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7.3.0 Water Hammer At San Onofre

Learning Objectives:

1. Describe three types of water hammer and their causes.
2. Describe corrective actions that were taken to prevent previous steam generator water hammer problems.
3. Describe the damage caused by the water hammer event at San Onofre Nuclear Generating Station Unit 1 (SONGS-l).
4. Describe how multiple check valve failures contributed to the initiation of the water hammer at SONGS-1.
5. Discuss how check valve testing required by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code could have prevented the SONGS-1 water hammer incident.

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7.3.1 History of Water Hammer at Nuclear Power Plants

During the early 1970s, the NRC became aware of the increasing frequency of water hammer events in nuclear power plant systems and became concerned about the potential challenges to system integrity and operability that could result from these incidents. For pressurized water reactors, the major contributor to these incidents was a phenomenon called steam generator water hammer (SGWH). Although the significance of these events varied from plant to plant, the NRC was concerned that a severe SGWH could cause a complete loss of feedwater and affect the ability of a plant to remove decay heat and cool down after a reactor trip.

Following the SGWH that occurred at Indian Point Unit 2 in 1972, which resulted in a circumferential weld failure in one of the feedwater lines, the NRC required all utilities to submit design and operational information describing design features for avoiding SGWH. In 1978, the generic subject of water hammer was classified as an unresolved safety issue (USI A-1) and received increased NRC and industry attention.

SGWH can occur following a reactor trip when the steam generator top feedring drains and refills with cold auxiliary feedwater. NRC attention was directed at the feedring design and internal steam generator (SG) components near the feedwater (FW) nozzle. Experience had revealed that internal damage to the feedring and supports could occur. Modifications implemented to prevent SGWH generally involved installation of J-tubes to prevent the draindown of feedrings, short horizontal runs of FW piping adjacent to SG feedwater nozzles to minimize the magnitude of water hammers, and limits on auxiliary feedwater (AFW) system flow rates to avoid the rapid refill of SGs with cold water. In general, attention focused on the internal structure and design of the steam generator rather than on conditions in the FW lines and flow control components.

The NRC was aware of the possibility of developing condensation-induced water hammer extending back into the feedwater piping as a result of line voiding because of a water hammer occurrence at the KRSKO plant in Yugoslavia in 1979. Limited information on that event suggests that leaky check valves or pre-operation pump testing (i.e., start and trip test), or both, were the underlying causes. Similar occurrences had not been reported for U.S. plants, and apparently check valve failures were not considered a significant contributor to feedwater system water hammer by the NRC. Implicit in the reliance the NRC placed on J-tubes to prevent steam generator feedring voiding to prevent SGWH, was the assumption that feedwater system check valves do not leak. It appears that the NRC did not consider feedwater piping water hammer due to failed check valves to be a substantial contributor and did not pursue this issue further.

7.3.2 Water Hammer

This section discusses the water hammer which occurred at SONGS- 1, its underlying causes, and the damage incurred. Since failed check valves in the feedwater piping were the underlying cause, this section also discusses valve maintenance and in-service testing related to these valves. To clarify the discussions that follow, a brief review of water hammer phenomena and commonly accepted definitions are provided.

Hydraulic instabilities occur frequently in piping networks as a result of changes in fluid velocity or pressure. Some of the better understood occurrences include induced flow transients due to starting and stopping pumps, opening and closing valves, water filling voided (empty) lines, and pressure changes due to pipe breaks or ruptures. As a consequence of the change in fluid velocity or pressure, pressure waves are created which propagate throughout the fluid within the piping network and produce audible noise, line vibrations and, if sufficient energy transfer occurs between the pressure wave and the pressure boundary, structural damage to piping, piping supports, and attached equipment. More specifically, this pressure transient is a fluid shock wave in which the pressure change is the result of the conversion of kinetic energy into pressure waves (compression waves) or the conversion of pressure into kinetic energy (rarefaction waves). Regardless of the underlying causes, this phenomenon is generally referred to as water hammer.

A water hammer event can be characterized as one of the following three major types:

1. “Classical water hammer” generally identifies a fluid shock, accompanied by noise, which results from the sudden, nearly instantaneous stoppage of a moving fluid column. Unexpected valve closures, backflow against a check valve, and pump startup into voided lines where valves are closed downstream are common underlying causes of classical water hammer and are generally well understood.

   Analytical methods have been developed to predict loads for this type of fluid hammer and include the effects of initial pressure, fluid inertia, piping dimensions and layout, pipe wall elasticity, fluid bulk modulus, valve operating characteristics (time to open or close), etc.

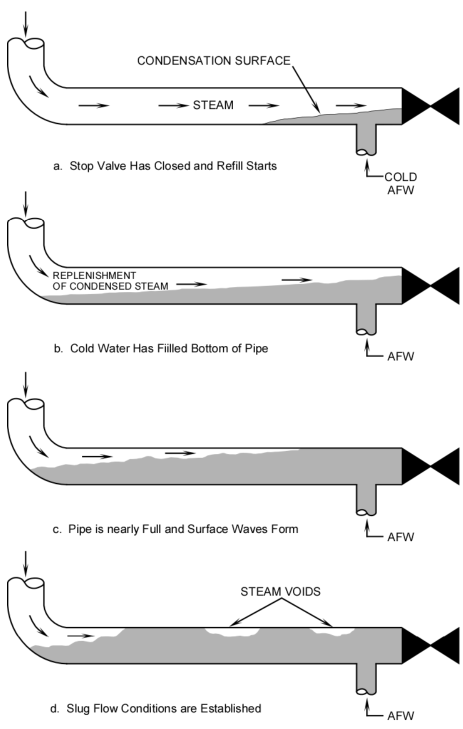


Figure 7.3-1 Filling of a Voided Feedwater Line

2. “Condensation-induced water hammer” results when cold water (such as auxiliary feedwater) comes in contact with steam. Conditions conducive to this type of water hammer are an abundant steam source and a long empty horizontal pipe run being refilled slowly with cold water. The cold water draws energy from the steam, with the rate of energy transfer being governed by local flow conditions. As the steam condenses, additional steam will flow countercurrent to the cold water, and as the pipe fills up (i.e., the void decreases) the steam velocity increases, setting up waves on the surface of the water, eventually entraining water and causing slug flow. Slug flow entraps steam pockets and promotes significant heat transfer between the steam and colder water. Figure 7.3-1 illustrates in simplified form the flow conditions which would

come about during the refilling of a voided horizontal feedwater line. Once slug flow conditions commence, a steam pocket will suddenly condense, creating a localized depressurization instantaneously. The resulting pressure imbalance across the slug (approximately 700 psi at SONGS-1) causes the slug to accelerate away from the source of pressure and toward the region of condensation.

Condensation is extremely rapid, and predicting its exact location is impossible. When the water slug suddenly strikes water in a previously filled pipe, it produces a traveling pressure wave which imposes loads of the magnitude that would be induced by classical water hammer in the piping network. This phenomenon, called condensation-induced water hammer, occurred at SONGS-1.

Predicting loads associated with this type of water hammer is extremely difficult because of the interactive and complex hydrodynamic and heat transfer phenomena which precede the sudden condensation. Void fraction (or how empty the pipe is) and subcooling (or how much colder the water is than the saturation temperature of the steam when steam and water come in contact) are two important parameters currently used in models for predicting this type of water hammer occurrence and its associated loads.

3. “Steam generator water hammer” is a condensation-induced water hammer which has occurred principally in pressurized water reactors (PWRs) with steam generators having top feedrings for feedwater injection. The underlying causes are similar to those discussed above (i.e., the voiding of the horizontal feedring and feedwater piping immediately adjacent to the steam generator and the subsequent injection of cold water). Damage from SGWH has generally been confined to the feedring and its supports and to the steam generator feedwater nozzle region. However, damage to feedwater line snubbers and supports has also occurred. An SGWH resulted in a fractured weld in a feedwater line at Indian Point Nuclear Power Plant Unit 2 in 1972.

7.3.3 San Onofre Water Hammer Incident

San Onofre Nuclear Generating Station Unit 1, operated by the Southern California Edison Company (SCE), is a 450-MWe Westinghouse pressurized water reactor located on the Pacific Ocean, approximately four miles south of San Clemente, California. The plant received an NRC operating license in 1967.

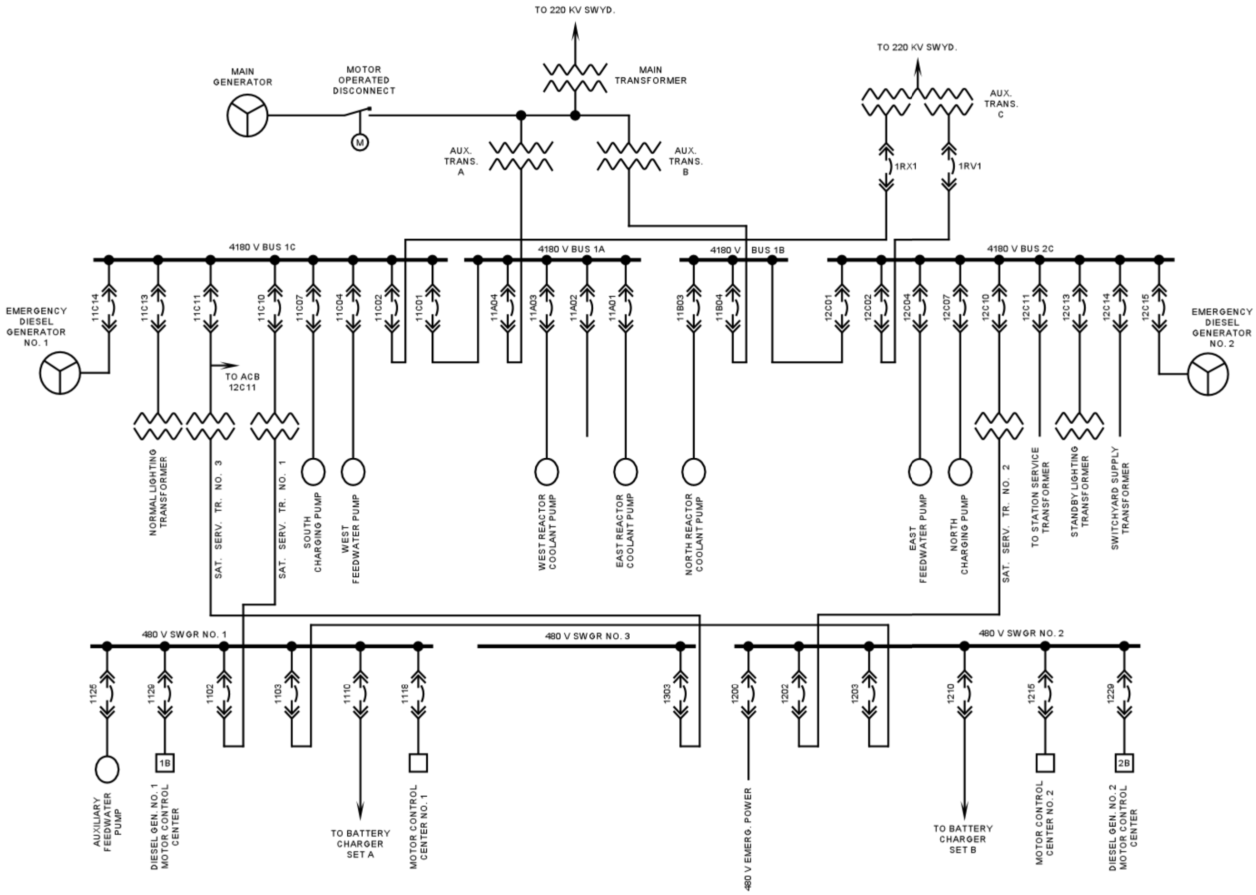


Figure 7.3-2 San Onofre Electrical System

At 4:51 a.m. on November 21, 1985, with the plant operating at 60 percent power, a ground fault was detected by protective relays associated with a transformer which was supplying power to one of two safety-related 4160-V electrical buses (see Figure 7.3-2). The resulting isolation of the transformer caused the safety-related bus to de-energize and tripped all feedwater and condensate pumps on the east side of the plant. The pumps on the west side of the plant were unaffected, since their power was supplied from another bus. The continued operation of the west feedwater and condensate pumps, in combination with the failure of the east feedwater pump discharge check valve to close, resulted in the overpressurization and rupture of an east-side flash evaporator low pressure heater unit. The operators, as required by emergency procedures dealing with electrical systems, tripped the reactor and turbine-generator. As a result, the plant experienced its first complete loss of steam generator feedwater and in-plant ac electrical power since it began operation.

