SECTION 3

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SECTION 3

3.0 DESIGN CRITERIA – STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

The extent to which the design criteria for the station structures, systems, and components important to safety meet the NRC “General Design Criteria for Nuclear Power Plants,” specified in Appendix A of 10CFR50, is discussed in Appendix 3D.

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

The structures, components, and systems which are required to avoid or mitigate the consequences of abnormal operational transients or accidents are classified in this section.

3.2.1 Seismic Classification

The USAR terminology used for defining the earthquakes and the seismic classification of the equipment is the same as was used in the PSAR and FSAR. This has been done to provide continuity between the various rationale, criteria, and design commitments made in the PSAR, FSAR, and the final design evaluations presented in the USAR.

The Maximum Probable Earthquake is 0.08g. It is the conservatively determined earthquake and associated ground motion that might reasonably or probably be expected to occur at the nuclear plant site. The Maximum Probable Earthquake is similar to the Operating Basis Earthquake (OBE) terminology presently being used by the NRC.

The Maximum Possible Earthquake is 0.15g. It is the conservatively determined earthquake and associated ground motion which could conceivably or possibly occur at the site. The Maximum Possible Earthquake is similar to the Safe Shutdown Earthquake (SSE) terminology presently being used by the NRC.

3.2.1.1 Definitions

Class I structures, systems and components for seismic design purposes are defined (in General Design Criterion 2 of Appendix A to 10CFR50, Appendix A to 10CFR100, and NRC Reg. Guide 1.29 [Rev. 2, 2/76]) as those structures, systems, and components important to safety that are designed to remain functional in the event of a Maximum Possible Earthquake. These structures, systems, and components are those necessary to ensure:

a. The integrity of the reactor coolant pressure boundary.

b. The capability to shut down the reactor and maintain it in a safe shutdown condition, or

c. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10CFR100.

Class I structures, systems and components are also defined as nuclear safety related (Q) and are relied upon to remain functional during design basis events. Additionally (for piping) Seismic Class I design requirements extend to the first seismic restraint beyond the nuclear safety related boundary. This seismic classification complies with Regulatory Guide 1.29 (Revision 2, February 1976).

Class II structures, systems, and components are defined as those structures, systems, and components which are not classified as Class I. These structures, systems, and components are designed in accordance with the Uniform Building Code, 1967 edition for Seismic Zone 1 loads. Load combinations and allowable stresses for Class II structures are described in Subsections 3.8.1.3.2 and 3.8.1.5, respectively.

Class II structures, systems and components whose continued function is not required but whose failure could reduce the functioning of any safety-related structure, system or component to an unacceptable safety level and are designed and constructed so that a Maximum Possible Earthquake will not cause failure. These structures, systems and components are identified as Seismic II/I.

3.2.1.2 Seismic Class I Structures, Systems, and Equipment

Structures, systems, and equipment classified as Seismic Category I are those which are essential to:

1) Prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary,

2) Provide immediate and long-term operation following a Maximum Possible Earthquake,

3) Permit shutdown of the reactor and maintain it in the safe shutdown conditions, and

4) Prevent the uncontrolled release of radioactivity.

Class 1E electric equipment is Seismic Category I equipment. Seismic analysis classification and design loadings of Seismic Class I structures are presented in Table 3.2-1.

The following are Seismic Category I equipment and systems:

Rector vessel and internals including control rods and control rod drives

Reactor coolant system components (steam generators, pressurizer, pumps, etc.) and interconnecting piping and supports

All piping and connections to the reactor coolant system, including the second isolation valve

Containment isolation valves and penetrations

Main steam and feedwater piping from steam generator to and including the isolation valve

Atmospheric vent, main steam safety valves, and associated piping

Auxiliary feedwater turbines, pumps, and system

Spent fuel storage facilities, including spent fuel pool equipment

Emergency Ventilation System

The following are Seismic Category I equipment and systems: (Continued)

Post-LOCA Hydrogen Purge Dilution System

Containment Air Recirculation System

Containment vessel polar crane (unloaded condition)

Station 250/125 volt battery

Service Water System, parts which service Class I systems

Component Cooling Water System serving the engineered safety features only

Containment Spray System

ECCS Room Cooling Coils

Containment air cooling units

Low Pressure Injection and Decay Heat Removal Sysem

Containment Vessel Vacuum Breaker System

Core Flooding System

High-Pressure Injection System

4160-V Switchgear (Essential Buses)

480-V Unit Substations (Essential Buses)

480-V Motor Control Centers (Associated with Engineered Safety Features)

Inverters, including 480 VAC/125 VDC rectifiers, 125 VDC to 120 VAC Essential Instrumentation Buses

Essential Instrumentation Bus Panels

Engineered Safety Features Actuation System Cabinets

Reactor Protection System Cabinets

Safety Features Actuation Systems Control Panel

Reactor Manual Trip

Containment Vessel Penetration Assemblies

The following are Seismic Category I equipment and systems: (Continued)

Power Cable Systems (Associated with Engineered Safety Features)

Instrumentation and Control Cable Systems (Associated with Engineered Safety Features)

Emergency Diesel Generators and their Associated Equipment

Steam Generator Blowdown Piping from Steam Generators to Auxiliary Building Boundary

Reactor Vessel Head Vent

3.2.1.3 Partially Class I Structures and Systems

All Class I structures are separated by an expansion joint from all Class II structures. Class I equipment and systems located in Class II structures have reinforced concrete enclosures designed to withstand the loads for Class I structures. Accordingly these Class II structures are designated as partially Class I structures.

3.2.2 System Quality Group Classification

Quality group classification complies with Regulatory Guide 1.26 (Revision 3, February 1976) as described below.

Table 3.2-2 delineates the system Quality Group classifications of each component of those fluid systems that are required to prevent or mitigate the consequences of accidents or malfunctions within a reactor coolant pressure boundary or to permit safe shutdown of the reactor and maintenance with safe shutdown condition. The containment structure pressure boundary is shown in Figure 3.2-1.

The system piping and instrument diagrams for the fluid systems delineate, with the symbol “Q”, the boundary of all “Q” listed components.

The extent and configuration of overpressure protection provided for systems and components relative to referenced codes and standards are illustrated on the piping and instrumentation diagrams.

TABLE 3.2-1

Seismic Analysis Classification and
Design Loadings for Seismic Class I Structures

Structures	*General Type of Seismic Analysis	Loadings		
		Earthquake	Tornado Wind and/or Pressure	Missile Tornado and/or Internal
CONTAINMENT				
Shield Building	Type I and II	X	X	X
Containment Vessel	Type I and II	X	X	X
Containment Vessel Penetrations	Type II	X	X	X
Containment Vessel Interior Structures	Type I and II	X		X
Fuel Transfer Tubes	Type I	X		
Class I Equipment Supports	Type II	X		
AUXILIARY BUILDING				
Exterior Structure	Type I and II	X	X	X
Interior Structures	Type I and II	X	X	X
Penetration Rooms	Type I and II	X	X	X
Station Control Rooms	Type I and II	X	X	X
Spent Fuel Storage Pool Area	Type I and II	X	X	X
Cable Spreading Room	Type I	X	X	X
Switchgear Rooms	Type I and II	X	X	X
Emergency Diesel Generator Rooms	Type I and II	X	X	X
Station Battery Room	Type I and II	X	X	X
Class I Equipment Supports	Type II	X		

TABLE 3.2-1 (Continued)

Seismic Analysis Classification and
Design Loadings for Seismic Class I Structures

Structures	*General Type of Seismic Analysis	Loadings		
		Earthquake	Tornado Wind and/or Pressure	Missile Tornado and/or Internal
INTAKE STRUCTURE				
Service Water Pumps Enclosure	Type I	X	X	X
Service Water Pipe Tunnel	Type I	X	X	X(1)
Valve Rooms Number 1 and 2	Type I	X	X	X
YARD STRUCTURES				
Borated Water Storage Tank	Type I	X		
Borated Water Storage Tank Foundation	Type I	X		
Emergency Diesel Fuel Oil Storage Tanks	Type I	X		X(1)
Electrical Manholes (3001, 3004, 3005, 3006, 3020, 3041 and 3042)	Type I	X	X(2)	X(2)

*Earthquake Analysis

Type I – Response Spectrum Analysis

Type II – Time History Analysis

(1)Underground Structure Protected by Fill Material

(2)Excluding 3006 and 3020 (See Section 3.3)

TABLE 3.2-2

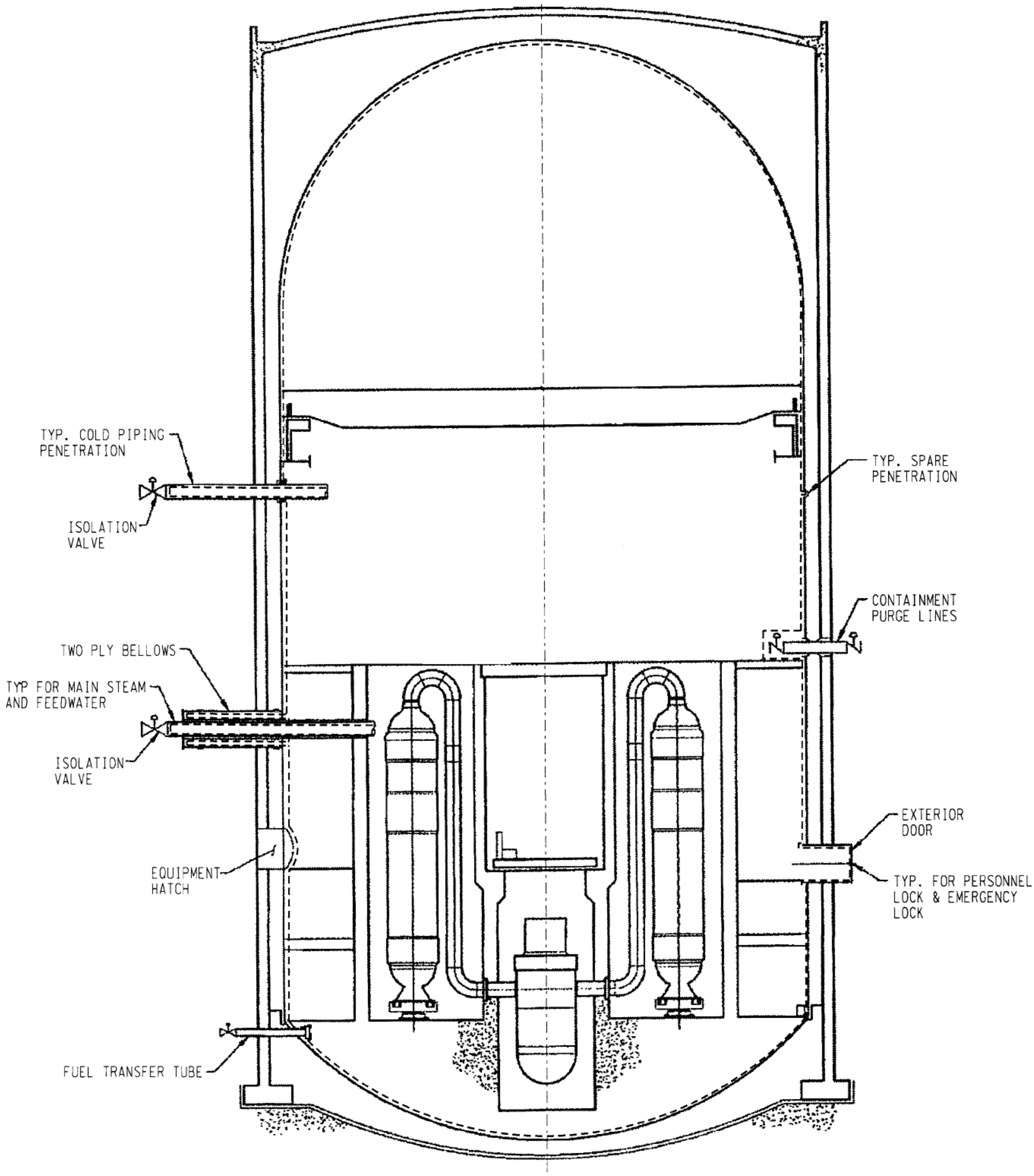
Code Classification

Component	Group A	Group B	Group C	Group D
Pressure Vessels	Reactor Vessel – ASME III, A, 1968 Steam Generator – ASME III, Class I, 2001 Edition, 2003 Addenda Pressurizer – ASME III, A, 1968 Reactor Coolant Pumps (casings) ASME III, A, 1968	Letdown Coolers – ASME 1977 Core Flooding Tanks – ASME III, C, 1968 Seal Return Coolers – ASME III, Class 2, 1971 Decay Heat Coolers – ASME 1968	Component Cooling Heat Exchangers – ASME VIII, 1968 Waste Gas Surge and Decay Tanks – ASME III, C, 1968	
Atmospheric Storage Tanks		Borated Water Storage Tank – AWWA D100	Component Cooling Surge Tank – ASME VIII, 1968	Emergency Feedwater Facility Diesel Oil Tank- API-620, 2013, NFPA-30, 2012
0-15 psig Tanks			Diesel Oil Day Tanks – ASME III, Class 3, 1971	
Piping	Original Reactor Coolant System Boundary – B31.7, Class I; ASME III Class I, 1971 Replacement Reactor Coolant System Boundary – ASME III Class I 2001 Edition, 2003 Addenda	Main Steam & Feedwater Systems from Steam Generator to Contain- ment Isolation Valves – ASME III, Class 2, 1971** L.P. Injection System – ASME III, Class 2, 1971 H.P. Injection Systems – ASME III, Class 2, 1971 Containment Spray System – ASME III, Class 2, 1971 Core Flooding System ASME III, Class 2, 1971	Component Cooling System – ASME III, Class 2 & 3, 1971 Service Water System ASME III, Class 2 & 3, 1971 Auxiliary Feedwater Systems – ASME III, Class 2, 1971**, B31.1, 100% radiography Emergency Feedwater System (Q portion)- ASME III, Class 2, 1971 (6”-EBB, 3”-EBB), B31.1 Critical, 100% radiography (3”-EBD),	Emergency Feedwater System (non-Q portion)- B31.1, 1971

TABLE 3.2-2 (Continued)

Component	Code Classification Group A	Group B	Group C	Group D
		Steam Generator Blowdown System from Steam Generators to Containment Isolation Valves – ASME III, Class 2, 1971		
	Reactor Vessel Head Vent ASME III Class 1, 1971			
Pumps	Reactor Coolant Pumps – ASME III, A as applicable	H.P. Injection Pumps – Draft Pump and Valve code, Class 2, 1968 Decay Heat Pumps – Draft Pump and Valve Code, Class 2, 1968 Containment Spray Pumps – Draft Pump and Valve Code, Class 2, 1968	Component Cooling Pumps – Draft Pump and Valve Code, Class 3, 1968 Service Water Pumps – ASME III, Class 3, 1971 Auxiliary Feed Pumps – ASME III, Class 3, 1971	

**The Replacement Once through Steam Generators included feedwater and auxiliary feedwater components. The ROTSGs and MFW/AFW components are designed and fabricated in accordance with Class I requirements of the ASME Boiler and Pressure Vessel Code Section III., 2001 Edition, 2003 Addenda.



NOTES:

LOCATION AND DIMENSION OF PENETRATIONS SHOWN ARE NOT TO SCALE.

----- INDICATES PRESSURE BOUNDARY

DAVIS-BESSE NUCLEAR POWER STATION
CONTAINMENT STRUCTURE SCHEMATIC DIAGRAM
OF CONTAINMENT PRESSURE BOUNDARY
FIGURE 3.2-1

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DB 07-16-14

DFN=G/USAR/UFIG321.DGN/TIF

3.3 WIND AND TORNADO DESIGN CRITERIA

The meteorology of the station site is generally continental in nature. Tornadoes are rather common in the area, but the probability of a tornado striking the station is very low. The station was designed so that the Seismic Class I structures would withstand tornado effects including credible missiles generated by a tornado except for the Borated Water Storage Tank and electrical manholes 3006 and 3020 (See Section 3.8.1.1.5).

The prevailing winds are from the west and southwest. Studies of the meteorological characteristics of the site have been made using long-term data from Toledo and Cleveland and, in addition, an on-site meteorological monitoring program was initiated in 1968 with the installation of a 300-foot high instrumented tower.

3.3.1 Wind Criteria

All Seismic Class I structures were designed for tornado forces as described in Subsection 3.3.2. Wind loads did not control the design of Seismic Class I structures due to the very low wind stresses in comparison with the tornado loads.

High winds, when they occur, are usually associated with summer thunderstorms or winter time cyclonic storms.

The highest winds are usually associated with thunderstorms (tornadoes excepted) during the passage of a line squall or cold front. Fastest mile data for Toledo (ref. 1) are:

Toledo (11 Years)

Month	Speed (mph)	Direction
January	40	W
February	40	NW
March	56	W
April	72	SW
May	45	W
June	49	SW
July	42	W
August	47	W
September	45	W
October	40	SW
November	65	SW
December	45	SW

The Task Committee on Wind Forces of the American Society of Civil Engineers (ref. 2) has estimated for the area a “fastest mile” of wind, at 30 feet above ground, of approximately 90 mph, with a 100 year recurrence.

H.C.S. Thom (ref. 3) has estimated the extreme fastest mile of wind, at 30 feet above the ground, for the following mean recurrence intervals:

Interval (Years)	Annual Extreme-mile (mph)
2	50
10	60
25	80
50	84
100	90

Minimum wind pressures used in the design of Station structures are based on Figure 1(b) of Reference (2) for a wind velocity of 90 mph, 30 feet above the ground with a 100 year recurrence.

