), 3 LPCI pumps	ADS, HPCI, 1 CS system (2 pumps), 1 LPCI pump
HPCI	ADS, 2 CS systems (4 pumps) 4 LPCI pumps (2 in each loop)	ADS, 2 CS systems (4 pumps), 2 LPCI pumps (in one loop)
ADS[3] (SPC ATRIUM™-10 Fuel)	5 of 6 ADS valves, HPCI, 2 CS systems (4 pumps), 4 LPCI pumps (2 in each loop)	5 of 6 ADS valves; HPCI, 2 CS systems (4 pumps), 2 LPCI pumps (in one loop)

(1) Other postulated failures are not specifically considered because they all result in at least as much ECCS capacity as one of the above assumed failures.

(2) Systems remaining, as identified in this table, are applicable to all non-ECCS line breaks. For a LOCA from an ECCS line break, the systems remaining are those listed for the recirculation suction line break, less the ECCS in which the break is assumed.

(3) For FANP ATRIUM™-10 fuel, a single failure in the ADS is modeled in separate calculations.

(4) One additional CS pump (50% flow) would also be available if DG failure was C or D diesel. Note, no additional CS pump would be available if DG failure was A or B diesel. The analyses assume no core spray cooling or inventory makeup credit for this pump.

TABLE 6.3-8
HPCI SYSTEM DESIGN PARAMETERS
(values in parenthesis are those used for the original design analyses, before power uprate)

HPCI PUMP		
Total Pump Design Flow	5070 GPM	
NPSHR	21 feet	
Developed Head, Maximum	3060 feet @ 1225 psia reactor pressure (2940 feet @ 1172 psia reactor pressure) 525 feet @ 165 psia reactor pressure	
Brake Horsepower	4800 HP @ 3060 feet developed head (not to exceed 4621 HP @ 2940 feet developed head) 1000 HP @ 525 feet developed head	
Design Pressure	1500 psig	
Water Temperature Range	40°F to 140°F	
HPCI TURBINE		
	High Pressure	**Low Pressure**
Reactor Pressure	1225 psia (1172 psia)	165 psia
Steam Inlet Pressure	1210 psia (1157 psia)	150 psia
Turbine Exhaust Pressure, Maximum	25 to 65 psia	25 to 65 psia
Design Inlet Pressure	1250 psig + Saturated Temperature	
Design Exhaust Pressure	200 psig + Saturated Temperature	
HPCI ORIFICE SIZING		
Coolant Loop orifices. Sized to obtain the required flow split between the lube oil cooler and the gland seal condenser.		
Minimum Flow Orifices. Sized to ensure a minimum flow or 500 gpm with the pump minimum flow bypass valve open.		
Test Return Orifice. Sized to simulate pump discharge pressure required when the HPCI System is injecting design glow with the reactor vessel pressure at 165 psia. A second orifice for normal system testing with the reactor vessel at 1015 psia may be provided to avoid a high pressure drop across the pump test return valve.		
Leak-off Orifices. Sized for 1/8 inch diameter minimum, 3/16 inch diameter maximum.		
Steam Exhaust Drain Pot Orifice. Sized for 1/8 inch diameter minimum, 3/16 inch diameter maximum.		

TABLE 6.3-8 HPCI SYSTEM DESIGN PARAMETERS (values in parenthesis are those used for the original design analyses, before power uprate)	
HPCI VALVES	
Steam Supply Valve – Open and/or close against full pressure within 20 seconds.	
Pump Discharge Valves – Open and/or close against full pressure within 20 seconds.	
Pump Minimum Flow Bypass Valve – Open and/or close against full pressure within 10 seconds.	
Steam Supply Isolation Valves – Close against full pressure at a minimum rate of 12 inches per minute.	
Cooling Water Pressure Control Valve – Self-contained downstream sensing control valve capable of maintaining constant downstream pressure of 75 psia.	
Pump Suction Relief Valve – 100 psig Relief Setting, 10 gpm @ 10% Accumulation.	
Cooling Water Relief Valve – Size to prevent over pressurizing piping, valves and equipment in the coolant loop in the event of failure of pressure control valves.	
Pump Test Return Valve – Capable of throttling against 1000 psi differential pressure.	
Relief Valve, Barometric Condenser – Capable of retaining 10 inches of mercury vacuum at 140°F ambient, with a set pressure of 5-7 psig and flow gpm at 10 percent accumulation.	
Turbine Exhaust Isolation Valve – Open and/or close against 50 psi differential pressure at a temperature of 360°F. Physically located at the highest point in the exhaust line on a horizontal run and as close to the containment as practicable.	
Check Valve – Located at the highest point in the line on a horizontal run, with adjacent piping arranged to provide a continuous downward slope from the upstream side of the check valve to the turbine exhaust drain pot and downstream of the check valve to the suppression pool.	
Isolation Valve, Steam Warmup/Drain Line – Open and/or close against differential pressure of 1210 psi with minimum travel of 12 inches per minute. The valve and valve associated equipment shall be capable of proper functional operation during maximum ambient conditions.	
Vacuum Breaker, Isolation Valves – Open and/or close against a differential pressure of 200 psi at a minimum rate of 12 inches per minute.	
Vacuum Breaker, Check Valves – Open with a minimum pressure drop (less than 0.5 psi) across the valve seat.	
Rupture Disk Assemblies – Utilized for turbine casing protection, shall include a mated vacuum support to prevent rupture disc reversing under vacuum conditions.	
Rupture Pressure: 175 psig ± 10 psig	
Flow Capacity: 750,000 lb/hr at 200 psig	
DC MOTOR POWER, MAXIMUM	
Hydraulic Oil Pump Motor:	7.5 Hp
Gland Seal Condenser Vacuum Pump Motor:	1.5 Hp
Gland Seal Condenser Drain Pump Motor:	3.0 Hp

TABLE 6.3-8 HPCI SYSTEM DESIGN PARAMETERS (values in parenthesis are those used for the original design analyses, before power uprate)		
CONDENSATE STORAGE		
135,000 Gallons Total reserve storage, per unit, for both HPCI and RCIC Systems.		
TURBINE EXHAUST VERTICAL REACTION FORCE		
Unbalanced pressure due to discharge under the suppression pool water level, which requires vertical hold down, is 20 psi.		
AMBIENT CONDITIONS		
	Temperature	**Relative Humidity**
Normal Plant Operation	60 to 100°F	95%
Accident Mode	148°F	100%

TABLE 6.3-9

MANUAL VALVES IN THE ECCS

SYSTEM	DWG./ FIGURE	VALVE NUMBER	SIZE/TYPE	FUNCTION	LOCATION	(UNIT 2)	METHOD OF ASSURING CORRECT POSITION
Containment Instrument Gas (ADS Valve Pneumatic Supply)	M-126, Sh.1	126015	1" GB	Maintenance	RB-749		Locked open.
	M-126, Sh.2	126017	1" GB	Maintenance	RB-749		Locked open.
	M-2126 Sh. 1	126020	1" GB	Test Vent	RB-749		If valve is left inadvertently open and the vent uncapped, an abnormal demand will be placed on the affected ADS supply header. Low header pressure will automatically open the supply valve from the high pressure nitrogen storage bottles and isolate the compressor from the header. Loss of nitrogen from the storage bottles will eventually result in actuation of the low pressure alarm in the control room. At that point, plant operating personnel will be alerted to take appropriate action.
		126021	1" GB	Test Conn.	DW-752		Locked closed. Also see 126020.
		126022	1" GB	Header isol.	DW-752		None
		126024G	1" GB	Isolation	DW-752		None
		126024J	1" GB	Isolation	DW-752		None
		126024M	1" GB	Isolation	DW-752		None
		126026	1" GB	Maintenance	RB-719	(749)	Locked open.
		126028	1" GB	Maintenance	RB-719	(749)	Locked open.
		126030	1" GB	Test Vent	RB-719		See 126020
		126031	1" GB	Test conn.	DW-719	(738)	Locked closed. Also see 126020
		126032	1" GB	Header isol.	DW-719	(738)	None
		126034K	1" GB	Isolation	DW-752		None
		126034L	1" GB	Isolation	DW-752		None
		126034N	1" GB	Isolation	DW-752		None
		126063	1" GB	Charging conn.	RB-749		Double isolation. Also see 126196.
		126064	1" GB	Charging conn.	RB-749		Double isolation. Also see 126196.
		126065	1" GB	Charging conn.	RB-719	(749)	Double isolation. Also see 126196.
		126066	1" GB	Charging conn.	RB-719	(749)	Double isolation. Also see 126196.
		126078 thru 090	1" GB	Maintenance	RB-749		None; only one of 26 high pressure nitrogen
		126091 thru 100	1" GB	Maintenance	RB-733	(749)	bottles would be lost if valve inadvertently closed.
		126153	1" GB	Test conn.	DW-719	(738)	See 126020.
		126155	1" GB	Test conn.	DW-752		See 126020.
		126159	1" GB	Maintenance	RB-719	(749)	See 126078.

TABLE 6.3-9

MANUAL VALVES IN THE ECCS

SYSTEM	DWG./ FIGURE	VALVE NUMBER	SIZE/TYPE	FUNCTION	LOCATION	(UNIT 2)	METHOD OF ASSURING CORRECT POSITION
Containment Instrument Gas (ADS Valve Pneumatic Supply) (continued)		126160	1” GB	Maintenance	RB-719	(749)	See 126078.
		126161	1” GB	Maintenance	RB-719	(749)	See 126078.
		126162	1” GB	Test vent	RB-749		See 126020.
		126163	1” GB	Test vent	RB-719	(749)	See 126020.
		126196	1” GB	Charging/Vent	RB-749	(226176)	Double isolation. If valves left inadvertently open, an abnormal demand will be placed on the affected ADS supply header. Loss of nitrogen from the storage bottles will eventually result in actuation of the low pressure alarm in the control room. At that point, plant operating personnel will be alerted to take appropriate action.
		126197	1” GB	Charging/Vent	RB-749	(226177)	See 126196.
		226175	1” GB	Charging/Vent	(U2 only)	RB-749	Administrative controls. (Normally open).
		226176	1” GB	Charging/Vent	(126196)	RB-779	See 126196.
		226177	1” GB	Charging/Vent	(126197)	RB-779	See 126196.
Nuclear Boiler – Automatic Depressurization System (ADS)	M-141, Sh.1	141007G, K,J,L,M & N	1” GB	Drain	DW-752		See 126020.
	M-141, Sh.2	141801, 807, 816, 811, 806, 817	1” GB	Vent	DW-739		None
RHR – Low Pressure Coolant Injection Mode (LPCI) and Suppression Pool Cooling Modes	M-151, Sh.1 M-151, Sh.2 M-151, Sh.3 M-151, Sh.4 M-2151, Sh. 1 M-2151 Sh. 3	151011	1” GB	Drain	RB-645		Administrative controls.
		151012	1” GB	Drain	RB-645		Administrative controls.
		151013	1” GB	Vent	RB-645		Administrative controls.
		151014	1” GB	Vent	RB-645		Administrative controls.
		151015	1” GB	Vent	RB-670		Administrative controls.
		151016	1” GB	Vent	RB-683		Administrative controls.
		151017	1” GB	Test conn.	RB-683		Administrative controls.
		151018	1” GB	Drain	RB-683		Administrative controls.
		151024	1” GB	Test conn.	DW-719		Double isolation.
		151026	1” GB	Test conn.	DW-719		Double isolation.
		151028	1” GB	Drain	RB-683		Administrative controls.
		151029	1” GB	Vent	RB-749		Administrative controls.
		151030	1” GB	Vent	RB-645		Administrative controls.
		151031	1” GB	Vent	RB-645		Administrative controls.
		151032	1” GB	Drain	RB-645		Administrative controls.
		151033	1” GB	Drain	RB-645		Administrative controls.

TABLE 6.3-9

MANUAL VALVES IN THE ECCS

SYSTEM	DWG./ FIGURE	VALVE NUMBER	SIZE/TYPE	FUNCTION	LOCATION	(UNIT 2)	METHOD OF ASSURING CORRECT POSITION
RHR – Low Pressure		151034	1” GB	Vent	RB-670		Administrative controls.
Coolant Injection Mode		151035	1” GB	Vent	RB-683		Administrative controls.
(LPCI) and Suppression		151036	1” GB	Test conn.	RB-683		Administrative controls.
Pool Cooling Modes		151037	1” GB	Drain	RB-683		Administrative controls.
(continued)		151048	1” GB	Drain	RB-645		Administrative controls.
		151049	1” GB	Vent	RB-683		Administrative controls.
		151051	1” GB	Test conn.	DW-719		Double isolation.
		151053	1” GB	Test conn.	DW-719		Double isolation.
		151056	1” GB	Vent	RB-683		Administrative controls.
		151058	1” GB	Drain	RB-683		Administrative controls.
		151065	1” GB	Vent	RB-683		Administrative controls.
		151070	12” GB	Isolation	RB-683		Locked closed; double isolation.
		151075	1” GB	Drain	RB-645		Administrative controls.
		151084	1” GB	Inst. Conn.	RB-749		Locked open.
		151085	1” GB	Inst. Conn.	RB-749		Locked open.
		151086	1” GB	Inst. Conn.	RB-749		Locked open.
		151087	1” GB	Inst. Conn.	RB-749		Locked open.
		151091 thru 92	1” GB	Vent	RB-683	(U1 only)	Administrative controls. Double isolation.
		151097	1” GB	Sample conn.	RB-683	(251101)	Administrative controls. (Normally open)
		151098	1” GB	Sample conn.	RB-683	(251102)	Administrative controls. (Normally open)
		151101	1” GB	Vent	DW-719	(251113)	Administrative controls. Double isolation.
		151102	1” GB	Vent	DW-719	(251114)	Closed capped. Double isolation.
		151103	1” GB	Drain	DW-704		Locked closed. Double isolation.
		151104	1” GB	Drain	DW-704		Closed capped. Double isolation.
		151105	1” GB	Drain	RB-683		Administrative controls. Double isolation.
		151106	1” GB	Drain	RB-683		Closed capped. Double isolation.
		151107	1” GB	Vent	DW-719		Administrative controls. Double isolation
		151108	1” GB	Vent	DW-719		Closed capped. Double isolation.
		151109	1” GB	Drain	DW-704		Locked closed. Double isolation.

TABLE 6.3-9

MANUAL VALVES IN THE ECCS

SYSTEM	DWG./ FIGURE	VALVE NUMBER	SIZE/TYPE	FUNCTION	LOCATION	(UNIT 2)	METHOD OF ASSURING CORRECT POSITION
RHR – Low Pressure Coolant Injection Mode (LPCI) and Suppression Pool Cooling Modes (continued)		151110	1" GB	Drain	DW-704		Closed capped. Double isolation.
		151111	1" GB	Drain	RB-683		Administrative controls. Double isolation.
		151112	1" GB	Drain	RB-683		Closed capped. Double isolation.
		151119A thru D	3/8" BV	Vent	RB-645		Administrative controls.
		151120A thru D	1" BV	Vent	RB-645		Administrative controls.
		151806	1" GB	Test Conn.	RB -683		Closed capped. Double isolation.
		151807	1" GB	Test Conn.	RB-683		Administrative controls. Double isolation.
		151808	1" GB	Test Conn.	RB-683		Administrative controls. Double isolation.
		151809	1" GB	Test Conn.	RB-683		Closed capped. Double isolation.
		151810	1" GB	Drain	RB-683		Administrative controls.
		151815	1" GB	Vent	RB-683		Administrative controls.
		151821	½" BV	Vent	RB-749		Administrative controls.
		1F018A thru 18D	4" GT	Maintenance	RB-645		Locked open.
		1F034A thru 034D	20" GT	Maintenance	RB-645		Locked open.
		1F045A thru 045D	1" GB	Drain	RB-645		Administrative controls.
		1F058A,B	1" GB	Vent & Test	RB-683		Administrative controls. Double isolation
		1F058B	1" GB	Vent & Test	RB-683		Administrative controls. Double isolation
		1F059A	1" GB	Vent & Test	RB-683		Administrative controls. Double isolation
		1F059B	1" GB	Vent & Test	RB-683		Administrative controls. Double isolation
		1F060A	24" GB	Block	DW-719		Locked open, position indication on 1C601.
		1F060B	24" GB	Block	DW-683		Locked open, position indication on 1C601.
		1F071A thru 071D	1" GB	Drain	RB-645		Administrative controls
		1F072A thru 072D	1" GB	Drain	RB-645		Administrative controls
		1F082	2" GB	Isolation	RB-683		Double isolation..
		1F102A thru 102D	1" GB	Vent	RB-645 & RB-670		Administrative controls.
		1F124A, B	1" GB	Drain	RB-645		Administrative controls.
		1F128A, B	1" GB	Drain	RB-670		Administrative controls.
		1F018A thru 018D	4" GT	Min. Flow	RB-645		Administrative controls
		RVPP15102A thru 102D	1" GB	Inst. Conn.	RB-645 & RB-683		Closed, plugged.
		251093	1" GB	Vent	(U2 only)	RB-683	Administrative controls. Double isolation.
		251094	1" GB	Vent	(U2 only)	RB-683	Administrative controls. Double isolation.
		251101	1" GB	Sample conn.	(151097)	RB-683	Administrative controls.
		251102	1" GB	Sample conn.	(151098)	RB-683	Administrative controls.
		251113	1" GB	Vent	(151101)	DW-719	Administrative controls. Double isolation.
		251114	1" GB	Vent	(151102)	DW-719	Closed capped. Double isolation.

TABLE 6.3-9

MANUAL VALVES IN THE ECCS

SYSTEM	DWG./ FIGURE	VALVE NUMBER	SIZE/TYPE	FUNCTION	LOCATION	(UNIT 2)	METHOD OF ASSURING CORRECT POSITION
RHR – Low Pressure Coolant Injection Mode (LPCI) and Suppression Pool Cooling Modes (continued)		151125	1" GB	Isolation	DW-704		Administrative controls.
		151126	1" GB	Isolation	DW-704		Administrative controls.
		151127	1" GB	Isolation	DW-704		Administrative controls.
		151128	1" GB	Isolation	DW-704		Administrative controls.
		151129	1" GB	Isolation	DW-704		Administrative controls.
		151131	1" GB	Drain	DW-704		Administrative controls.
		151132	1" GB	Drain	DW-704		Administrative controls.
Fuel Pool Cooling & Cleanup (crosstie to LPCI)	M-153, Sh. 1	153070A	8" GB	Isolation	RB-799		Double isolation (with 151070).
		153070B	8" GB	Isolation	RB-799		Double isolation (with 151070).
		153808	1" GB	Vent	RB-749		Closed plugged.
Core Spray	M-152, Sh.1	152001 thru 004	1" GB	Drain	RB-645		Administrative controls.
		152006	1" GB	Test conn.	RB-670		Administrative controls.
		152007	1" GB	Test conn.	RB-670		Administrative controls.
		152008	1" GB	Drain	RB-683		Administrative controls.
		152009	1" GB	Drain	RB-683		Administrative controls.
		152010	1" GB	Vent	RB-749		Administrative controls.
		152011	1" GB	Test conn.	DW-752		Double isolation.
		152012	1" GB	Test conn.	DW-752		Double isolation.
		152013	1" GB	Test conn.	DW-752		Double isolation.
		152014	1" GB	Test conn.	DW-752		Double isolation.
		152015	1" GB	Vent	RB-749		Administrative controls
		152016	1" GB	Vent	RB-645		Double isolation.
		152021	16" GT	Isolation	RB-645		Locked closed; double isolation
		152024	1" GB	Drain	RB-645		Administrative controls.
		152025	1" GB	Drain	RB-645		Administrative controls.
		152026	1" GB	Drain	RB-645		Administrative controls.
		152027	1" GB	Drain	RB-645		Administrative controls.
		152028	3" GB	Isolation	RB-670		Double Isolation (in combination with check valve 152005)
		152029	1" GB	Bypass	RB-749		Locked closed.
		152030	1" GB	Bypass	RB-749		Locked closed.
		1F002A	16" GT	Isolation	RB-645		Locked closed; double isolation
		1F002B	16" GT	Isolation	RB-645		Locked closed; double isolation
		1F007A	12" GT	Block	DW-752		Locked open; position indication on 1C601.
		1F007B	12" GT	Block	DW-752		Locked open; position indication on 1C601.
		1F008A	1" GB	Vent	RB-645		Administrative controls.