The subsequent four-minute loss of in-plant electrical power started the emergency diesel generators (which by design did not load), deenergized all safety-related pumps and motors, significantly reduced the number of control room instruments available, produced spurious indications of safety injection system actuation, and caused the NRC red phone on the operator’s desk to ring. Restoration of in-plant electric power was delayed by the unexpected response of an automatic sequence that should have established conditions for delayed remote-manual access to offsite power still available in the switchyard.

The loss of steam generator feedwater was the direct result of the loss of power to the two main feedwater and one auxiliary feedwater pump motors, and the designed three-minute startup delay of the steam-powered auxiliary feedwater pump. The loss of the feedwater pumps, in combination with the failure of four additional feedwater check valves to close, allowed the loss of inventory from all three steam generators and the partial voiding of the long horizontal runs of feedwater piping within the containment building. The subsequent automatic start of feedwater injection by the steam-powered auxiliary feedwater pump did not result in the recovery of steam generator levels because the backflow of steam and water to the leak in the evaporator carried the auxiliary feedwater with it.

Later, operators isolated the feedwater lines from the steam generators, as required by procedure, which resulted in refilling the feedwater lines in the containment building. Before all feedwater lines were refilled, a severe water hammer occurred that bent and cracked one feedwater pipe in the containment building, damaged its associated pipe supports and snubbers, broke a feedwater control valve actuator yoke, and stretched the studs, lifted the bonnet, and blew the gasket of a four-in. feedwater check valve. The damaged check valve developed a significant steam/water leak, the second leak in the event.

Despite these problems, operators later succeeded in recovering level indications in the two steam generators not directly associated with the feedwater piping leak. With the re-establishment of steam generator levels, the operators safely brought the plant to a stable cold shutdown condition, without a significant release of radioactivity to the environment (an existing primary-to-secondary leak was not exacerbated) and without significant additional damage to plant equipment.

A brief description of how the SONGS-l mechanical and electrical systems involved in this event function and interact is provided. Understanding the major differences between this plant and more recently designed pressurized water reactors will clarify the basis for operator actions.

7.3.4 Plant Conditions Leading to Water Hammer

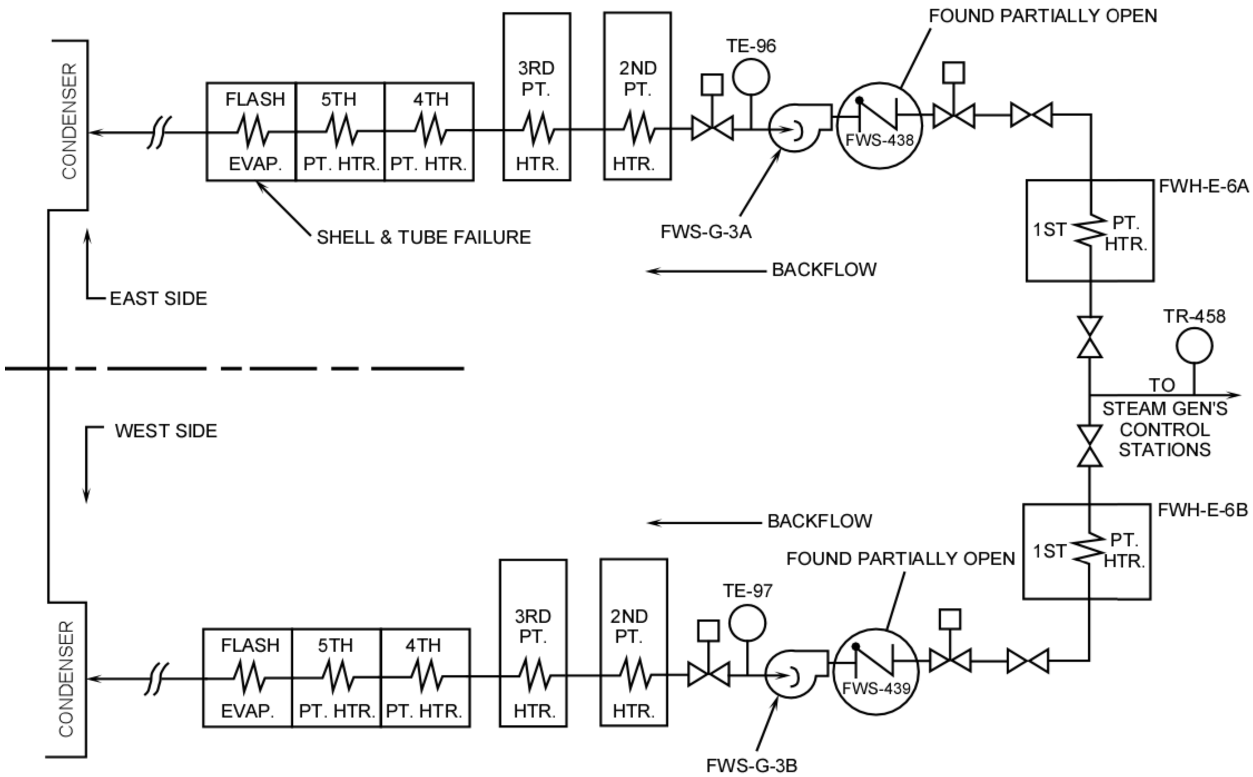


Figure 7.3-6 SONGS-1 Feedwater Flow Diagram

The plant conditions at SONGS-1 which led to a steam condensation-induced water hammer included the voiding of long horizontal lengths of feedwater lines, which allowed the backflow of steam from all steam generators before operators isolated the FW lines (by closing motor-operated valves MOV-20, 21, and 22), and the subsequent refilling of the FW lines with relatively cold (i.e., less than 100^0F) AFW. Figures 7.3-3, 7.3-4, 7.3-5, 7.3-6, 7.3-7 and 7.3-8 illustrate the flowpaths, valves and other equipment affected by this water hammer.

Upon detection of the fault on the C auxiliary transformer, relay protection de-energized 4.16-kV bus 2C, de-energizing east-side main feedwater (MFW) pump FWS-G-3A. The continued operation of west-side MFW pump FWS-G-3B, due to the unusual electrical alignment, combined with the failure of east-side MFW pump discharge check valve FWS-438 to seat, resulted in the overpressurization and failure of the east flash evaporator tube and shell. The subsequent unit trip de-energized the west-side MFW pump and denied power to electric-driven AFW pump AFW-G-IOS. With the cessation of flow to the steam generators, the failure of check valve FWS-438, and the failure of the check valves in the SG feedwater supply lines (valves FWS-346, FWS-345, and FWS-398), a path was provided for the blowdown of all three steam generators through their respective feedwater lines to the atmosphere through the failed flash evaporator.

The drop in the steam generator water levels following the unit trip initiated the AFW system, but the electric pump was de-energized, and steam-driven AFW pump AFW-G-10 took 3.5 minutes to deliver flow because of a programmed warmup period for the turbine. Thus, for three to four minutes no flow was being provided to the steam generators, and the leaking check valves permitted the horizontal feedwater lines to void. Further, the initiation of AFW flow at a rate of about 135 gpm from the steam-driven pump was not effective in halting the voiding, because flow was being carried away from the steam generators by the steam blowing down through the failed check valves in all three FW control stations and out the leak in the flash evaporator.

Following restoration of unit power, the motor-driven AFW pump started automatically, increasing the indicated AFW flow rate to a preset rate of 155 gpm per steam generator. However, all three steam generator levels continued to drop since the FW check valves remained open, the main steam system had not been isolated, and steam generator blowdown had not been isolated. Subsequently, in accordance with an emergency operating procedure for reactor trip response, operators isolated the failed FW check valves by shutting the three FW control isolation valves, MOV-20, 21, and 22, at approximately 4:55 a.m. Isolation of the feedwater trains occurred before the water hammer in the FW line to 8GB.

Subsequent to the isolation of the main FW lines, and recognition in the control room that both AFW pumps were delivering water, the operators became concerned about overcooling of the reactor coolant system and the decrease in pressurizer level. The operators decreased the AFW flows from 155 gpm to zero, and then increased them to 40 gpm. Refilling the FW lines downstream of the flow control stations was thus halted and then resumed at a much lower flow rate.

The slow refilling of the FW lines within the containment building continued from when AFW flow was first throttled to when the water hammer was reported to have occurred seven minutes later by a plant equipment operator. As noted previously, conditions conducive to steam condensation-induced water hammer in the feedwater lines were present for quite some time.

The gross failure of upstream check valves, which permitted water to drain from the feedwater lines and be replaced with steam, was the underlying cause for water hammer. Leaky check valves have been previously cited in reports of other water hammer occurrences. Five check valves are known to have been failed during the SONGS-1 event.

7.3.5 Water Hammer-Induced Damage

The following sections detail water hammer-induced damage to loop B feedwater piping and supports, to the loop B FW flow control station, and to the loop B AFW piping and describe the existing damage to feedwater system check valves.

7.2.5.1 Piping and Piping Support Damage

Damage to the loop B FW piping was confined to plastic yielding of the northeast elbow and to a visible crack on the outside of the pipe, extending approximately 80 inches axially. The crack penetrated approximately 30 percent of the pipe wall at its deepest point from the outside and approximately 25 percent on average. Damage to supports was severe in some instances. This section provides a description of the damage visible after the FW piping insulation was removed.

Figure 7.3-9 shows the loop B FW piping layout and identifies the piping support stations where damage occurred. This figure also provides directional orientation and indicates piping dimensions. Figure 7.3-10 shows principal areas of damage and indicates how the pipe moved.

The water hammer forces were sufficiently large to damage pipe supports and piping and to transmit loads through the containment building penetration structure outward to the loop B feedwater regulating station. No damage was evident to the steam generator B feedring or nozzle region that can be attributed to water hammer, nor was there evident damage to or movement of the piping between support HOOC and the steam generator B feedwater nozzle. Table 7.3-1 and Figures 7.3-9 and 7.3-10 illustrate the piping and support damage.