Height above ground (ft)	Velocity (mph)
31 - 50	96
51 - 100	111
101 - 300	137
301 - 500	164
510 - 1,200	180

For any other height H, the following equations were used:

$$V = 90\left[\frac{H}{30}\right]^{0.223}$$

Design equations from Reference (2):

$$q = 0.002558V^2$$
$$P = q(C_D)(G)$$

Where:
- V = Wind Velocity (mph)
- P = Pressure (lbs/sq ft)
- q = Dynamic Pressure (lbs/sq ft)
- C_D = Drag Coefficient (see Ref. 2)
- G = Gust Factor (see Ref. 2)

3.3.2 Tornado Criteria

3.3.2.1 General

There are few reliable measurements of the pressure drop associated with a tornado funnel. The greatest reliably measured pressure drops have been on the order of 1.5 psi or less.

The design pressure drop is assumed to be 3 psi in 3 seconds. This is 100% greater than the greatest pressure drop ever reliably measured, which is conservative.

Because of the complexity of the airflow in a tornado, it has not been possible to calculate the velocity or trajectory of missiles that would truly represent tornado conditions. It is assumed that objects of low cross-sectional density, such as boards, metal siding, and similar items, are picked up and carried at the maximum wind velocity of 300 mph.

The behavior of heavier, oddly shaped objects, such as an automobile, is less predictable. The design values of 50 mph for a 4000 lb automobile lifted 25 ft in the air is considered to be representative of what could happen in a 300 mph wind as the automobile is lifted, tumbled along the ground, and ejected from the tornado funnel by centrifugal force. These missile velocities are consistent with reported behavior of such items in previous tornadoes (Ref. 23).

The following Category I structures are analyzed for tornado loading (not coincident with the LOCA or earthquake):

1. Shield Building and Containment Vessel
   Note: Containment Vessel only analyzed for differential pressure (see section 3.8.2.1).

2. Auxiliary Building

3. Intake structure

4. Valve rooms number 1 and 2

5. Service Water tunnel

6. The five electrical manholes (3001, 3004, 3005, 3041, and 3042)

The above structures are analyzed for tornado loading using the following criteria:

1. Differential pressure between the inside and outside is assumed to be 3 psi positive pressure for the shield building and 3.0 psi maximum for the Auxiliary Building subcompartments, as described in Section 3.3.2.2.1.

2. Differential pressure between the inside and outside is assumed to be +1.5 psi for the intake structure and +3.0 psi for the remaining structures mentioned above. Basically, the intake structure is an open structure with one closed room (23 feet X 54 feet X 13.3 feet high) which contains seismic Class I equipment. This room has sufficient openings (12 square feet) for ventilation to justify a reduction in the design pressure as documented in Ref. 106.

3. The lateral force on Class 1 structures is from a 300 mph wind (see section 3.3.2.2). This lateral force is based on the average wind velocity of the design basis tornado with a rotational wind velocity of 300 mph and a progression wind velocity of 60 mph (see Final Safety Analysis Report Question and Response 3.3.2 for use of 300 mph instead of 360 mph).

4. A list of credible external missiles is provided in Table 3.5-2.

A discussion of the probability of tornado occurrence is presented in Section 2.3.1.2.6.

For additional information concerning Shield Building load combination factors, see Appendix 3A.

Seismic Class II structures are separated from Seismic Class I structures, and any damage to Seismic Class II buildings, due to the tornado does not damage Class I systems.

3.3.2.1.1 Tornado Model

The tornado model was patterned after the Dallas tornado of April 2, 1957, as studied by Hoecker (ref. 41). The model is basically that given in WCAP-7897 (ref. 42) but with a more rigorous extrapolation to the parameters desired for a design tornado than given by Bates and Swanson (ref. 43).

Hoecker summarized his findings by the use of a “pressure-time profile” for an average translational velocity of 27 mph and as a function of percentage of total pressure drop.

The two exponential equations used by Hoecker to determine the time-pressure profile cross each other at a radius of 1240 feet instead of the 300 feet at which they cross when a translational velocity is 27 mph. Therefore, it is only necessary to use one equation since the starting tangential velocity corresponding to this distance is 66 mph, which is less than the minimum 75 mph considered by Bates and Swanson.

By incorporating these two assumptions, namely that the vertical component is equal to one-third of the tangential and the radial component is a function of radial distances between minimum and maximum tangential components being considered, a complete windfield was defined by using the following equations:

$$V_t = \frac{249\,(\mathrm{Exp}\,(-48.3 V_1^3 / R)\ V_1^3 D)^{1/2}}{R^3} \qquad (1)$$

$$V_r = \frac{-(1240 - R)\,R}{(1240 - 300)} \qquad (2)$$

$$V_v = 1/3\ V_t \qquad (3)$$

Where:

V_t = tangential velocity, fps

V_r = radial velocity, fps

V_v = vertical velocity, fps

V_1 = translational velocity, fps

D = total pressure drop, psf

R = radius, ft

Equation (1) has been left in a general form for use in future models to predict different total pressure drops or translational velocities. However, D was taken as 432 psf (3.0 psi) and V_1 = 88 fps (60 mph).

A degree of conservatism was added to the tornado model by assuming constant velocities from the ground to a height of 500 feet (Ref. 41).

3.3.2.2 Conversion of Tornado Loads Into Forces

The tornado wind velocity of 300 mph is used uniformly on Class I structures such as the Shield Building and Auxiliary Building. The equation

$$q = 0.002558V^2$$

is used to compute the wind forces (Ref. 2). The shape factors of 0.8 and 0.5 for windward and leeward side, respectively, are used. No gust factors are used for tornado loads.

The wind force, differential pressure (see section 3.3.2.1), and missile impact (see Table 3.3-2, Table 3.5-2 and Section 3.5) are considered to act concurrently for Class 1 structures.

The load combination used for the Tornado Accident is $U = 1.0\ D + 1.0\ L + 1.0\ W^1 + 1.0\ T_0 + 1.25\ H_0$ where:

D = Dead Load

L = Live Load

W^1 = Tornado Load (including differential pressure)

T_0 = Thermal Loads

H_0 = Thermal Force

3.3.2.2.1 Differential Pressure Drop in Auxiliary Building

The auxiliary building has been analyzed to determine the impact of the tornado depressurization. This analysis was based on the 3.0 psi tornado depressurization model, as shown in USAR Fig. 3.3-29. The auxiliary building subcompartments were modeled as volumes that are interconnected by pressure-dependent variable vent area junctions, such as through plant doors, panels, and hatches. This model was evaluated using the COMPARE computer analysis program. The volume thermodynamics are based on a homogeneous gas, assumed to undergo a quasi-static process. Flow between volumes is based on the one-dimensional solution of the momentum equation, including inertia effects. The turbine building was considered to be at atmospheric conditions for this analysis. No flow was considered available through HVAC ductwork. This analysis considered both isentropic and polytropic air flows. This program computed the compartment pressures and the relative differential pressure between interfacing subcompartments.

The computer program "COMPARE" determined that the maximum differential pressure between subcompartments was 3.0 psi. The auxiliary building has been evaluated for the tornado differential pressure loads as described in Section 3.8. The auxiliary building is adequate for these tornado differential pressures.

3.3.3 Essential Systems and Components for Safe Shutdown in the Event of a Tornado

Table 3.3-1 lists all of the essential systems and components that are required for a safe shutdown in the event of a tornado. The equipment listed includes required control, sensing, power and cooling lines. The equipment is located within the protective boundary provided by the barriers listed in Subsection 3.3.2.1.

Table 3.5-2 “Credible External Missiles” lists the kinetic energy due to external missiles.

Table 3.3-2 provides a listing of the depth of missile penetration. The penetrations are less than half of the thickness of the barriers.

TABLE 3.3-1

Essential Systems Required for a Safe Shutdown in the Event of a Tornado

No.	System Description
1.	Auxiliary feedwater system
2.	Service water system
3.	Component cooling system
4.	Decay heat removal system
5.	Makeup pumps
6.	ECCS room air cooling fans
7.	Containment air coolers
8.	Steam generators
9.	Pressurizer
10.	Deleted
11.	Auxiliary feed pump room vent fans
12.	Boric acid addition system
13.	Emergency diesel generators, air receivers, and day tanks
14.	Diesel generator rooms vent fans
15.	Service Water System and MCC's E12C, F12C, and EF12C (intake structure)
16.	Emergency diesel generator buses C1, D1, and diesel vent fans
17.	E1 and F1 substations, MCC's E12A, E14, E15, EF15, F12A, F14 and F15
18.	Batteries, chargers, rectifiers, and assoc. panels
19.	MCC E11A
20.	MCC E11B, E11C
21.	MCC E11D, E12D, F11C, and F11D
22.	MCC E12B, F12B
23.	MCC F11A and F11B

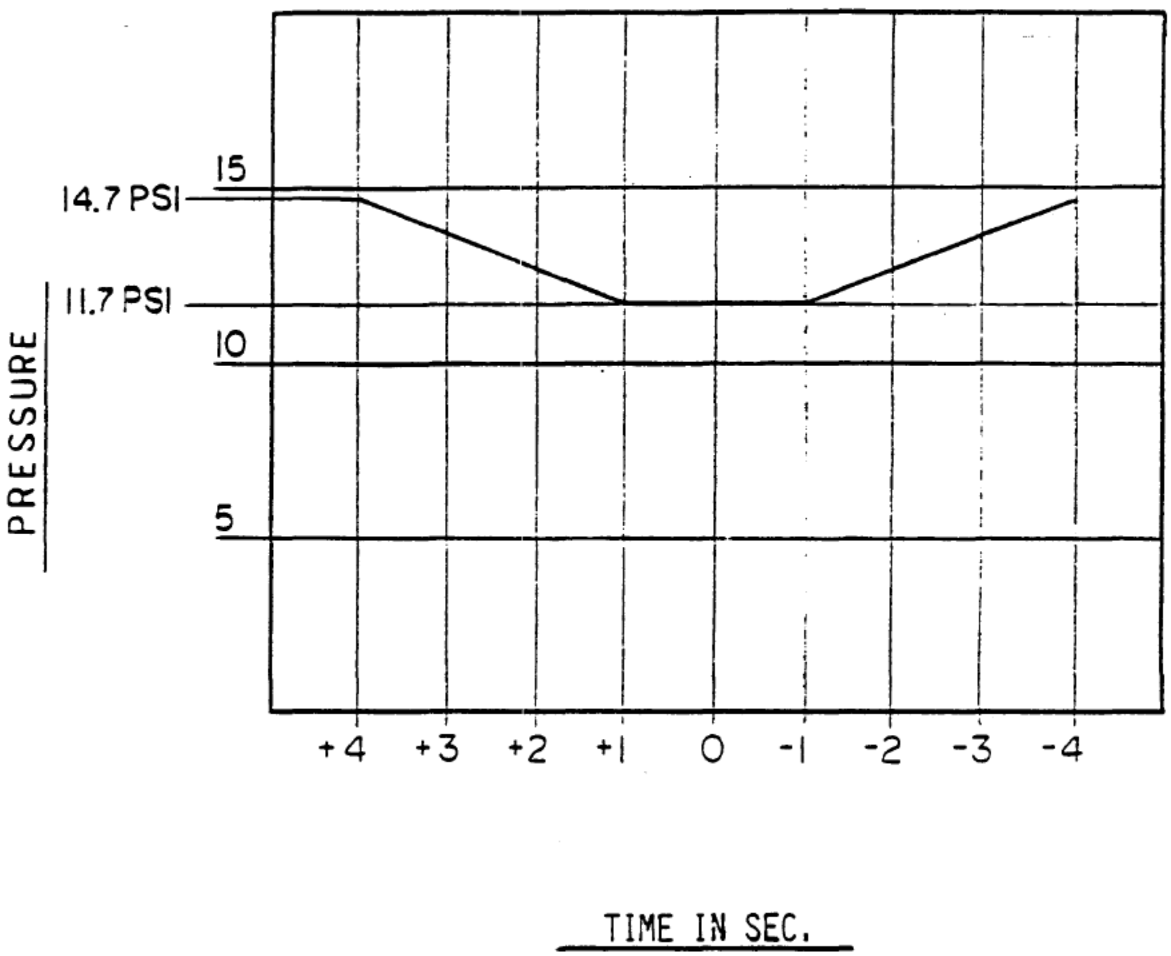
TABLE 3.3-2

Missile Penetrations

Missiles	Maximum Velocity (fps)	Maximum Height (ft)	x (in)
Wooden plank	279	60	2.69
Utility pole	182	10	3.11
1” Steel rod	192	10	3.37
6” Pipe, sch. 40	162	10	1.96
12” Pipe sch. 40	153	--	1.24
3” Pipe sch. 40	192	10	3.89

x = Penetration in inches

Note: Only horizontal missiles are postulated (no vertical velocity component).



DAVIS-BESSE NUCLEAR POWER STATION
TORNADO-PRESSURE DROP CURVE
FIGURE 3.3-29

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3.4 WATER LEVEL (FLOOD) DESIGN CRITERIA

All of the Seismic Class I structures are designed to withstand the static and dynamic forces associated with the design flood level established for the station.

3.4.1 Maximum Probable Water Levels

The maximum probable static water level is Elevation 583.7 (See Section 2.4). The maximum water level due to wave runup on the break wall is calculated to be Elevation 590.5(4). This station is protected along the north, east, and partially on the south side by an earthen breakwall built up to Elevation 591.0. There is no effective dynamic force applied on the critical structures associated with the maximum probable hydrodynamic water level and waves except the front wall of the intake structure which is designed for this loading condition. All Seismic Class I structures are designed for maximum probable static water level of Elevation 584.0.

The Auxiliary Building and Containment Vessel have no access openings below ground floor Elevation 585.0. Consequently, a maximum probable floodwater level of 583.7 does not have adverse effects on these structures. The structures are protected from water intrusion by a complete waterproof envelope below Elevation 583.6. Articulated joints between structures have continuous flexible water stops embedded in concrete walls and floors which provide redundant protection from flooding.

The intake structure is designed to withstand the effects of flooding and wave run-up (see Subsections 3.4.2 and 3.4.2.3). Water stops are provided at construction joints of Seismic Class I structures which prevent water from entering the structure. Watertight doors at both access openings complete the barrier against water entering the service water pump room. Floor drains and a sump collect seepage which might enter the room during a flood. Seismic Class I systems and structures are therefore completely protected from adverse effects of flooding.

The Seismic Class I service water tunnel may be flooded due to postulated failures of either the water treatment structures and systems or failure of Seismic Class II pipe within the tunnel. This tunnel is sealed at both ends, thereby preventing flooding of either the auxiliary building or the intake structure. The Seismic Class I systems within the tunnel are designed to remain operational while flooded.

Electrical manholes and conduits may become flooded during postulated high water conditions. Additionally, some conduits within the duck banks may experience long term wetting. The ends of all conduits from manholes terminating in Seismic Class I structures which cannot be flooded are turned up above Elevation 583.7, thereby eliminating the possibility of draining into these structures. Temporary flooding or long term wetting of seismic class electrical manholes and conduits does not impair functioning of Seismic Class I electrical systems.

The Seismic Class I Borated Water Storage Tank is founded at Elevation 585 above flood level, and consequently is not adversely affected by a flood level of 583.7 feet.

The intake structure is directly protected by a flood control concrete wall. This structure is designed to accept the wave action directly applied on the front face. The dynamic water and wave loads are determined by the Westergaard parabola equations. The equations are based on the assumption of parabolic load distribution presented in Reference 5.

The Emergency Diesel Fuel Oil Storage tank(s) access and vent openings are located at an elevation of 594.5 feet or greater. Consequently, a maximum probable floodwater level of 583.7 feet does not have an adverse effect on these structures.

3.4.1.1 Seismic Category I Systems and Equipment Below El. 583.5’

The following Seismic Class I equipment and portions of the respective systems are located below Elevation 583.5.

- ECCS room sump pumps
- ECCS room coolers
- High Pressure Injection pumps
- Containment Spray pumps
- Decay Heat pumps
- Decay Heat removal coolers
- Waste Gas Surge tank
- Waste Gas Decay tanks
- Hydrogen Dilution System blower
- Makeup pumps
- Auxiliary feed pump turbine units
- Auxiliary feed pump room vent fans
- Containment air coolers
- Core Flooding Tanks
- Letdown coolers
- Motor control centers: MCCE11A; MCCE11D, MCCF11C; MCCF11D; MCCE12C; MCCF12C; and BEF12C
- Service Water pumps
- Reactor vessel and internals
- Reactor coolant pumps
- Steam Generators (bottom)
- Service Water strainers
- Containment isolation valves

3.4.2 Static Loading

The design of the Seismic Class I structures for the loads associated with the maximum probable water level is discussed below.

3.4.2.1 Containment

A triangularly distributed saturated soil pressure is assumed to span between Elevation 565.0 and Elevation 584.0. The cylindrical wall is treated as a beam on an elastic foundation with both ends, Elevation 565.0 and Elevation 584.0, being fixed connections. The fixity at Elevation 584.0 is replaced by an internal moment and radial shear, and by considering the vertical continuity at Elevation 584.0, all of these internal forces are incorporated in the design of the shell.

3.4.2.2 Auxiliary Building

The saturated soil pressure of 41 feet with a triangular distribution is used to check the exterior walls of the Auxiliary Building for the sliding forces and overturning moments.

3.4.2.3 Intake Structure

The exterior walls are designed for a triangularly distributed saturated soil pressure. These forces are not put in the seismic mathematical mass model because they have an insignificant effect on the dynamic response of the model. Also, the high static water level is not concurrent with the earthquake. However, soil loads are increased in accordance with reference (6). The Intake Structure is protected against the wave action and its runup to elevation 591.0.

3.5 MISSILE PROTECTION CRITERIA

Note that this section includes both internal missiles and tornado missiles. Not all information in this section may be applied to tornado missiles.