TABLE 6.3-9

MANUAL VALVES IN THE ECCS

SYSTEM	DWG./ FIGURE	VALVE NUMBER	SIZE/TYPE	FUNCTION	LOCATION	(UNIT 2)	METHOD OF ASSURING CORRECT POSITION
Core Spray (continued)		1F008B	1” GB	Vent	RB-645		Administrative controls.
		1F010A thru 010D	3” GB	Isolation	RB-645		Locked open.
		1F013A	1” GB	Test conn.	RB-749		Double isolation.
		1F013B	1” GB	Test conn.	RB-749		Double isolation.
		1F014A	1” GB	Test conn.	RB-749		Locked closed; double isolation.
		1F014B	1” GB	Test conn.	RB-749		Locked closed; double isolation.
		1F016A thru 016D	1” GB	Vent	RB-645		Locked closed.
		1F017A, B	1” GB	Maintenance	RB-719		If valve is inadvertently left closed, indication of reactor core differential pressure would be abnormally low.
		1F020A thru 020D	2” GB	Bypass Isolation	RB-645		Locked closed.
		1F046A thru 046D	1” GB	Drain	RB-645		Locked closed.
		1F025A, B	1” GB	Fill Sys.	RB-683		Administrative controls.
		1F027A, B	1” GB	Fill Sys.	RB-683		Administrative controls.
		1F028A, B	2” GB	Fill Sys.	RB-683		Locked open.
		RV-PP-15236A thru 236D	1” GB	Instr. Conn.	RB-645		Closed / plugged.
		252031 and 033	1” GB	Vent	(U2 only)	DW-752	Closed / capped; double isolation.
		252032 and 034	1” GB	Vent	(U2 only)	DW-752	Locked closed; double isolation
		252035 and 037	1” GB	Test Conn.	(U2 only)	RB-749	Locked closed /capped; double isolation
		252036 and 038	1” GB	Test Conn.	(U2 only)	RB-749	Closed / capped; double isolation.
High Pressure Coolant Injection (HPCI)	M-155, Sh.1	155001	1” GB	Drain	RB-645		Double isolation.
		155002	1” GB	Drain	RB-645		Double isolation.
		155003	1” GB	Test conn.	RB-645		Double isolation.
		155004	1” GB	Test conn.	RB-645		Double Isolation
		155007	2” GB	Maintenance	RB-645		Administrative controls.
		155008	2” GB	Maintenance	RB-645		Administrative controls.
		155009	1” GB	Vent	RB-645		Administrative controls.
		155010	1” GB	Vent	RB-670		Double isolation.
		155011	1” GB	Vent	RB-670		Double isolation.
		155014	1” GB	Test conn	RB-739		Double isolation.
		155015	1” GB	Test conn.	RB-739		Double isolation.
		155016	1” GB	Drain	RB-645		Double isolation.
		155017	2” GB	Fill sys.	RB-645		Locked open.
		155018	2” GB	Fill sys.	RB-719		Administrative controls.
		155019	1” GB	Drain	RB-645		Double isolation.

TABLE 6.3-9

MANUAL VALVES IN THE ECCS

SYSTEM	DWG./ FIGURE	VALVE NUMBER	SIZE/TYPE	FUNCTION	LOCATION	(UNIT 2)	METHOD OF ASSURING CORRECT POSITION
High Pressure Coolant Injection (HPCI) (continued)	M-155, Sh.1	155020	1” GB	Drain	RB-645		Double isolation.
		155021	1” GB	Maintenance	RB-670		Administrative controls.
		155022	1” GB	Maintenance	RB-670		Administrative controls.
		155023	1” GB	Maintenance	RB-670		Administrative controls.
		155025	1” GB	Vent	RB-670		Administrative controls.
		155026	1” GB	Vent	RB-670		Administrative controls.
		155027	1” GB	Maintenance	RB-670		Administrative controls.
		155028	1” GB	Vent	RB-670		Administrative controls.
		155029	1” GB	Maintenance	RB-670		Administrative controls.
		155030	1” GB	Vent	RB-670		Administrative controls.
		155031	1” GB	Maintenance	RB-670		Administrative controls.
		155032	1” GB	Drain	RB-670		Double isolation.
		155033	1” GB	Maintenance	RB-670		Administrative controls.
		155034	1” GB	Drain	RB-670		Double isolation.
		155035	1” GB	Maintenance	RB-670		Administrative controls.
		155036	1” GB	Drain	RB-645		Double isolation.
		155037	1” GB	Bypass Isolation	RB-645		Administrative controls.
		155038	1” GB	Bypass Isolation	RB-749		Locked closed.
		155039	1” GB	Vent	RB-670		Administrative controls.
		155040	1” GB	Drain	RB-670		Double isolation.
		155800	1” GB	Test conn.	RB-739		Double isolation.
		155801	1” GB	Test conn.	RB-739		Double isolation..
		155802	¼” BV	Test conn.	RB-670	(255052)	Locked closed / plugged
		1F010	16” GT	Isolation	RB-645		Locked open.
		1F013	1” GB	Test conn.	RB-670		Administrative controls.
		1F014	1” GB	Test conn.	RB-683		Double isolation.
		1F015	1” GB	Test conn.	RB-683		Double isolation.
		1F016	1” GB	Test conn.	RB-670		Double isolation.
	M-155,	1F017	1” GB	Test conn.	RB-670		Double isolation.
	Sh.1	1F023A thru 023D	1” GB	Maintenance	RB-683		Administrative controls. If valve is inadvertently left closed, a steam line high dP or “instrument line break signal” may be actuated, resulting in automatic closure of HPCI steam supply valve 1F002 or 1F003.
		1F036	1” GB	Drain	RB-645		Locked open.

TABLE 6.3-9

MANUAL VALVES IN THE ECCS

SYSTEM	DWG./ FIGURE	VALVE NUMBER	SIZE/TYPE	FUNCTION	LOCATION	(UNIT 2)	METHOD OF ASSURING CORRECT POSITION
High Pressure Coolant Injection (HPCI) (continued)	M-155, Sh.1	1F037	1" GB	Drain	RB-645		Locked open.
		1F044	1" GB	Test conn.	RB-645		Administrative controls.
		1F055	1" GB	Test conn.	RB-645		Double isolation.
		1F056	1" GB	Test conn.	RB-645		Double isolation.
		1F064	1" GB	Test conn.	RB-645		Double isolation.
		1F065	1" GB	Test conn.	RB-670		Double isolation.
		1F090	1" GB	Test conn.	RB-670		Administrative controls.
		1F091	1" GB	Test conn.	RB-670		Administrative controls.
		1F092	1" GB	Test conn.	RB-670		Administrative controls.
		255052	¼" BV	Test Conn.	RB-670	(155802)	Locked closed / capped
HPCI Turbine-Pump	M-156, Sh.1	156001	3/4" GB	Leakoff	RB-645		Administrative controls.
		156002	3/4" GB	Leakoff	RB-645		Administrative controls.
		156003	3/4" GB	Leakoff	RB-645		Administrative controls.
		156006	3/4" GB	Leakoff	RB-645		Administrative controls.
		156007	1" GB	Drain	RB-645		Administrative controls.
		156008	1" GB	Drain	RB-645		Double isolation.
		156009	1" GB	Drain	RB-645		Double isolation.
		156010A	1" GB	Vent	RB-645		Double isolation.
		156010B	1" GB	Vent	RB-645		Double isolation.
		156011	1" GB	Drain	RB-645		Administrative controls.
		156018	1" GB	Drain	RB-645		Administrative controls.
		156019	1" GB	Drain	RB-645		Administrative controls.
		15617	1" GB	Vent	RB-645		Administrative controls.
		1F043	1" GB	Test conn.	RB-645		Administrative controls.
		1F058	2" GB	Maintenance	RB-645		Administrative controls.
	M-156 Sh. 1	1F0161A thru O61C	1" GB	Vent	RB-645		Administrative controls.
		1F063A	1" GB	Vent	RB-645		Double Isolation.
		1F063B	1" GB	Vent	RB-645		Double Isolation.

TABLE 6.3-9

MANUAL VALVES IN THE ECCS

SYSTEM	DWG./ FIGURE	VALVE NUMBER	SIZE/TYPE	FUNCTION	LOCATION	(UNIT 2)	METHOD OF ASSURING CORRECT POSITION

LEGEND

TYPE:
GB = Globe Valve
GT = Gate Valve
BV = Ball Valve

LOCATION:
RB = Reactor Bldg.
DW = Drywell

The number represents the floor elevation.

TABLE 6.3-10

SAFETY-RELATED VALVES IN THE DRYWELL SUBJECT TO SPRAY IMPINGEMENT

P&ID(1)	OPR	TYPE	NUMBER
143	AO	HV	F019
143		TEST	F021
143		TEST	F022
143		MAN	F059
151		PSV	F126
141		TEST	F017
141		TEST	F018
141		DRN	F095A
141		DRN	F095B
141		DRN	F096A
141		DRN	F096B
141		ROOT	1RV-PP-14107A1
141		ROOT	2RV-PP-14107A1
141		ROOT	1RV-PP-14107B1
141		ROOT	2RV-PP-14107B1
141		ROOT	1RV-PP-14107A2
141		ROOT	2RV-PP-14107A2
141		ROOT	1RV-PP-14107B2
141		ROOT	2RV-PP-14107B2
141		ROOT	1RV-PP-14107A3
141		ROOT	2RV-PP-14107A3
141		ROOT	1RV-PP-14107B3
141		ROOT	2RV-PP-14107B3
144		TEST	F002
144		TEST	F003
149	AO	HV	F088
155	AO	HV	F100
126		CK	126072
126		CK	126152
126		CK	126154
143		CK	F013A
143		CK	F013B
126		TEST	126021
126		MAN	126022
126		MAN	126024G
126		MAN	126024J
126		MAN	126024M
126		TEST	126031
126		MAN	126032
126		MAN	126034K
126		MAN	126034L
126		MAN	126034N
126		TEST	126047

TABLE 6.3-10

SAFETY-RELATED VALVES IN THE DRYWELL SUBJECT TO SPRAY IMPINGEMENT

P&ID(1)	OPR	TYPE	NUMBER
126		TEST	126073
126		TEST	126075
126		TEST	126153
126		TEST	126155
141		DRN	141006A
141		DRN	141006B
141		DRN	141006C
141		DRN	141006D
141		DRN	141006E
141		DRN	141006F
141		DRN	141006G
141		DRN	141006H
141		DRN	141006J
141		DRN	141006K
141		DRN	141006L
141		DRN	141006M
141		DRN	141006N
141		DRN	141006P
141		DRN	141006R
141		DRN	141006S
141		DRN	141007G
141		DRN	141007J
141		DRN	141007K
141		DRN	141007L
141		DRN	141007M
141		DRN	141007N
141		DRN	141008A
141		DRN	141008B
141		DRN	141008C
141		DRN	141008D
141		DRN	141014
141		DRN	141015
143		VNT	F001A
143		VNT	F001B
143		VNT	F002A
143		VNT	F002B
143		MAN	F014A
143		MAN	F014B
141		VNT	141800
141		VNT	141801
141		VNT	141802
141		VNT	141803
141		VNT	141804

TABLE 6.3-10

SAFETY-RELATED VALVES IN THE DRYWELL SUBJECT TO SPRAY IMPINGEMENT

P&ID(1)	OPR	TYPE	NUMBER
141		VNT	141805
141		VNT	141806
141		VNT	141807
141		VNT	141810
141		VNT	141811
141		VNT	141812
141		VNT	141813
141		VNT	141814
141		VNT	141815
141		VNT	141816
141		VNT	141817
141	MO	HV	F001
141	MO	HV	F002
141	MO	HV	F005
141		ISO	141018
143		VNT	F025A
143		VNT	F025B
143		VNT	F026A
143		VNT	F026B
143		DRN	F027A
143		DRN	F028A (Unit 1 Only)
143		VNT	F034A
143		VNT	F034B (Unit 1 Only)
143		VNT	F035A
143		VNT	F035B (Unit 1 Only)
143		DRN	F036A
143		DRN	F036B
143		DRN	F038A
143		DRN	F038B
143		TEST	143027A
143		TEST	140327B
143		TEST	143028A
143		TEST	143028B
143		VNT	143030A
143		VNT	143030B
143		VNT	143029A
143		VNT	143029B
143		VNT	143015A
143		VNT	143015B
143		VNT	143016A
143		VNT	143016B
143		VNT	143017A
143		VNT	143017B

TABLE 6.3-10

SAFETY-RELATED VALVES IN THE DRYWELL SUBJECT TO SPRAY IMPINGEMENT

P&ID(1)	OPR	TYPE	NUMBER
143		VNT	143018A
143		VNT	140318B
143		VNT	143019A
143		VNT	143019B
143		VNT	143020A
143		VNT	143020B
143		VNT	143021A
143		VNT	143021B
143		VNT	143022A
143		VNT	143022B
143		VNT	143023A
143		VNT	143023B
143		VNT	143024A
143		VNT	143024B
143		VNT	143025A
143		VNT	143025B
143		VNT	143026A
143		VNT	143026B
144		DRN	144002A
144		DRN	144002B
151		TEST	F062
151		TEST	F061
151		DRN	151053
151		DRN	151051
151	AO	HV	F122A
151	AO	HV	F122B
151		TEST	F063
151		TEST	F064
151		VNT	151062
151		VNT	151061
151		TEST	151026
151		TEST	151024
151		VNT	151093 (Unit 1 Only)
151		VNT	151094 (Unit 1 Only)
151		DRN	151095
151		DRN	151096
2151		VNT	251097 (Unit 2 Only)
2151		VNT	251098 (Unit 2 Only)
2151		VNT	251099 (Unit 2 Only)
2151		VNT	251100 (Unit 2 Only)
152		TEST	152011
152		TEST	152012
152		TEST	152013

TABLE 6.3-10

SAFETY-RELATED VALVES IN THE DRYWELL SUBJECT TO SPRAY IMPINGEMENT

P&ID[(1)]	OPR	TYPE	NUMBER
152		TEST	152014
2152		VNT	252031 (Unit 2 Only)
2152		VNT	252032 (Unit 2 Only)
2152		VNT	252033 (Unit 2 Only)
2152		VNT	252034 (Unit 2 Only)
152	AO	HV	F037A
152	AO	HV	F037B
141		CK	F024A
141		CK	F024B
141		CK	F024C
141		CK	F024D
141		CK	F036A
141		CK	F036B
141		CK	F036C
141		CK	F036D
141		CK	F036E
141		CK	F036F
141		CK	F036G
141		CK	F036H
141		CK	F036J
141		CK	F036K
141		CK	F036L
141		CK	F036M
141		CK	F036N
141		CK	F036P
141		CK	F036R
141		CK	F036S
141		CK	F040G
141		CK	F040J
141		CK	F040K
141		CK	F040L
141		CK	F040M
141		CK	F040N
148		CK	F007
148		MAN	F008
144		MAN	F103
2155		VNT	255040 (Unit 2 Only)
2155		VNT	255041 (Unit 2 Only)
126	MO	HV	12603
143		DRN	F051A
143		DRN	F051B
144		DRN	F029
144		DRN	F030

TABLE 6.3-10

SAFETY-RELATED VALVES IN THE DRYWELL SUBJECT TO SPRAY IMPINGEMENT

P&ID(1)	OPR	TYPE	NUMBER
141		MAN	141016
141	MO	HV	F016
187	AO	HV	18792B2
187	AO	HV	18792B1
187	AO	HV	18792A2
187	AO	HV	18792A1
126		CK	126074
144	MO	HV	F106
144	MO	HV	F100
144	MO	HV	F101
149	MO	HV	F007
143	MO	HV	F032A
143	MO	HV	F032B
113	MO	HV	11345
113	MO	HV	11346
141		PSV	F037A
141		PSV	F037B
141		PSV	F037C
141		PSV	F037D
141		PSV	F037E
141		PSV	F037F
141		PSV	F037G
141		PSV	F037H
141		PSV	F037J
141		PSV	F037K
141		PSV	F037L
141		PSV	F037M
141		PSV	F037N
141		PSV	F037P
141		PSV	F037R
141		PSV	F037S
141		PSV	14137A
141		PSV	14137B
141		PSV	14137C
141		PSV	14137D
141		PSV	14137E
141		PSV	14137F
141		PSV	14137G
141		PSV	14137H
141		PSV	14137J
141		PSV	14137K
141		PSV	14137L
141		PSV	14137M

TABLE 6.3-10

SAFETY-RELATED VALVES IN THE DRYWELL SUBJECT TO SPRAY IMPINGEMENT

P&ID(1)	OPR	TYPE	NUMBER
141		PSV	14137N
141		PSV	14137P
141		PSV	14137R
141		PSV	14137S
151	MO	HV	F022
151		CK	F019
144	MO	HV	F102
144	MO	HV	F001
187	AO	HV	18782A2
187	AO	HV	18782A1
187	AO	HV	18782B2
187	AO	HV	18782B1
141		PSV	F013A
141		PSV	F013G
141		PSV	F013E
141		PSV	F013C
141		PSV	F013J
141		PSV	F013P
141		PSV	F013M
141		PSV	F013S
141		PSV	F013L
141		PSV	F013B
141		PSV	F013R
141		PSV	F013H
141		PSV	F013K
187		VNT	187828
187		DRN	187827
187		VNT	187830 (Unit 1 Only)
187		VNT	187817 (Unit 1 Only)
187		VNT	187816 (Unit 1 Only)
187		VNT	187826
187		VNT	187831 (Unit 1 Only)
187		VNT	187809 (Unit 1 Only)
187		VNT	187808 (Unit 1 Only)
2187		DRN	287824 (Unit 2 Only)
2187		VNT	287825 (Unit 2 Only)
2187		VNT	287823 (Unit 2 Only)
2187		VNT	287822 (Unit 2 Only)
2187		DRN – Unit 1 VNT – Unit 2	187829 287829
2187		VNT	287842 (Unit 2 Only)
2187		VNT	287843 (Unit 2 Only)
157#		PSV	15704A1, B1, C1, D1, E1

TABLE 6.3-10

SAFETY-RELATED VALVES IN THE DRYWELL SUBJECT TO SPRAY IMPINGEMENT

P&ID(1)	OPR	TYPE	NUMBER
157#		PSV	15704A2, B2, C2, D2, E2
141		PSV	F013F
141		PSV	F013D
141		PSV	F013N
155	MO	HV	F002
152	AO	CK	F006B
152	AO	CK	F006A
152		MAN	F007B
152		MAN	F007A
151	MO	HV	F009
151		MAN	F067
151	AO	CK	F050A
151	AO	CK	F050B
151		MAN	F060A
151		MAN	F060B
141		CK	F010A
141		CK	F010B
141	MO	HV	F011A
141	MO	HV	F011B
141	AO	HV	F022A
141	AO	HV	F022B
141	AO	HV	F022C
141	AO	HV	F022D
143	MO	HV	F023A
143	MO	HV	F023B
143	MO	HV	F031A
143	MO	HV	F031B
141		VNT	141022A
141		VNT	141022B
141		VNT	141023A
141		VNT	141023B
151		VNT	151101 (Unit 1 Only)
151		VNT	151102 (Unit 1 Only)
151		DRN	151103
151		DRN	151104
151		VNT	151107
151		VNT	151108
151		DRN	151109
151		DRN	151110
2151		VNT	251113 (Unit 2 Only)
2151		VNT	251114 (Unit 2 Only)
148		TEST	148003
148		TEST	148004

TABLE 6.3-10

SAFETY-RELATED VALVES IN THE DRYWELL SUBJECT TO SPRAY IMPINGEMENT

P&ID(1)	OPR	TYPE	NUMBER
151		ISO	151125
151		ISO	151126
151		ISO	151127
151		ISO	151128
151		ISO	151129
151		CK	151130
151		DRN	151131
151		DRN	151132

GENERAL NOTES:

1. See Table 1.8-4 for a cross-reference between P&ID and FSAR figures.
2. All above listed valves are subject to spray impingement.
3. Unit 2 valves are listed only if they are in addition to the corresponding Unit 1 valves.
4. # These items are located in the wetwell, but are subject to containment spray.
5. Tables do not include safety-related solenoid valves and associated air line valves to air operators since they do not perform any function required for safe shutdown.