7.2.5.2 Feedwater Loop B Flow Control Station Damage

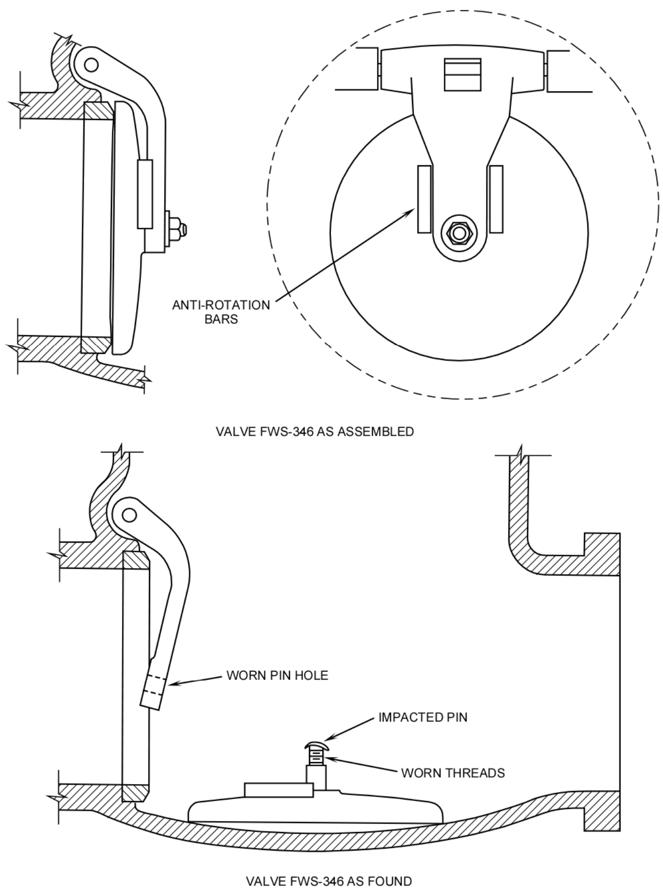


Figure 7.3-12 Check Valve FWS-346

Figure 7.3-11 shows the typical internal arrangement of a swing check valve. The water hammer originating in the feedwater line within the containment building generated a water slug which transmitted a pressure wave upstream to the loop B flow control station. Check valves FWS-346 and FWS-378, downstream of the control valves, were designed to prevent backflow, although post-event inspection revealed that the closure disk for FWS-346 (see Figure 7.3-12) was lying in the bottom of the valve chamber. Thus, any closed valve upstream of the check valve would be subjected to the water hammer loads. In addition to check valve FWS-378, flow control valve FCV-457 and motor-operated valve MOV-20 were subjected to the water hammer loads, because they had been closed by operators following the emergency operating procedures.

Because check valve FWS-378 was intact and operational, it was subjected to water hammer loads and absorbed much of the water hammer energy, whereupon the bonnet studs yielded and the gasket was forced outward against the studs. The failure of the gasket relieved much of the internal pressure, thereby minimizing damage to other equipment and valves at this station. Valve FCV-457 did incur damage to the flow actuator yoke and a bent valve stem.

7.2.5.3 AFW Piping Damage

The AFW injection points to the main feedwater piping at SONGS-1 lie in the "breezeway" upstream of the containment building steel shell. The AFW lines run horizontally and then vertically to tie into the main feedwater lines. Water hammer loads were imposed on AFW loop B piping. Although pipe movement extended several hundred feet upstream, there was no evidence of piping damage.

7.2.5.4 Valve Malfunctions

Post-event disassembly and examination of valves that contributed to water hammer conditions confirmed that check valve failures were the underlying causes for the occurrence of water hammer. Inspection findings identified the valve conditions listed in Table 7.3-2.

7.3.6 Valve In-Service Testing

The ASME Boiler and Pressure Vessel Code, Section XI, which specifies valve in-service testing (IST) requirements for valves like the SONGS-1 feedwater check valves, states:

> Valves shall be exercised to the position required to fulfill their function unless such operation is not practical during plant operation. Valves that cannot be exercised during plant operation shall be specifically identified by the owner and shall be full-stroke exercised during cold shutdowns. Full-stroke exercising during cold shutdowns for all valves not full-stroke exercised during plant operation shall be on a frequency determined by the intervals between shutdowns as follows: for intervals of 3 months or longer, exercise during each shutdown; for intervals of less than 3 months, full-stroke exercise is not required unless 3 months have passed since last shutdown exercise.

Additionally, the NRC staff position on cold shutdown testing of valves is as follows:

1. The licensee is to commence testing as soon as the cold shutdown condition is achieved, but not later than 48 hours after shutdown, and continue until complete or until the plant is ready to return to power.
2. Completion of all valve testing is not a prerequisite for returning to power.
3. Any testing not completed during one cold shutdown should be performed during any subsequent cold shutdowns, starting from the last test performed at the previous cold shutdown.

All feedwater system check valves are periodically tested in the closed position. The main and bypass feedwater regulating check valves are normally tested in cold shutdown (mode 5) and the feedwater pump discharge check valves are tested in hot standby (mode 3).

There are 121 valves that are subject to IST during cold shutdown. Although IST was performed during each outage, all of the valves were not tested. Consequently, the feedwater valves had been tested only one time since October 1984. The available opportunities for valve IST were not always fully utilized due to higher priority operational requirements.

Surveillance test procedures for verification of check valve closure for the main feed pump discharge check valves (FWS-438 and FWS439) require one main feed pump to be running while the other pump is stopped. The discharge valve at the idle pump is then opened and the pressure is monitored between the pump and its discharge check valve. An increase in pressure or an operator observation that the pump is rotating backwards would indicate that the check valve is not closed. While providing reasonable assurance of check valve closure, this testing method also subjects the low pressure pump suction piping to some relatively high pressures if the check valve fails to close (as in the November 1985 event), and thus damage is possible to such components as the flash evaporator. Testing with the idle pump suction valve shut would provide a more rigorous test.

Surveillance test procedures for verifying closure of other main feedwater check valves require testing to be performed during cold shutdown with the steam generators filled to a level above the feedrings. The motor-operated valve upstream of each check valve is closed, and the drain valve between this valve and the associated check valve is opened. The column of water in the steam generator provides approximately *4.5* psi of differential pressure across the valve to provide the closing force on the check valve disc. The procedure states that the section of piping between the motor-operated valve and check valve is to be drained, and that “little or no flow” from the drain should be verified. This test procedure leaves the surveillance operator to make the decision about how much flow is “little” and thus indicative of positive verification of check valve closure. The IST records do not provide a means of determining whether flow occurs or its extent, or for verifying complete valve cavity drainage before a determination is made that “little or no flow” has occurred.

Valves FWS-345 and FWS-346 failed the IST on February 24, 1985, when tested during mode 5 (cold shutdown). Maintenance work orders were prepared to repair both valves. However, on February 26, 1985, “Non-routine and Increased Frequency IST” was performed during mode 3 (hot standby), and the valves passed. During mode 3 the steam generator pressure increased the differential pressure available to seat the check valves (to approximately 700 psi) and thereby enabled them to pass. The work orders were then cancelled, and no corrective maintenance was performed.

7.3.7 Valve Failure Findings

Check valve failures caused by partial disassembly while in service do not appear to be unique to SONGS-1 or to the valve manufacturer (MCC Pacific). A limited review of licensee event reports (LERs) indicates that these valve failures are not unique.

Failures of FWS-438 and FWS-439, the main feed pump discharge check valves, may have been due to inadequate valve design, since the disc-retaining nut of each valve was not provided with a positive locking device that should have reduced the probability of the disc working loose, wedging into the valve seat, and failing open. Additionally, excessive clearances between the hinge and disc assemblies allowed the discs to rotate past the anti-rotation devices.

The failure of FWS-346, the B feedwater header check valve, may have been caused by the inadequate hardness of the disc-attaching stud, which allowed the threads to strip and the end to mushroom over, conditions contributing to the ultimate valve failure. However, the service conditions (i.e., flow-induced vibration) experienced by this valve may also have been a major contributor to failure. The failures of FWS-345 and FWS-398, the A and C feedwater header check valves, may have been due to similar service conditions.

The cracks in the seating surface of FWS378, the four-in, check valve in the B loop bypass line, appear to be service related. However, these cracks may have been caused by the significant forces on the valve from the water hammer.

Failure of the yoke of FCV-457, the loop B feedwater regulating valve, was probably due to lack of sufficient support or bracing of the valve operator during the pipe movement caused by water hammer loading.

7.3.8 Flash Evaporator Unit

During the event, the east condensate header was overpressurized, resulting in catastrophic failure of the east flash evaporator tubes and shell. The evaporator unit is in a shell which also houses two stages of low pressure feedwater heaters and drain coolers. The flash evaporators had not been used for several years, and extraction steam to them had been isolated. The evaporator condenser is part of the condensate system flowpath. The design pressure of the flash evaporator condenser and fourth- and fifth-point low pressure feedwater heater tubes is 350 psig, while the shell-side design pressure is 15 psig. The low pressure feedwater heaters were in service during the water hammer event.

When bus 2C was de-energized and the east main feed pump tripped, failed discharge check valve FWS-438 allowed the west main feedwater pump to pressurize the east condensate header.

This pressure caused a tube failure in the east evaporator condenser. The flash evaporator shell was subsequently overpressurized, resulting in the failure of the shell. After the loss of all in-plant ac power, the remaining (west) main feed pump coasted down, and the failed main feedwater regulating valve check valves (FWS345, 346, and 398) allowed backflow from all steam generators through failed valve FWS-438 to the failed tube in the east flash evaporator condenser. This backflow continued until the operators closed motor-operated feedwater header isolation valves MOV-20, 21, and 22, and main feedwater regulating valves FCV-456, 457, and 458.

Helium leak checks were performed on all east feedwater heaters, revealing no leakage beyond that expected from normal operation. The west feedwater heaters were leak tested before the unit was returned to service. The failure of the flash evaporator had no direct safety significance.

7.3.9 Turbine Breakable Diaphragms (Rupture Disks)

During the event, steam was observed issuing from the low pressure turbine breakable diaphragms. Each low pressure turbine has four breakable diaphragms designed to protect the turbine casing from overpressurization. The diaphragms, made of thin lead, are designed to break if the turbine exhaust pressure, normally subatmospheric, reaches 5 psig. The diaphragms are supported against external atmospheric pressure and normally seal the turbine casing against air in-leakage. All diaphragms were intact prior to the water hammer event.

Four of the diaphragms ruptured during the event, three on low pressure turbine 1 and one on low pressure turbine 2. Rupture of the diaphragms is not considered unusual for conditions existing after a loss of all ac power with continued energy addition into the main condenser, and is of no safety significance.