This section defines and postulates the existence of selected missiles and the source of energy which creates them. The event which generates these postulated missiles is considered to be singular, affecting only the postulated missile itself. Although the missile generation event may occur simultaneously with a postulated accident such as a LOCA or be coincident with a seismic event, for purposes of identification of missiles and their analysis, the assumption is made that the event is a separate occurrence. While it is conceivable that missiles could be generated during the LOCA as a result of blowdown forces, they are not identifiable and therefore their postulation is not considered. Dynamic effects associated with a LOCA are discussed in Section 3.6.

3.5.1 Missile Criteria

The design bases for missile protection is found in NRC General Design Criterion 4. In order to comply with the intent of this criterion the following criteria are given as design bases:

(1)a. Protection is provided for potential missiles that could result in a LOCA.

b. Protection is provided for potential missiles that could result in the loss of ability to control the consequences of a LOCA including both the necessity for core cooling and for retention of containment integrity.

c. Protection is provided for potential missiles that could jeopardize functions necessary to bring the reactor to a safe shutdown condition during normal or abnormal conditions.

The relationship that missiles have to single failure criteria is:

(2)a. That if a missile is generated and causes failure of an adjacent system then that is considered to be a single failure for which that adjacent system must be designed. No other failures of the adjacent system is assumed for design purposes.

b. That a missile which may be generated from the reactor coolant system, coincident with a LOCA, is considered a part of the LOCA for single failure assumption purposes, but may be considered as a separate subject for missile analysis purposes.

Protection against a potential missile may be provided by but not necessarily limited to any one or combination of the following protection methods:

(3)a. Compartmentalization – enclosure of missile source or equipment requiring protection in compartments whose walls preclude penetration by the missile.

b. Barriers – erection of missile barriers either at the source or at the equipment to be protected.

c. Separation – sufficient separation of redundant systems or of components in a safety network so that a potential missile cannot damage both redundant strings of the system and preclude safe shutdown of the reactor plant.

d. Distance – location of equipment beyond the range of a potential missile.

e. Restraints – securing potential missiles by means of restraints.

f. Strategic Orientation – facing equipment and components of equipment in a direction that points the potential missile away from equipment to be protected.

g. Equipment Design – design structure or equipment to withstand impact of potential missile without loss of function.

The following Seismic Class I structures are designed to withstand external and internal missiles:

a. Containment structures (Shield Building, Containment Vessel and Containment Internals)

b. Auxiliary Building

c. Intake structure

d. Valve rooms number 1 and 2

e. Service Water Tunnel

f. Electrical manholes (3001, 3004, 3005, 3041 and 3042)

The above discussion provides methods by which protection may be given for postulated missiles, however, a very basic premise for protection is to design components and equipment so that they have a low potential for generation of missiles. In general, that design which results in reduction of missile generation potential is also the same as that design which promotes the long life and usability of a component, and that is to design well within permissible limits of accepted codes and standards. The following general methods are used in the design, manufacture, and inspection of equipment:

(4)a. All pressurized equipment and sections of piping that from time to time may become isolated under pressure are provided with pressure relief code valves. These valves are preset to assure that any pressure build up in equipment or piping sections does not exceed the design limits of the materials involved.

b. All components and equipment of various systems have been designed and built to the standards established by the ASME Code or other equivalent industrial standards. A stringent quality control program is also applied during manufacture, testing and installation.

c. Volumetric and ultrasonic testing of materials used in components and equipment coupled with periodic inservice inspections add further to the assurance that any material flaws that could permit the generation of missiles are detected.

The selection of postulated missiles to be analyzed to disclose those that should be protected against is based on the examination of components and equipment for energy sources that could be converted to kinetic energy of the potential missile. The following energies are considered:

(5)a. Stored strain energy – this type of energy is associated with nuts, bolts, studs, etc.

b. Contained fluid energy – associated with equipment and components that contain fluids under pressure.

c. Rotational energy – associated with equipment such as the reactor coolant pump motor flywheel. (Consideration for the reactor coolant pump motor flywheel is given in Subsection 5.2.3).

It should be noted that missiles generated due to these energy sources are those that can be identified as component parts of systems and therefore are those for which measures can be taken for protection at the source as well as at the point where damage might occur. Randomly generated missiles cannot be identified or analyzed for protection on normal bases, therefore, protection must be given at the point where damage might occur.

Various postulated missiles located throughout the station may be disqualified from further consideration for missile protection if any of the following design conditions can be identified and associated with the potential missiles involved.

(6)a. If sufficient distance exists between the postulated missile and the equipment and components that if damaged would not result in conditions (1) a, b, or c. Sufficient distance is defined as a distance that if traveled by the missile renders it incapable of causing (1) a, b, or c.

b. If barriers inherent to the station design can be identified and associated with potential missiles such that due to the barrier the postulated missile is rendered incapable of causing (1) a, b, or c. A valid barrier in this case is defined as a structure that is capable of absorbing the effect of the missile impact and not resulting in any of conditions (1) a, b, or c.

c. If enclosure of a postulated missile source can be identified and associated with the missile such that the walls of the enclosure can absorb the effects and preclude the penetration of the missile without causing any of conditions (1) a, b, or c.

d. If restraints inherent to station design can be identified and associated with a postulated missile so that due to the restraint the missile is incapable of being generated.

e. If postulated missile sources associated with components and equipment have been oriented as a result of inherent station design such that the path of the missile is away from equipment to be protected. The path of the missile must be directed into an area such that conditions (1) a, b, or c do not occur if the potential missile is generated.

f. If equipment or components are specifically designed against generation of missiles or are specifically designed to be capable of withstanding missile impact without resulting in the occurrence of conditions (1) a, b, or c.

3.5.2 Missile Identification and Characterization

Systems, components, and equipment of systems have been examined in order to identify and postulate specific missiles and to identify the associated sources of energy that would lead to the generation of these postulated missiles. In addition each postulated missile has been classified.

In general, missiles are characterized by:

Mass
Velocity
Size
Density
Orientation or location
Impact area

Also missiles may be identified (or categorized) according to the potential energy source which serves as the generation mechanism such as:

1. Stored strain energy

2. Contained fluid energy

    a. Jet propelled type missile

    b. Piston type missile

3. Rotational Energy

All Seismic Class I structures are designed to withstand an end-on impact of the missiles as outlined in Tables 3.5-1 and 3.5-2. Analytical techniques described in Subsections 3.5.7 and 3.5.8 are used to analyze Seismic Class I structures, and the Seismic Class I structures are found to be adequate.

3.5.2.1 Postulated Missiles

The tornado missiles which are considered in Subsection 3.3.2.1 were used to check the exterior walls and slabs of Seismic Class I structures. All exterior walls and slabs have more than double the thickness required for missile penetrations in order to prevent spalling of the concrete and generation of secondary missiles. Only horizontal tornado missiles are postulated (no vertical velocity component).

A source of potential missiles found within the secondary shielding is the Reactor Coolant System pressurizer. The primary source of energy associated with the pressurizer is contained fluid energy. Attachments to the pressurizer in contact with this energy are the following: spray line electric motor operated (EMO) valve stem, sample line 3/4” valve stem and sample ¾” EMO valve stem. The generation mechanism associated with these potential missiles is postulated by assuming that the threads of a threaded valve stem fail. A material flaw or some other defect may permit this failure to occur and in turn allow the transfer of contained fluid energy into the kinetic energy of the missile generated. It should be noted that the above failure is not likely to occur due to an overpressurized condition of the pressurizer. Adequate overpressurization

protection for the pressurizer is provided by two safety valves directly connected to the pressurizer. Additional credit against the generation of these potential missiles should be given to the standards established by the ASME Code to which these components must adhere and the various tests and periodic inservice inspections of the components are subjected to.

Although contained fluid energy is one source of energy associated with the pressurizer it is not the only source of energy that could result in the generation of missiles. Another energy source associated with the pressurizer is stored strain energy. This energy source relates to the following components of the pressurizer: 4” valve bonnet stud, 6” valve bonnet stud, 16” manway cover stud, heater bundle stud, 3/4” valve stem stud. Stored strain energy in the above components would be transferred to the kinetic energy of the missile generated in the event of a severe failure in the bolting material. Such a failure could be postulated by assuming a material flaw or some other defect would permit the stored strain energy to detach and directly propel the potential missile. The use of proper installation methods ensures that the stress level in the stud or bolt material does not exceed the safe level as prescribed by the ASME Code. The above, coupled with the use of proper materials and periodic inservice inspections should assure that the probability of missile generation from these sources is very low.

A second source of contained fluid energy within the secondary shielding is associated with the once through Steam Generator. The following attachments of the Steam Generator are in contact with contained fluid energy: vent valve stem and wheel, sample line 1” valve stem and wheel, sample line 1” EMO valve stem and wheel. The generation mechanisms for these potential missiles would be the same as those previously described for valve stems associated with the pressurizer. It should be mentioned that these valve stems and the related valve are inspected and shop tested in accordance with the standards established by the ASME Code. The tests and inspections coupled with periodic inservice inspections should assure a low probability of potential valve stem missiles.

Stored strain energy is also associated with the Steam Generator and is considered as an energy source that could generate potential missiles. Sources of stored strain energy associated with components connected to the Steam Generator are the following: 1-1/2” vent valve bonnet stud, 4” feedwater inlet flange stud, 16” I.D. manway stud, 6” handhole stud, 3” inspection port stud, and 1” valve bonnet stud. Again, the generation mechanisms for these components would be the same as those previously described for the pressurizer. Proper precautions in accordance with established regulations have been taken to assure that these bolts and studs are not torqued beyond the limits established for the materials. These precautions and periodic inservice inspections should assure a low probability of missile generation.

A third source of contained fluid energy and stored energy found within the secondary shielding is related to the reactor vessel and Control Rod Drive Assemblies. Components containing stored strain energy would include: reactor closure head nut, reactor closure head stud with nut, 1” valve bonnet stud, and control rod drive nozzle flange bolt. A missile generated from one of these components could be assumed to be produced due to the failure of threaded parts or undetected material flaw. Generation mechanisms such as these could possibly permit stored strain energy to be converted to kinetic energy of the missile generated.

The second source of energy mentioned above is contained fluid energy. The control rod drive mechanisms located on the top of the reactor vessel closure head are in contact with this energy. Various components of a control rod drive assembly are under the influence of this energy and could be postulated as potential missiles. It has been postulated, for the purpose of

analysis, that contained fluid energy might detach a control rod drive closure cap or an entire Control Rod Drive Assembly.

A control rod drive closure cap has been analyzed as a piston type missile. It is assumed that the generation mechanism is the 2200 psi normal reactor vessel pressure acting on the cross sectional area of the closure cap as a constant force over a certain distance. This is considered as a piston type action. By equating the work done (force x distance) with the kinetic energy equation, the velocity of the missile can be derived. Since the insert closure cap is retained by a closure nut which is threaded to the inside of the motor tube, it should be noted that for a closure cap and nut to become a missile, the threads of the closure nut must completely fail. This is an event that is unlikely to occur for any Control Rod Drive Assembly that has been placed into normal operating conditions. This is assumed since all Control Rod Drive Assemblies are preassembled according to the accepted industry standards and shop tested at a pressure above 3100 psi, well above the pressure of normal operating conditions.

Any material defects that would permit the control rod drive closure cap to become a missile would be forced to failure prior to time the assembly is placed into normal operation. This preservice test coupled with periodic inservice inspections should assure that the probability of a control rod drive closure cap becoming a missile is low.

An entire Control Rod Drive Assembly has been postulated, for the purpose of analysis, as becoming a possible jet propelled missile. An individual Control Rod Drive Assembly is bolted to each nozzle of the reactor vessel closure head. Thus, each Control Rod Drive Assembly is in contact with the contained fluid energy of the Reactor Vessel. This class of missile (the most significant type in terms of available kinetic energy) would be subjected to a significant jet of escaping fluid should this missile ever be generated. The jet imparts an impulse to the missile as the jet expands into the Reactor Building. The velocity of the jet is assumed to be constant at the maximum orifice velocity. As the assembly moves further from the rupture, the jet expands and the weight of the fluid actually striking the Control Rod Drive Assembly decreases. The angle of jet expansion is an important assumption in determining the expected velocity of this potential missile. For subcooled conditions in the fluid, the liquid remains in a stream and an angle of expansion of 10° from the normal is assumed; for steam flow, a jet expansion of 30° from the normal is assumed. For the purpose of a conservative analysis a jet expansion angle of 10° has been used.

It is assumed that an entire Control Drive Assembly is ejected from the reactor vessel closure head due to a severe failure in either the control rod drive nozzle bolting material, the control rod drive nozzle, or the Control Rod Drive Assembly material itself. In the unlikely event of a severe material failure in any one of the above locations, a complete or partially complete Control Rod Drive Assembly could be forced out of the reactor vessel closure head and be propelled by the force of a jet of escaping reactor coolant. This is the worst case missile for which the missile shielding that is located above the Control Rod Drive Assemblies has been designed.

It should be noted that the materials such as the control rod drive nozzle bolting material, the control rod drive nozzle or the Control Rod Drive Assembly material itself are selected to be compatible with and operate in the environment associated with the Reactor Vessel. Quality standards relative to material selection, fabrication, and inspection are specified to ensure safety functions of the assemblies essential to accident prevention. All welding has been performed by personnel qualified under ASME Code, Section IX, Welding Qualifications. The above coupled with proper field installation and periodic inservice inspections should assure that missiles being generated from Control Rod Drive Assemblies are of very low probability.

Another type of potential missile to be considered is potential instrumentation missiles. Instrumentation and instrument connection attached to the reactor coolant piping, Steam Generator, and pressurizer have been considered as potential jet propelled missiles. The energy source for these potential missiles is contained fluid energy. Instrument connections to piping and various components are welded in place. Should a weld fail, the instrument connection and in some cases (i.e., temperature elements) the instrument itself could become a jet propelled missile. Protection against these missiles is afforded by application of quality construction methods that assure the proper installation of equipment and components, thus reducing the probability of missiles being generated from this source.

Further consideration for valves that are part of the primary pressure boundary and other pressurized systems throughout the station are given here. Valve design inherently minimizes the possibility of a valve part becoming a missile by providing multiple interferences between the stem, disk, bonnet, and operator. The use of proper installation methods ensures that the stress level in the stud or bolt material does not exceed the safe level indicated by the ASME Code. Cast stainless steel valve bodies are radiographed and dye penetrant tested to disclose any unacceptable material flaws. Finally, the complete valve is hydrotested according to specifications of the ASME Code.

The Reactor Coolant pumps contain the principal source of rotational energy located within the Reactor Building. The primary consideration for missiles being generated from this source has been placed on the pump motor flywheel. Treatment of the flywheel topic is given in Subsection 5.2.3. The discussion there covers the general method of implementing the General Design Criterion 4 with regard to the flywheels of the reactor coolant pump motors. The method of implementing Criterion 4 is based on the principles of fracture mechanics and on the requirement of sufficiently small probability of flywheel failure. These methods are used as an assessment of the acceptability of flywheels so that the consequences of failure need not be protected against.

The following rotating equipment and pressurized containers were considered for possible missiles which could damage essential equipment:

a. Decay Heat Removal Pumps

b. Containment Spray Pumps

c. H.P. Injection Pumps

d. Service Water Pumps

e. Cooling Tower Makeup Pumps

f. Diesel Fire Pump Unit

g. Auxiliary Feed Pump Turbine Units

h. H.P. Nitrogen Storage Tanks (yard)

i. Cryogenic Nitrogen Storage Unit (yard)

j. Main Steam Isolation Valve Nitrogen Accumulators

k. Waste Gas Decay Tanks

l. Waste Gas Surge Tanks

m. Diesel Generator Air Receivers

n. N_2 Cylinders for CAC Service Water Outlet Air Operated Valves (AOVs)

Other equipment, because of its small size and/or location, was considered incapable of generating missiles dangerous to essential equipment.

Removable slabs have been provided in the interior floors of the auxiliary building and the top slab of the intake structure and were evaluated for potential missiles. The dead load of the slabs is greater than the internal pressures and this will prevent them from becoming internal missiles.

3.5.3 Discussion of Missile Orientation

Potential missiles considered within the secondary shielding have trajectories that can be assumed. It is not feasible to attempt to justify any single path for the trajectory of a potential missile. It is recognized that a potential missile, if it is generated, could travel in any number of directions depending upon the means of generation. However, by examining the general orientation of potential missiles, the expected path has been assumed, should the potential missile be generated. Based on the assumed path, protection against the missile can be considered.

Initial consideration was given to the orientation of potential missiles associated with the Reactor Vessel. The first potential missile considered is the reactor closure head stud with nut. It is assumed that a missile generated from this source travels in the vertical direction. Other missiles associated with the Reactor Vessel are confined to the Control Rod Drive Assemblies. Since control rod drives are also oriented in the vertical direction, it is postulated that missiles generated from this source travel in the vertical direction.

Potential missiles related to the Steam Generators may also travel in assumed paths delineated by the general orientation of the potential missile. Postulated missiles such as the feedwater flange studs, 16” I.D. secondary manway studs, 6” hand hole and 3” inspection port cover studs are assumed to travel on a horizontal path in addition to the paths delineated by their general orientation.

The pressurizer has potential missiles that if generated would travel in the horizontal or vertical direction. Missiles traveling a horizontal path would include, 16” manway cover stud, heater bundle stud, and spray line 2-1/2” electric motor operated valve stem. Potential missiles assumed to travel a vertical path would include 4” and 5” valve bonnet studs which are part of the safety valves located on top of the pressurizer.

Potential instrumentation missiles that are part of the reactor coolant pressure boundary can be expected to travel vertical or horizontal trajectories. Each hot leg leading to a Steam Generator has a section of piping containing flow meter instrumentation. Directly above these sections are located instrumentation for monitoring pressure and temperature of flow through the piping. The orientation of these attachments are such that potential instrumentation missiles travel a path in a horizontal plane defined by the level of the attachment to the piping. Instrumentation attached

to the vertical sides of the Steam Generators are also assumed to travel in a horizontal plane defined by the level of the attachment on the Steam Generator.