FIGURE 6.3-1A REPLACED BY DWG. M-155, SH. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
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FIGURE 6.3-1A REPLACED BY DWG. M-155, SH. 1

FIGURE 6.3-1A, Rev. 56

AutoCAD Figure 6_3_1A.doc

FIGURE 6.3-1B REPLACED BY DWG. M-156, SH. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

FIGURE 6.3-1B REPLACED BY DWG. M-156, SH. 1

FIGURE 6.3-1B, Rev. 55

AutoCAD Figure 6_3_1B.doc

FIGURE 6.3-2 REPLACED BY DWG. M1-E41-4, SH. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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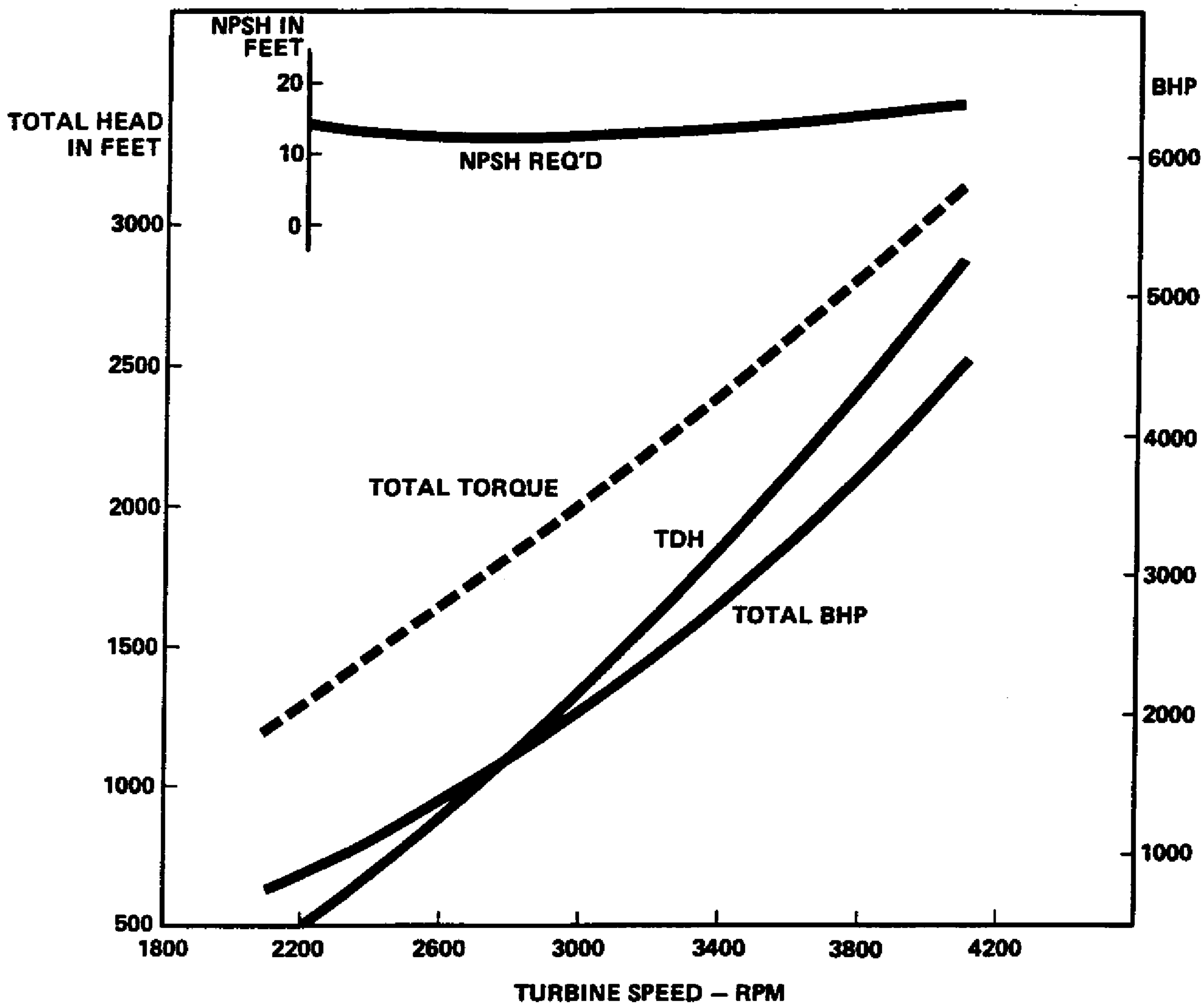
FINAL SAFETY ANALYSIS REPORT

FIGURE 6.3-2 REPLACED BY DWG. M1-E41-4, SH. 1

FIGURE 6.3-2, Rev. 52

AutoCAD Figure 6_3_2.doc



CONSTANT 5000 GPM WITH 100 GPM BOOSTER BY-PASS

Information Used for the Original, Pre-upate Design Analyses

FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
HIGH PRESSURE COOLANT INJECTION PUMP SPEED CHARACTERISTICS FIGURE 6.3-3A, Rev 50

AutoCAD: Figure Fsar 6_3_3A.dwg

FIGURE 6.3-4 REPLACED BY DWG. M-152, SH. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

FIGURE 6.3-4 REPLACED BY DWG. M-152, SH. 1

FIGURE 6.3-4, Rev. 55

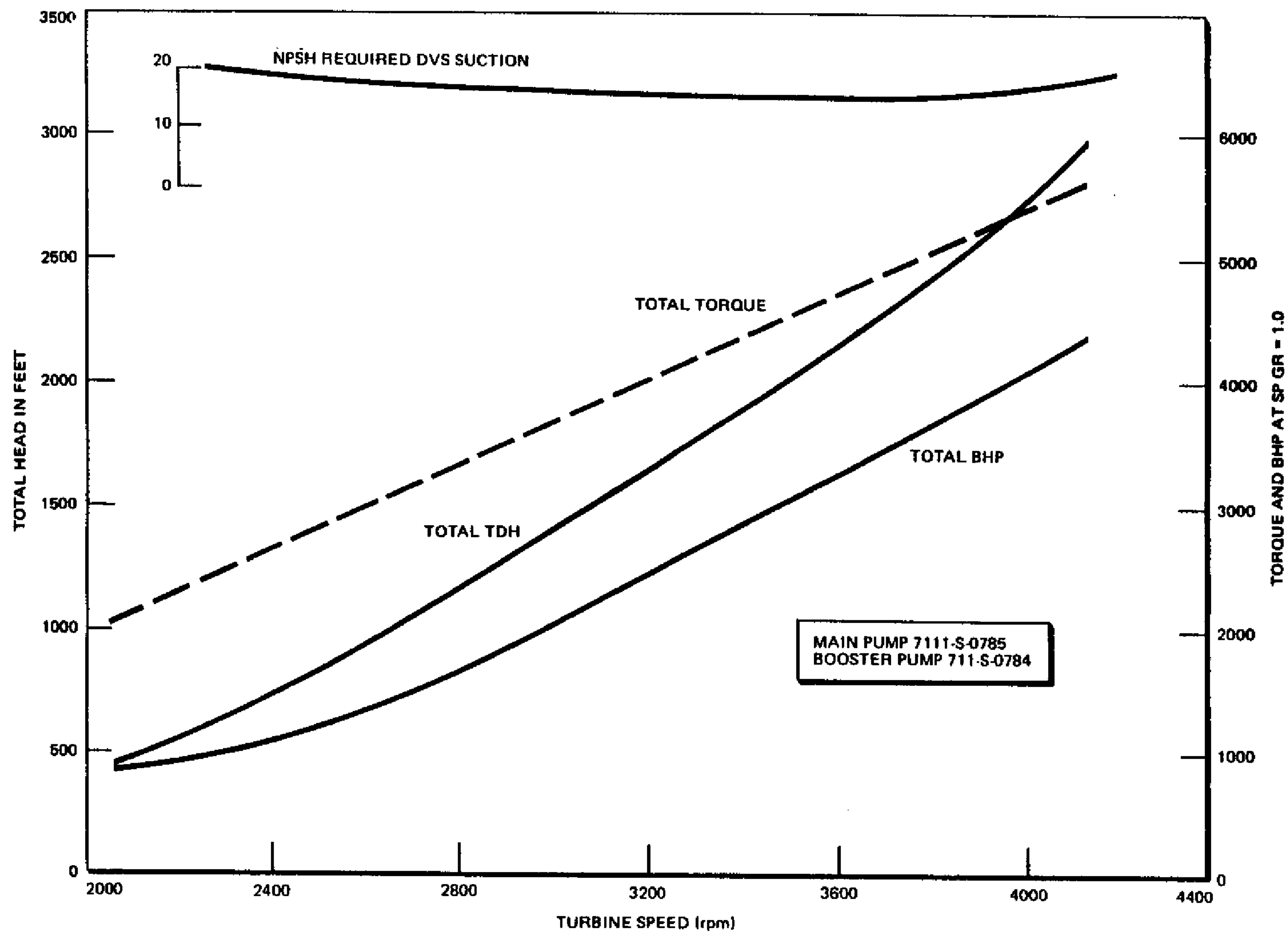
AutoCAD Figure 6_3_4.doc

FIGURE 6.3-5 REPLACED BY DWG. M1-E21-15, SH. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
FIGURE 6.3-5 REPLACED BY DWG. M1-E21-15, SH. 1 FIGURE 6.3-5, Rev. 51

AutoCAD Figure 6_3_5.doc



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PUMP CHARACTERISTIC CURVES
FOR HPCI PUMPS

FIGURE 6.3-77, Rev 49

AutoCAD: Figure Fsar 6_3_77.dwg

FIGURE RENUMBERED FROM 6.3-79 TO 6.3-79A

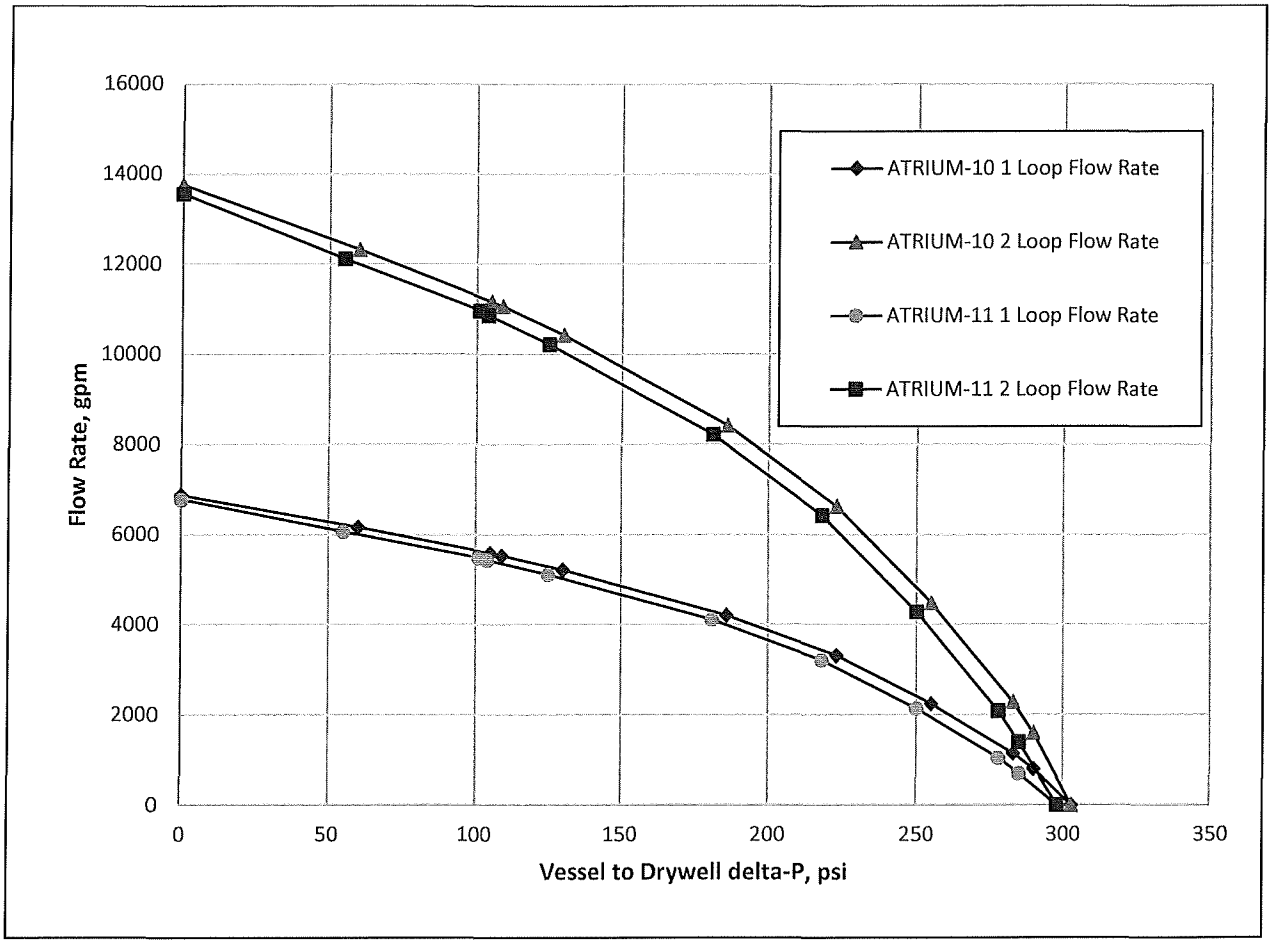
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

FIGURE RENUMBERED FROM 6.3-79 TO 6.3-79A

FIGURE 6.3-79, Rev. 54

AutoCAD Figure 6_3_79.doc

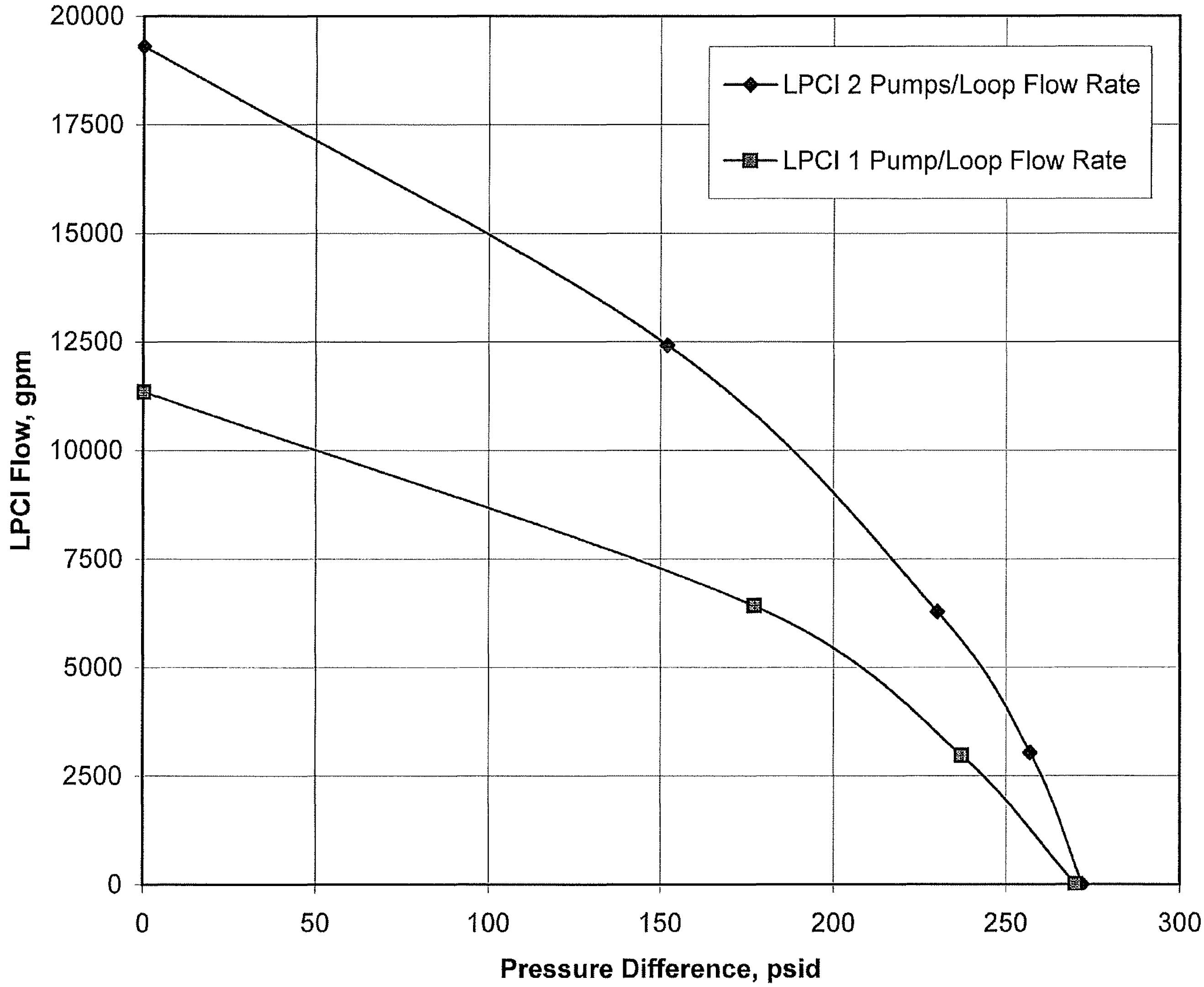


FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LOW PRESSURE CORE SPRAY FLOW
VS. HEAD USED IN LOCA ANALYSIS

Figure 6.3-79C, Rev 3



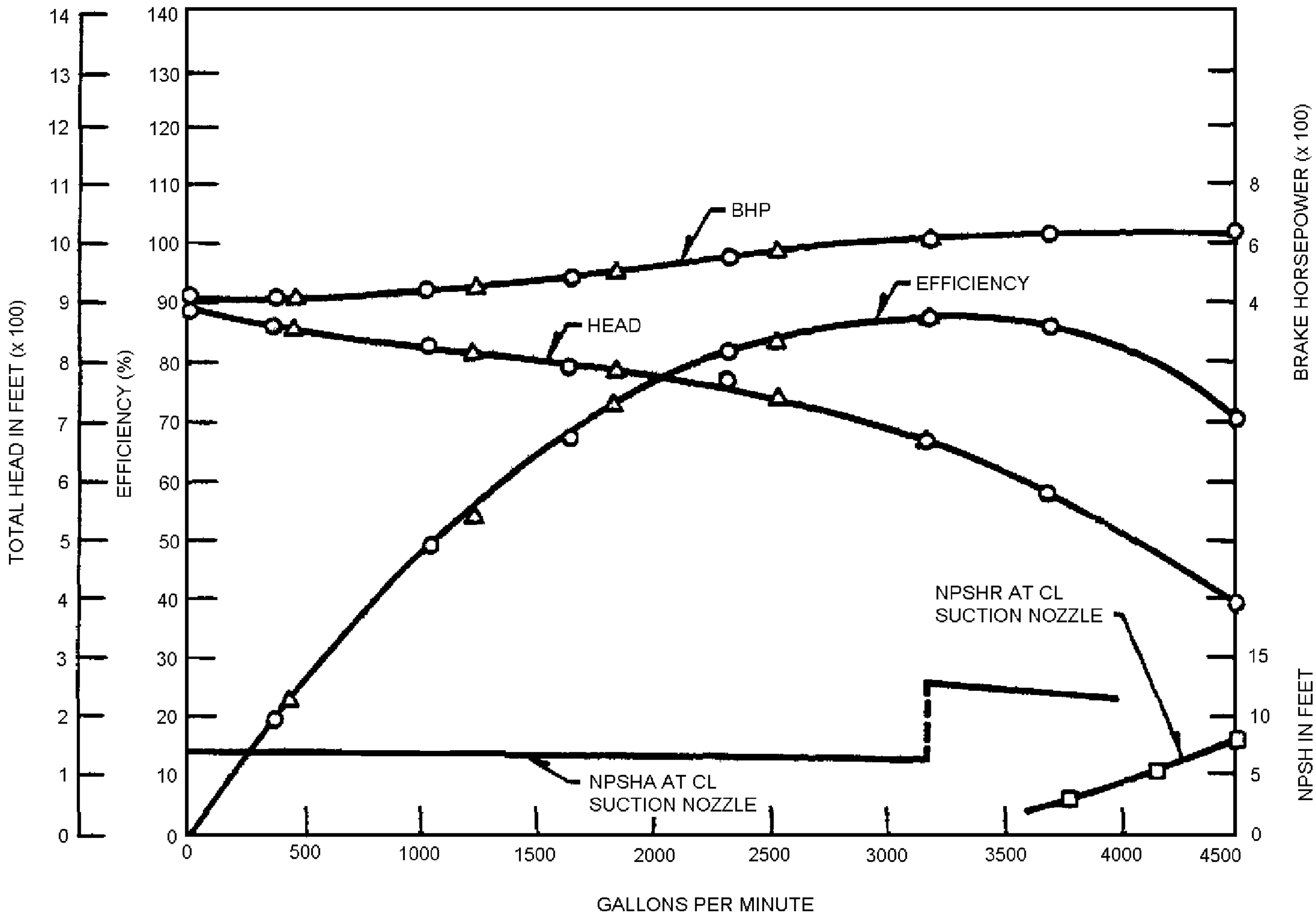
FSAR REV.70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RHR (LPCI) FLOW VS. HEAD CHARACTERISTICS USE IN LOCA ANALYSIS