7.3.10 Summary

On November 21, 1985, Southern California Edison's San Onofre Nuclear Generating Station Unit 1, located south of San Clemente, California, experienced a partial loss of in-plant ac electrical power while the plant was operating at 60 percent power. Following a manual reactor trip, the plant lost all in-plant AC power for four minutes and experienced a severe incidence of water hammer in the feedwater system which caused a leak, damaged plant equipment, and challenged the integrity of the plant's heat sink. The most significant aspect of the event involved the failure of five safety-related check valves in the feedwater system. These failures appeared in less than a year, without detection, and jeopardized the integrity of safety systems. The event involved a number of equipment malfunctions, operator error, and procedural deficiencies.

TABLE 7.3-1 Description of Feedwater Pipe Damage Following SONGS-1 Water Hammer	
Support Locations	Description of Component, Damage, Motion. Etc.
HOOC HOOB HOOA	This snubber station, the closest to the SG B, showed no visible damage or pipe movement. The feedwater pipe turns vertically, and at an angle, to rise approximately 10 feet to mate with the SG feedwater inlet nozzle.
HOOD HOO5 HOO6	These support stations were the first that showed damage (or movement) caused by water hammer. Dent in pipe that resulted when the pipe hit the concrete corner and then rebounded.
HOOG	Movement of approximately 12 inches, slippage of vertical support pads off channel beam structures and downward drop of FW pipe.
HOOH	Horizontal and vertical support pads displaced southward approximately 12 inches.
120	Evidence of first lateral motion (eastward); deformed vertical structure, and then axial rebounding which displaced pipe supports approximately 12 inches southward.
HOOK	Damage incurred at the support structure downstream of the southeast elbow. The damage incurred by the structure illustrates the magnitude of pipe motion which occurred during the water hammer pulse.
HOOL	Lateral movement (westward) of pipe which resulted in sheared vertical support structure. Concrete and support plate damaged by water hammer, nuts were loosened and bolts were missing in wall plates.
HOOM	Piping and support damage just downstream of where FW B line takes a 90-degree bend to exit the containment building.

TABLE 7.3-2 Inspection Findings		
Valve	Description	As Found
FWS-345 SG A	MFW Reg Check	Disc separated from hinge arm, disc stud broken (threaded portion).
FWS-346 SG B	MFW Reg Check	Disc separated from hingearm, disc stud deformed.
FWS-398 SG C	FW Reg Check	Disc nut loose. Disc partially open. Disc caught inside of seat ring.
FWS-438	FWP Discharge Check	Disc nut loose. Disc partially open. Disc caught inside of seat ring. (Figure 7.3-13)
FWS-439	FWP Discharge Check	Disc nut loose. Disc partially open. Anti-rotation lug lodged under hinge arm.