Valves located within the secondary shielding that are components of the reactor coolant pressure boundary are in most cases check valves. Of primary concern is the orientation of bonnets of check valves, bonnets, stem and wheel of manual valves, and bonnet, stem and motor of electric motor operated valves. Only where considered practical and without limiting the operation of the valve, the above components are oriented so that the path of potential missiles are directed into an area where missile damage does not endanger components and equipment necessary for the safe shutdown of the reactor.

3.5.4 Internally-Generated Missiles

The internally generated missiles considered in the analysis of Class I structures are listed in Table 3.5-1.

3.5.5 Externally-Generated Missiles

The external missiles considered in the analysis of the Class I structures are listed in Table 3.5-2.

3.5.6 Off-Site Generated Missiles

There are no missiles that might result from off-site activities. For further discussion see Chapter 2.

3.5.7 Discussion of Analytical Techniques Used for Potential Missiles Generated from Energy Sources

Analytical techniques used for each missile classification is explained by listing the basic equations used for the analysis of missiles under each classification. The level of conservatism associated with the equations is discussed. The basic approach to missile analysis has been to assume a worst case missile is generated and to determine a missile velocity. Once velocity has been conservatively calculated the missiles energy may be estimated and the potential effects may be assessed.

Class I Missiles:

Class I missiles are postulated to be originated by stored strain energy.

The following equations are used in the evaluation of Class I missiles:

a. Equation for velocity $V = \sigma \left[\frac{g}{EW}\right]^{1/2}$

Where V = velocity of missiles ft/sec

σ = ultimate tensile stress, lb/ft^2

E = modulus of Elasticity, lb/ft^2

g = constant = 32.2 $\frac{\text{lbm-ft}}{\text{lbf-sec}^2}$

W = mass density of projectile, lbm/ft^3

m = mass of projectile, lbm

b. Equation for kinetic energy $K.E. = \frac{mv^2}{2g}$

These equations provide a conservative analysis of missile energy because, (1) the ultimate tensile stress (σ ult.) for the material is used resulting in a larger amount of energy than would actually be present at fracture and (2) all strain energy is converted to kinetic energy with no consideration for energy losses due to friction, relaxation, heating of the material or air resistance.

Class II Missiles:

Class II missiles are postulated to be originated by contained fluid energy and are piston type missiles.

The following equations are used in the evaluation of Class II missiles:

a. Equation for velocity $V = \left[\frac{2PAoL}{m}\right]^{1/2}$

Where V = velocity of missile at end of piston stroke, ft/sec

P = system pressure, lb/ft^2

Ao = missile area under pressure, ft^2

L = length of stroke, ft

m = mass of missile, $\text{lb-sec}^2\text{/ft}$

b. Equation for kinetic energy $K.E. = \frac{mv^2}{2}$

These equations provide a conservative analysis of the missile energy since no consideration is given for energy losses due to friction or air resistance.

Class III Missiles:

Class III missiles are postulated to be originated by contained fluid energy and are jet propelled missiles.

The following equations are used in the evaluation of Class III missiles:

a. Equation for velocity

$$\left(1-\frac{V}{V_f}\right)-\ln\left(1-\frac{V}{V_f}\right)=K_1-\left[\frac{K_2}{r_o+\tan\beta}\right]$$

Where $K_1=\left(1-\frac{V_o}{V_f}\right)-\ln\left(1-\frac{V_o}{V_f}\right)+\frac{K_2}{r_o}$

$K_2 = \frac{P_f A_o A_m}{M \pi \tan \beta}$

V = missile velocity at distance X, ft/sec

V_f = jet velocity, ft/sec

r_o = radius of throat, ft

x = distance traveled, ft

β = angle of jet expansion (^{o} from normal)

V_o = initial velocity of missile, ft/sec

P_f = density of fluid jet, lb sec^2 ft^4

A_o = missile area under pressure, throat area, ft^2

A_m = cross-sectional area of missile, ft^2

M = mass of missile, lb-sec^2 / ft

b. Equation for kinetic energy K.E. = $\frac{mv^2}{2}$

These equations provide a conservative estimate of the missile energy since no consideration is given for energy losses due to friction or air resistance.

3.5.8 Analytical Techniques on Penetration

3.5.8.1 Analysis

The penetration analysis of the missiles is carried out using the following modified Petry formulas: See References 6, 7 and 18.

$$D = KA_pV^1$$

$$V^1 = \log_{10}\left[1 + \frac{V^2}{215000}\right]$$

$$D^1 = D\left[1 + e^{-4\left(\alpha^1 - 2\right)}\right]$$

$$\alpha^1 = \frac{T}{D}$$

3.5.8.2 Missile Energy Losses

Along the projectile path, there is a certain amount of energy loss due to air drag which is proportional to the square of the velocity.

$$F_D = \frac{C_D \rho A V^2}{2}$$

Upward Flight:

$$dv = -gdt - \left[\frac{C_D \rho A V^2}{2}\right] dt$$

Downward Flight:

$$dv = gdt - \left[\frac{C_D \rho A V^2}{2}\right] dt$$

Among all the missiles, only the turbine missile is considered for the energy losses due to air drag.

3.5.8.3 Notations

D = Depth of penetration in a slab of infinite thickness, feet.

D^1 = Penetration in a slab of finite thickness T.

T = Thickness of slab, feet.

A_P = Sectional pressure obtained by dividing the weight of missile (lbs) by its cross-sectional area (psf).

V = Velocity of missile, ft/sec.

K = Penetration material coefficients, experimentally determined (See Table 3.5-3).

V^1 = Velocity coefficient factor.

α^1 = Ratio of slab thickness to penetration thickness.

F_D = Drag force, lbs.

C_D = Drag coefficient (dimensionless) used as 1.0.

ρ = Air density, lb-sec^2/ft^4.

w = Air density, lb/ft^3 used as 0.074 lb/ft^3.

A = Projected area, ft^2.

W = Weight of missile, lbs.

M = Missile mass, lb-sec^2/ft.

g = Acceleration of gravity, ft/sec.^{2}

TABLE 3.5-1

Credible Internal Missiles
Reactor Vessel and Control Rod Drive

Missile Class	Description	Weight (lbs)	Missile Impact Area (in^2)	Maximum Velocity (ft/sec)	Kinetic Energy (ft-lbs)
I	Reactor Closure head nut	80.0	38.0	97.6	11840.8
I	Reactor Closure head stud w/nut	660.0	71.0	97.6	97686.6
I	1” Valve Bonnet Stud	0.5	.6	74.5	43.0
I	Control Rod Drive Nozzle Flange Bolt	3.0	3.1	97.6	444.0
II	CRD Closure Cap	8.0	7.0	229.1	6518.6
III	CRD Assembly	1000.0	64.0	90.0	125,777.0
	STEAM GENERATORS				
I	1 1/2” Vent Valve Bonnet Stud	2.0	0.8	74.5	172.4
I	4” ID Feedwater Stud	2.0	0.8	74.7	171.0
I	16” ID Primary Manway Stud	17.4	3.1	74.7	1509.0
I	16” ID Secondary Manway Stud	4.0	1.2	74.7	343.0
I	1” Valve Bonnet Stud	0.5	0.6	74.5	43.1
I	6” ID Primary Handhole Stud	4.1	1.2	74.7	355.0
I	6” ID Secondary Handhole Stud	2.2	0.8	74.7	189.0
I	3” ID Inspection Port Stud	0.9	0.4	74.7	79.0
II	Vent Valve Stem and Wheel	5.0	.4	45.9	163.3
II	Sample line 1” Valve Stem and Wheel	4.0	0.3	36.4	82.3
II	Sample line 1” EMO Valve Stem and Wheel	4.0	0.3	36.4	82.3
III	Temperature Sensing Connection (Thermowell)	6.0	1.7	81.2	610.0

TABLE 3.5-1 (Continued)

Credible Internal Missiles
Reactor Vessel and Control Rod Drive

Missile Class	Description	Weight (lbs)	Missile Impact Area (in^2)	Maximum Velocity (ft/sec)	Kinetic Energy (ft-lbs)
	PRESSURIZER				
I	4" Valve Bonnet Stud	3.0	1.0	74.5	258.5
I	5" Valve Bonnet Stud	3.0	2.4	74.5	258.5
I	16" Manway Cover Stud	7.5	3.1	74.5	646.4
I	Heater Bundle Stud	25.0	7.0	74.5	2154.6
I	3/4" Valve Stem Stud	0.8	0.4	74.5	68.9
II	Spray Line 2 1/2" EMO Valve Stem	7.0	0.8	62.6	425.6
II	Sample Line 3/4" Valve Stem	4.0	0.3	77.6	374.3
II	Sample Line 3/4" EMO Valve Stem	4.0	0.3	77.6	374.3
	INSTRUMENTS				
III	Resistance Temperature Element	1.0	0.2	79.0	97.0
III	Resistance Temperature Element & Plug	2.0	4.0	156.0	756.0
	SYSTEM PIPING				
I	Core Flooding Line, 14" C.V. Bonnet Stud	2.0	1.7	74.5	172.4
I	Core Flooding Line, 14" Valve Bonnet Stud	3.5	4.0	74.5	301.6
II	Core Flooding Line, 14" C.V. Pivot Stud	10.0	1.75	118.7	2187.0
II	Core Flooding Line, 14" Power Operated Valve Stem	98.0	5.0	94.7	13,656.9
I	Low Pressure Injection, 12" C.V. Bonnet Stud	2.0	1.7	74.5	172.4

TABLE 3.5-1 (Continued)

Credible Internal Missiles
Reactor Vessel and Control Rod Drive

Missile Class	Description	Weight (lbs)	Missile Impact Area (in^2)	Maximum Velocity (ft/sec)	Kinetic Energy (ft-lbs)
	SYSTEM PIPING (Cont'd)				
II	Low Pressure Injection, 12" C.V. Pivot Stud	10.0	1.8	112.6	1968.0
I	Reactor Vessel Outlet to L.P. System 14" Valve Bonnet Stud	2.5	2.0	74.5	215.5
I	Reactor Vessel Outlet to L.P. System, Relief Valve Bonnet Stud	0.5	0.3	74.5	43.1
I	Reactor Vessel Outlet to L.P. System Relief Valve Stem Ass'y	40.0	12.5	35.3	768.0
II	Reactor Vessel Outlet to L.P. System 14" EMO Valve Stem	50.0	3.1	70.6	3870.0
I	Reactor Vessel Inlet from H.P. System 2 ½" C.V. Bonnet Stud	1.0	0.8	74.5	86.2
II	Reactor Vessel Inlet from H.P. System, 2 ½" C.V. Pivot Stud	3.0	0.8	78.6	287.1
I	Steam Generator Outlet Line to R.C. Pump Inlet, 1" Drain Valve Bonnet Stud	0.8	0.6	74.5	68.9
II	Steam Generator Outlet Line to R.C. Pump Inlet, 1" Drain Valve Stem Ass'y	4.0	0.3	89.2	494.1
I	Reactor Coolant Pump Seal Water Return to H.P. System, 1" EMO Valve Bonnet Stud	2.0	0.6	74.5	172.4
II	Reactor Coolant Pump Seal Water Return to H.P. System, 1" EMO Valve Stem	12.0	1.8	88.9	1471.0
I	Letdown Cooler Inlet and Outlet Lines, 2 ½" EMO Valve Bonnet Stud	1.6	0.6	74.5	137.9

TABLE 3.5-1 (Continued)

Credible Internal Missiles
Reactor Vessel and Control Rod Drive

Missile Class	Description	Weight (lbs)	Missile Impact Area (in^2)	Maximum Velocity (ft/sec)	Kinetic Energy (ft-lbs)
	SYSTEM PIPING (Cont'd)				
II	Letdown Cooler Inlet and Outlet Lines, 2 ½" EMO Valve Stem	6.0	0.6	160.9	2651.0
I	R.C. Pump Seal Water Inlet & Outlet Lines, 1 ½" C.V. Bonnet Stud	1.0	0.8	74.5	86.2
I	R.C. Pump Seal Water Inlet & Outlet Lines, 1" Outlet Valve Bonnet Stud	1.5	1.0	74.5	129.3
II	R.C. Pump Seal Water Inlet & Outlet Lines, 1" Outlet Valve Stem	5.0	1.0	73.0	414.0
II	R.C. Pump Seal Water Inlet & Outlet Lines, 1 ½" C.V. Pivot Stud	0.5	.5	164.5	203.2
II	R.C. Pump Vent & Drain Lines, 1 ½" Vent & Drain Valve Stem	5.0	1.7	97.3	734.6
I	R.C. Pump Vent & Drain Lines, 1 ½" Vent & Drain Valve Bonnet Stud	2.0	.8	74.5	172.4

TABLE 3.5-2

Credible External Missiles

Description Tornado Missiles	Weight (lbs)	Missile area (sq ft)	Maximum Velocity (fps)	Kinetic Energy (ft-lbs)
(1) A tornado-driven missile equivalent to a 12-foot-long piece of wood 8 inches in diameter traveling end on at a speed of 250 mph.	105	0.35	367	2.20×10^5
(2) A tornado-driven missile equivalent to a 4000 pound automobile traveling through the air at 50 mph and at not more than 25 feet above the ground, is assumed.	4000	20.0	73.5	3.36×10^5
(3) A tornado-driven missile equivalent to a 10-foot-long piece of pipe 3.5 inches O.D. traveling end on at a speed of 100 mph, is assumed. Pipe type 3" I.D. schedule 40.	75.8	0.063	147	2.54×10^4
(4) 4" x 12" plank x 12' long with a density of 50 lb/ft^3	200	0.33	279	2.42×10^5
(5) Utility pole 13.5"ϕ x 35' long with a density 43 lb/ft^3	1495	0.99	182	7.69×10^5
(6) One-inch solid steel rod 3' long with a density of 490 lb/ft^3	8	0.0054	192	4.58×10^3
(7) 6" schedule 40 pipe 15' long with a density of 490 lb/ft^3	284	0.24	162	1.16×10^5
(8) 12" schedule 40 pipe 15' long with a density of 490 lb/ft^3	743	0.89	153	2.70×10^5
(9) 3" schedule 40 pipe 15' long with a density of 490 lb/ft^3	114	0.067	192	6.52×10^4

Note: Only horizontal missiles are postulated (no vertical velocity component).

TABLE 3.5-3

VALUES OF PENETRATION COEEFICIENT (K)
FOR VARIOUS MATERIALS (FROM REFERENCE 18)

Material	**Ft^3 $lb.^{-1}$**
Limestone	5.38×10^{-3}
Concrete[1]	7.99×10^{-3}
Reinforced concrete[2]	4.76×10^{-3}
Specially-reinforced concrete[3]	2.82×10^{-3}
Stone masonry	11.72×10^{-3}
Brickwork	20.48×10^{-3}
Sandy soil	36.7×10^{-3}
Soil with vegetation	48.2×10^{-3}
Soft soil	73.2×10^{-3}

[1]Mass concrete with a crushing strength of 2,200 pounds per square inch.

[2]Normal reinforced concrete with a crushing strength of 3.200 pounds per square inch and 1.4 percent of reinforcement.

[3]Specially-reinforced concrete with a crushing strength of 5,700 pounds per square inch and 1.4 percent of reinforcement.

3.6 PROTECTION AGAINST DYNAMIC AND ENVIRONMENTAL EFFECTS ASSOCIATED WITH POSTULATED RUPTURE OF PIPING

3.6.1 Introduction

This section describes the special measures taken in the design and construction of the plant to protect the public against the consequences of dynamic effects associated with the postulated rupture of piping both inside and outside containment. The containment, and all essential equipment, both inside and outside containment are protected against the effects of pipe whip, jet impingement, missiles, and environmental effects.

In the design of nuclear safety systems, it is necessary to ensure that all components which are required for the safe shutdown of the plant do not fail as a result of a failure in a high energy or moderate energy piping system. The criteria for, and a listing of, the nuclear safety systems are given in Subsections 3.6.2.1 and 3.6.2.7.1, respectively.

The purpose of these criteria is to provide protection of nuclear power station equipment and structures from adverse effects due to pipe-rupture accidents. It is intended to comply with 10CFR50 Appendix A, including Criterion 4 - Environmental and Missile Design Bases. Compliance with 10CFR100 is the overriding safety consideration for postulated pipe breaks.

In response to NRC letters, dated September 26, 1972 (Reference 99) and December 15, 1972 (Reference 40), the provisions to protect safety-related structures and systems outside containment are presented.

3.6.2 Dynamic and Environmental Effects Protection Criteria

3.6.2.1 General

The basic design criteria for pipe ruptures set the following requirements:

a. The primary coolant system is to be protected from pipe whip, jet effects, and missiles that could cause a loss-of-coolant accident.

b. The containment vessel is to be protected from any pipe rupture effects resulting from a loss-of-coolant accident and from the rupture of lines required to be in service during normal plant operation (Modes 1-3).

c. Each redundant engineered safety features system is to be protected from damage by pipe rupture from its parallel system.

d. A concurrent rupture in another loop of the primary system is prevented.

e. Protection is to provide for those Seismic Class I structures and systems which are located outside the containment and are necessary for safe shutdown, against pipe whip and environmental effects of the rupture of high-energy or Seismic Class II piping located in the near vicinity.

f. Consistent with Section 7.4, safe shutdown is defined as that condition in which the reactor is 1% subcritical and the reactor coolant system temperature and pressure are in the normal operating range (e.g., RCS pressure approximately 2150 psig and RCS temperature of approximately 530°F).