FIGURE 6.3-80C, Rev 3

AutoCAD: Figure Fsar 6_3_80C.dwg



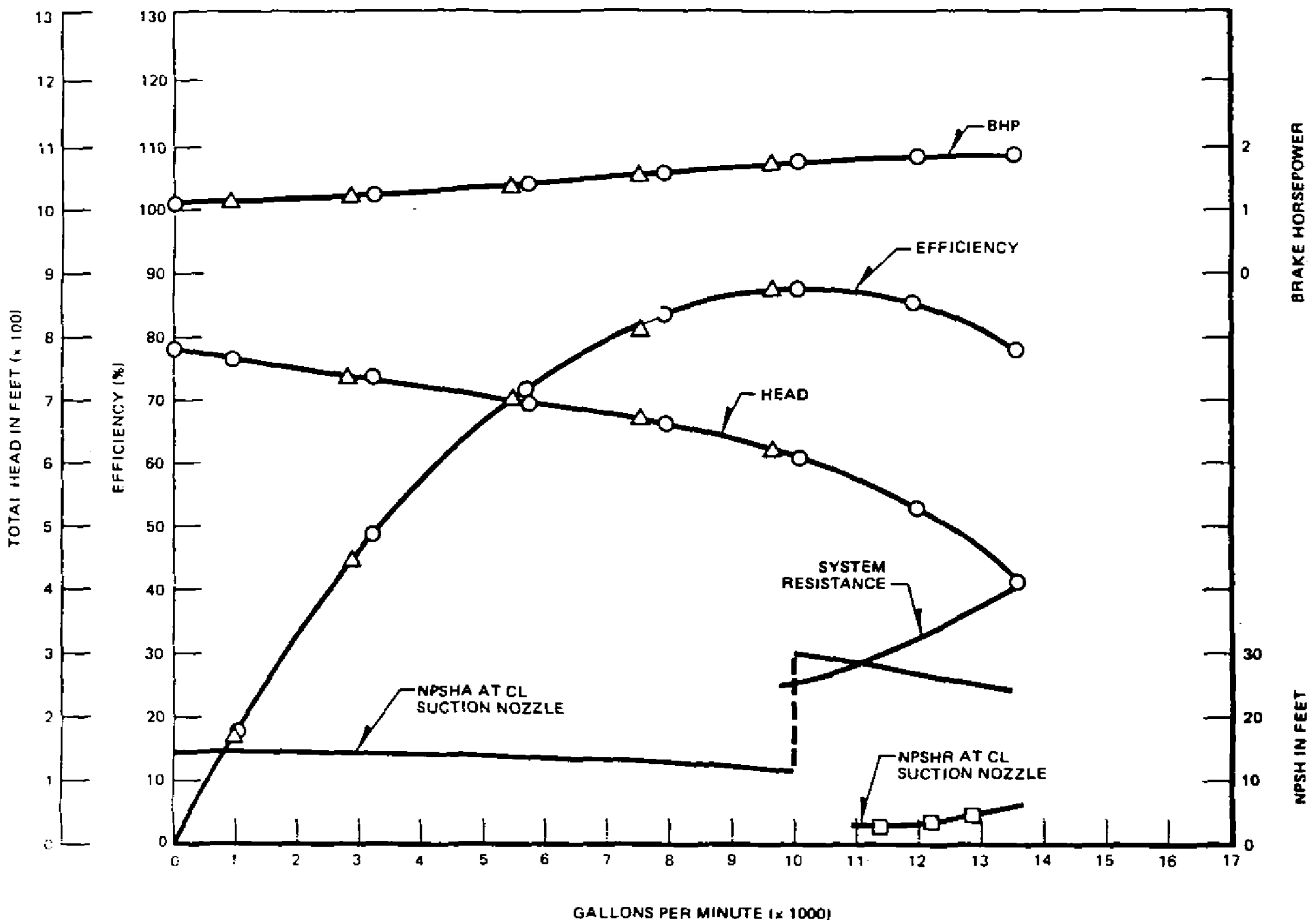
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

CHARACTERISTIC CURVES FOR
CORE SPRAY PUMP

FIGURE 6.3-118, Rev. 52

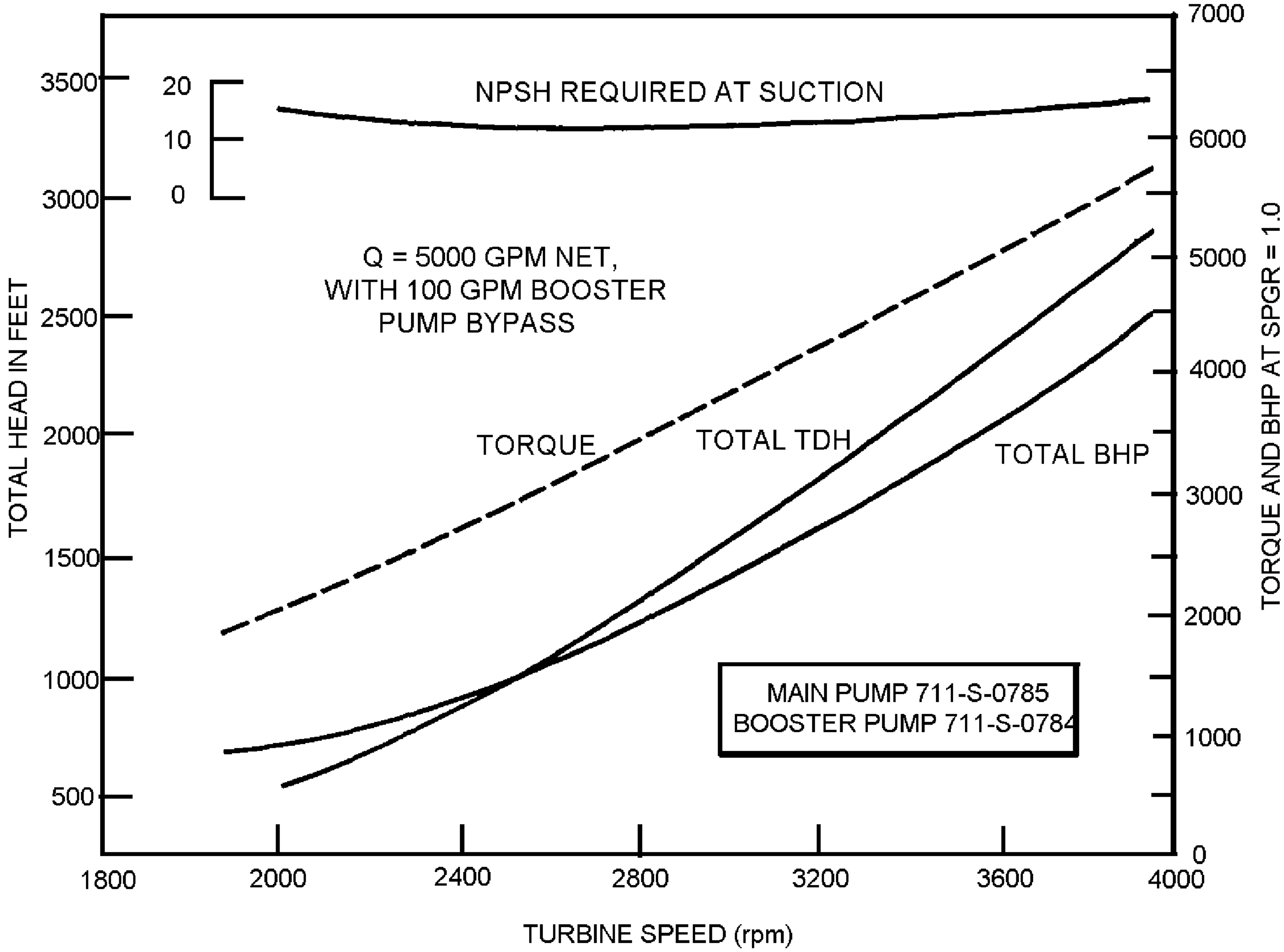
Auto Cad: Figure Fsar 6_3_118.dwg



FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2 FINAL SAFETY ANALYSIS REPORT
CHARACTERISTIC CURVES FOR LPCI PUMP FIGURE 6.3-119, Rev. 51

Auto Cad: Figure Fsar 6_3_119.dwg



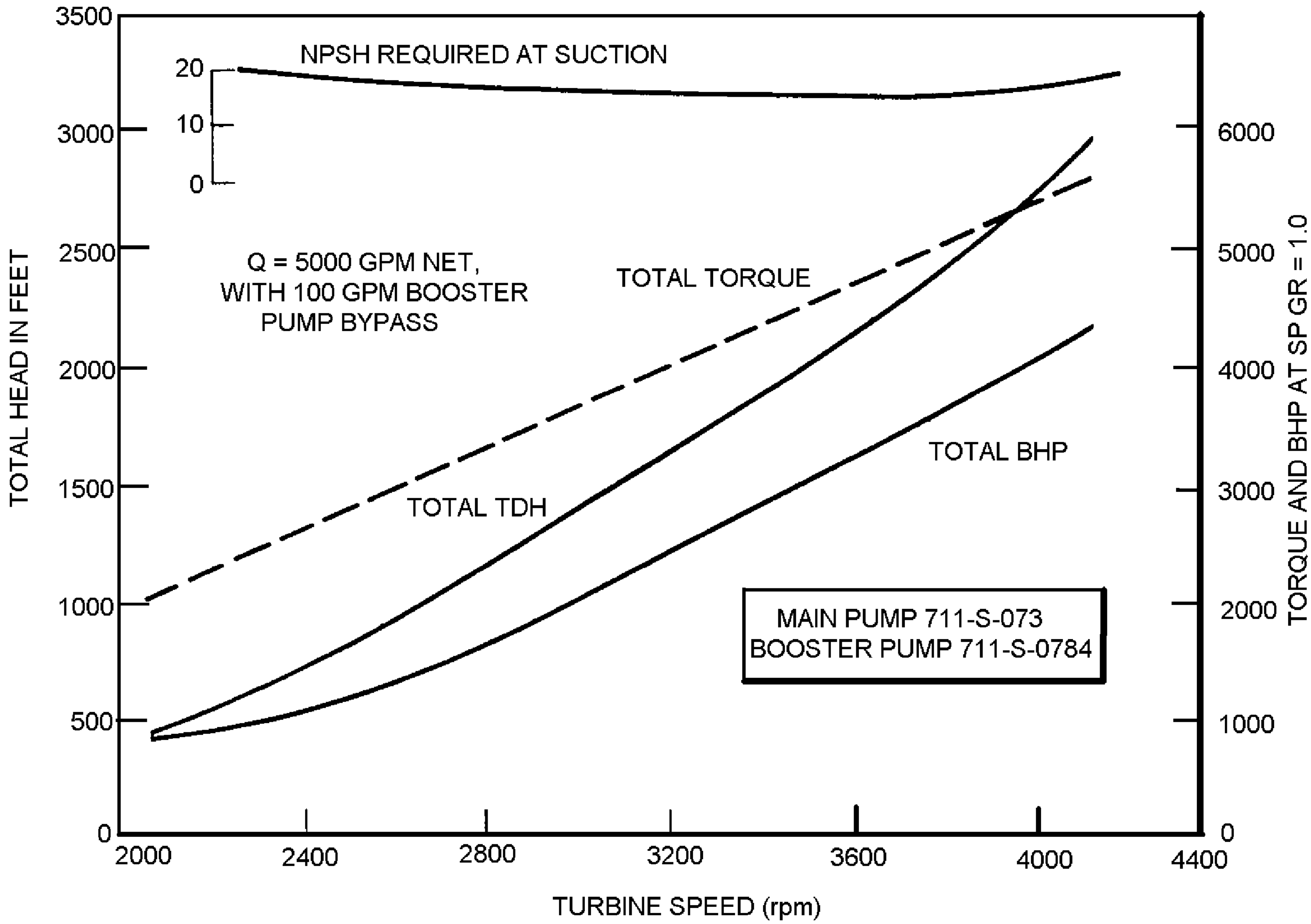
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

SPEED CHARACTERISTIC CURVES FOR
HPCI PUMPS - UNIT 1

FIGURE 6.3-120, Rev. 54

Auto Cad: Figure Fsar 6_3_120.dwg



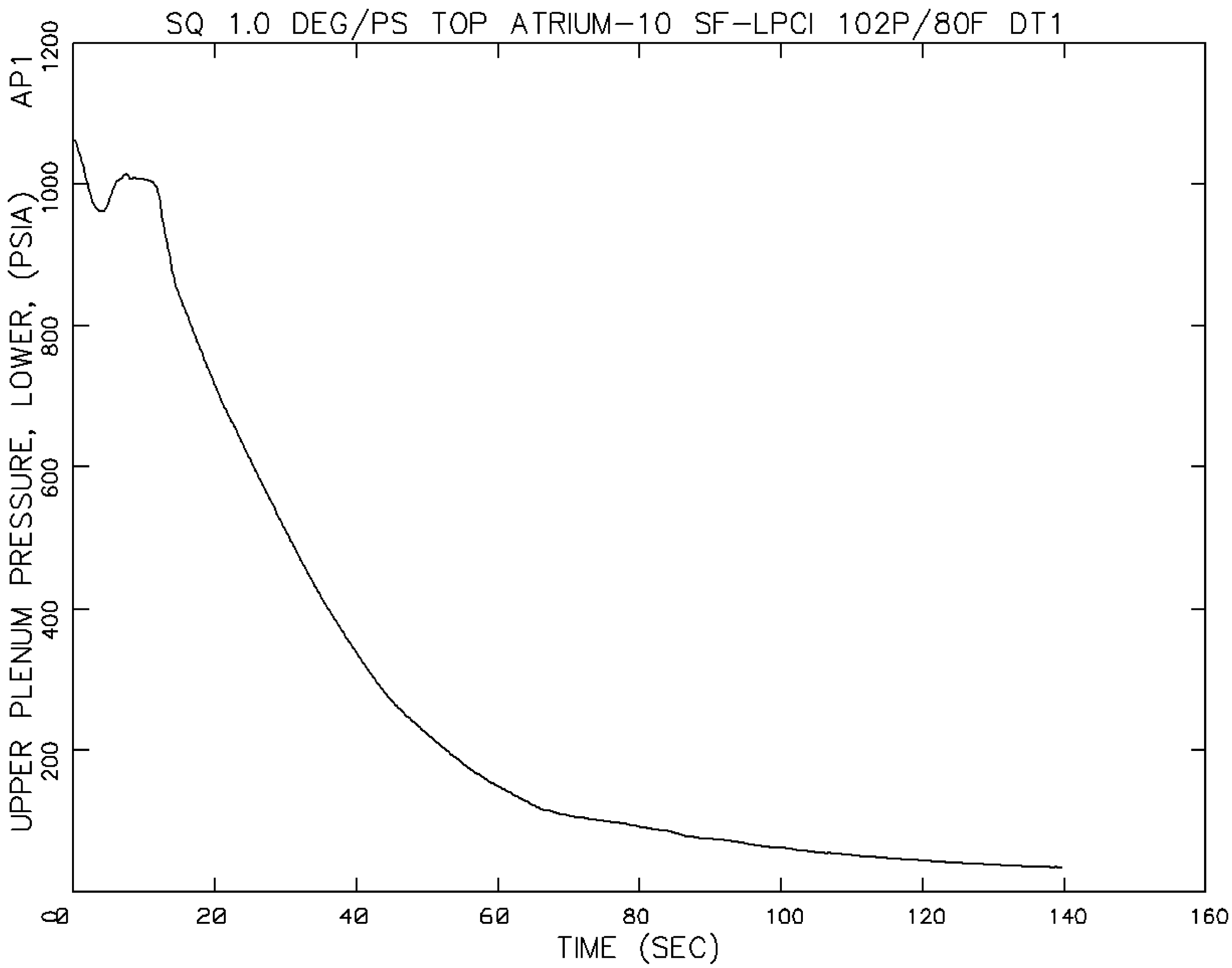
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