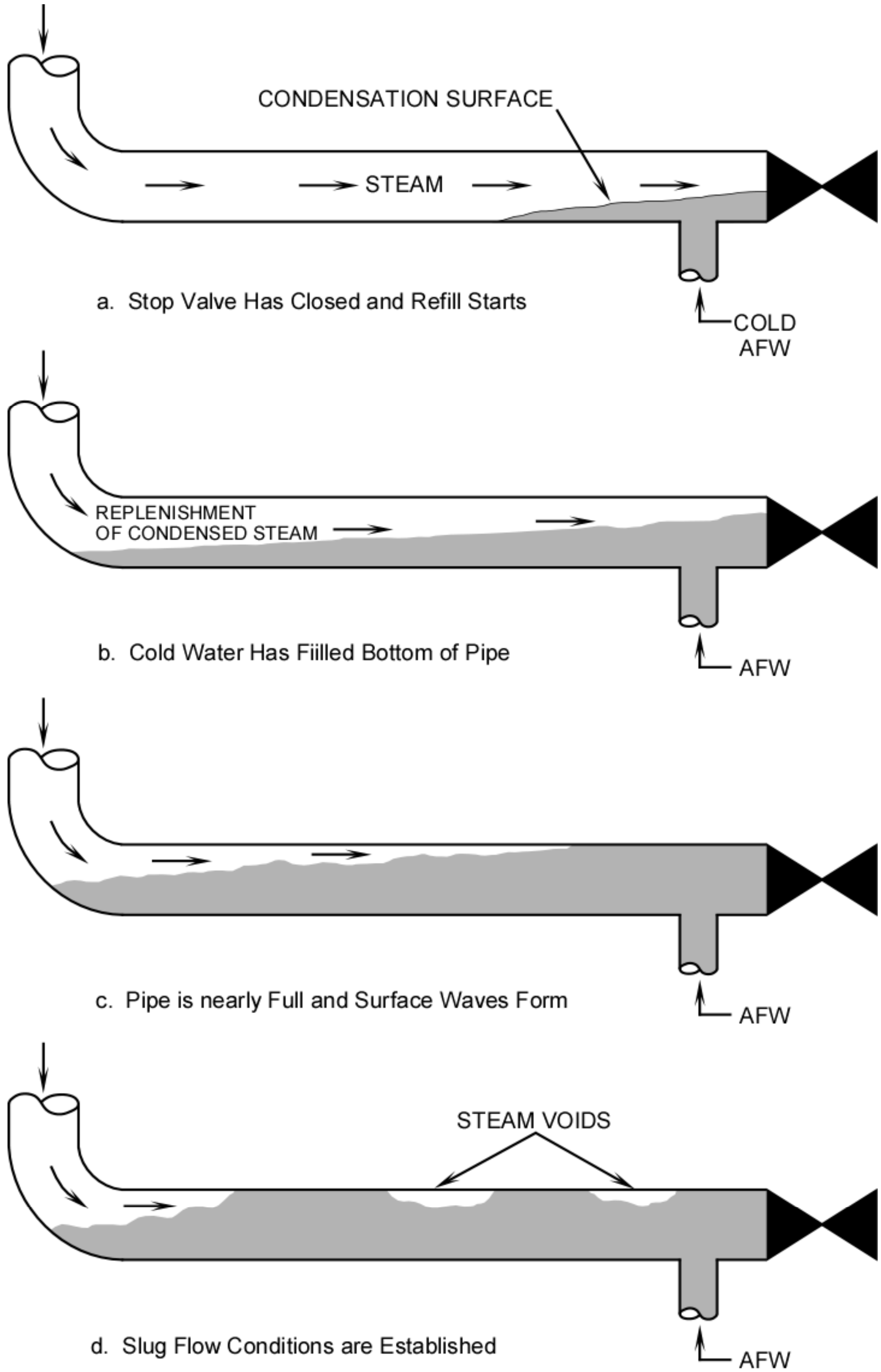


Figure 7.3-1 Filling of a Voided Feedwater Line

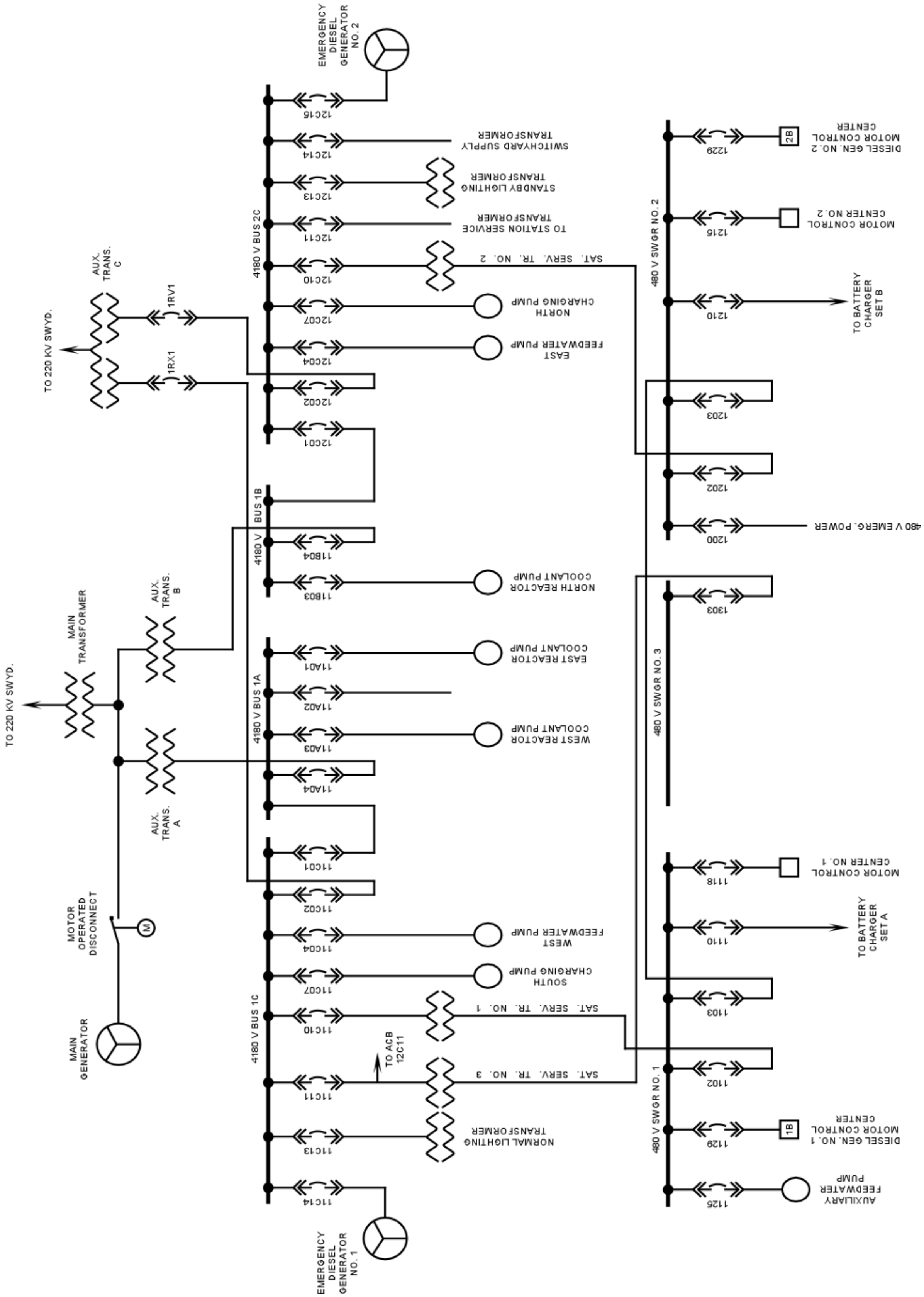


Figure 7.3-2 San Onofre Electrical System

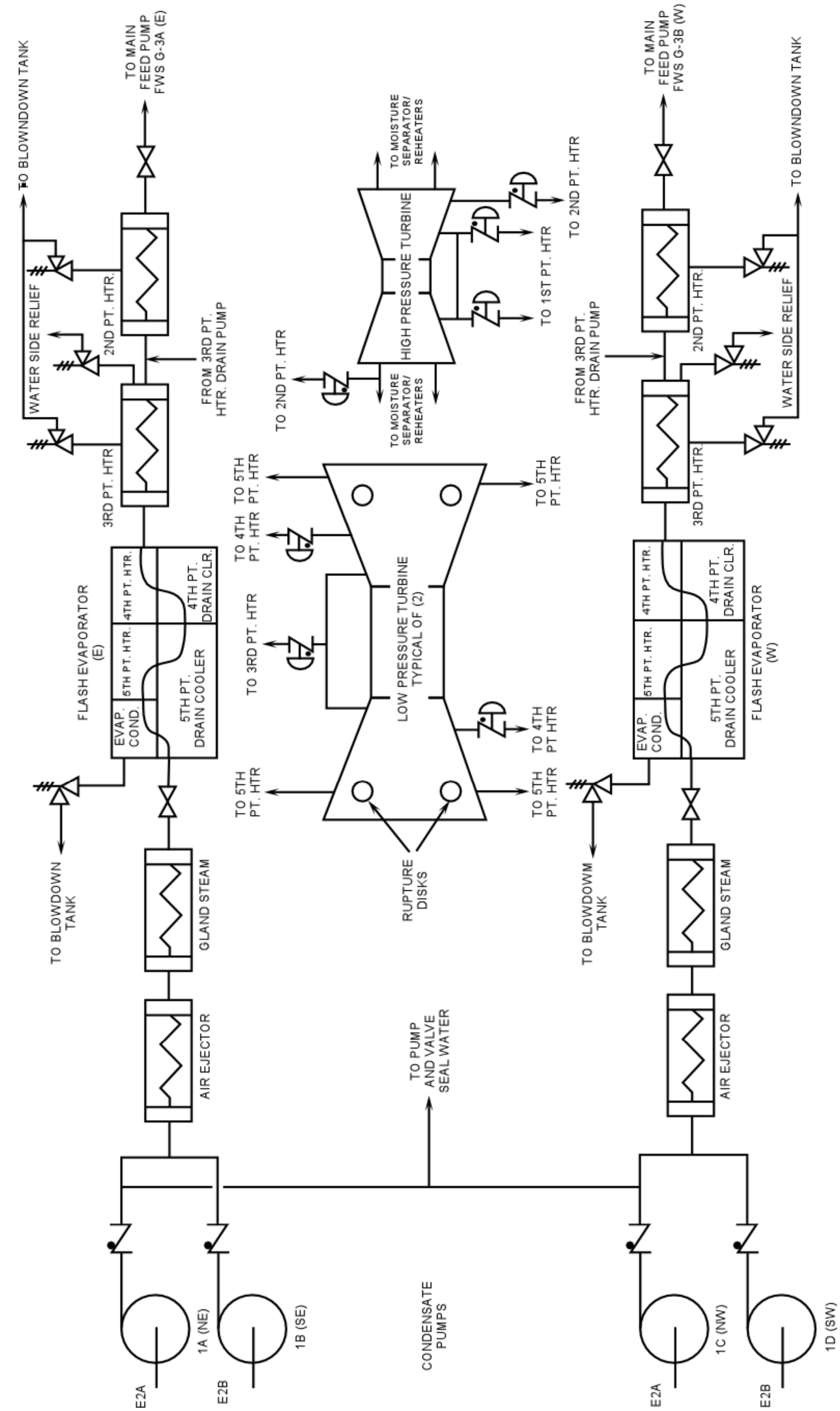


Figure 7.3-3 Condensate System

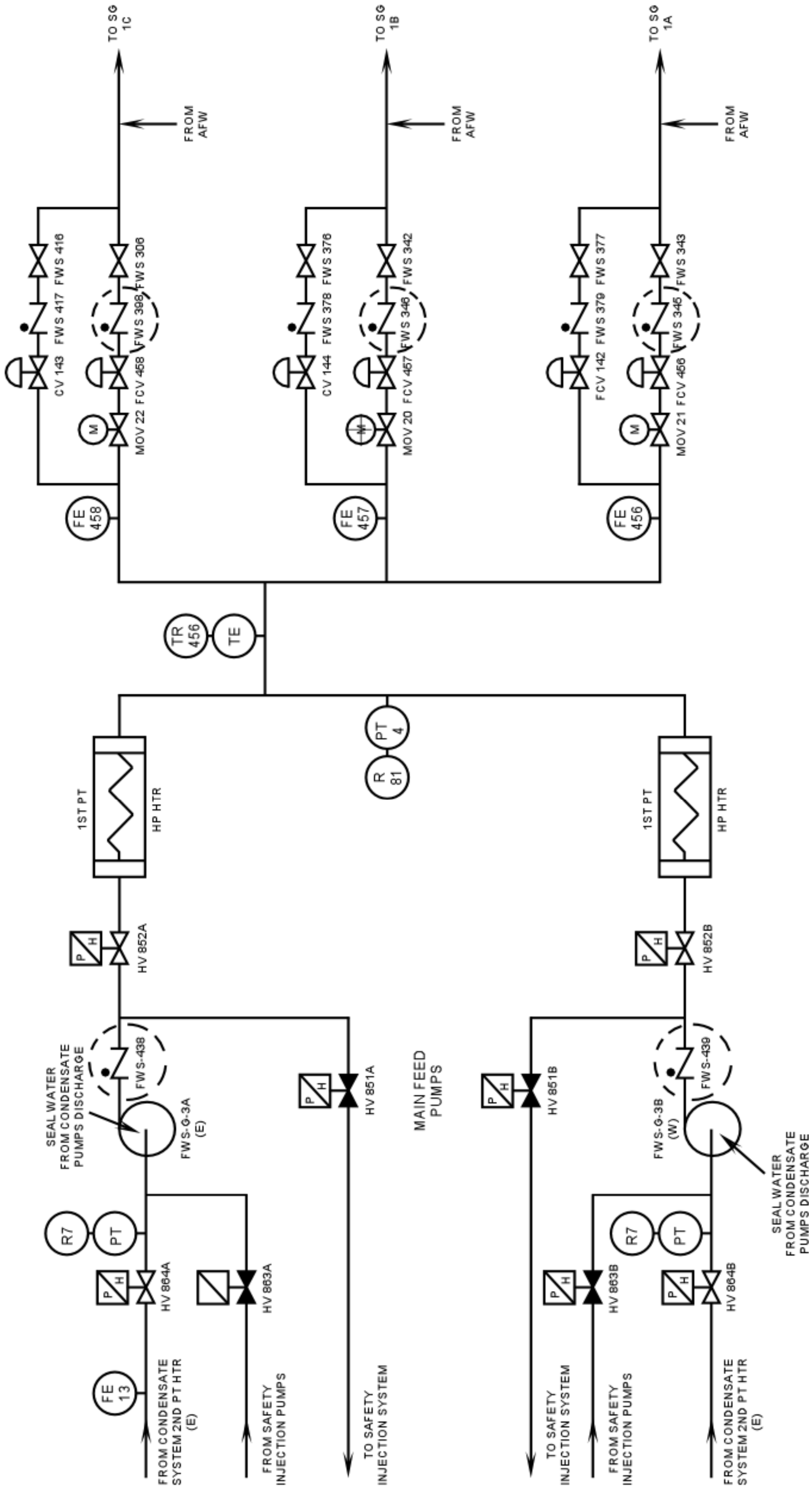