To meet these design criteria, the following assumptions are made:

a. A severed pipe inside containment vessel, operating above 275 psig or 200°F, may experience significant translocational motion in any direction, if unrestrained, and cause physical damage to other piping or components within close proximity. Jet impingement can cause damage to unprotected components within close proximity.

b. A severed pipe outside containment vessel operating above 275 psig and 200°F may experience significant translocational motion in any direction, if unrestrained, and cause physical damage to other piping or components within close proximity. Jet impingement can cause damage to unprotected components within close proximity (see Table 3.6-1).

c. A break in a pipe outside containment, carrying fluid above 275 psig or 200°F, can cause the loss of safety-related equipment in the vicinity of the break due to environmental effects (see Table 3.6-1).

d. Seismic Class II piping systems outside the containment vessel with fluid conditions below the criteria of assumptions ‘b’ and ‘c’ can cause the loss of safety-related equipment in the vicinity of the break due to water flooding.

e. Lines 1-inch nominal diameter and smaller cannot cause physical damage to large adjacent components and piping.

f. A ruptured pipe cannot damage an equal or larger pipe of equal or heavier wall thickness by whipping.

g. Operating conditions prior to the pipe rupture are considered normal steady-state or hot standby.

h. Other passive failures in addition to the postulated pipe break are not assumed to occur.

i. Only those high energy systems which are in service on a long-time basis (greater than six hours) are assumed to rupture.

j. Offsite power is assumed not available if the pipe rupture results in a unit trip.

k. Coincident with the postulated pipe break, a single failure in the active component of a required safety-related system is assumed.

l. A postulated critical crack size is taken to be one-half the pipe inside diameter in length and one-half the wall thickness in width.

To categorize these criteria more fully, see Tables 3.6-3, 3.6-5, and 3.6-7. Table 3.6-3 references figures which show all high energy lines analyzed for whipping. Postulated pipe break and whip restraint locations are indicated. Pipe breaks within Containment Vessel are postulated in accordance with NRC Regulatory Guide 1.46. Breaks outside containment are in accordance with Reference 40. The elimination of Arbitrary Intermediate Breaks, as permitted per Generic Letter 87-11, has been adopted for evaluating break locations within piping systems

both inside and outside containment. Criteria given in Standard Review Plan 3.6.1 and 3.6.2 including Branch Technical Position MEB 3-1 were also used in determining the pipe break locations for pipe whip restraint design as discussed in section 3.6.2.2.1. Also, provisions of revised GDC4 were applied to exclude the dynamic effects of a RCS pipe rupture. For the stresses for postulated pipe break points see Note 1 in Table 3.6-2.

3.6.2.2 Pipe Restraint Criteria to Prevent Pipe Whip Impact

The basic philosophy used to prevent high-energy pipes from whipping is to provide restraints of sufficient capacity and at such spacing that pipe whipping cannot develop. These restraints are independent of operating and seismic supports. Consequently, they are designed for pipe-rupture loads only. It should be noted that pipe-rupture loads are approximately two orders of magnitude larger than the combined operating and earthquake loads. Pipe ruptures either guillotine or side-split types are postulated to occur (See Note 1 in Table 3.6-2). Allowable pipe spans are calculated assuming that all the energy developed during the accident experience is transferred to the pipe restraint. The pipe restraints are then designed to absorb this energy. Restraints are further designed to prevent the pipes from shearing off and generating missiles. Jet effects from ruptured pipe are considered in designing pipes, walls, and shields. The concurrent effects of jets and pressure differentials are also considered in designing walls and shields.

3.6.2.2.1 Pipe Restraint Design Criteria to Prevent Pipe Whip Impact Within the Containment Vessel

The basic philosophy used to prevent high-energy pipes from whipping is to provide restraints of sufficient capacity and at such spacing that pipe whipping cannot develop. These restraints are independent of operating and seismic supports. Consequently, they are designed for pipe-rupture loads only. It should be noted that pipe-rupture loads are approximately two orders of magnitude larger than the combined operating and earthquake loads.

1. Restraints have been located for breaks postulated in accordance with NRC Regulatory Guide 1.46 for the following systems:

L.P. Injection Lines	Core Flooding Lines*
Letdown Line	Decay Heat Removal Lines*

* It should be noted that portions of the non-Class 1 piping of this system is exempt from high energy pipe break requirements due to the short period of time when it operates at high-energy conditions.

2. Restraints have been located for breaks postulated in accordance with Standard Review Plan Section 3.6.1 and/or 3.6.2 including Branch Technical Position MEB 3-1 (Reference 53) for the following systems:

Pressurizer Surge Line* *	Auxiliary Feedwater Lines
Steam Generator Blowdown Lines	Main Feedwater Lines
Reactor Head Vent*	Main Steam Lines
High Pressure Injection Lines (Class 2 Portion)	

* Whip restraint R-16 was removed from the reactor head vent line at its attachment point on OTSG 1-2. Whip restraints R-10, R-11, R-13, R-14, and R-15 were removed from the reactor head vent line in accordance with ECP 10-0470.

** The Pressurizer Surge Line whip restraints have been abandoned in place (Reference Figure 3.6-30). The Pressurizer Surge Line pipe whip and jet spray have been evaluated for a “break anywhere” criteria and it has been determined that no safety related components, equipment, or structures will be adversely affected.

3. The Reactor Coolant System has been evaluated using the criteria of Standard Review Plan 3.6.3, Leak-Before-Break evaluation procedures. This criteria, in conjunction with General Design Criterion 4 (GDC-4) of 10CFR50 Appendix A, allows the exclusion of the dynamic effects of a postulated pipe rupture. Consequently, the pipe whip restraints installed are no longer required.

   The RCS Leak Detection Systems meet the intent of the regulatory positions provided in Regulatory Guide 1.45 and therefore the prerequisites for applying GDC-4 were satisfied (Reference 98).

For all other high-energy systems, pipe ruptures of either guillotine or side-split type are postulated to occur anywhere on the pipe surface and restraints are located accordingly.

Allowable pipe spans are calculated, assuming that all the energy developed during the accident experience is transferred to the pipe restraint. The pipe restraints are then designed to absorb this energy. Restraints are further designed to prevent the pipe from shearing off and generating missiles. Jet effects from ruptured pipe are considered in designing pipes, walls, and shields. The concurrent effects of jets and pressure differentials are also considered in designing walls and shields.

3.6.2.2.2 Pipe Restraint Design Criteria to Prevent Pipe Whip Impact Outside the Containment

The basic philosophy used to prevent high energy pipes from whipping is to provide restraints of sufficient capacity and with such spacing that pipe whipping cannot develop. These restraints are independent of operating and seismic supports. Consequently, they are designed for pipe rupture loads only. Pipe rupture of either guillotine or side-split type are postulated to occur (see Note 1 in Table 3.6-2) for the 36 inch main steam and 18 inch main feedwater lines. Allowable pipe spans are calculated, assuming that the force developed during the accident experience is transferred to the pipe restraint. The pipe restraints are then designed to withstand this force. The restraints are further designed to prevent the pipe from shearing off and generating missiles.

Jet effects from ruptured pipe are considered in designing pipes, walls, and shield. The concurrent effects of jets and pressure differential are also considered in designing walls and shields. Steam generator blowdown pipe whip restraint locations are based on the protection of specific safety-related equipment, not on the maximum span between restraints.

3.6.2.3 Features Provided to Shield Vital Equipment Inside and Outside the Containment Vessel

Subsection 3.6.2.2 describes the restraints provided to prevent pipes from whipping and from generating missiles. The pipe systems are further investigated and designed for jet effects.

Shield walls are provided to protect the Containment Vessel and vital equipment from the effects of jets. These shield walls are designed for the jet effects, acting concurrently with compartment pressure differentials. Section 3.8 defines the structural design criteria for shield walls.

The piping systems which are outside the Containment Vessel are investigated and designed for jet effects. Concrete shielding is provided for vital equipment and systems to guard against the jet impingement forces.

3.6.2.4 Physical Separation of Piping and Components of Redundant Engineered Safety Features for Inside and Outside the Containment Vessel

Wherever practical, physical separation of piping has been utilized to protect the redundant portions of the engineered safety features systems from the rupture and subsequent whipping of the pipes. Where physical separation was riot possible, proper restraints have been provided in accordance with the criteria discussed in Subsection 3.6.2.2 above.

3.6.2.5 Analytical Methods Used

3.6.2.5.1 General

1. Inside Containment Vessel

The following criteria relate to the structural aspects of pipe-rupture-restraint design within the Containment Vessel. The basic philosophy of determining which pipe systems must be restrained is defined in Subsection 3.6.2.1 and Table 3.6-3. Restraints against whipping are not provided where redundant systems or physical separation of systems eliminates the need for them. For longitudinal breaks, pipes are designed to span between restraints without breaking (see discussion in Subsection 3.6.2.5.1.1). This approach eliminates the need to investigate the effects of pipes impacting on safety features and other vital systems since the missile is not allowed to develop. The basic analytical method that was used for stress analysis of the integral supports is given in Subsection 5.2.1.11. The ASME Code, Section III, with certain exceptions, is used as a design basis. These exceptions are noted in the following discussion.

The structural analysis consists primarily of an energy-balance approach. A plastic mechanism is allowed to form. The displacement of this mechanism reaches its limit, by conservation of energy principles, when the external work equals the internal work done on the structure. The dynamic model used is usually considered to be single-degree-of-freedom; that is, all energy is distributed to the one mass-spring system. Since not all the energy is distributed to the one mass-spring system, this approach results in a conservative restraint design. The additional feature of being able to account for all the energy entering the system provides a sound basis for defending this approach.

At the locations where the thermal movement, and therefore the gap between the pipe and the restraint, is small, the equivalent static load approach, as described in Subsections 3.6.2.5.1(2) and 3.6.2.6.3, is used in the design of pipe restraints.

External work equations ordinarily take the form of F(t)dx but may readily include kinetic expressions where mass and velocity of the ruptured pipe are known. Internal work equations are of the same F(t)dx form. This form is graphically represented by the area under a resisting force/displacement diagram.

2. Outside Containment Vessel

The following criteria relate to the structural aspects of pipe-rupture-restraint design outside the containment vessel. The basic philosophy of determining which pipe systems must be restrained is defined in Subsection 3.6.2.1 and Table 3.6-3. Restraints against whipping are not provided where redundant systems or physical separation of systems eliminates the need for them. For longitudinal breaks, pipes are designed to span between restraints without breaking (see discussion in Subsection 3.6.2.5.1.1). This approach eliminates the need to investigate the effects of pipes impacting on safety features and other vital systems since the missile is not allowed to develop.

The structural analysis consists primarily of an equivalent static load approach which utilizes a dynamic amplification factor (DAF), as described in Subsection 3.6.2.5.6.1.2.b, when determining reaction forces. Plastic hinges are allowed to form. However, restraints are spaced so that failure mechanism is not formed.

Several pipe whip restraints were designed by the assumption of full transfer of energy solution of the energy-balance approach. When using this approach, an actual dynamic amplification factor (DAF) value is not selected, but is rather a byproduct of the methodology as the resistance force of the restraint is a function of its elastic and plastic behavior in absorbing the input energy (Ref 45). A plastic mechanism is allowed to form. The displacement of this mechanism reaches its limit, by conservation of energy principles, when the external work equals the internal work done on the structure. The dynamic model used is usually considered to be single-degree-of-freedom; that is, all energy is distributed to the one mass spring system. Since not all of the energy is actually distributed to a one mass-spring system, this approach results in a conservative restraint design. The additional feature of being able to account for all energy entering the system provides a sound basis for defending this approach.

3.6.2.5.1.1 Criteria for Locating Restraints

In postulating pipe break restraint locations the span length is determined such that failure does not occur in case of a longitudinal rupture in the pipe. Design margins against such failure are shown in Table 3.6-9 for those piping systems requiring pipe whip restraints and not directly using the criteria of Regulatory Guide 1.46 and/or SRP 3.6.1 and 3.6.2 including Branch Technical Position MEB 3-1 for inside containment and the letter from A. Giambusso of December 15, 1972 for outside containment. From the table it can be seen that the minimum design margin of 3.17 exists when the yield stress is compared to the material minimum ultimate tensile strength for considering a plastic hinge.

The following formulae were used in determining the span between the restraints:

$$L = \frac{M_\mu}{F} \text{ (for cantilever span)}$$

$$L = \frac{8M_\mu}{F} \text{ (for fixed ends span)}$$

where: F= Rupture Force

$$M_{\mu} = \frac{F_y}{S_p}$$

F_y = Yield stress @ operating temperature

S_p = Plastic section modulus

3.6.2.5.2 Jet Impingement Forces

The pressure loading, on a structure or a barrier at a distance x, due to a jet force is taken as follows:

$$P_j(x) = \frac{F_j}{A_j(x)}$$

where:

F_j = Total jet force and is assumed to be constant throughout its travel.

$A_j(x)$ = Expanded jet area at location x along the jet axis.

The geometry of jet expansion can be seen in Figure 3.6-1. Region 1 extends to the asymptotic area at which point the jet expansion area is calculated according to Moody's method. In region 2, the jet area remains constant, and in region 3 the jet expands at half angle ø = 10°. The extent of region 1 is taken as:

$$X_1 = 5De$$

where De is the inside diameter of the pipe as defined in Figure 3.6-1, and the jet area at location X_1 is given by the equation:

$$A_j(x_1) = \pi R^2{}_{j1} \qquad (R_{j1} \text{ is radius at location } x_1)$$

Region 2 extends to the location of x_2 given by

$$A_j(x_1) = A_j(x), (X = x_2)$$

where $A_j(x)$ is the jet area in region 3 and is calculated by one of the following equations:

1. Guillotine Break;

$$A_j(x) = A_e\left(1 + \frac{2x}{De}\tan ø\right)^2$$

where ø =10° is the half angle of jet expansion

2. Longitudinal (slot) Break;

$$A_j(x) = A_e\left(1+\frac{2x}{\lambda}\tan\ \varnothing\right)\left(1+\frac{2x}{w}\tan\ \varnothing\right)$$

where λ and w are slot dimensions, and A_e is the break area.

3. Circumferential Crack;

$$A_j(x) = A_e\left(1+\frac{2x}{w}\tan\ \varnothing\right)\left(1+\frac{x}{\lambda}(1+2\tan\ \varnothing\right)$$

In region 1, an additional conservative assumption is made that the jet area increases uniformly from A_e at x = 0 to $A_j(x)_1$ @ $x = x_1$

After the jet area A_j is calculated by the above method, the jet pressure is calculated as follows

$$P_j = \frac{F_j}{A_j}$$

$$F_j = K_j P_0 A_e$$

P_0 = maximum operating pressure inside the ruptured pipe

A_e = effective area of the break

K_J = jet impingement coefficient

and the jet impingement load on the target is given by;

$$F_t = P_j x A_{te}$$

where A_{te} is the effective target area.

If the target with physical area A_t, cancels all the fluid momentum in the jet, then

$A_{te} = A_t$

For the case where target is oriented at angle 0 with respect to the jet axis and there is no flow reversal

$$A_{te} = A_t \sin\ \theta$$

$$F_j = KA_b(P_0 - P_a)$$

F_j = impingement force on wall perpendicular to flow, lbs

$(P_0 - P_a)$ = upstream gauge pressure, psig

A_b = break area, in²

3.6.2.5.3 Plastic Mechanisms for Inside Containment Vessel

A plastic formation is used in the analysis. Three plastic hinges are allowed to form when analyzing fixed-end or continuous runs of pipe. Cantilevers develop only one plastic hinge. In some instances, because of the pipe configuration, it was not possible to develop plastic hinges because of support conditions. Each case was thoroughly investigated to determine the plastic condition.

3.6.2.5.4 Design Stresses for Inside Containment Vessel

Stresses of 1.2Fy at the design temperature are used. This increase over static yield stress is explained by allowing some dynamic increase in yielding and/or by allowing a limited amount of strain hardening. Consequently, plastic shape factors for the combined effects of these stresses are used in the analyses.

3.6.2.5.5 Allowable Strains for Inside Containment Vessel

The primary strain criterion used for pipe restraints is 20 percent of ultimate uniform strain at ultimate tensile stress with an additional allowable bending (surface) strain of 20 percent of ultimate uniform strain.

These allowable strains provide a safety ratio of 5.0 when compared to ultimate uniform strain and are considered reasonable for pipe-rupture loads which are postulated to occur only once during the station lifetime.

3.6.2.5.6 Dynamic Amplification Factors for Inside Containment Vessel

When utilizing the Equivalent Static Load Approach, dynamic amplification factors are determined in each design case, considering the effects of gaps. Where large gaps are required, the kinetic energy at impact is frequently the largest significant aspect of the analysis and consequently is not neglected. Dynamic amplification factors are determined, making the following assumptions: the forcing function is assumed to be rectangular and is equal to (1.2)(P)(A) where P is the maximum operating pressure and A is the pipe internal cross-sectional area. A rise time on the force is not employed.

All energy, including that which is impactive (kinetic energy at impact) and/or impulsive (energy developed after impact), is absorbed by the restraint. The restraint, then, is designed for 100 percent of the energy which develops during the postulated pipe break. Investigation of momentum transfer indicates that there is a small conservatism with this approach when performing an inelastic analysis. Dynamic amplifications are determined by performing a single-degree-of-freedom energy-limit analysis. The ratio of the resisting force to the driving force is defined as the Dynamic Amplification Factor (DAF). Impact is assumed to be plastic. This assumption is borne out of the design. Energy-absorbing materials of sufficient capability or the restraint itself are then used to meet the DAF used in design.

3.6.2.5.6.1 Pipe Whip Restraint Design Parameters

A discussion of parameters used in the design of pipe whipping restraints both inside and outside of the containment is presented in the succeeding paragraphs. These parameters include: (1) the thrust coefficient, (2) dynamic amplification factor and (3) gap rebound effects.

1. Thrust Coefficient:

For impingement and reaction forces, a thrust coefficient for saturated steam and water is used to account for the effect of the change in momentum of the escaping fluid. This factor is 1.2. Impingement force thrust coefficients based mainly on theoretical considerations are presented in Table 3.6-4. From this table it is evident that a coefficient of 1.2 is reasonable for calculating the impingement force of a jet from a sharp edged orifice, such as a pipe break. The thrust coefficient of 1.2 is used both inside and outside of the containment.