SPEED CHARACTERISTIC CURVES FOR
HPCI PUMPS - UNIT 2

FIGURE 6.3-121, Rev. 54

Auto Cad: Figure Fsar 6_3_121.dwg

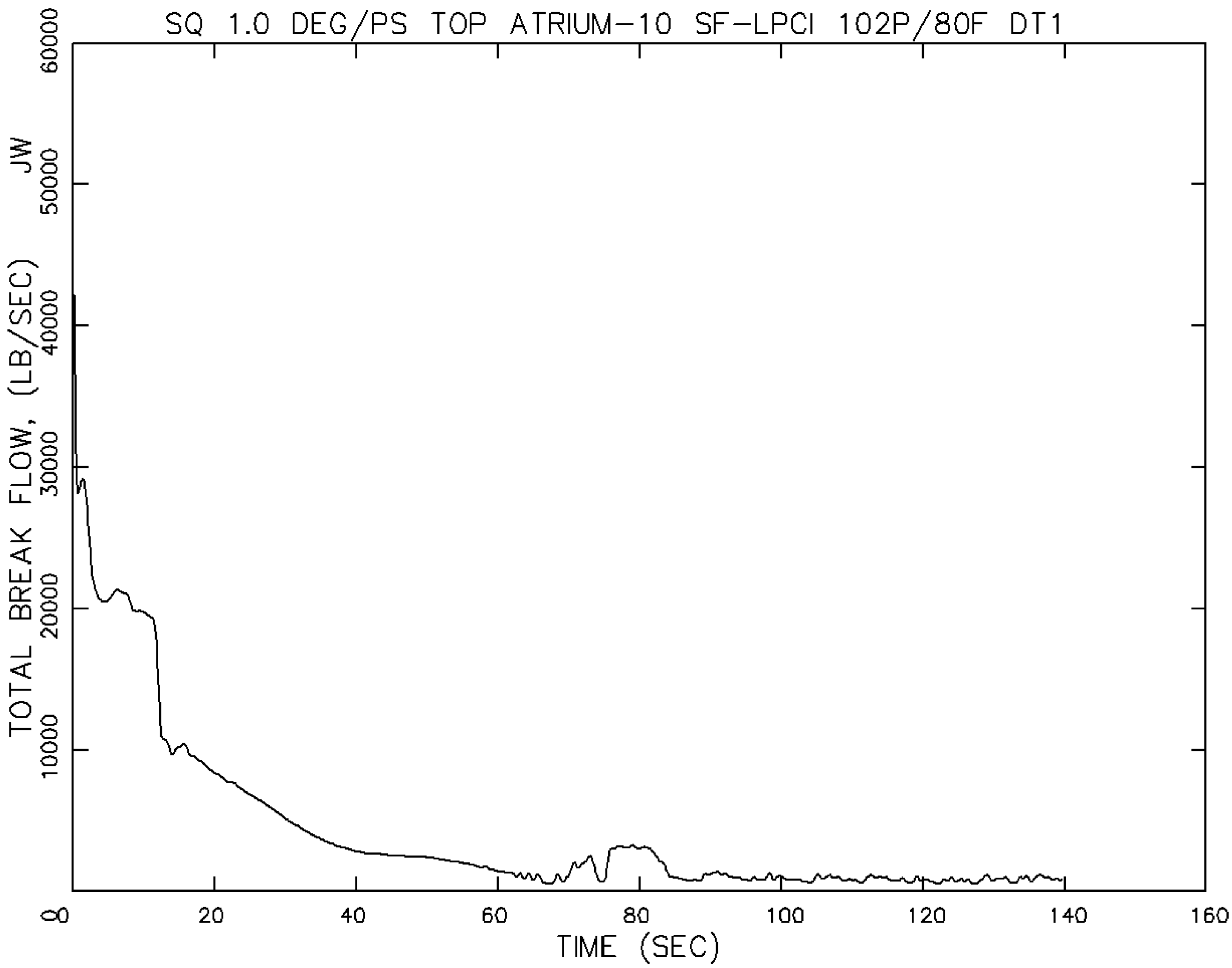


Limiting TLO Recirculation Line Break
Upper Plenum Pressure (Lower)

FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
LIMITING TLO RECIRCULATION LINE BREAK UPPER PLENUM PRESSURE (LOWER)
FIGURE 6.3-201, Rev 3

AutoCAD: Figure Fsar 6_3_201.dwg



Limiting TLO Recirculation Line Break
Total Break Flow Rate

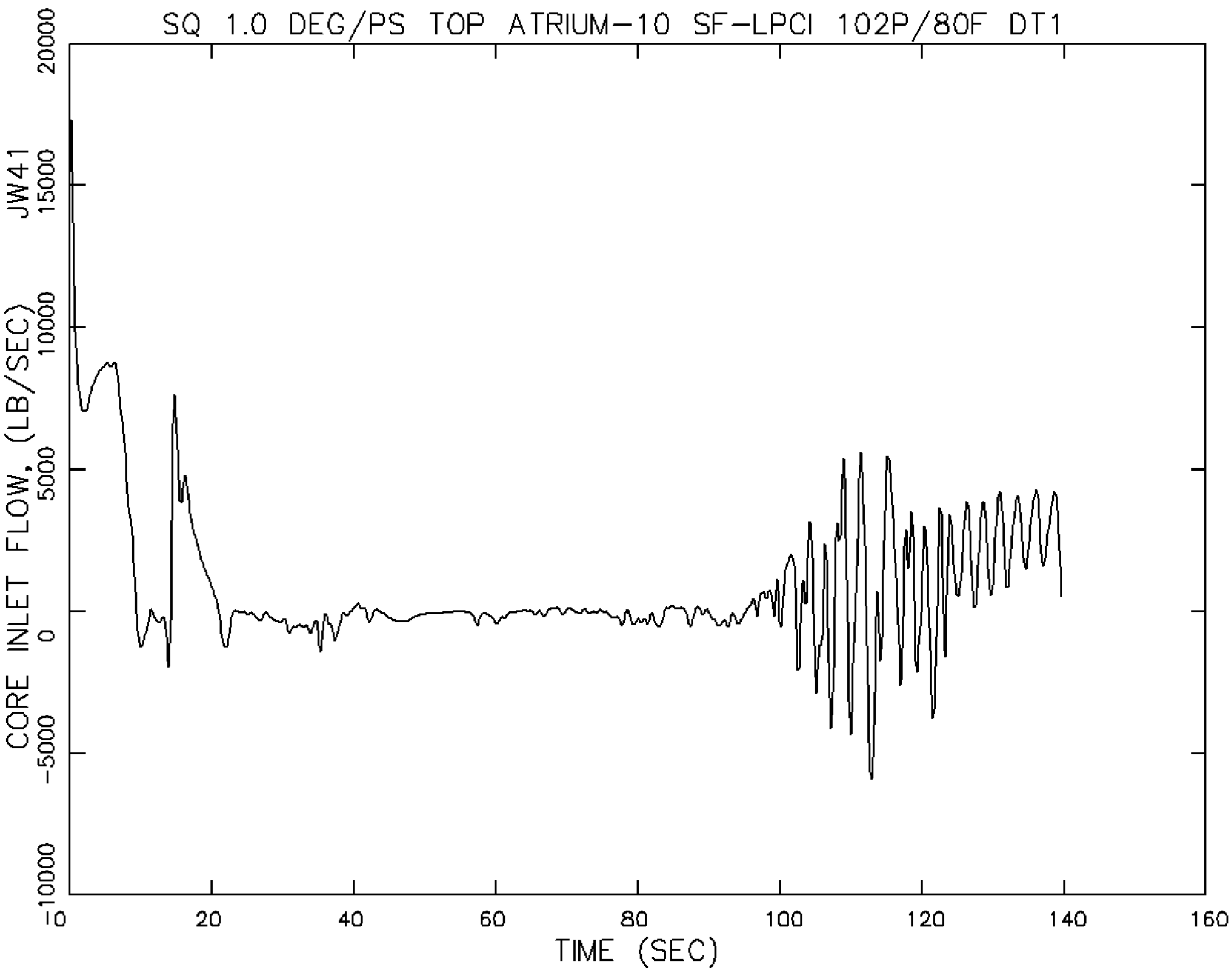
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LIMITING TLO RECIRCULATION LINE BREAK
TOTAL BREAK FLOW RATE

FIGURE 6.3-202, Rev 3

AutoCAD: Figure Fsar 6_3_202.dwg



Limiting TLO Recirculation Line Break
Core Inlet Flow Rate

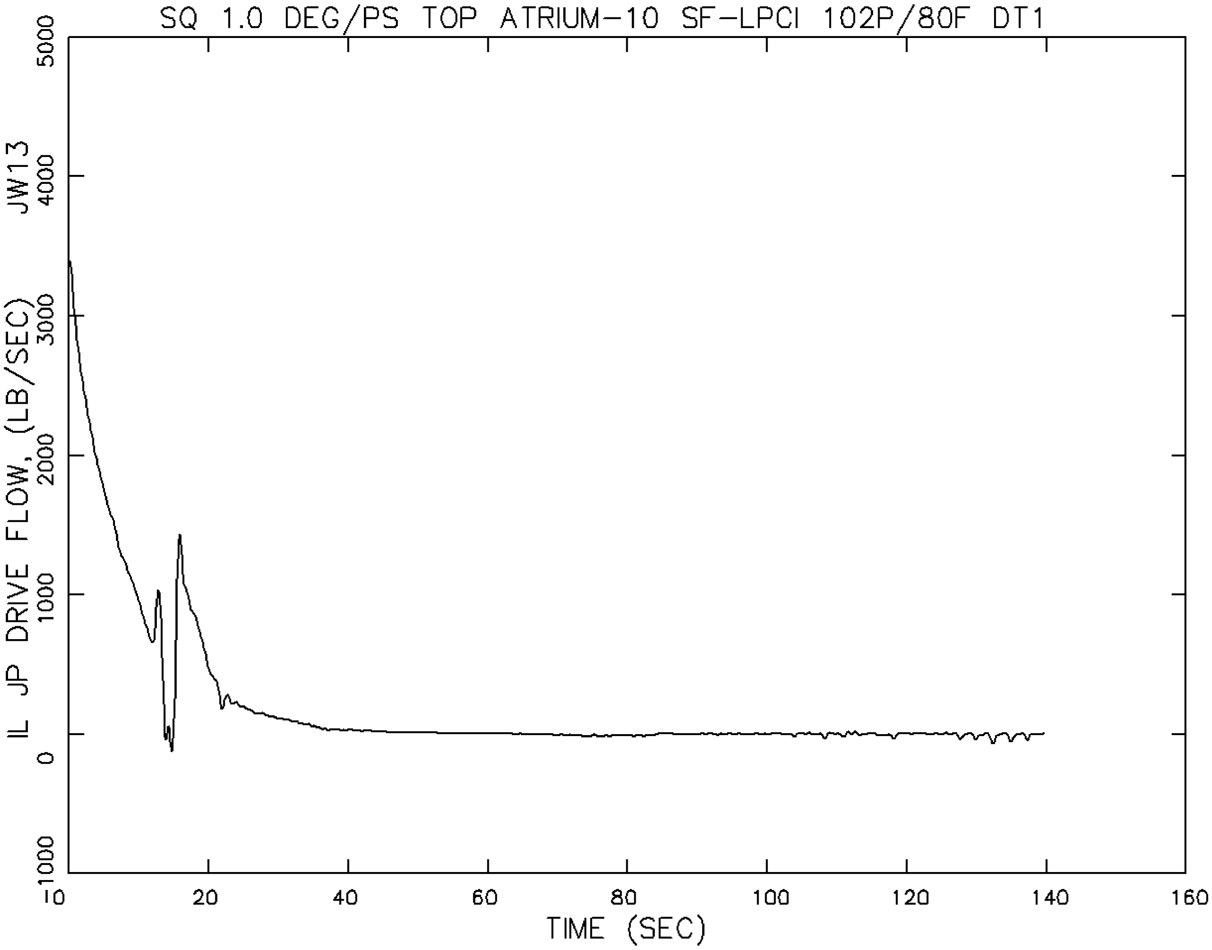
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LIMITING TLO RECIRCULATION LINE BREAK
CORE INLET FLOW RATE

FIGURE 6.3-203, Rev 3

AutoCAD: Figure Fsar 6_3_203.dwg



Limiting TLO Recirculation Line Break
Intact Loop Jet Pump Drive Flow Rate

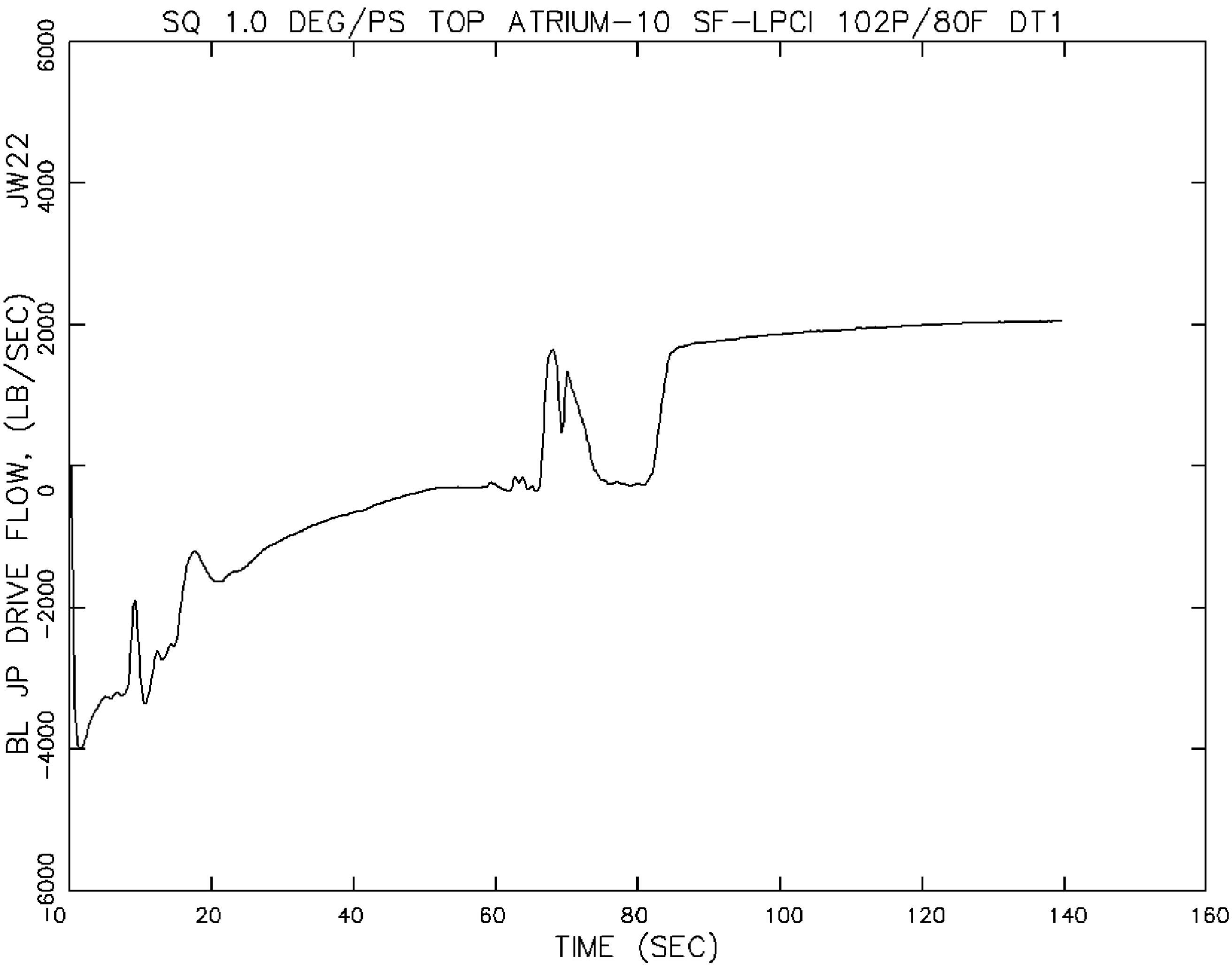
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LIMITING TLO RECIRCULATION LINE BREAK
INTACT LOOP JET PUMP DRIVE FLOW RATE

FIGURE 6.3-204, Rev 3

AutoCAD: Figure Fsar 6_3_204.dwg



Limiting TLO Recirculation Line Break
Broken Loop Jet Pump Drive Flow Rate

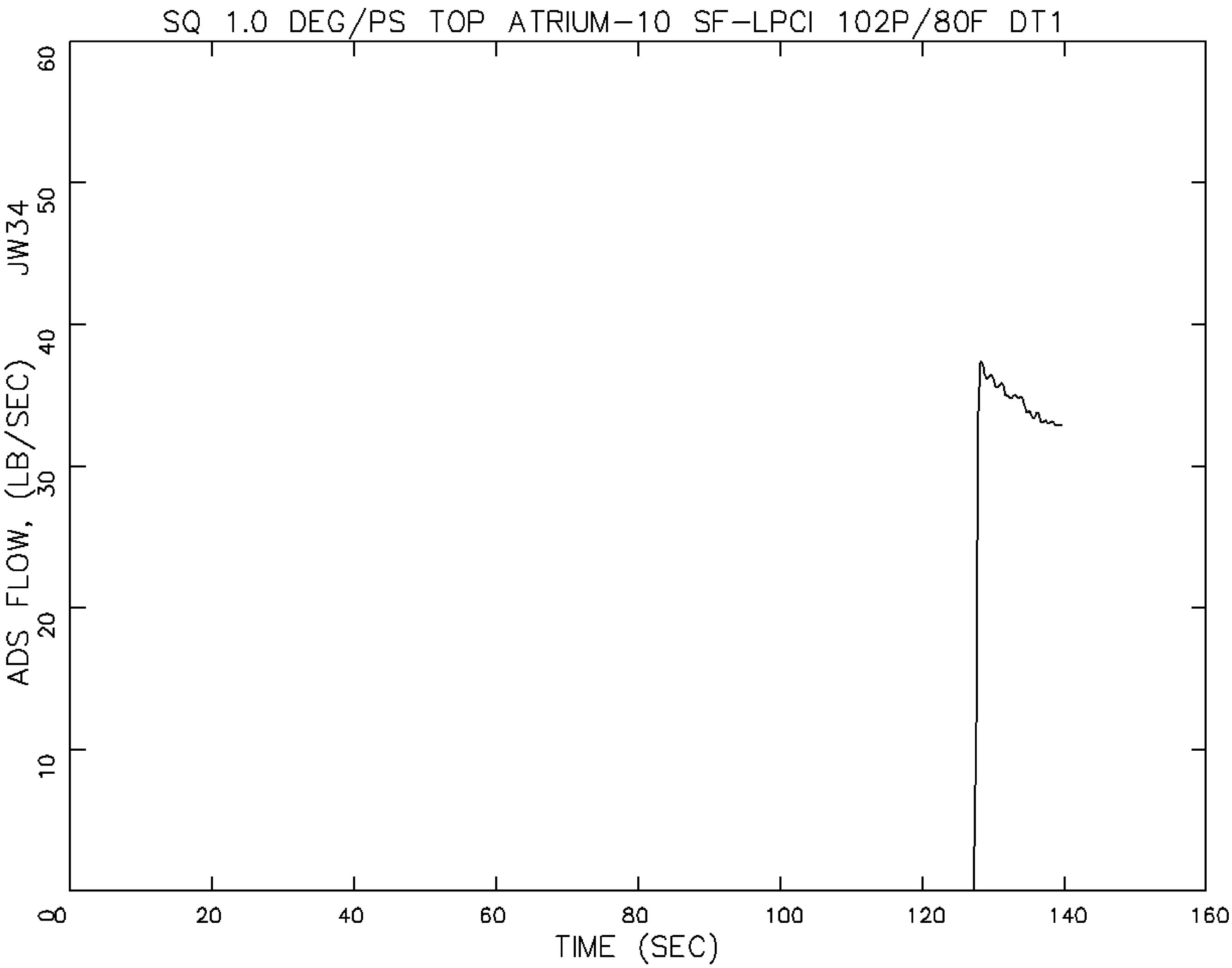
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LIMITING TLO RECIRCULATION LINE BREAK
BROKEN LOOP JET PUMP DRIVE FLOW RATE