Figure 7.3-4 Main Feed System

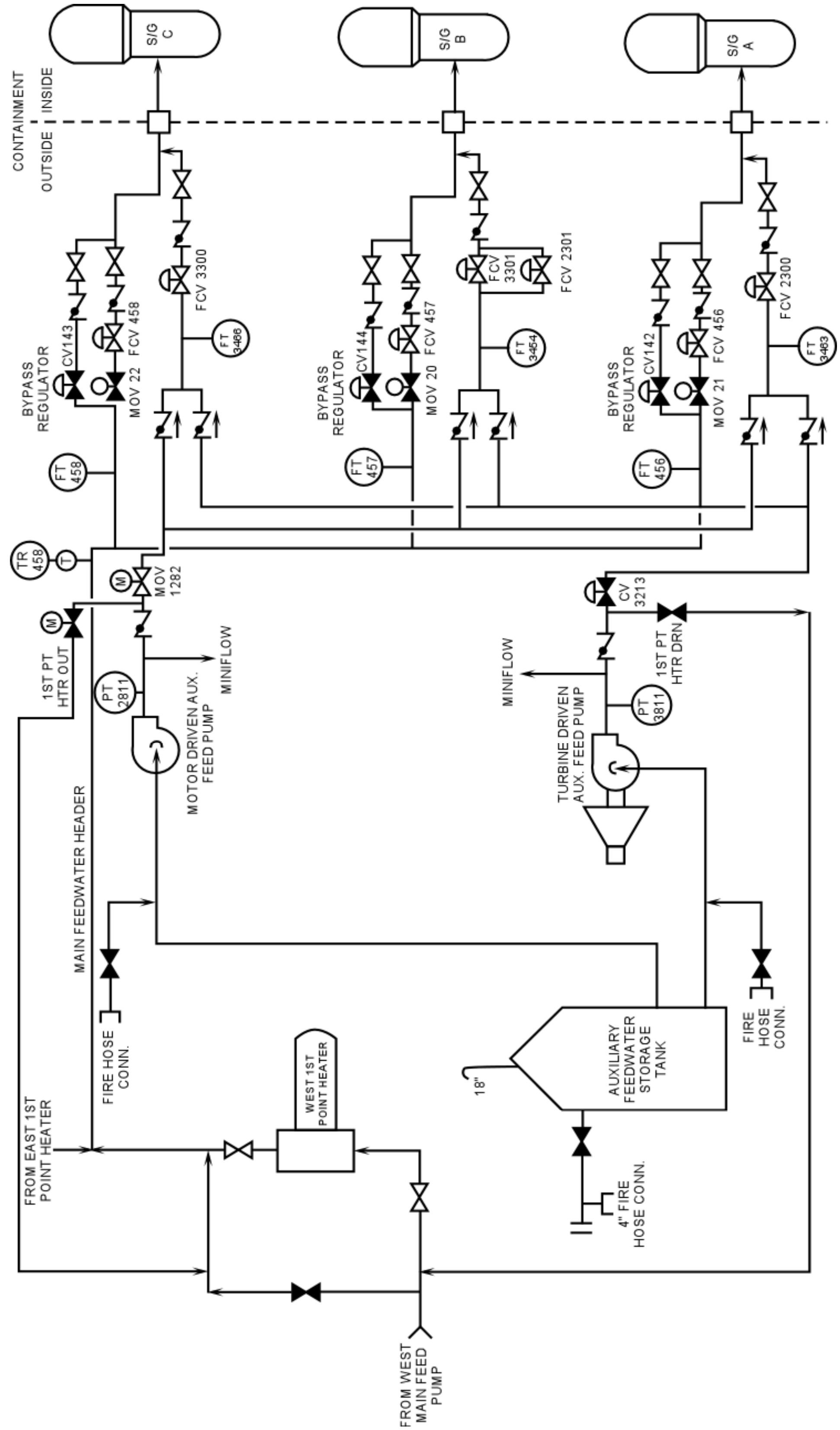


Figure 7.3-5 Auxiliary Feedwater System

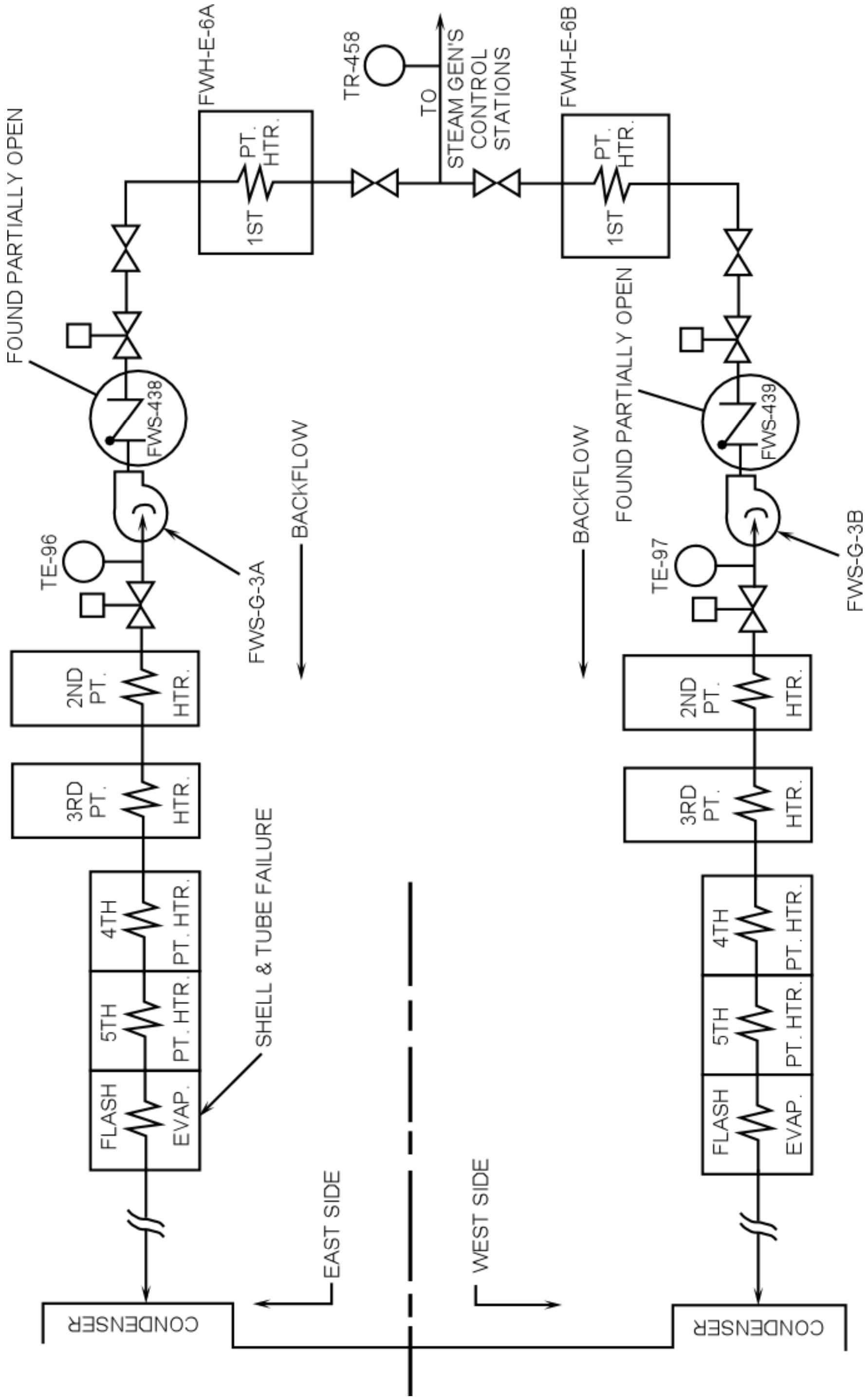


Figure 7.3-6 SONGS-1 Feedwater Flow Diagram

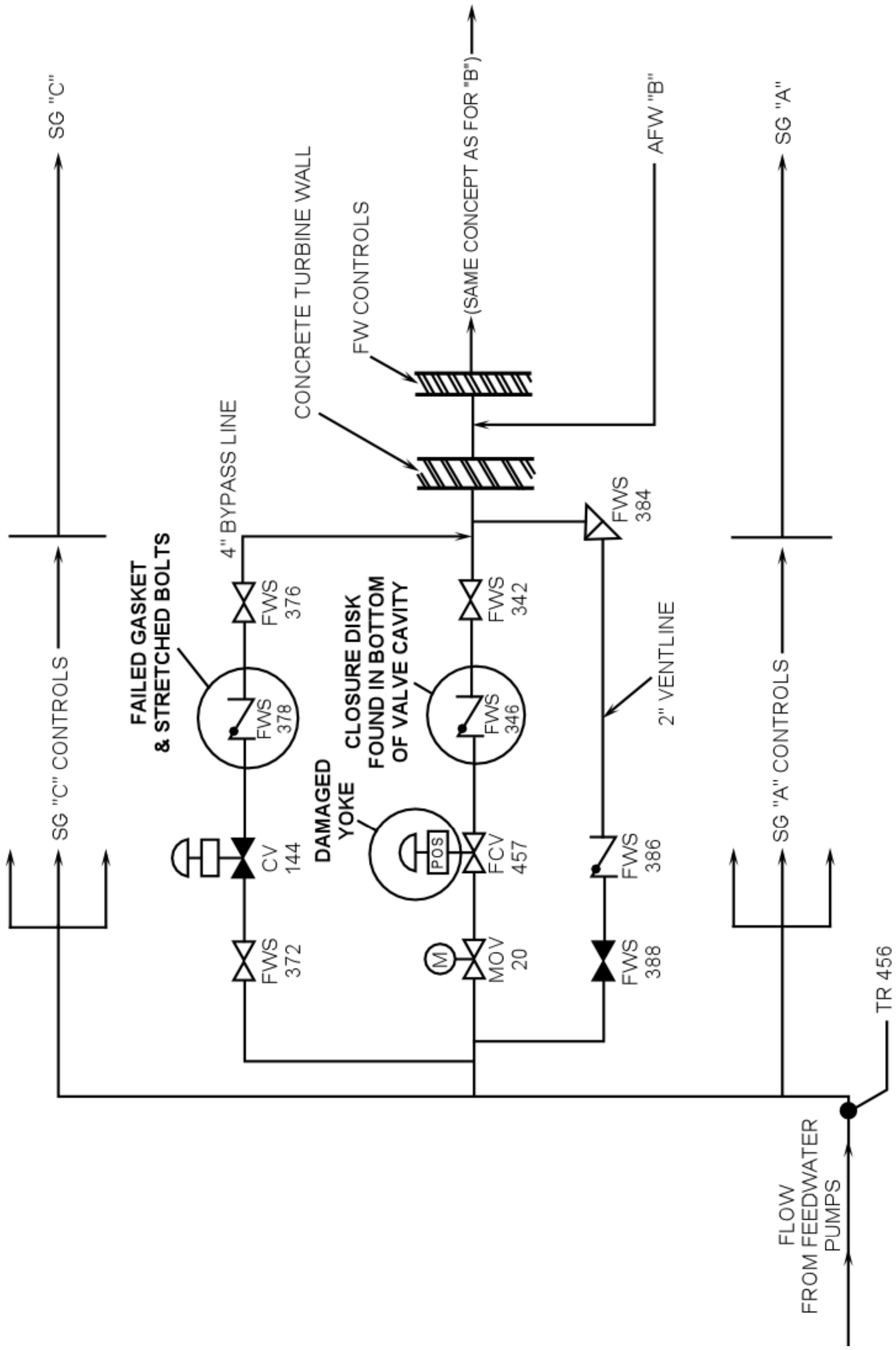


Figure 7.3-7 SONGS-1 Loop B Steam Generator Flow Control Station