2. Dynamic Amplification Factor: (Equivalent Static Load Approach)

   a. Inside Containment:

Dynamic amplification factors are determined in each design case considering the effects of gaps. Where large gaps are required, the kinetic energy at impact is frequently the largest significant aspect of the analysis and consequently cannot be neglected. Dynamic amplification factors are usually determined making the following assumptions:

1. The forcing function is assumed to be rectangular and is equal to 1.2PA where P is the operating pressure and A is the pipe internal cross-sectional area.

2. A rise time on the force is not employed.

3. All energy, including that which is ordinarily considered as impactive and/or impulsive, is carried across the gap. The restraint, then, is designed for 100% of the energy which develops during the postulated pipe break. Investigation by momentum transfer indicates that there is a small conservatism with this approach when performing an inelastic analysis.

4. Dynamic amplification factors are determined by performing a single degree of freedom energy limit analysis. This analysis is performed in accordance with Section 5.5b of Biggs, John M., “Introduction to Structural Dynamics,” McGraw Hill Inc., 1964. The ratio of the resisting force to the driving force is the dynamic amplification factor by definition.

5. Impact is assumed to be plastic. This assumption is borne out in the design. For the majority of pipe restraints and for all large line pipe restraints inside containment, the Dynamic Amplification Factor (DAF) is established on a case-by-case basis using energy-balance methods as discussed above. For small line pipe restraints, a generic DAF was selected as discussed below.

A comparison of order of magnitude values for dynamic effects (DAF) for differing gaps is presented below. A single degree of freedom dynamic model is used in the analysis with results as shown. The force/time function is assumed to be rectangular, i.e., no rise time is used. Resisting functions are triangular in the elastic range and trapezoidal in the plastic range.

Dynamic Amplification Factors (DAF)

	Gap Size in Inches				
Design Criteria	0.01	0.10	1.0	2.5	5.0
Ductility=5.0	1.18	1.90	9.03	21.0	40.8
Ductility=20.0	1.04	1.21	2.85	5.60	10.15
20% Ultimate Uniform Strain @ Minimum Ultimate Tensile Stress	1.01	1.09	1.89	3.23	5.48
Energy Absorbing Material * Thickness of 2 x gap Dimension, 20% Ultimate Uniform Strain @ Minimum Ultimate Tensile Stress	-	-	1.47	1.67	1.83
Energy Absorbing Material * Thickness of 3 x gap Dimension, 20% Ultimate Uniform Strain @ Minimum Ultimate Tensile Stress	-	-	1.38	1.52	1.58

*Energy Absorbing Material (EAM) is not used in current whip restraint designs. Current whip restraint designs are based on the capacity of the structural steel members of the restraint structure itself.

The representative small line piping restraints were designed with a DAF equal to 1.67. Thus the total design force is equal to the product of the thrust coefficient, dynamic amplification factor, operating pressure and pipe flow area, i.e., F = 1.2 x 1.67 x P x A = 2.0 PA. This is the design force used inside the containment for small piping. The table indicates that a DAF other than 1.67 can be used depending on the size of the gap. The DAF of 1.67 used for small piping is justified based upon the following:

1. Gap dimension is small.

2. Insulation thickness is reduced at whip restraints.

3. Hangers and anchors are located so that thermal and seismic movements are small.

2.b. Outside Containment:

When utilizing the Equivalent Static Load Approach, a dynamic amplification factor, nominally 1.25, is included in determining reaction forces. This factor depends on the natural frequency and ductility factors of the supports and structures involved.

Dynamic amplification factors based upon the following Bechtel design criteria are presented in Table 3.6-8. The dynamic amplification factor depends on:

(1) Ductility Factor =

Max. Allowable Deflection/Yield Deflection

(2) Ratio of Load Duration/Period of Structure

The factors presented in Table 3.6-8 were derived for the thrust time curves supplied by various reactor manufacturers.

It is evident that there is not much difference in the values presented in Table 3.6-8. Thus, for most circumferential or longitudinal pipe breaks, the values in either part of the table are acceptable for design purposes.

In the case of a concrete structure a reasonable value for the ductility factor is 3. The range of frequencies for individual structural members is large. However, the maximum frequency can be expected to be less than 50 cps. Thus, from a design standpoint, a conservative dynamic load factor is 1.25.

In the case of steel members, ductility factors are much greater than 3. The natural frequency of such members is approximately 10 cps. On the basis of the data presented in Table 3.6-8 it is evident that a dynamic amplification factor of 1.25 is conservative.

No additional loads on pipe supports are assumed because of the possible momentum of the pipe in moving from its normal position to the pipe stop on the restraint. Although this momentum energy can be large, the full break and the jet thrust from a break cannot be developed without some movement in the pipe.

Thus the total design force is equal to the product of the thrust coefficient, dynamic amplification factor, operating pressure and pipe flow area, i.e., $F = 1.2 \times 1.25 \times P \times A = 1.5\ PA$. This is the design force used outside the containment when utilizing the Equivalent Static Load Approach.

3. Gap Rebound Effects:

For a discussion of gap rebound effects reference is made to BN-TOP-2 Rev. 2, May, 1974, "Design for Pipe Break Effects", Bechtel Power Corporation, San Francisco, California, Appendices A and C. This topical report concludes that gap rebound effects are only of concern for the case where the rupture force is smaller than the pipe resistance. In all cases for this design the rupture force is equal to or greater than the pipe resistance. Maximum response occurs after the first quarter cycle of response. Even when rebound is of concern, the topical report recommends a 10% increase in the rupture force for inelastic design and not the 50%.

4. Conclusion:

The value of 3.78 is equal to the product of the factors 1.26 x 1.5 x 2.0. Apparently the dynamic amplification factor of 2.0 is for an elastic design which is not the case for these restraints which are all based on inelastic design. For rebound effects a factor of 1.0 rather than the 1.5 factor is used in design since gap rebound effects are not a problem. A factor of 1.2 instead of 1.26 is used to account for the change in momentum of the escaping fluid. The difference between 1.2 and 1.26 is so small (<5%) as to be negligible.

BN-TOP-2 (previously referenced) procedures were used for a design comparison with restraint design based on the inside and outside of containment criteria stated above. An analysis of a representative number of restraints using BN-TOP-2 as a common criteria for both inside and outside of containment demonstrated their adequacy.

5. Additional Analyses:

Additional analyses were performed to justify that the pipe whip restraints are adequate as designed. The worst case (most critical condition), both inside and outside of containment was analyzed with results as follows:

a. Inside Containment:

Pipe Line Analyzed:	26” 0.D. Main Steam Line (Largest gap and smallest ductility)
Design Force:	Fx = 1.2 x 1.2 x 1.67 x P x A thrust coefficient = 1.2 rebound factor = 1.2 Dynamic Amplification Factor = 1.67 Operating Pressure = P Pipe Flow Area = A
Design Allowable:	Stress = 1.2 Fy Strain = 50% ultimate uniform strain μ = 80
Results:	Actual Stress = 0.95Fy μ = 2.4

b. Outside Containment:

Pipe Line Analyzed:	36” 0.D. Main Steam Line
Design Force:	Fx = 1.2 x 1.2 x 2.0 x P x A thrust coefficient = 1.2 rebound factor = 1.2 Dynamic Amplification Factor = 2.0 Operating Pressure = P Pipe Flow Area = A

Design Allowable: Stress = 1.2 Fy
Strain = 50% ultimate uniform strain
μ = 80

Results: Actual Stress = 1.08 Fy
μ = 38

Based upon the analyses of the worst cases (most critical conditions) listed above, the design criteria outlined in Section 3.6 covering inside and outside of containment is adequate for design of all pipe whip restraints. It should also be noted, that if all pipe whip restraints are analyzed based on the above criteria the actual stresses and strains will be below allowables.

3.6.2.5.7 Safety Margins

1. Inside the Containment Vessel:

   Safety margins are best indicated in this method of analyses by comparing the total energy-absorbing capabilities of the structures to the actual work done on the structure. When working to strain limits of 20 percent ultimate uniform strain at ultimate tensile stress, a factor of 5.0 is obtained.

2. Outside the Containment Vessel:

   Safety margin is assured in this method of analyses by the following:

   a. Gaps between the pipe and sleeve have been kept to the minimum required to allow for thermal movement of the pipe.

3. Insulation has been cut down to the minimum required.

   a. Stresses in the restraints have been kept within the yield stresses of the material.

   b. Buckling effects have been thoroughly investigated because of the high stress levels being used.

   c. Notch effects have been carefully investigated and excluded from design.

3.6.2.5.8 Other Considerations for Inside Containment Vessel

1. Buckling Effects:

   Buckling effects are thoroughly investigated because of the high stress levels being used. Obviously, a lower collapse load would be reached if buckling should occur; however, in all cases investigated in pipe-restraint design buckling does not occur at the 1.2Fy stress level, because of the heavy structural sections used in the design.

2. Notch Effects (stress concentration):

   Should a notch effect develop in a given restraint, the capability to distribute strain energy uniformly over the members would be lost. This could develop in

bolted connections where the bolts are the weakest link in the structure. The effect of notches would be to mobilize all the plasticity in a localized area without distributing this energy uniformly throughout the structure with the result that a failure at the notch might develop. Consequently, notch effects are carefully investigated and excluded from the design.

3. Hammering Effects:

In reviewing the proposed method of analysis, it should be remembered that the elasto-plastic deflection developed associates with that first time when the mass-spring reaches its limit (stops moving). This is actually the first quarter cycle, during which time most of the energy expended on the structure is in the plastic range. Investigation of the rebound effects in plastic impact problems indicates that the critical consideration for design is that point where the mass-spring system first comes to rest. Subsequent cycles invariably produce a reduced response and therefore are not critical. This is the case for plastic impact with a constant level of applied force and is not applicable to elastic models with cyclic loading where resonance might amplify the dynamic effects.

3.6.2.5.9 Design Loads for Structures Outside the Containment

The restraint structures that utilize the Equivalent Static Load Approach, are designed to withstand the equivalent static force computed as follows:

$F = (K_j)(DAF)(P)(A)$

where:

F = Equivalent Static Force
K_j = Thrust multiplication factor
DAF = Dynamic Amplification Factor (1.25)
A = Break flow area
P = Maximum operating pressure

Thrust multiplication factor K, accounts for the increased momentum of the escaping fluid and the differential between the pipe internal pressure and the outside ambient pressure. The values of thrust multiplication factors are shown in Table 3.6-4 for various conditions.

3.6.2.5.10 Types of Ruptures

Two types of instantaneous ruptures of the high energy piping are assumed for the restraints. These are double-ended and longitudinal split ruptures.

1. Double-Ended Rupture

A double-ended rupture is an assumed instantaneous break occurring at any point along the length of the pipe with such a configuration that the pipe lines retain no structural integrity of their own. A break of this type would occur perpendicular to the longitudinal center line of the pipe.

2. Longitudinal Split Rupture

   A longitudinal split is assumed to be an instantaneous break, occurring along the axis of the pipe, opening the wall of the pipe equivalent in area to that of the cross section of the pipe. The length of the wall split is assumed to be twice the diameter of the pipe.

3.6.2.5.11 Types of Restraints

Two redundant support and restraint systems have been provided. One is for normal, upset, and emergency conditions, and the other is for accident conditions. This has been discussed in more detail in Subsection 3.6.2.6.

3.6.2.5.12 Design Stresses for Outside the Containment

Stresses in the restraint structures are not allowed to exceed 90 percent minimum code-specified yield strength of the material.

3.6.2.6 Design Loading Combinations, Design Condition Categories, and Design Stress Limits

3.6.2.6.1 Pipe Supports, Inside the Containment Vessel

The pipes are supported by redundant systems. The first support system is designed for normal, upset, and emergency conditions, excluding pipe rupture. This system is designed to function while remaining in the elastic range for the Loading conditions and at the stress allowables defined in Section 3.8.

During the postulated accident condition, the pipe is driven across a thermal gap and impacts on the restraint. Subsection 3.6.2.5 describes the analytical methods used, loading conditions, and the allowable stress and strain limits employed during the dynamic transient. Following the dynamic transient, a steady-state blowdown condition exists during which time the restraint again functions in an elastic manner. Investigation of the pipe-rupture incident confirms that progressive plastic deformations do not occur.

Dynamic system analysis are performed for loading conditions with the exception of pipe rupture loads. Pipe stresses and strains are held within the ASME Code allowables for these loading conditions.

During the pipe-rupture incident, pipe system analyses are not performed other than for the reactor coolant system. The simplified approach described in Subsection 3.6.2.5 is used in designing the restraints. Investigation of the change in geometry experienced by the pipe during the pipe-rupture transient indicates that stresses and strains for the ruptured section of pipe are of the same magnitude as those for the restraint; that is, the stress for this condition of loading is at yield stress and the strains are limited to 20 percent of ultimate strain. These criteria ensure that the pipe is not severed during the accident and, consequently, that pipe missiles are not generated.

3.6.2.6.2 Reactor Coolant System Component Supports

a. The integral supports for the principal Reactor Coolant System components are designed to meet the stress and deformation requirements of Section III Class 1 of the ASME Code. (See Table 5.2-4 for design stress limits and categories.)

b. The nonintegral supports were analyzed for the following load cases and corresponding allowable stresses:

1. Reactor Vessel Nozzle Supports

Case I: Dead Weight + Operating Basis Earthquake

Shear Allowable = 0.4 Sy
Membrane Allowable = 0.6 Sy
Bearing Allowable = 0.9 Sy
Bending Allowable = 0.66 Sy

Case II: Dead Weight + Design Basis Earthquake

Shear Allowable = 0.48 Sy
Membrane Allowable = 0.72 Sy
Bearing Allowable = 1.08 Sy
Bending Allowable = 0.8 Sy

Case III: Dead Weight + Design Basis Earthquake + LOCA

Shear Allowable = 0.42 Su
Membrane Allowable = 0.7 Su
Bearing Allowable = 1.08 Su
Bending Allowable = 1.05 Su

Effects of Asymmetric LOCA Loadings: Dead Weight + Asymmetric Cavity Pressure LOCA

$$\text{Membrane Allowable} = \text{Lesser of} \begin{cases} 2.4 \quad \text{Sm} \quad (1.6\ \text{Sy}) \\ 0.7 \quad \text{Su} \end{cases}$$

Membrane + Bending Allowable = 1.5 x Membrane Allowable

2. Steam Generator Support Cone & Support Platform

Design Condition Allowable Stress Intensity for Cone

$P_m \leq S_m$
$P_L \leq 1.5 * S_m$
P_m (or P_L) + $P_b \leq 1.5 * S_m$

Level D Allowable Stress Intensity for Cone

$P_m \leq 0.7 * S_u$
$P_L \leq 1.05 * S_u$
P_m (or P_L) + $P_b \leq 1.05 * S_u$

Design Condition Allowable Stress Intensity for Platform (Linear Type Support)

Bearing stress	$\sigma_{bearing}$	$\leq$	S_y
Shear stress	$\tau_{average}$	$\leq$	$0.6\ S_m$
Partial penetration weld normal stress	σ_{p_norm}	$\leq$	$0.3\ S_{u\_weld\_metal}$
Partial penetration weld shear stress	τ_{p_weld}	$\leq$	$0.3\ S_{u\_weld\_metal}$
Partial penetration base shear stress	τ_{p_base}	$\leq$	$0.4\ S_{y\_base\_metal}$
Fillet weld shear stress	τ_{f_weld}	$\leq$	$0.3\ S_{u\_weld\_metal}$
Fillet base shear stress	τ_{f_base}	$\leq$	$0.4\ S_{y\_base\_metal}$
Bolt average tensile stress	σ_{bolt_ave}	$\leq$	$0.5\ S_u$
Bolt shear stress	τ_{bolt_shear}	$\leq$	$0.62\ S_u/3$

Bolt combined stress ratio $$\left(\frac{\sigma}{0.5S_u}\right)^2 + \left(\frac{\tau}{0.62S_u/3}\right)^2 \leq 1.0$$

Level D Allowable Stress Intensity for Platform (Linear Type Support)

For Level D condition per F-1334, the above Design stress limits can be increased by the Level D factor f_D as follows:

If $S_u > 1.2\ S_y$ then f_D is equal to the smaller of 2.0 and $1.167\ S_u/S_y$
If $S_u \leq 1.2\ S_y$ then $f_D = 1.4$

3.6.2.6.3 Design Condition Limits for Pipe Supports Outside the Containment

The pipes are supported by redundant systems. The first support system is designed for normal, upset, and faulted conditions excluding pipe rupture. These support systems are designed to function while remaining in the elastic range for the stress allowables defined in Section 3.8.

During the postulated accident condition, the pipe is driven across a thermal gap and impacts on the restraint. Subsection 3.6.2.5 describes the analytical methods used and the loading conditions and allowable stress limits employed during the dynamic transient.

Dynamic system analyses are performed for loading conditions with the exception of pipe rupture loads. Pipe stresses and strains are held within the ASME Code allowables for these conditions.

During the pipe-rupture incident, pipe system analyses are not performed on the systems. The simplified approach described in Subsection 3.6.2.5 is used in designing the restraints. Investigation of the change in geometry experienced by the pipe during the pipe-rupture transient indicates that stresses and strains for the ruptured section of pipe are of the same magnitude as those for the restraint. These criteria ensure that the pipe is not severed during the accident and, consequently, that pipe missiles are not generated.

3.6.2.7 Protection Against Environmental Effects Outside the Containment Vessel

3.6.2.7.1 High Energy Fluid Piping Systems

High energy fluid piping systems outside the containment vessel, within Seismic Class I structures were subjected to an analysis to determine the environmental effects of a postulated rupture on safe shutdown. Those systems exceeding the 275 psig and/or 200°F criteria are listed in Table 3.6-5. (See Table 3.6-3 for systems with pipe whip and jet impingement protection and Table 3.6-1 for failure criteria).