FIGURE 6.3-205, Rev 3

AutoCAD: Figure Fsar 6_3_205.dwg



Limiting TLO Recirculation Line Break
ADS Flow Rate

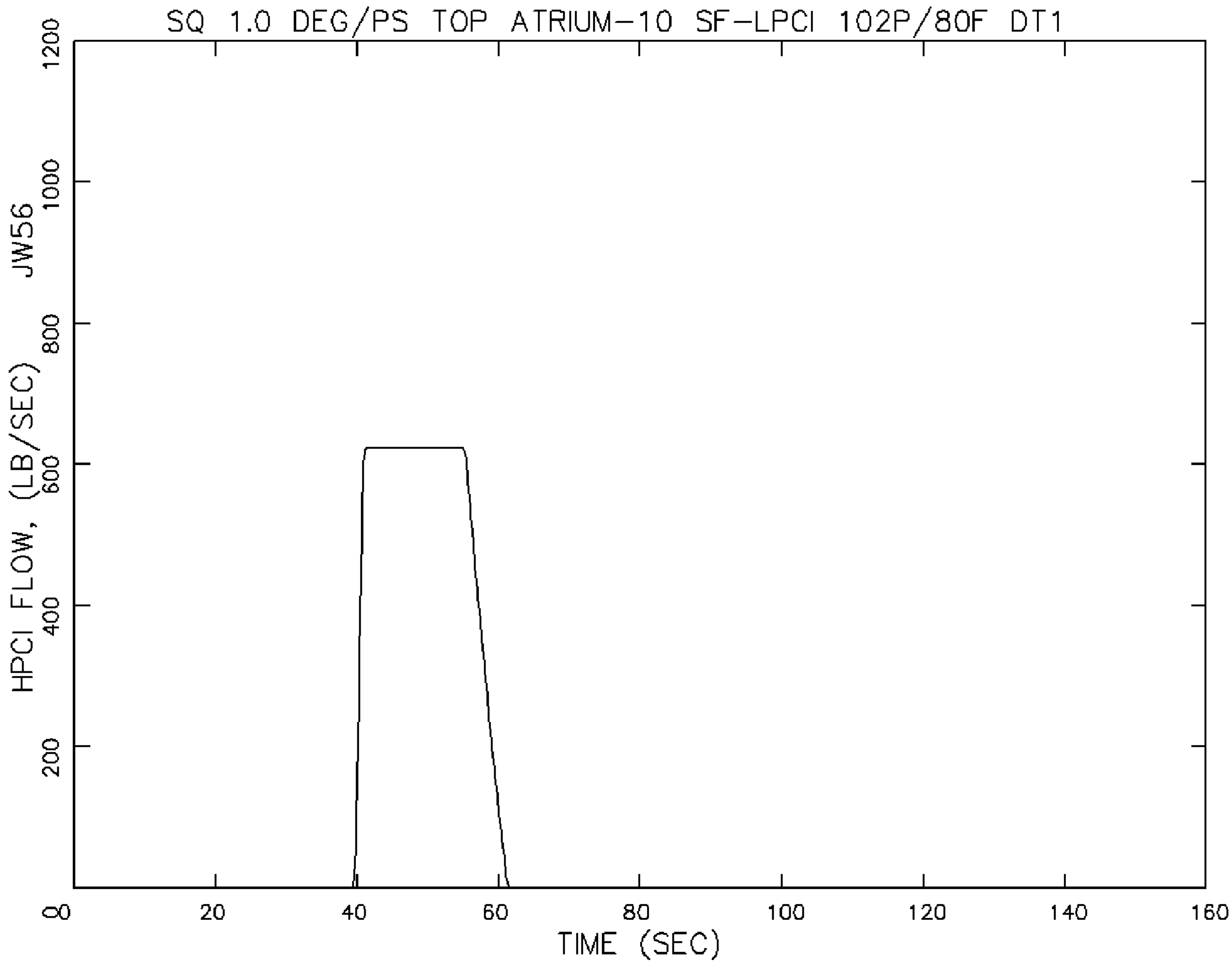
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LIMITING TLO RECIRCULATION LINE BREAK
ADS FLOW RATE

FIGURE 6.3-206, Rev 3

AutoCAD: Figure Fsar 6_3_206.dwg



Limiting TLO Recirculation Line Break
HPCI Flow Rate

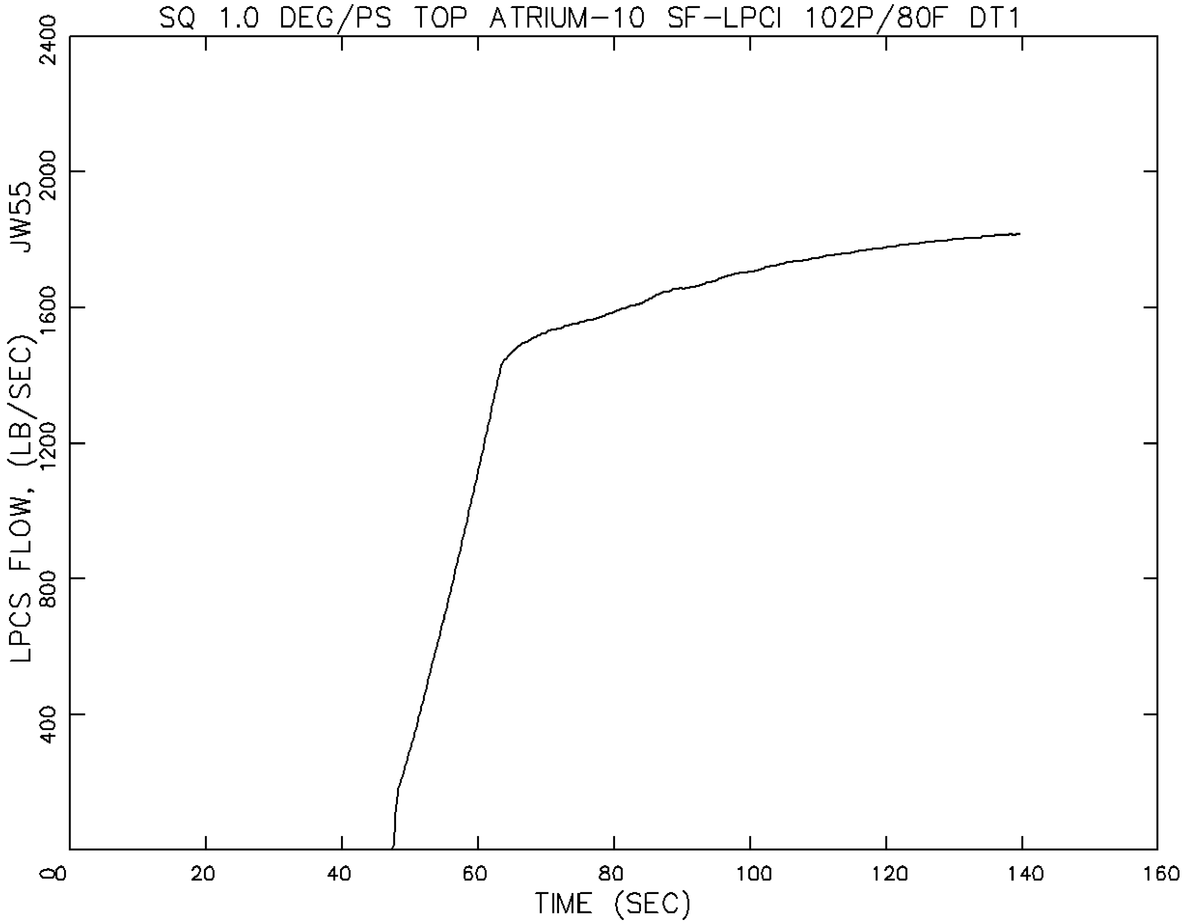
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LIMITING TLO RECIRCULATION LINE BREAK
HPCI FLOW RATE

FIGURE 6.3-207, Rev 3

AutoCAD: Figure Fsar 6_3_207.dwg



Limiting TLO Recirculation Line Break
LPCS Flow Rate

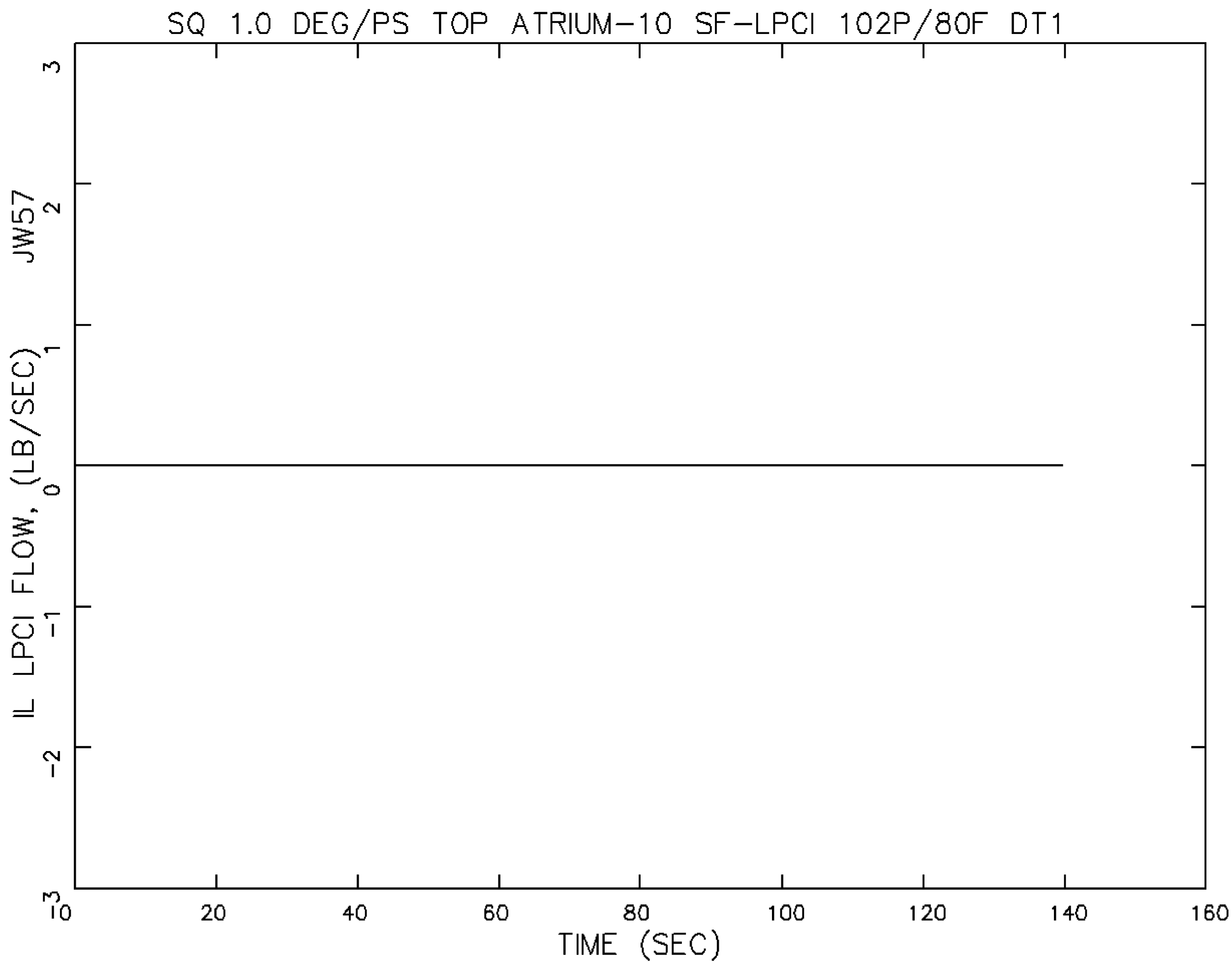
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LIMITING TLO RECIRCULATION LINE BREAK
LPCS FLOW RATE

FIGURE 6.3-208, Rev 3

AutoCAD: Figure Fsar 6_3_208.dwg

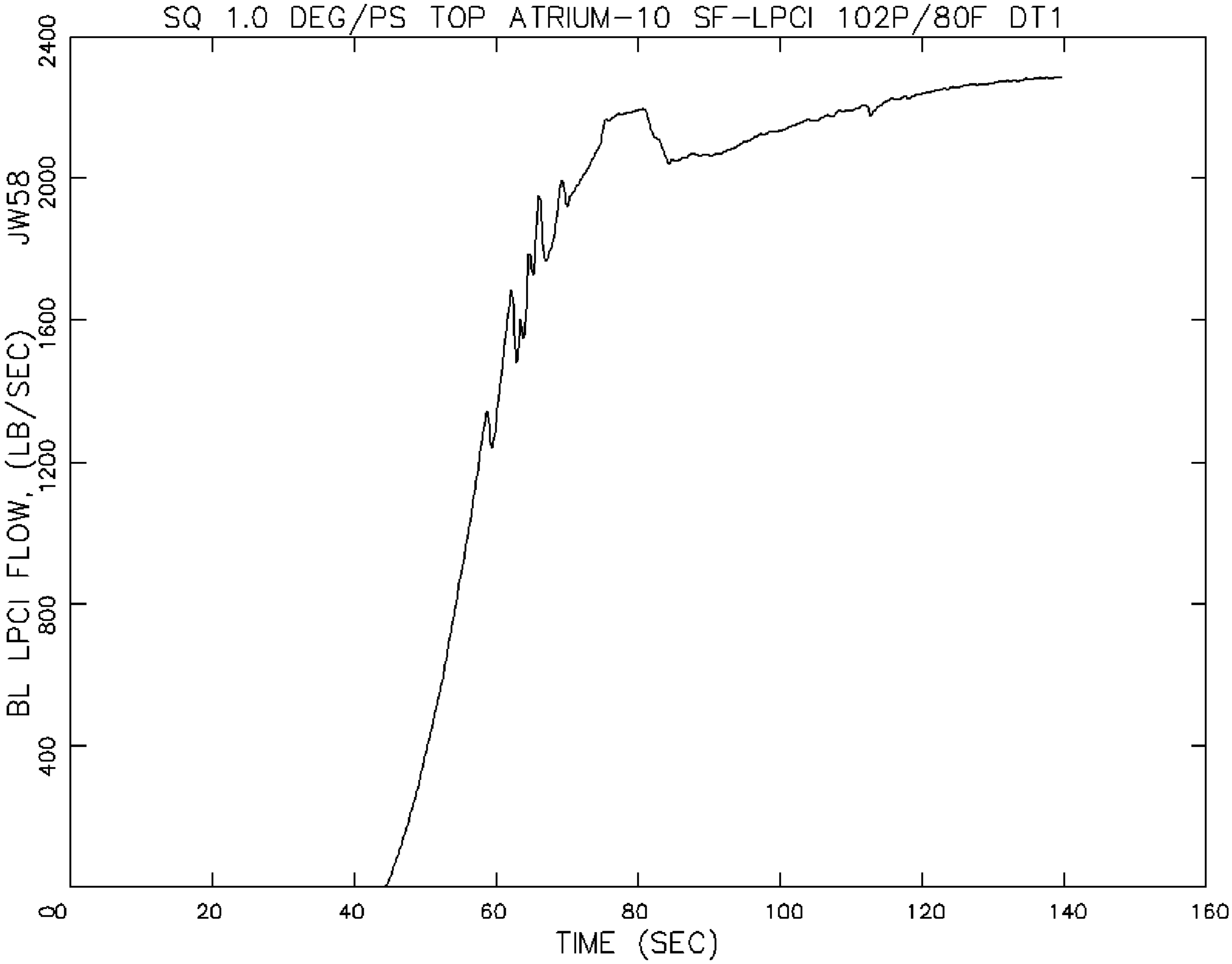


Limiting TLO Recirculation Line Break
Intact Loop LPCI Flow Rate

FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
LIMITING TLO RECIRCULATION LINE BREAK INTACT LOOP LPCI FLOW RATE FIGURE 6.3-209, Rev 3

AutoCAD: Figure Fsar 6_3_209.dwg



Limiting TLO Recirculation Line Break
Broken Loop LPCI Flow Rate

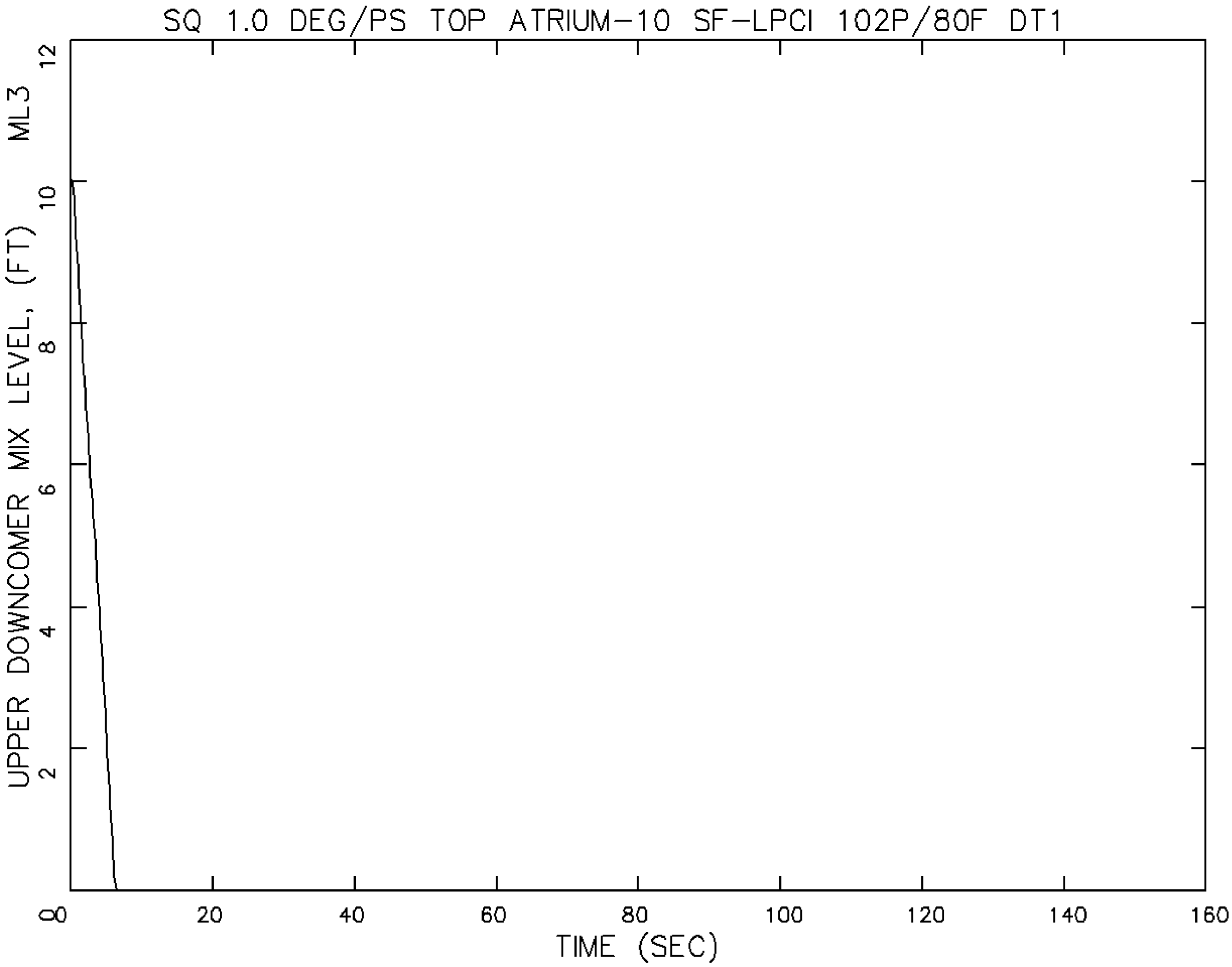
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LIMITING TLO RECIRCULATION LINE BREAK
BROKEN LOOP LPCI FLOW RATE