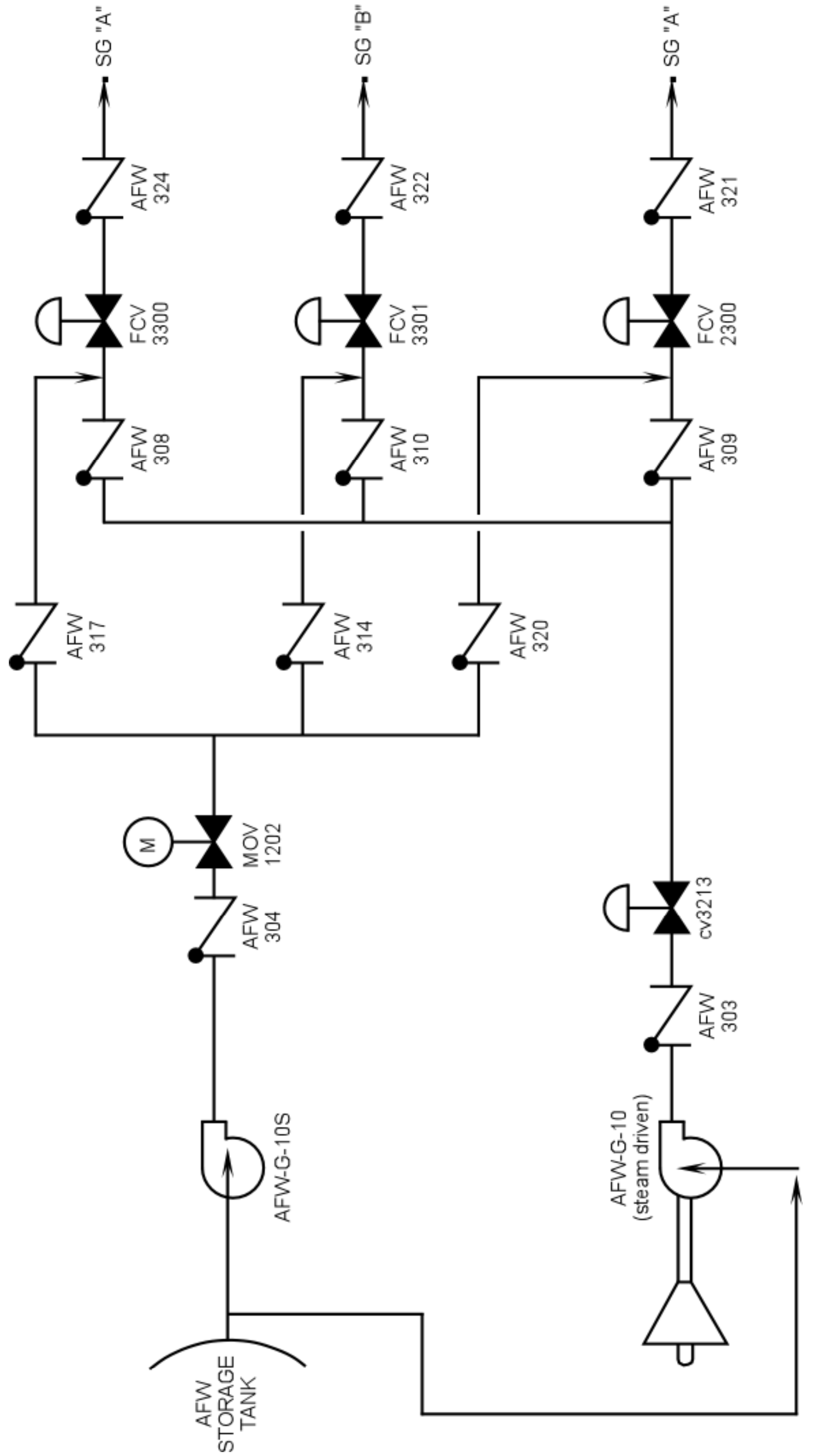


Figure 7.3-8 SONGS-1 Auxiliary Feedwater System

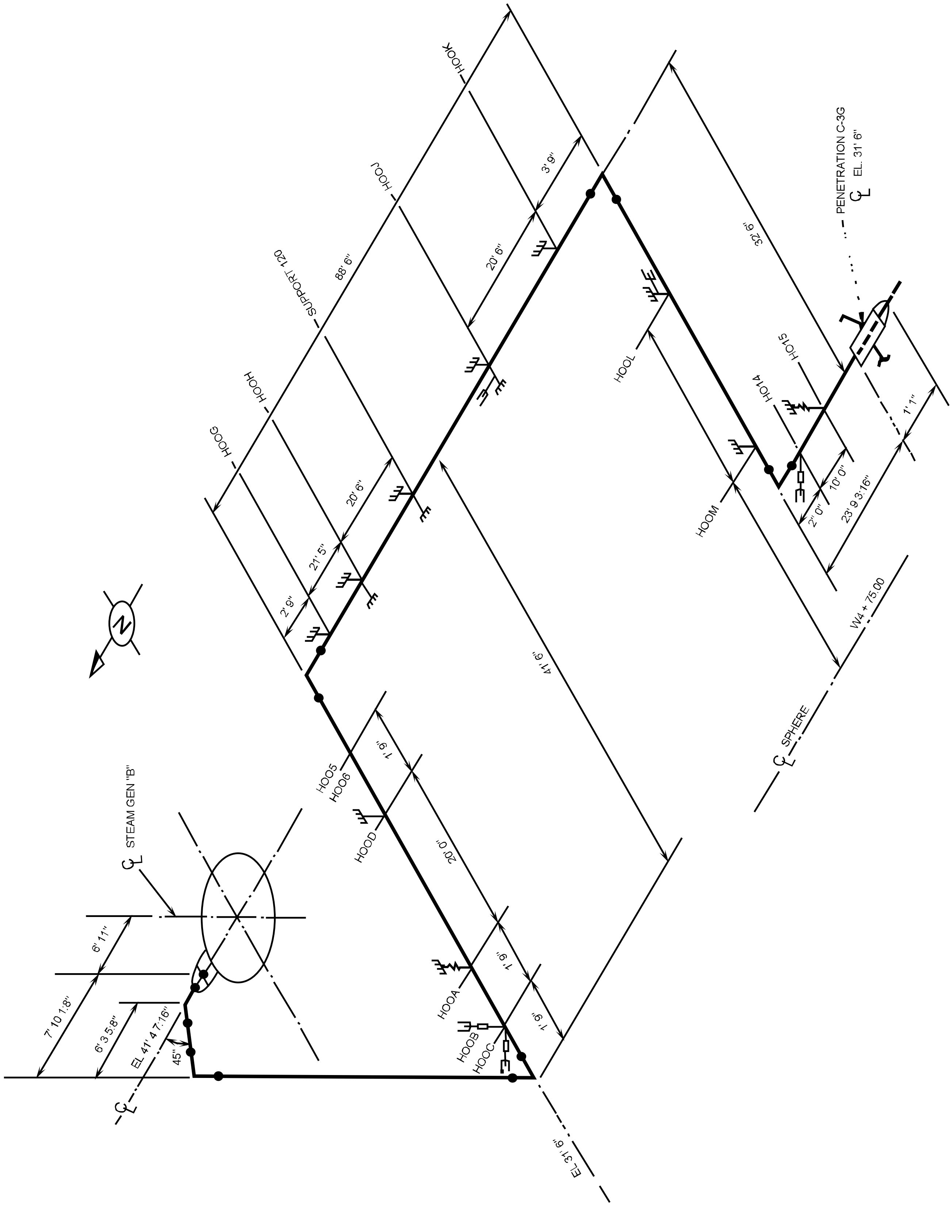


Figure 7.3-9 FW Loop B Piping and Support Layout

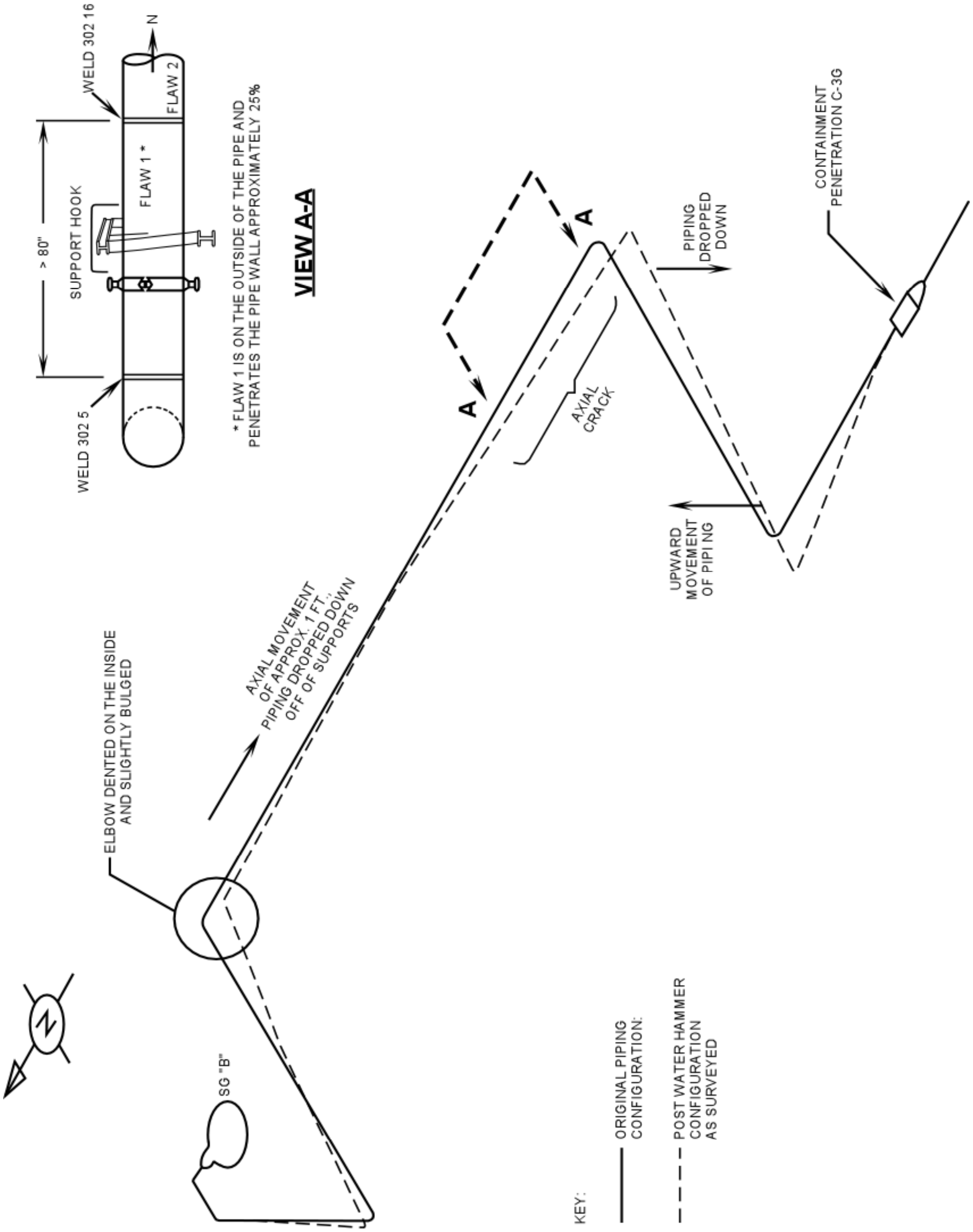


Figure 7.3-10 Overview of Feedwater Piping and Support Damage Due to Water Hammer