Limiting high energy fluid piping systems outside containment within the Turbine Building were subjected to an analysis to determine environmental effects on safe shutdown equipment located in adjoining Auxiliary Building rooms that can communicate with the Turbine Building atmosphere. The details of this analysis are provided in Section 3.6.2.7.1.16.

3.6.2.7.1.1 Compartment Pressurization

The following describes the analytical techniques used to evaluate high-energy pipe rupture.

Initial Compartment Conditions:

The masses of air and water as steam in the compartments are determined using the initial input conditions of temperature pressure, relative humidity, and compartment volumes. The specific humidity of saturated air at the compartment temperature is read from a correlation table of temperature and water vapor in saturated air. The compartment specific humidity is obtained by:

SH = (RH)(SSH)

where:

SH = specific humidity of compartment air, lb steam/lb air

RH = relative humidity of compartment air

SSH = specific humidity of saturated air at compartment temperature, lb steam/lb air.

The vapor pressure of the water is determined by:

$$P_w = \frac{(SH)(PT)}{0.623 + SH}$$

where:

P_w = vapor pressure of water at compartment temperature, psia

P_t = total compartment pressure, psia.

The air pressure in the compartment is determined by:

$$P_a = P_t - P_w$$

The mass of air in the compartment is evaluated using the perfect gas law equation:

$$MA = \frac{(144)(P_a)(V)}{(R/n)(T)}$$

where:

V = volume of compartment, ft^3

R = gas constant, 1545.3

T = compartment temperature, °R

n = molecular weight of air, 28.97 lb/lb mole.

The mass of water in the compartment, MS, is:

MS = (MA) (SH)

The masses of air and water in the remaining compartments are determined in the same manner.

The energy of the air in each compartment is calculated using 0°F as a base:

UA(I) = [CV] [MA(I)] [TP]

where:

CV = specific heat of air at constant volume, 0.171 Btu/lb -°F

TP = compartment temperature, °F.

The energy of the water vapor in each compartment is calculated by the equation:

US(I) = [MS(I)] [UG]

where:

UG = internal energy of the steam evaluated from the saturated steam tables at the compartment temperature.

Conservation of Mass and Energy in Compartments:

The inventory of the total mass and energy in the compartments is maintained from the inlet and exit flows during the time increment:

$$MA(I) = MA'(I) + \sum^{N} |MAI| - \sum^{N} |MAO|$$

$$MW(I) = MW'(I) + \sum^{N} |MWI| - \sum^{N} |MWO|$$

$$MS(I) = MS'(I) + \sum^{N} |MSI| - \sum^{N} |MSO|$$

$$MV(I) = MW(I) + MS(I)$$

$$MT(I) = MV(I) + MA(I)$$

$$UA(I) = UA(I) + \sum^{N} |UAI| \sum^{N} |UAO|$$

$$UW(I) = UW(I) + \sum^{N} \left(HI|MWI|\right) - \sum^{N} \left(HO|MWO|\right)$$

$$\mathit{US(I)} = \mathit{US(I)} + \sum^{N} \left(HGI|MSI|\right) - \sum^{N} \left(HGO|MSO|\right)$$

$$UW(I) = UW(I) + \sum^{N} \left(HI|MWI|\right) - \sum^{N} \left(HO|MWO|\right)$$

$$US(I) = US(I) + \sum^{N} \left(HGI|MSI|\right) - \sum^{N} \left(HGO|MSO|\right)$$

$$UV(I) = UW(I) + US(I)$$

$$UT(I) = UV(I) + UA(I)$$

Where:

Primed (‘) values refer to end of previous time step, all other values refer to current time step

MA(I) = mass of air in compartment

MW(I) = mass of water in compartment (I), lb

MV(I) = mass of water and steam in compartment (I), lb

MT(I) = total mass in compartment (I), lb

MAI = mass of air entering compartment, lb

MAO = mass of air leaving compartment, lb

MWI = mass of water entering compartment, lb

MWO = mass of water leaving compartment, lb

MSI = mass of steam entering compartment, lb

MSO = mass of steam leaving compartment, lb

UAI = total energy of air entering compartment, Btu

UAO = total energy of air leaving compartment, Btu

HI = enthalpy of water entering compartment (I), Btu/lb

HO = enthalpy of water leaving compartment (I), Btu/lb

HGI = enthalpy of steam entering compartment (I), Btu/lb

HGO = enthalpy of steam leaving compartment (I), Btu/lb

UA(I) = energy in air in compartment (I), Btu

UW(I) = energy in water in compartment (I), Btu

US(I) = energy in steam in compartment (I), Btu

UV(I) = energy in vapor in compartment (I), Btu

UT(I) = total energy in compartment (I), Btu

Compartment Pressure Calculations:

The compartment pressure is calculated using the total mass and energy in the compartment after the flow from the upstream compartments and/or the blowdown has been added to the compartment inventory of mass and energy. A convergence procedure is used to arrive at the equilibrium thermodynamics conditions in the compartment using temperature as the trial argument. The equilibrium thermodynamic state is considered determined when the trial temperature provides properties such that the ratio of the difference between the trial energy balance and the energy inventory is less than 0.001. The state properties of the steam and water mixture at the trial temperature are obtained from the saturation tables. The mass of steam is then determined by:

$$MS = \left[(V) - (MW_1)(VL)\right] / VG$$

where:

V = volume of compartment, ft^3

VL = specific volume of water, ft^3/lb

VG = specific volume of steam, ft^3/lb

MW_1 = mass water from previous iteration, lb

The mass of water (MW) is determined by:

$$MW = MV - MS$$

A trial energy balance is calculated:

$$\begin{aligned} ETRIAL = {} & (MS)(UG) + (MW)(UL) \\ & + (0.171)(MA)(TP) \end{aligned}$$

The procedure is repeated varying the value of TP until the relation:

$$(UT - ETRIAL)/UT \leq 0.001$$

is satisfied.

If, after establishing the thermodynamic equilibrium conditions $MW \leq 0$ the compartment is considered to be superheated, the equilibrium conditions are recalculated by setting the steam mass equal to the vapor mass and calculating the steam pressure at the search temperature by:

$$PS = (0.5961)\,(MS)\,(T)/V$$

PS = pressure of steam, psia

T = compartment search temperature, °R

V = compartment volume, ft^3

$$0.5961 = R/(\text{mole weight})\,(144) = 1545.3/(18)(144)$$

The internal energy of the steam at the pressure and temperature is obtained from the superheat tables and a trial energy balance calculated by:

$$ETRIAL = (MS)(UG) + (0.171)(MA)(TP)$$

The procedure is repeated varying the value of TP until the relation:

$$(UT - ETRIAL)/UT \leq 0.001$$

is satisfied.

The total pressure in the compartment is the sum of the steam pressure and the air pressure with the latter being calculated by:

$$PA = (0.37MA)(TP + 459.688)/V$$

where

$$0.37 = R/(\text{Mole Weight})(144) = 1545.3/(28.97)(144)$$

Flow Calculation:

Two flow equations are provided for calculating the flow between compartments. The Moody equation is used for the analysis of reactor cavity pressures resulting from the decompression of the primary coolant system and for other compartments where the blowdown results in single-component, two-phase flow fairly early in the transient. A compressible fluid flow equation is used for the analysis of steam generator compartment pressures for the main steam line breaks and for other compartments where the blowdown results in two-component, two-phase flow for all of the transient or that portion of the transient through the maximum peak pressure.

In the application of the Moody equation for calculating the flow from Compartment 1 to Compartment 2, the flow is assumed to be critical if the pressure in Compartment 2 is less than 0.55 times the pressure in Compartment 1. If the flow is critical, the throat pressure is set equal to 0.55 time Compartment 1 pressure.

For sub-critical flow, the form of the Moody equation is:

$$G = \left[\frac{2(g_c)(J)(HO - H2)}{\left[\frac{X(VG)}{K} + (1 - X)(VF)\right]^2 \left[X(K)^2 + 1 - X\right]} \right]^{1/2}$$

where HO is the stagnation enthalpy of the fluid in Compartment 1 and the remaining state terms are evaluated at Compartment 2 pressure. For isentropic flow, the formula is evaluated by the equations in Compartment 1:

HO = UV(I) / MV(I)

X = (HO – HF) / (HG – HF)

SO = SF + X (SFG)

and then in Compartment 2:

X = (SO – SF) / (SG – SF)

H2 = HF + X (HFG)

$K = (VG/VL)^{1/3} (1.224) (P1 - P2) / P2$

The state properties for Compartments 1 and 2 are obtained from the saturation tables at the pressures in the compartments. For critical flow, the form of the Moody equation is:

$$G = \frac{\left[2(g_c)(J)(HO - HT)\right]^{1/2}}{\left[(1 - x)(VF)^{2/3} + X(VG)^{2/3}\right]^{3/2}}$$

where for Compartment 1:

HO = UV(I) / MV(I)

X = (HO-HF) / (GH – HF)

SO = SF + X (SFG)

and for the throat:

PT = (0.55) PT (I)

X = (SO-SF) / (SG-SF)

HT = HF + X (HFG)

The state properties for Compartment 1 and the throat are obtained from the saturation tables at the respective pressures in the compartment and throat.

For both the sub-critical and critical flow conditions, the calculated value of the flow is decreased to sixty percent of the flow.

In the application of the compressible fluid flow equation, if the ratio of the pressure in the downstream compartment (Compartment 2) to the pressure in the upstream compartment (Compartment 1) is less than RC as obtained by:

$$RC = \left[\frac{2}{1+K}\right]^{\frac{K}{K-1}}$$

the flow is considered to be critical. The form of the flow equation is:

$$G = \left[(g_c)(K)(P1)(RH01)\left\{\frac{2}{K+1}\right\}^{\frac{K+1}{K-1}}\right]^{1/2}$$

The isentropic exponent K for the air, steam, and water mixture leaving the compartment is calculated by:

$$KA = \frac{(KGF)(PS(I))}{PT(I)} + \frac{PA(I)}{PT(I)}$$

where:

KA = isentropic value of K for air of 1.4

KGF = isentropic value of K for steam-water mixture

RH01 = MT(I)/VOL(I), lb/ft^3

P1 = Compartment 1 pressure, psia
g_c = gravity acceleration, 32.174 ft/sec^2

If the flow is sub-critical, the form of the flow equation is:

$$G = \left[\frac{2(g)(P1)(RH01)K\left[R^{\frac{2}{K}} - R^{\frac{K+1}{K}} \right]}{K-1} \right]^{1/2}$$

where the terms are as previously defined and R = P2/P1, P2 = Compartment 2 pressure.

The mass flow for both the compressible fluid flow equation and the Moody equation is calculated by:

total MF = (G)(A)(C),lb

air MAF = [MF][MA(I)]/MT(I), lb

water MWF = [MF][MW(I)]/MT(I), lb

steam MSF = [MF][MS(I)]/MT(I), lb

The energy transferred by the flow is:

air UAF = [(MAF)(CP)] [TC(I)], Btu

water UWF = (MWF)(HL), Btu

steam USF = (MSF)(HG), Btu

where:

A = area of flow path, ft^2

G = mass flow, lb-sec^2/ft

C = coefficient calculated external to code

CP = specific heat of air at constant pressure

HL = enthalpy of water at compartment temperature

HG = enthalpy of steam at compartment temperature

The flow coefficient “C” was calculated using the same methods as outlined in the COPRA Computer Program which has been previously submitted for NRC review in NS-731-TN, “Containment Pressure Analysis”. Power and Industrial Division, Bechtel Corporation, San Francisco, California (December 1968).

The computer code used to predict temperature effects due to main steam line, main feedwater piping, main steam to AFPT lines (below 643’ elevation), steam generator blowdown lines, and

auxiliary steam lines breaks/cracks in the Auxiliary Building is PCFLUD written by the Bechtel Corporation. This code is a derivative of the COPDA code which is described above. PCFLUD is formulated in a functionally similar manner to the COPDA code, and utilizes additional relations which are in conformance with NRC guidance as provided in NUREG 0588. Within compartments, PCFLUD assumes homogeneous mixing of all air, vapor, and liquid phases (with thermodynamic equilibrium between all phases). Convective and condensing heat transfer coefficients which are provided by PCFLUD (or user-supplied coefficient tables) allow modeling heat transfer into one-dimensional finite difference heat sinks. Condensing heat transfer is based on the UCHIDA condensing heat transfer correlation. Use of only 8 percent re-evaporation contributes to conservative results for the prediction of average room temperatures.

The computer code used to predict pressure effects due to main steam line, main feedwater piping, main steam to AFPT line (below 643' elevation), steam generator blowdown line, and auxiliary steam line breaks/cracks in the Auxiliary Building is PCFLUD, described above, and/or GOTHIC. Both codes are further described in section 3.11.2.2.1.

The computer program utilized in the Turbine Building High Energy Line Break analysis can model non-homogeneous fluid distributions in one, two, or three dimensions. This permits intra-volume flow to be modeled so that buoyancy driven mass and energy flow can be included. The heat transfer and condensation correlations utilized provide conservative estimates of environmental parameters. The most limiting high energy lines were selected for analysis based on their high energy content and proximity to structures containing safety related equipment. The mass and energy release of the breaks was based on an infinite constant source until break isolation using safety related equipment would occur. For the most limiting break, a Main Steam Line Break, the mass and energy release was based on Reference 67. This models the worst case single failure.

3.6.2.7.1.2 Structural Analysis for Shield Building and Auxiliary Building for High Energy Lines

The effects of a postulated rupture in a high energy piping system on structural safety were analyzed to assure, to the necessary extent, that safe shutdown condition of the reactor can be accomplished and mitigation of the consequences of the accident insured. The high energy piping system outside the containment is mainly housed in areas of the Auxiliary Building, a Class I structure. All structural elements in those areas are analyzed and checked for pipe whipping, jet impingement, temperature, and compartment pressurization. The structural elements are checked for rupture loads, and some structural elements are modified to withstand the effect of pipe breaks.

A general description of the Auxiliary Building is contained in Section 3.8 of the USAR. Detailed results of the calculation for the effects of compartment pressurization in the Auxiliary Building are discussed in various other Subsections of 3.6. Figures 3.6-2, 3.6-3, 3.6-4, 3.6-6, and 3.6-7 show the general arrangement of each floor in the Auxiliary Building

The containment shield building cylinder wall is checked for external pressure due to high energy line rupture in the Auxiliary Building. The detailed description of the computer application and analysis results is given in Paragraph c of this Subsection.

Vent areas between compartments are checked. Some new openings are added as apart of the modification, and some structural elements, such as beams, slab, walls, etc., are modified for the high energy line breaks.

Load combinations, design criteria, and analytical methods used in checking the compartment pressurization are discussed in Section 3.8.

a. Evaluation of Seismic Class I Structures

The Seismic Class I structures are evaluated for structural adequacy following a postulated high energy line rupture, using the design bases shown in Section 3.8.

The allowable stresses and strains in checking existing concrete structural elements are the same values mentioned in Section 3.6.

b. Structural Adequacy Evaluation for Auxiliary Building

The loads on structures resulting from a postulated high energy line break consist primarily of differential pressure acting on the various walls, slabs, and ceilings. Peak values were used in all areas of each floor. All those affected structural elements are checked for this predicted pressure and for maximum stress limit of structural element.

As a result of this checking, the maximum pressure capability for the external walls, slabs, beams, and ceilings are tabulated in Table 3.6-6 for the affected rooms.

All the slabs in the auxiliary building which are affected by this pressure due to postulated high energy line breaks are checked for their structural adequacy for downward and upward loads.

c. Structural Adequacy of Containment Shield Wall:

The compartment pressurization in the upper levels of the Auxiliary Building due to postulated high energy line rupture affects part of the containment shield wall. The possible pressure areas outside the containment wall are as shown in Figure 3.6-8.

In order to analyze the effect due to this external pressure on the containment wall, a Bechtel Computer Program ("CE-771, Fortran IV, Analysis of Axisymmetric Shell and Solid Subjected to Non-Axisymmetric Static Loading, Dynamic Load Base Acceleration") was used.

For inputting Fourier coefficients to allow for Fourier description of non-axisymmetric loading, a program called "POLYFOUR" was also used as an aid to CE-771 to find coefficients for arbitrary function. The possible largest pressure area on the containment shield wall is shown on Figure 3.6-8.

For the maximum value of stresses, moments, and displacements found from the output, see Figure 3.6-9.

3.6.2.7.1.3 Environmental Effects Due to HELB Outside Containment in the Auxiliary Building

High Energy Line Break analyses Outside Containment were performed to predict temperature and pressure conditions that would result at essential equipment locations following pipe breaks in the following lines in the Davis-Besse unit 1 Auxiliary Building:

Main Steam to Auxiliary Feedwater Pump Turbine (MSAFPT)
Main Steam (MS)
Main Feedwater (MF)
Steam Generator Blowdown (SGBD)
Auxiliary Steam (AS)

The purpose of these analyses was to provide equipment qualification (EQ) parameters for essential equipment that is affected by the breaks and required to function in order to mitigate the break. [Note: Analysis for pipe rupture dynamic effects (pipe whip, jet impingement) and subcompartment integrity for pressurization are presented in Sections 3.6.2.7.1.1 and 3.6.2.7.1.2]. Postulated break locations and HELB system identification criteria are outlined in USAR Section 3.6.2.1. This information, along with existing Mass and Energy release data for these systems was used as the primary input to the environmental analyses performed. Additional input or assumptions used in these analyses are presented in Table 3.6-10. Results of these analyses are presented in Table 3.6-11.

Additional analyses have been performed in support of Technical Specification Amendment 192 for environmental effects due to a HELB in MODE 3 with a greater initial steam generator inventory than previously assumed. See Subsection 15.4.4.3.3 for details.