FIGURE 6.3-210, Rev 3

AutoCAD: Figure Fsar 6_3_210.dwg



Limiting TLO Recirculation Line Break
Upper Downcomer Mixture Level

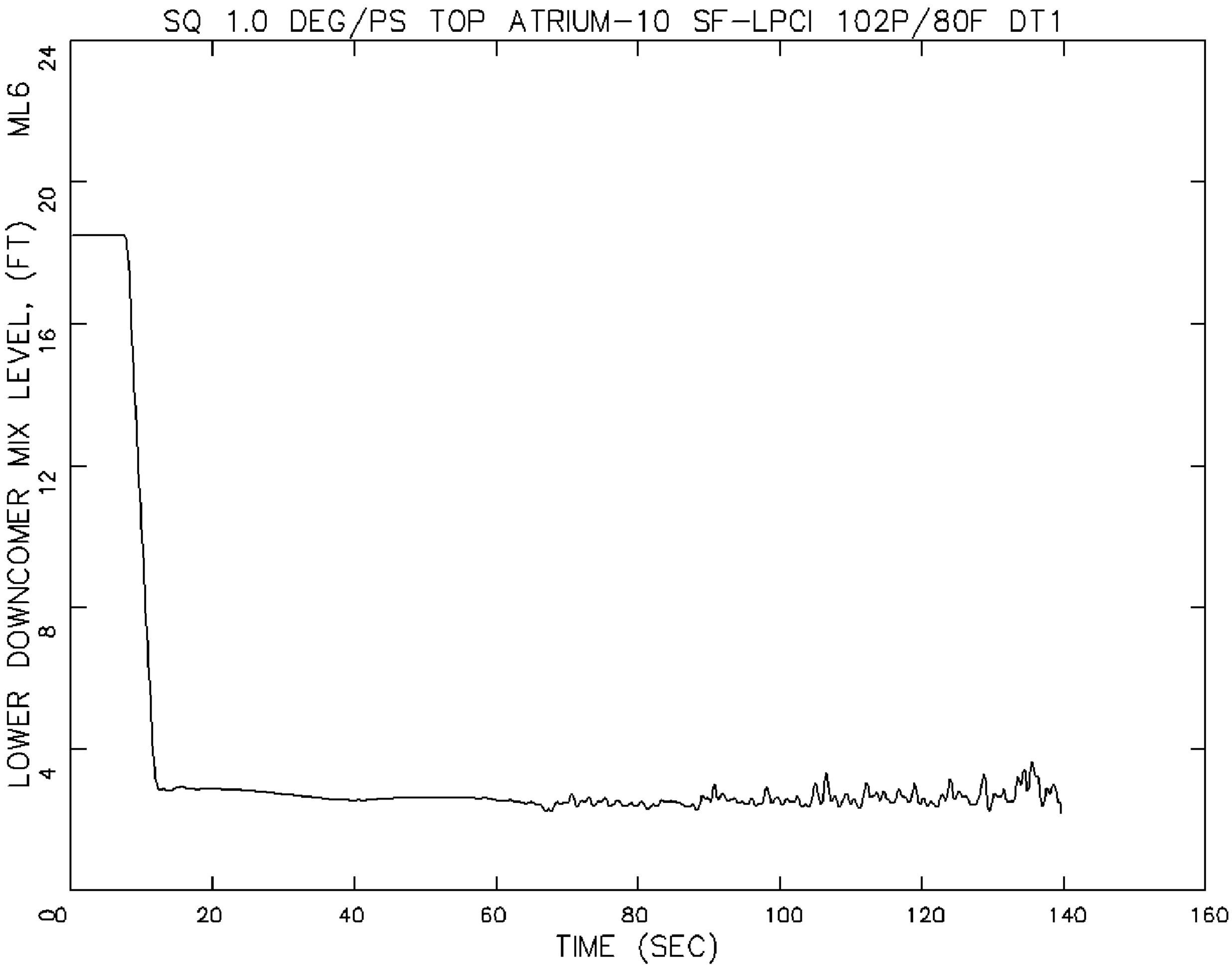
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LIMITING TLO RECIRCULATION LINE BREAK
UPPER DOWNCOMER MIXTURE LEVEL

FIGURE 6.3-211, Rev 3

AutoCAD: Figure Fsar 6_3_211.dwg



Limiting TLO Recirculation Line Break
Lower Downcomer Mixture Level

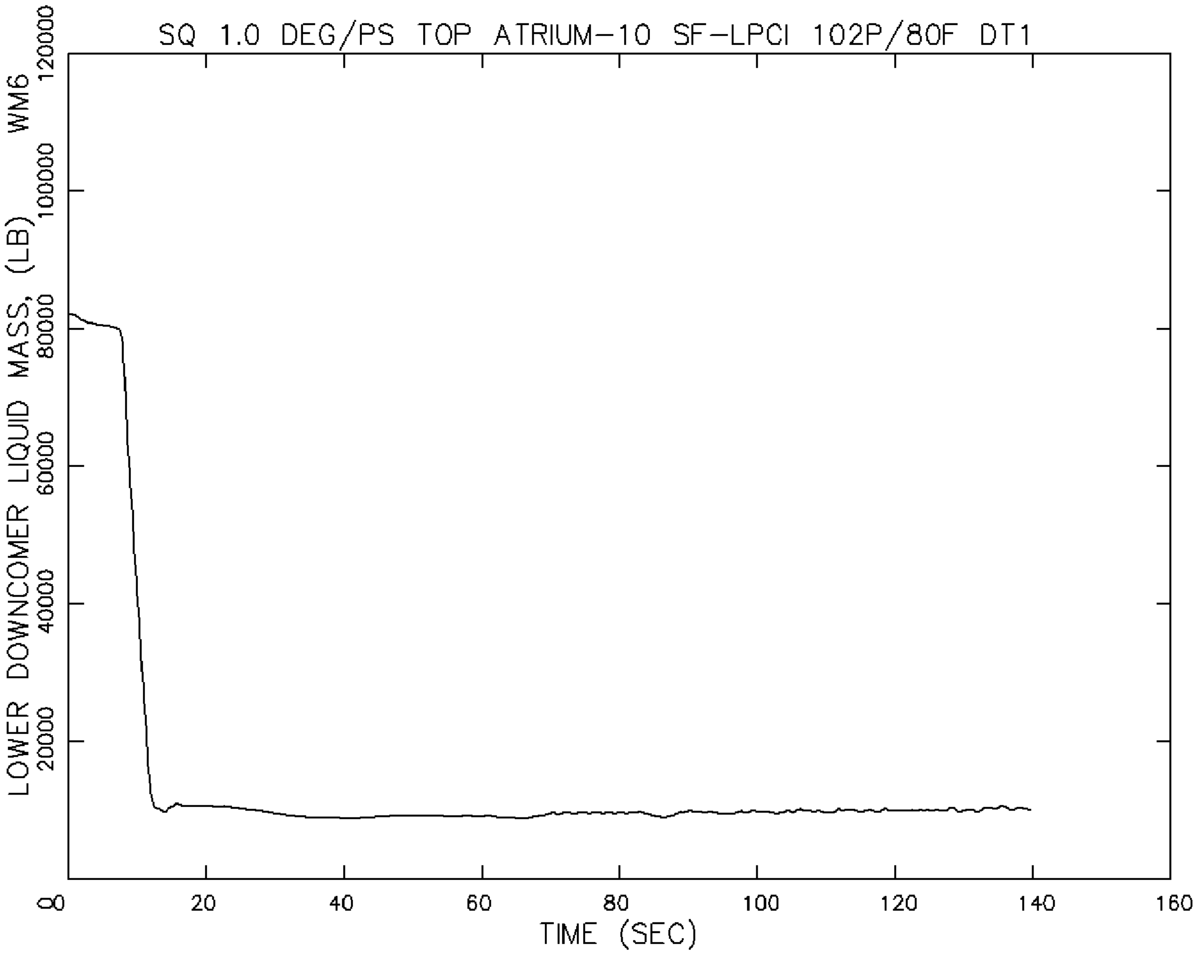
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LIMITING TLO RECIRCULATION LINE BREAK
LOWER DOWNCOMER MIXTURE LEVEL

FIGURE 6.3-212, Rev 3

AutoCAD: Figure Fsar 6_3_212.dwg



Limiting TLO Recirculation Line Break
Lower Downcomer Liquid Mass

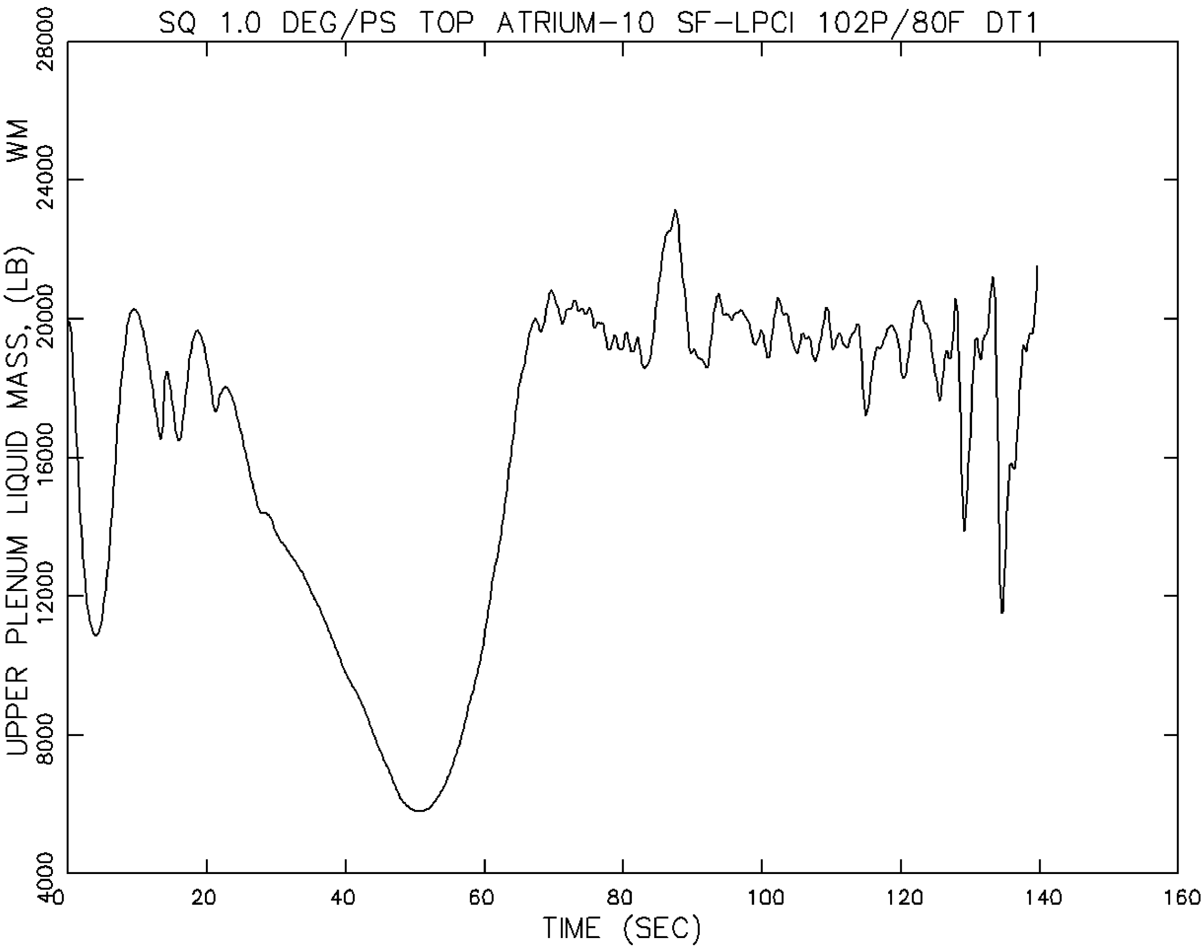
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LIMITING TLO RECIRCULATION LINE BREAK
LOWER DOWNCOMER LIQUID MASS

FIGURE 6.3-213, Rev 3

AutoCAD: Figure Fsar 6_3_213.dwg



Limiting TLO Recirculation Line Break
Upper Plenum Liquid Mass

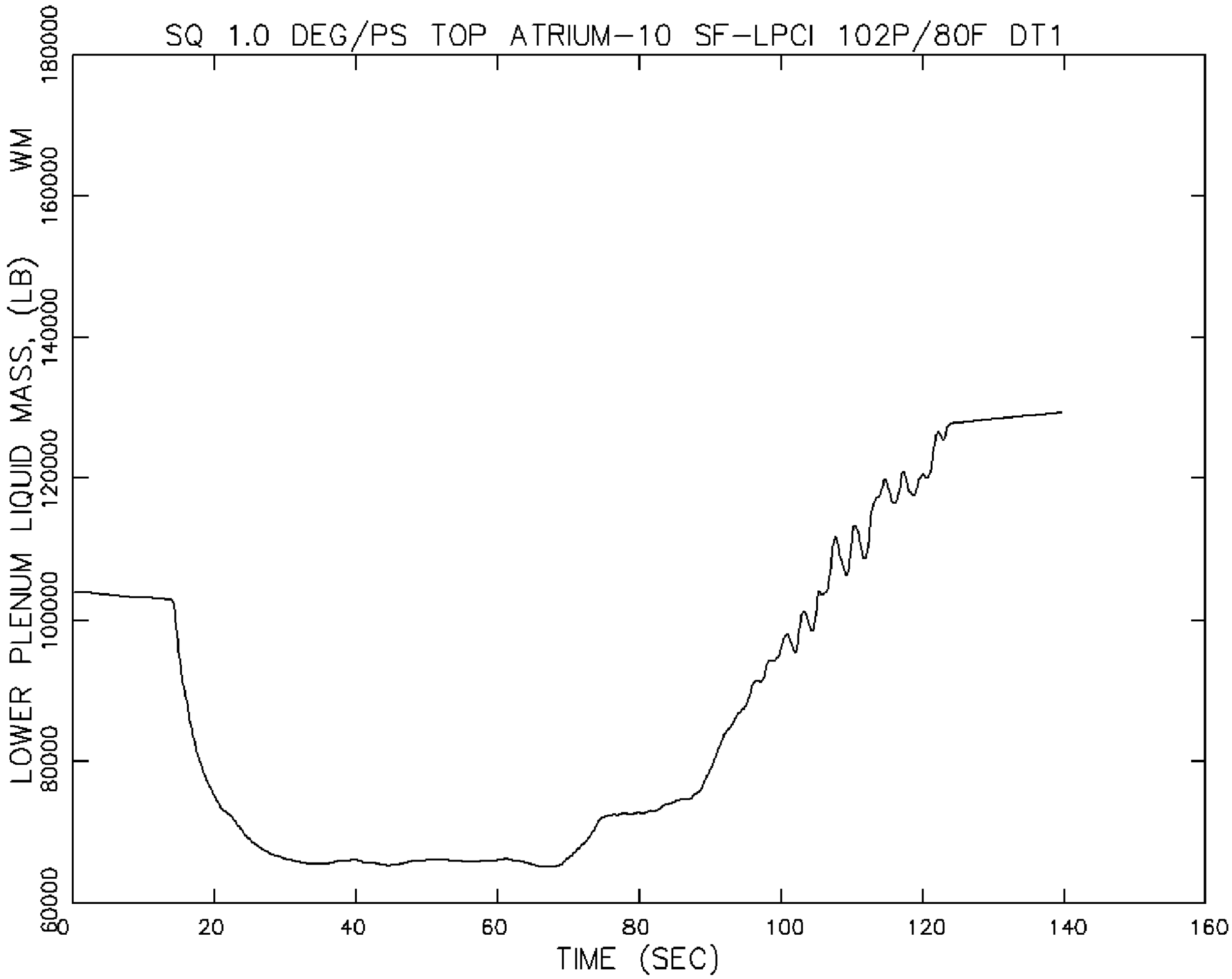
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LIMITING TLO RECIRCULATION LINE BREAK
UPPER PLENUM LIQUID MASS

FIGURE 6.3-214, Rev 3

AutoCAD: Figure Fsar 6_3_214.dwg



Limiting TLO Recirculation Line Break
Lower Plenum Liquid Mass

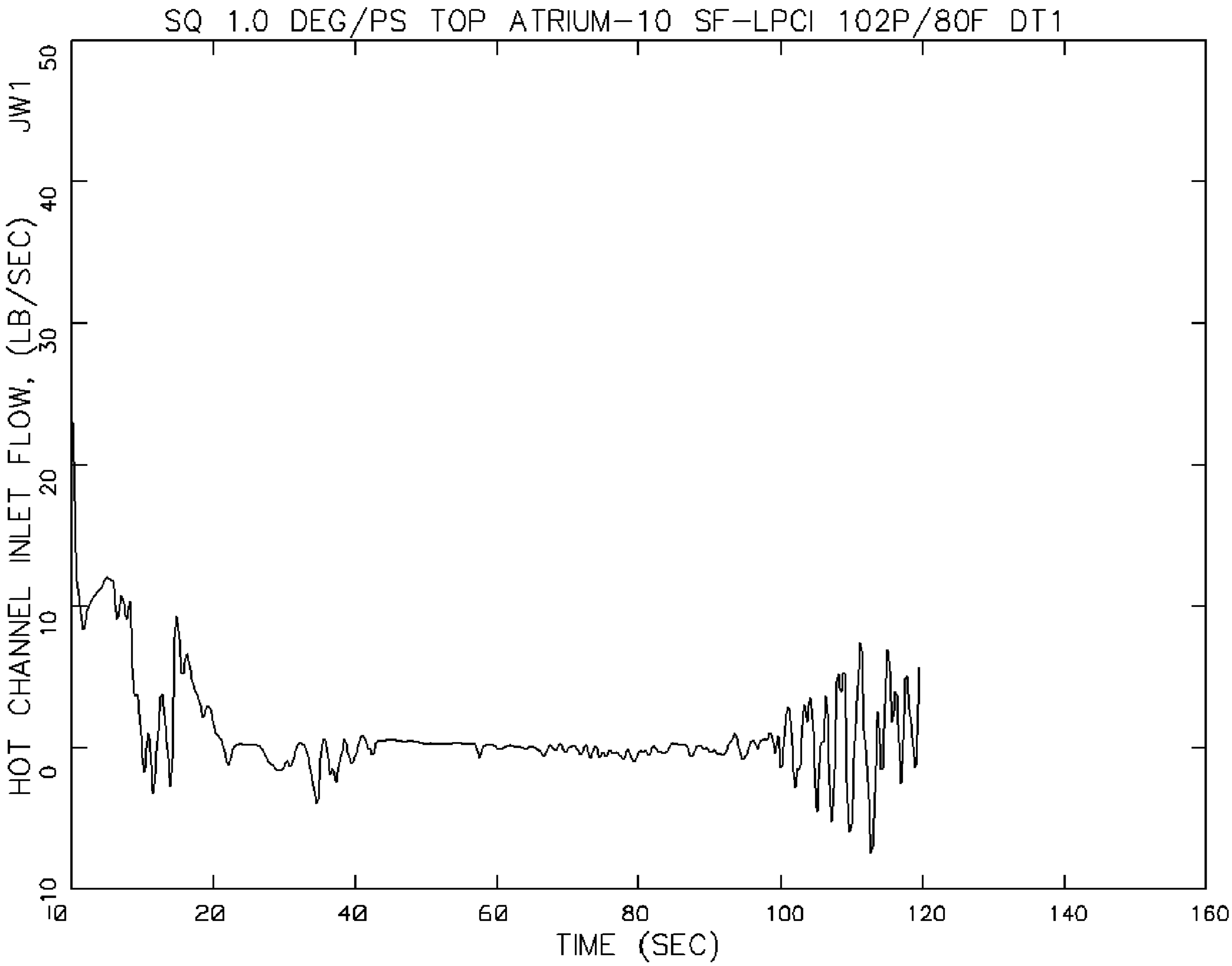
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LIMITING TLO RECIRCULATION LINE BREAK
LOWER PLENUM LIQUID MASS