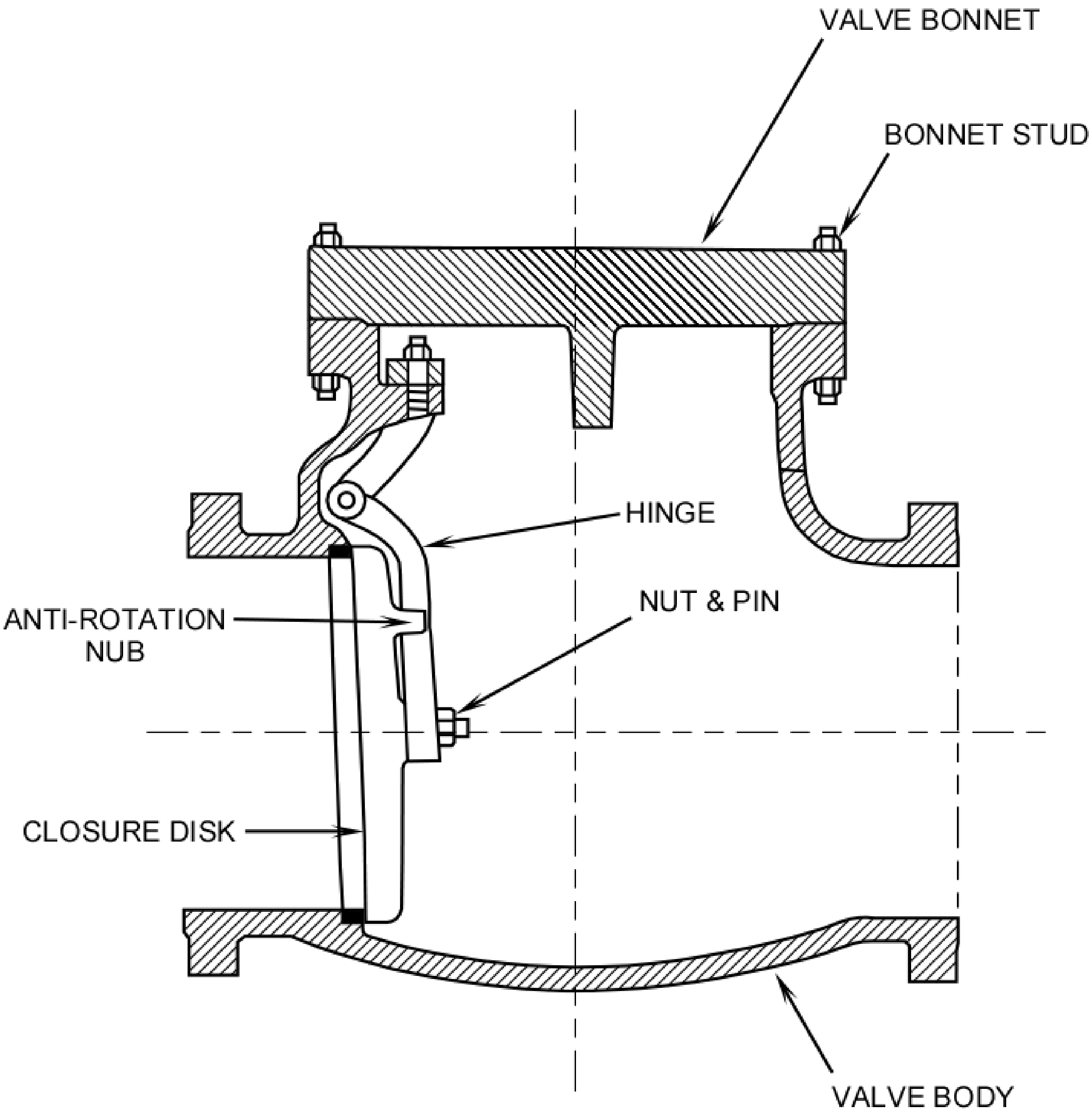
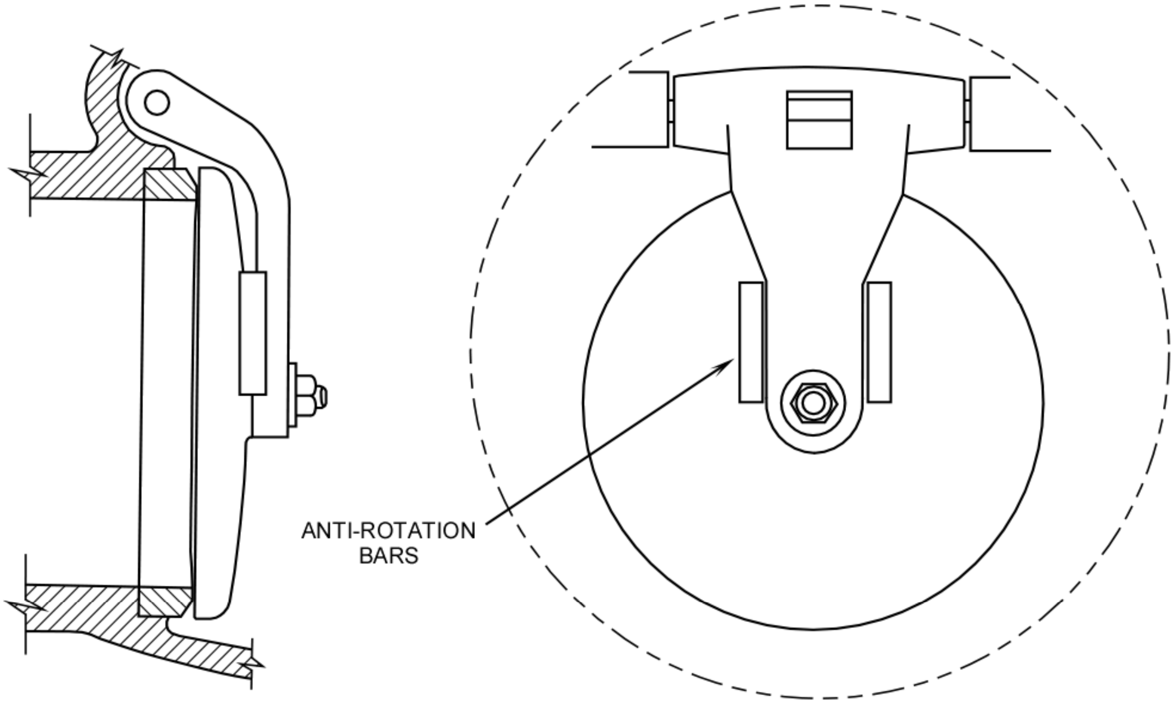
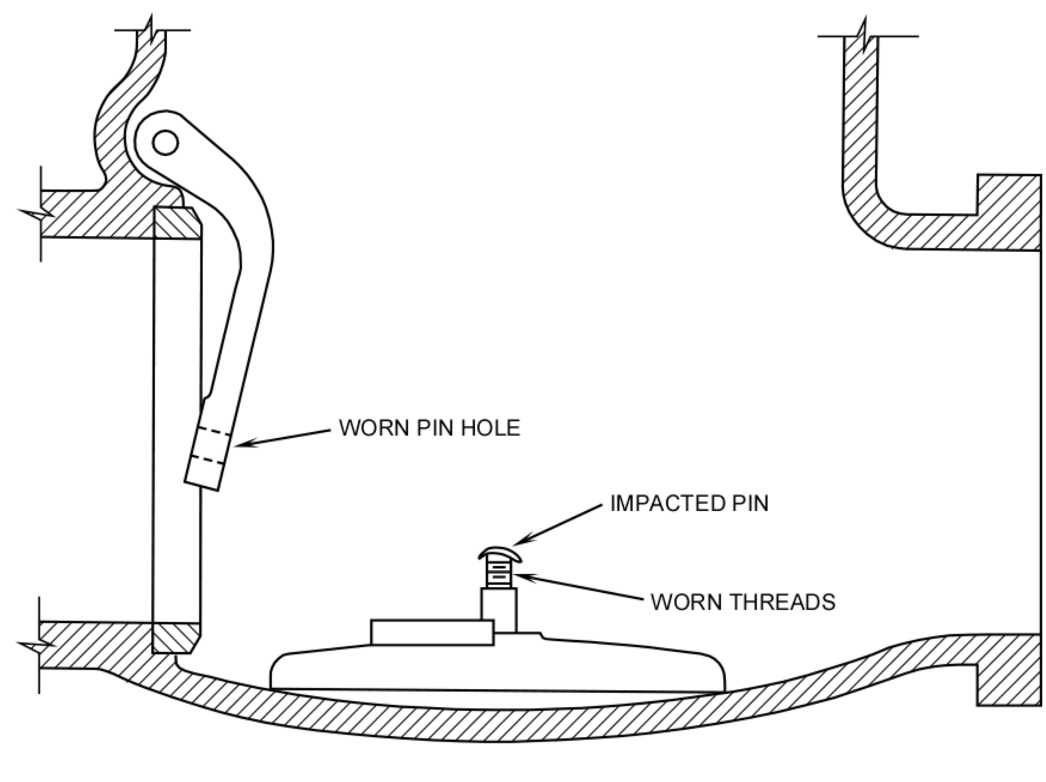


Figure 7.3-11 Typical Swing Check Valve

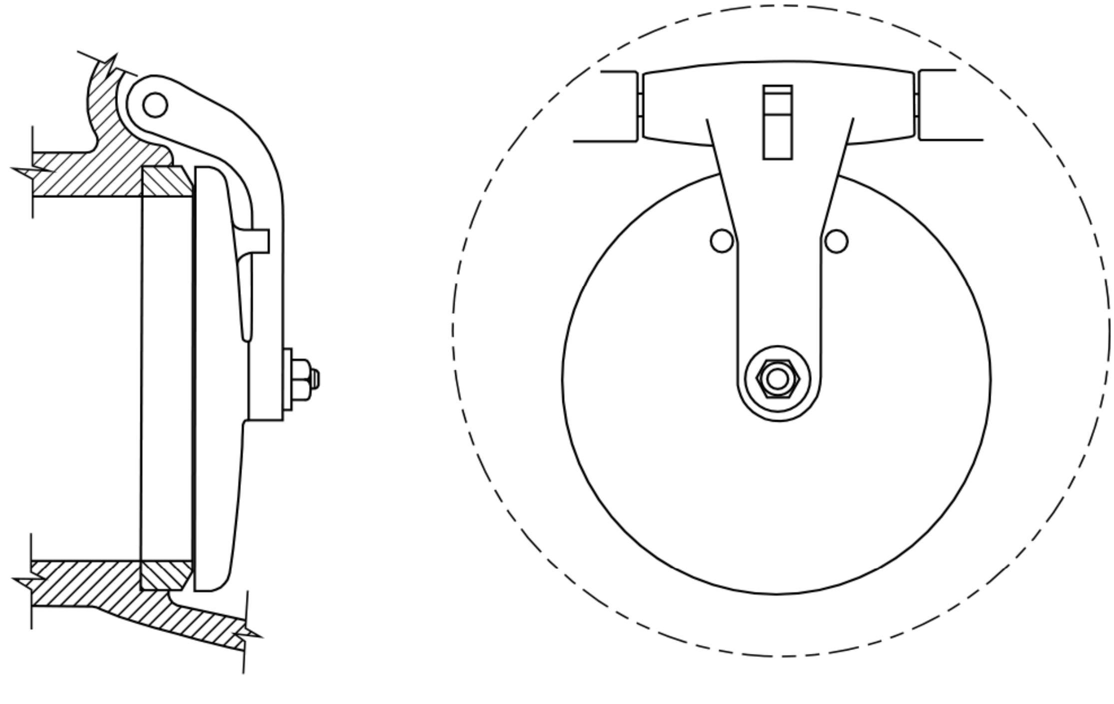


VALVE FWS-346 AS ASSEMBLED

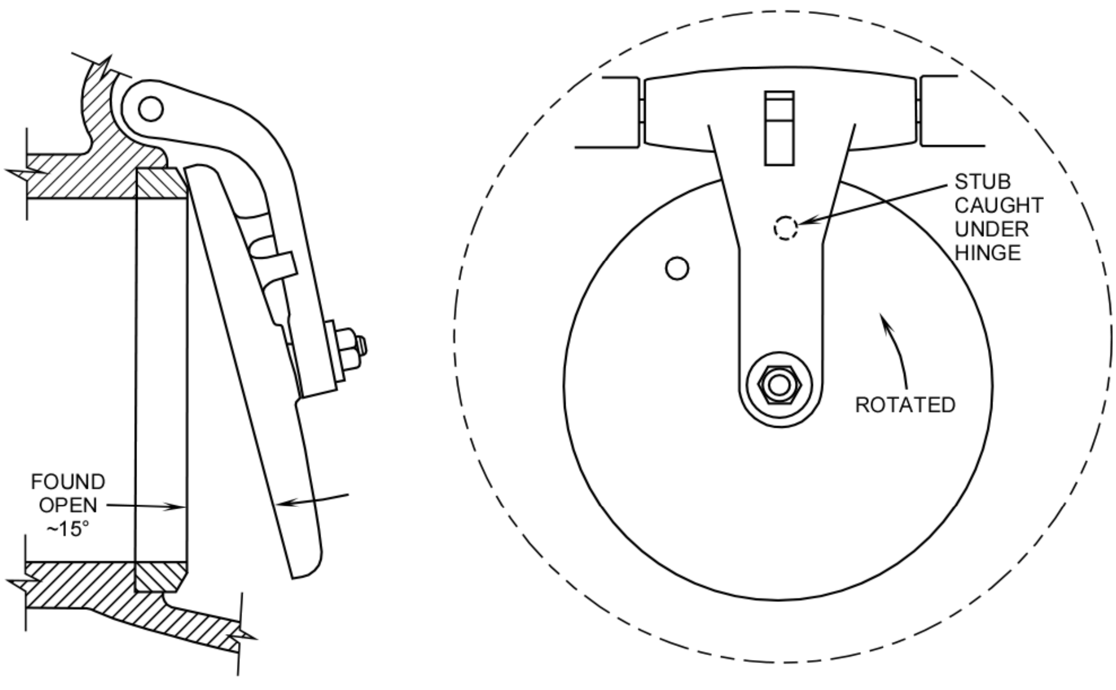


VALVE FWS-346 AS FOUND

Figure 7.3-12 Check Valve FWS-346



VALVE FWS-438 AS ASSEMBLED



VALVE FWS-438 AS FOUND

Figure 7.3-13 Check Valve FWS-348

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