FIGURE 6.3-215, Rev 3

AutoCAD: Figure Fsar 6_3_215.dwg



Limiting TLO Recirculation Line Break
Hot Channel Inlet Flow Rate

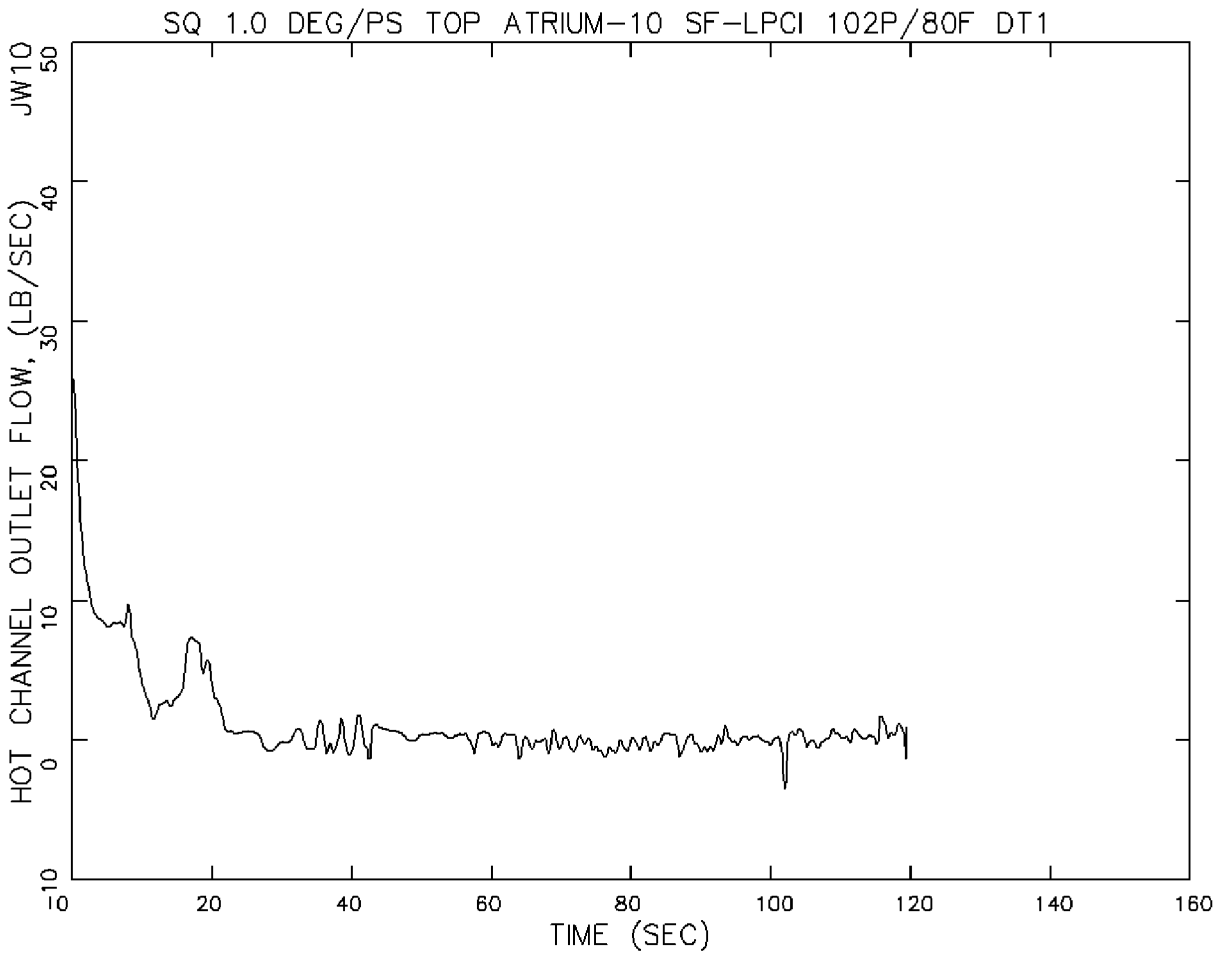
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LIMITING TLO RECIRCULATION LINE BREAK
HOT CHANNEL INLET FLOW RATE

FIGURE 6.3-216, Rev 3

AutoCAD: Figure Fsar 6_3_216.dwg



Limiting TLO Recirculation Line Break
Hot Channel Outlet Flow Rate

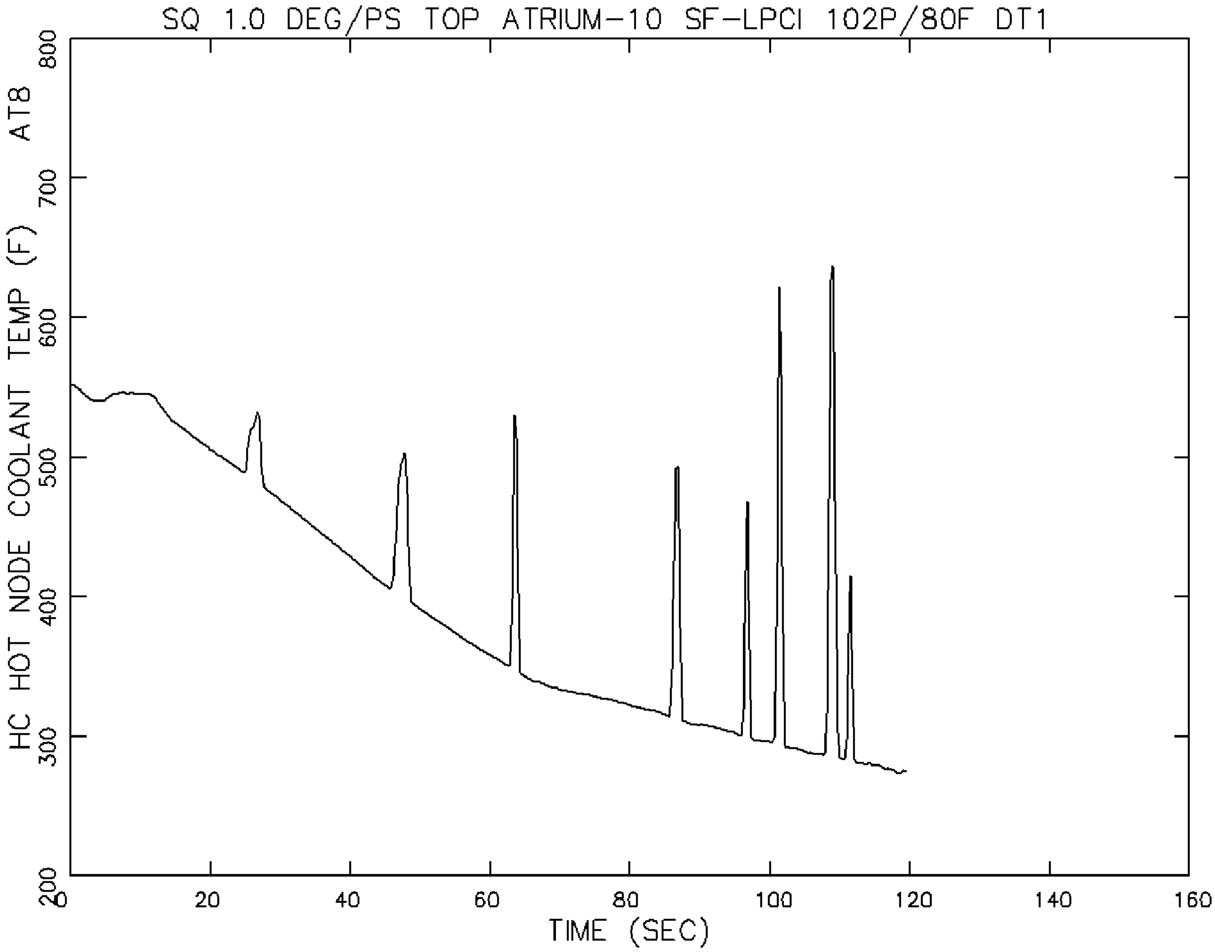
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LIMITING TLO RECIRCULATION LINE BREAK
HOT CHANNEL OUTLET FLOW RATE

FIGURE 6.3-217, Rev 3

AutoCAD: Figure Fsar 6_3_217.dwg



Limiting TLO Recirculation Line Break
Hot Channel Coolant Temperature
at the Limiting Node

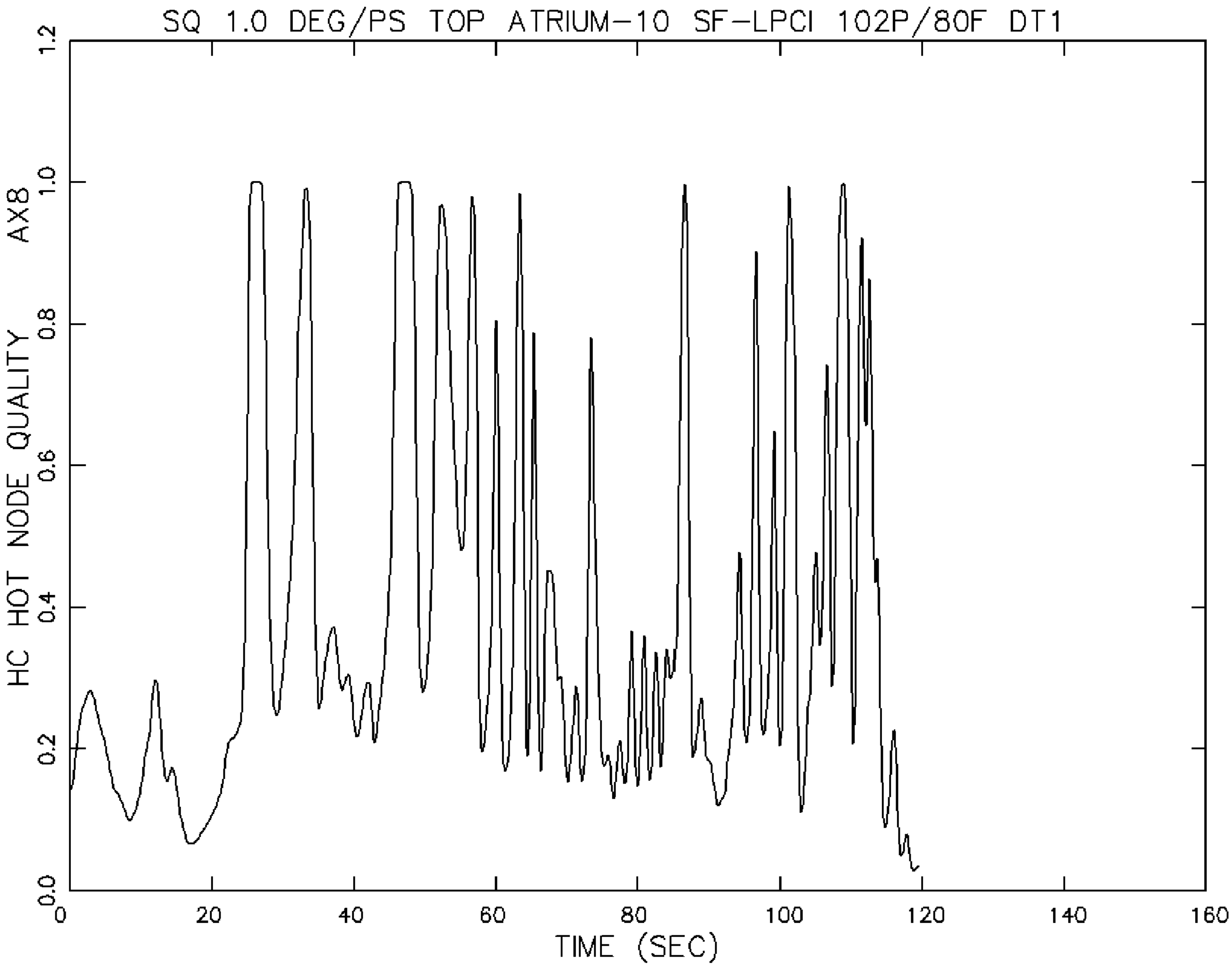
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LIMITING TLO RECIRCULATION LINE BREAK
HOT CHANNEL COOLANT TEMPERATURE
AT THE LIMITING NODE

FIGURE 6.3-218, Rev 3

AutoCAD: Figure Fsar 6_3_218.dwg



Limiting TLO Recirculation Line Break
Hot Channel Quality at the Limiting Node

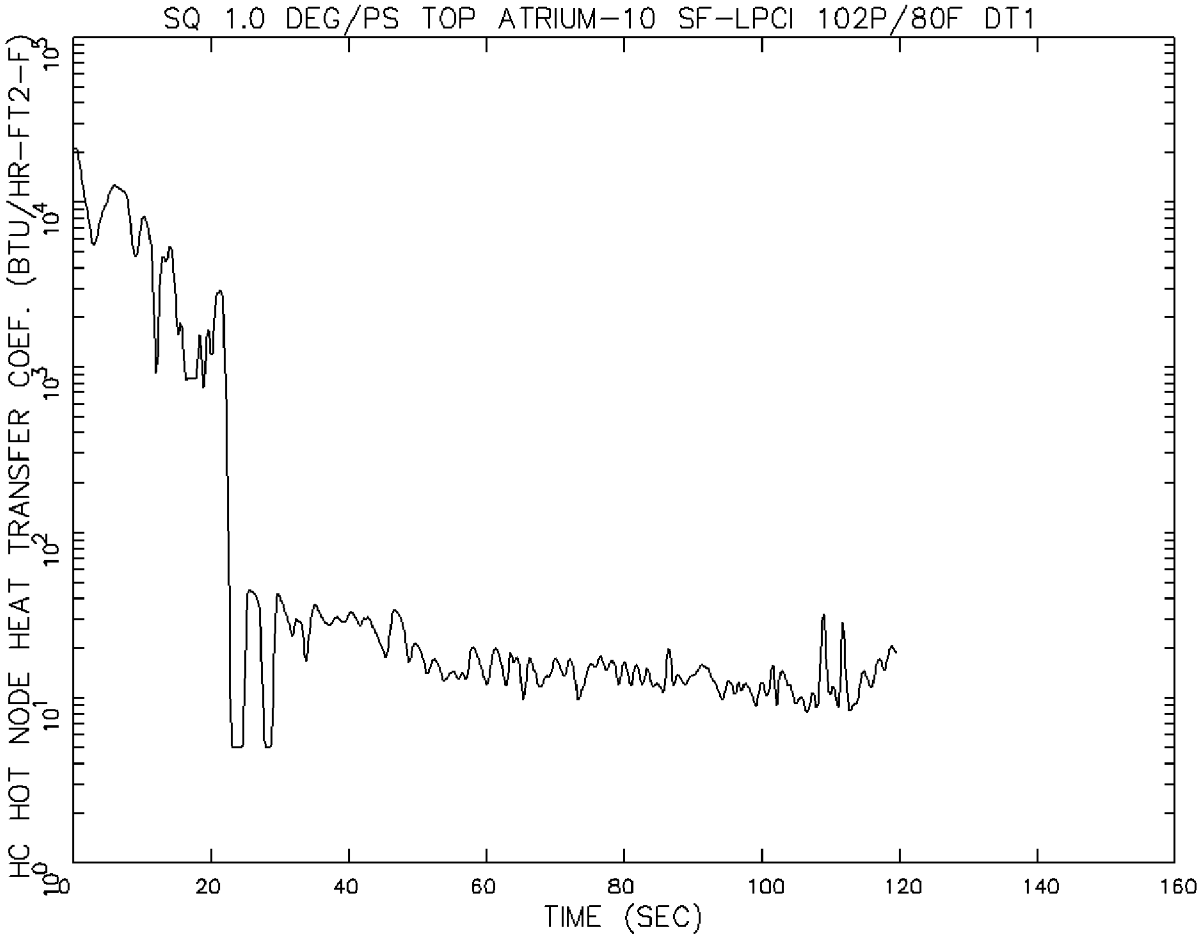
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LIMITING TLO RECIRCULATION LINE BREAK
HOT CHANNEL QUALITY AT THE LIMITING NODE

FIGURE 6.3-219, Rev 3

AutoCAD: Figure Fsar 6_3_219.dwg



Limiting TLO Recirculation Line Break
Hot Channel Heat Transfer Coeff.
at the Limiting Node

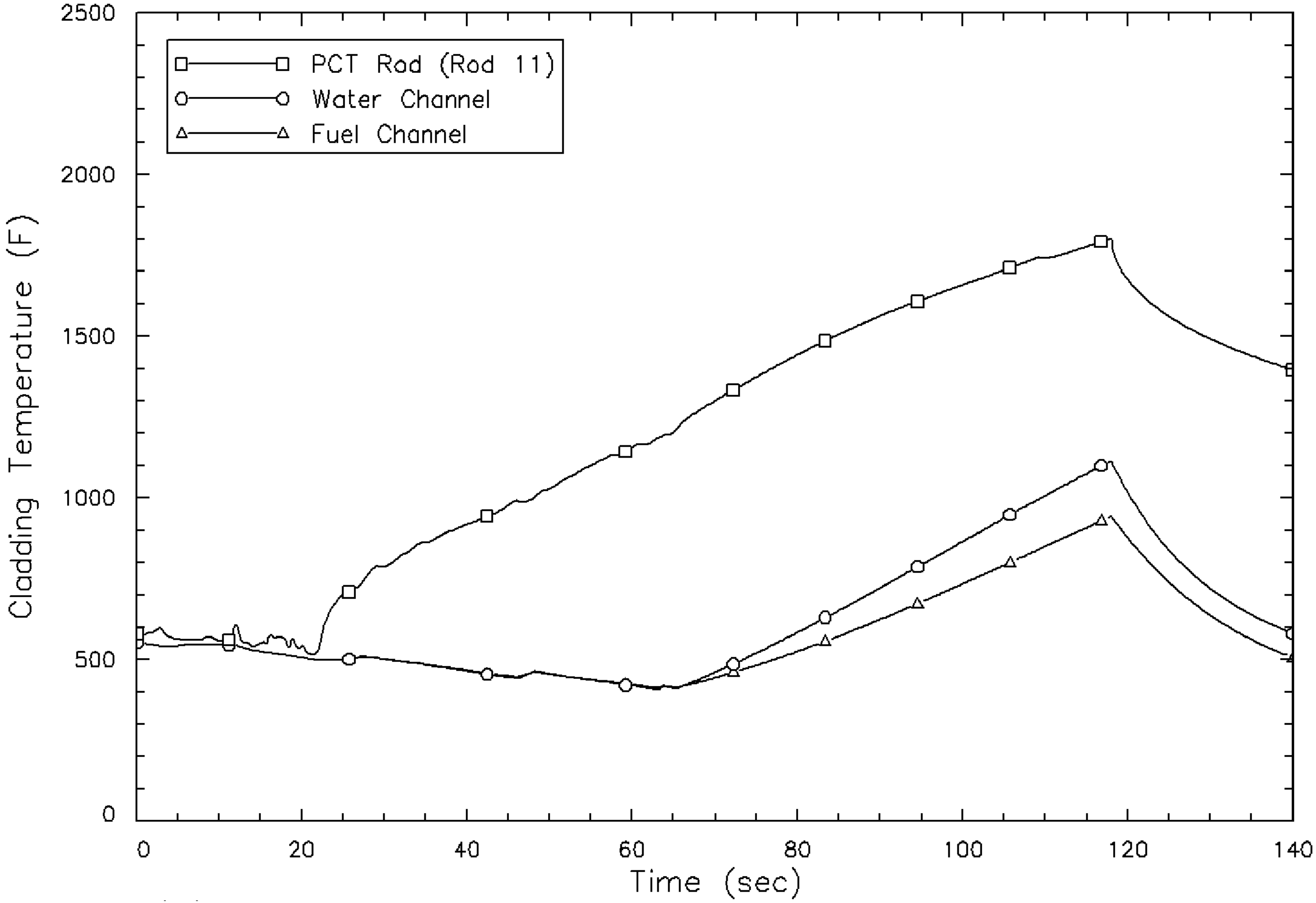
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LIMITING TLO RECIRCULATION LINE BREAK
HOT CHANNEL HEAT TRANSFER COEFF.
AT THE LIMITING NODE

FIGURE 6.3-220, Rev 3

AutoCAD: Figure Fsar 6_3_220.dwg



Limiting TLO Recirculation Line Break
Cladding Temperatures

FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LIMITING TLO RECIRCULATION LINE BREAK
CLADDING TEMPERATURES

FIGURE 6.3-221, Rev 3

AutoCAD: Figure Fsar 6_3_221.dwg