Results obtained from the HELB analyses were used to evaluate all equipment required to operate in order to provide an orderly safe shutdown capability, and/or mitigate the HELB initiating event. The Environmental Qualification Program described in Section 3.11 formed the basis of this evaluation. Individual equipment qualification books have been developed for each component and a Master Equipment list is maintained and controlled to provide a summary listing typically by Plant ID number, Description, EQ Package number, Manufacturer, Model, Location, but may contain more or less information provided the link between the plant equipment and the qualification documentation is maintained. Individual System/Component Evaluation Worksheets (SCEW) are provided in each equipment qualification book to define the unique evaluation for each component. This evaluation has taken into account system and component operability requirements, temperature and pressure margins from the test/analysis used to qualify the component, radiation effects, and potential consequences of submergence.

Failure modes and effects analyses were used to identify adverse failures of components that may be exempt from one or both of the Design Basis Events used in this evaluation (LOCA, HELB). These components have been annotated on the Master Equipment List for ease of identification and their SCEW sheets provided detailed notes to document this evaluation.

3.6.2.7.1.4 Main Steam to the Turbine Generator

Description:

The main steam system under normal operating conditions are 925.5 psia and 597.7°F at the steam generator. For a description of the main steam system, see Section 10.3. The physical arrangement of this system in the Auxiliary Building is shown in Figure 3.6-2.

Proximity to required safety-related equipment:

The following safety-related equipment or associated power and/or control cables are located in room number 601:

a. Steam Generator number 1-1 main steam line isolation valve.

b. Main steam warmup line isolation valve.

c. Main steam line isolation valve bypass valve.

d. Atmospheric vent valve.

e. Steam Generator number 1-1 main steam relief valves.

Similar safety-related equipment, as listed above, is located in room 602 (for Steam Generator 1-2).

Postulated Design Basis Break Locations:

For pressurization and environmental effects, pipe ruptures were originally considered to occur anywhere. However, Figures 3.6-10 and 3.6-11 are included to indicate selected pipe rupture locations in accordance with NRC's "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment", (Reference 40) or NRC Standard Review Plan 3.6.2, Branch Technical Position MEB 3-1 (Reference 53). The break locations are designated by the prefix (MS). A critical crack is not indicated since no unrelated safety equipment is affected, and equivalent pipe area cracks were considered to occur anywhere.

Compartment Pressurization:

Pressurization studies were conducted in rooms 601, 601A, 602, and 602A. The mass and energy release from the break were predicted with the RELAP5/MOD2-B&W (see Section 1.5.1, BAW-10164P-A) computer code. A double-ended guillotine rupture of the Main Steam line between the steam generator and the Main Steam Isolation Valve was modeled. An initial core power level of 2,827 MWt was analyzed. Several assumptions were made to maximize the mass and energy release through the break. For example, the Main Steam Line Break (MSLB) predictions resulted in some liquid carryout which produces a lower than expected enthalpy for the break effluent. In order to ensure a conservative calculation of the room's pressure and temperature, the enthalpy of the break affluent was adjusted to that corresponding to the saturated vapor enthalpy of the break donor volume or the calculated effluent, whichever is higher. Additional initial conditions and details are provided in Reference 67.

A reactor trip signal via the Steam and Feedwater Rupture Control System (SFRCS) on low pressure and an Anticipatory Reactor Trip System (ARTS) reactor trip (on turbine trip) is normally generated. However, for the MSLB transient, these trip functions were not credited. The depressurization of the steam generators and increase in the primary-to-secondary heat removal results in a high flux reactor trip. The depressurization of both steam generators continues until the SFRCS low steam line pressure trip is reached in the steam line of the affected steam generator. The SFRCS low steam line pressure trip generates a signal to close the Main Steam Isolation Valves (MSIV) and isolate the steam-side of the unaffected steam generator. The SFRCS low steam line pressure trip also results in a signal to start Auxiliary

Feedwater (AFW) to both steam generators. Normally, AFW would be isolated to the affected steam generator due to the sensing of low steam line pressure following MSIV closure. However, a single failure of the AFW isolation valve allows the continuation of AFW supply to the affected steam generator. The remaining inventory in the affected steam generator and the AFW supplied by the AFW pumps continue to be released to the containment until operator action is taken at 10 minutes to isolate AFW to the affected steam generator.

The worst-case single failure for the analysis of the mass and energy release data was determined to be an AFW isolation valve failure. The other single failures considered were: (1) failure of one train of HPI, (2) failure of one Main Steam Isolation valve to close and, (3) failure of the Main Feedwater Stop valve to close upon receiving an SFRCS signal.

Since pressurization without venting would exceed the design differential pressure, blowoff roof vents are provided over all four compartments (601, 601 A, 602, and 602A). The analyzed areas for the blowoff roof vents are provided in Reference 67. As a result of rupturing the vents, the peak pressures in the affected rooms quickly reach atmospheric pressure. For example, for a MSLB in Room 602, the room is able to reach atmospheric pressure in less than 10 seconds.

In addition, pressurized doors were provided between each steam line room and the Mechanical Equipment Room (603). A ventilation duct in Room 602 which is connected to the Auxiliary Building lower levels was completely enclosed in concrete. Ventilation ducts which passed from the Mechanical Equipment Room to Room 602 were removed and the wall sealed.

Environmental Effects:

The two 36" Main Steam lines in the Auxiliary Building originate at the main steam containment penetration on elevations 643' and enter the turbine building directly on that elevation. The line from steam generator (SG) 1 traverse rooms 601 and 601 A and the SG2 line traverses rooms 602 and 602A. MS line breaks are postulated in accordance with the Section 3.6 criteria in each room that the lines traverse. Breaks in 601 A and 602A need not be analyzed since the effects of breaks in these rooms are bounded by the effects of breaks in rooms 601 and 602. A break in room 601 is of no concern to room 601 from an EQ standpoint because the room contains no equipment essential to mitigation of that break. However, room 602 contains equipment essential to mitigation of room 601 breaks, so room 601 breaks need be analyzed for their effects on room 602. Similarly, room 602 breaks need to be analyzed for their effects on room 601 (Reference 55).

Since a steam line rupture upstream of the main steam line isolation valve negates the operation of any safety-related equipment associated with this line, the effects of steam flooding or temperature are incidental. The mass of water accumulated on the compartment floor due to the complete blowdown of the steam generator does not exceed floor loading limits. The floor drainage system eventually passes all water into the storm water systems. Wall thickness has been increased in several areas to prevent jet impingement damage. The entrance door from the Turbine Building into the control room area is not subject to direct jet forces due to a break in the Turbine Building.

Shutdown:

See Subsection 15.4.4 for shutdown procedure. Instrumentation on the main steam line would detect the rupture. Low pressure signals would close the main steam isolation valves from both steam generators, trip the turbine generator, close the main feedwater and startup control valves and feedwater stop valves to both steam generators.

3.6.2.7.1.5 Main Steam to the Auxiliary Feed Pump Turbines

Description:

For a description of the AFW system, see Section 9.2.7. The two 6" main steam supply lines to the AFW pump turbines (AFWPT) originate in the main steam penetration rooms on elevation 643' and terminate in the AFW pump rooms on elevation 565'. During normal station operation, main steam to the AFWPTs is isolated at the steam admission valves to the AFWPTs. The pressurized piping is normally at the main steam system pressure, and between main steam temperature and saturation temperature. These lines together traverse a total of 15 rooms in the auxiliary building. These rooms are 124, 235, 236, 237, 238, 304, 314, 400, 401, 404, 427, 500, 501, 601, and 602. During normal station operation, MS-106A and MS-107A remain open so the lines are considered to be high energy downstream of these valves and are thus subject to postulation of breaks in accordance with Section 3.6.2.1. Flow restrictors consisting of back to back 6" X 2-1/2" flow reducers have been installed in the AFWPT supply lines in rooms 600 and 601. Non-isolable breaks are postulated both upstream and downstream of the flow restrictors.

Breaks of 2 1/2" and 6" diameter pipes are below the sensitivity level of the SFRCS. Therefore the plant would continue to operate until the plant operators take manual action. Low pressure switches (PSL-5894A, PSL-5894B and PSL-5895A, PSL-5895B) are located on the steam supply lines to the Auxiliary Feedwater Pump Turbines (AFPT) in rooms 500 and 501, downstream of the 6" x 2 1/2" flow reducers and upstream of the main steam to AFPT isolation valves (MS-106, MS-107 and MS-106A, MS-107A). These switches will detect breaks down to the inlet of the steam admission valves (MS-5889A and MS-5889B) and actuate an annuciator in the control room. Additional low pressure switches (PSL-106A through D and PSL-107A through D) are located in rooms 237 and 238. The set points of these pressure switches are such that they would detect breaks between the check valves (MS-726, MS-727, MS-734, MS-735) and the sensor location upstream of the steam admission valves (MS-5889A, MS-5889B). However, these switches are not qualified for the harsh environment that would result from a break in the room in which they are located. Therefore, these switches are only assured to actuate for breaks downstream of the check valves (MS-726, MS-727, MS-734, MS-735) and outside the AFPT room. If these switches actuate, they initiate closure of the steam supply valves (MS-106A, MS-107A) and initiate a control room alarm. If these pressure switches fail to isolate the break, the control room will be alerted by the actuation of the other set of pressure switches (PSL-5894A, PSL-5894B and PSL-5895A, PSL-5895B) described above.

Since the break may not be isolated automatically, the analysis conservatively assumes that the blowdown continues for 10 minutes at full power during which time the operators would diagnose the problem and then initiate manual action. Mass and energy releases for this time are assumed to be constant. The flow rate from the break is choked flow. After 10 minutes, the analysis assumes that for non-isolable breaks the reactor is tripped, SFRCS is manually initiated and the transient is terminated by isolating the affected steam generator. Manually opening the AVV would divert flow from the break and lessen the severity of the blowdown to the rooms. The mass and energy releases for the blowdown through the 2 1/2" and 6" breaks is given in references 65, 66 and 67. The analysis provides blowdown for the break in the 2 1/2" line with and without the AVV available. The AVV on the affected steam generator is not available for breaks of the 6" line in rooms 601 and 602. The analyses assume a loss of offsite power coincident with the reactor trip. Single active failures are taken which provide the worst effects for the particular break.

Additional analyses has been performed in support of Technical Specification Amendment 192 for a line break in Mode 3. With an initial water level of 96% Operating Range, with the MDFP supplying the steam generators and with the SFRCS low pressure trip active, the mass of water released is more than assumed in this analysis. However, the energy content of the steam exiting the break is always lower at any given time in the transient because of Mode 3 conditions. As a result the environmental effects are bounded by the existing analysis. See Subsection 15.4.4.3.3 for additional information.

A 1 ½" minimum flow line has been added from the 6" steam supply lines in both Auxiliary feedwater pump rooms. These lines are joined in a common line which is ultimately routed to a feedwater heater in the turbine building. The minimum flow lines may be used to maintain a continuous steam flow through check valves MS734 and MS735. A break is postulated on either minimum flow line at the terminal end located in the associated auxiliary feedwater pump room or on the non-seismic portion located in the turbine building. The common line is normally automatically isolated following a turbine trip. However, breaks or cracks on this line are below the sensitivity level of SFRCS and will not automatically isolate. A break or crack within either auxiliary feedwater pump room will affect only the associated auxiliary feedwater pump, while breaks or cracks in the turbine building will not affect either auxiliary feedwater pump turbine. Breaks or cracks on the minimum flow lines in the auxiliary feedwater pump rooms or in the turbine building will not prevent access to unaffected auxiliary feedwater pump turbines. A break, crack, or malfunction on a minimum flow line could cause a slight reduction in steam supply pressure to an operating auxiliary feedwater pump turbine however, this will not affect the ability of the AFW system to cool the RCS to a point where the Decay Heat Removal System can be placed into operation.

Proximity to Required Safety Related Equipment:

Safety related equipment required to mitigate 2-1/2" AFWPT supply line breaks are presented on Figures 3.6-57 through 3.6-62. Cables servicing necessary equipment are qualified for the harsh environment through which they pass.

For the 6" AFWPT supply break, the associated safety-related equipment for rooms 601 and 602 are the AVV on the unaffected steam generator and the associated instrumentation and cables for their valve.

Postulated Design Basis Break Locations:

The piping stresses in the AFWPT supply line are below the pipe break criteria given in MEB 3-1 (reference 53) therefore breaks are only postulated to occur at terminal ends which are located in Rooms 237, 238, 401, 500, 501, 601, 602 and 326 (Heater Bay area, 585' level - adjacent to AFP room 238 ventilation/access opening). In addition, piping stresses were evaluated to determine the location of postulated cracks. Based on MEB 3-1 criteria, critical cracks need to be postulated in rooms 237, 238, 400, 401, 404, 500, 501, 601, and 602.

Figures 3.6-12, 3.6-12a, 3.6-13 and 3.6-14 are included to indicate selected pipe rupture locations in accordance with Reference 40. The break locations are designated by the prefix AS.

Compartment Pressurization:

The PCFLUD model used for the 2-1/2” AFWPT breaks and cracks in the auxiliary building is taken from Figure 3.6-56. The detailed assumptions for the specific break locations are given in the paragraphs below.

Compartment pressurization from postulated critical cracks is mild compared to full breaks. The energy releases associated with AFWPT supply line cracks are sufficiently small that little or no pressure overshoot will occur prior to rupture of a blowout panel or a door. The peak pressures for the line breaks or cracks are presented in Table 3.6-6.

AFWPT supply line breaks or cracks in rooms 237 and 238 (AFW pump rooms)

The AFWPT supply line break or crack in the AFW pump rooms are upstream of the steam admission valves. For these breaks the worst case single failure would be failure of the motor operated isolation valves (MS-106A, MS-107A). The pressurization transient would be limited to the affected AFW pump room and the turbine building. The adjacent AFW train is unaffected by the break since a wall and interconnecting pressure door separate the two AFW pump rooms. The AFW pump rooms directly vent into the turbine building. The only vent area modeled from the turbine building to the environment is a roll-up door. A ventilation path is available from the turbine building to the adjacent AFW pump room. The pressurization transient is terminated when the roll-up door ruptures and vents to the environment. Rooms 123 - 126 and room 235 are also connected to the AFW pump rooms through floor drains. These rooms are minimally affected by a break and are modeled as one volume.

Jet impingement forces from a break in the AFW pump rooms do not affect essential equipment or structures.

AFWPT supply line breaks or cracks in rooms 400,401 or 404 (equipment hatch area)

For the AFWPT supply line break or crack in 400,401, or 404, the worst case single failure would be failure of the motor-operated isolation valve (MS-106A). A break in room 401 communicates directly with rooms 300, 400 and 404, the fuel handling and equipment hatch area. Flow into room 405 is through a ventilation transfer grill from room 404. Room 300 relieves to the atmosphere through a roll-up door. The pressurization transient is terminated when the roll-up door in room 300 ruptures and vents to the atmosphere.

Jet impingement forces from the break do not affect essential equipment or structures.

AFWPT supply line breaks or cracks in rooms 500 and 501 (ventilation equipment rooms)

For a AFWPT supply line break or crack in rooms 500 or 501, the break upstream of the isolation valve is non-isolable. The worst case single failure assumed for non-isolable breaks is failure of the associated Atmospheric Vent Valve (AVV). The manual AVV actuator reach rods in rooms 500 and 501 would be inaccessible.

For a AFWPT supply line break or crack downstream of the isolation valves in rooms 500 and 501, the worst case single failure would be failure of the motor operated isolation valves, either directly or due to failure of an emergency diesel generator. In this case, the AVVs would be available to assist in steam generator depressurization.

Rooms 500, 501, and 515 are a series of large interconnected volumes. Following a postulated AFWPT supply line break in either room 500 or 501, door 517 in room 501 will open to the turbine building heater bay. This limits the maximum pressure in rooms 500, 501 and 515 to approximately the door blowout setpoint. Steam will also enter the Radwaste Ventilation System ductwork through duct openings in rooms 500, 501, and 515. The steam flow is limited by the very small differential pressure which will exist after the blowout door opens. In addition, fire dampers to other areas of the auxiliary building will isolate. Thus, only rooms 500, 501 and 515 will be significantly impacted by the break.

The two redundant steam lines enter these interconnected rooms. Adequate physical separation is maintained between redundant cross-connect lines. A pressure door between room 501 and the control room will easily withstand the peak room pressure. Jet impingement forces from the break do not affect essential equipment or structures.

AFWPT supply line breaks or cracks in rooms 601 and 602

Breaks in rooms 601 or 602 are the only unrestricted 6” AFWPT supply line breaks. They are upstream of the flow reducer and are non-isolable. Rooms 600 and 601 are not separated. Rooms 601, 602, and 602A are interconnected volumes separated only by partial height walls. The blowout panels for these rooms are located on the roof of the auxiliary building. The pressurization transient is terminated when the blowout panels in the affected room are actuated and release steam to the environment. Only the rooms adjacent to the break (600/601, 601A, 602, and 602A) are affected by the break.

Environmental Effects:

For the 2-1/2” and 6” breaks, compartment temperature responses were calculated using the same models as described previously for compartment pressures. The peak temperatures are given in Table 3.6-11. For the 2-1/2” break, the equipment required to operate to mitigate the consequences of the breaks are shown on figures 3.6-57 through 3.6-62. For the 6” AFWPT break, the associated safety-related equipment for rooms 601 and 602 are the AVVs. All the equipment and cables which are required to mitigate the consequences of the transients have been environmentally qualified for the resultant environments. Temperature effects from critical cracks have been evaluated. The environmental effects of the cracks are bounded by the more severe break environments.

Rooms 237 and 238 (AFW Pump Rooms)

The environmental effects of the steam flooding are most severe in the AFW pump room in which the break occurs. The steam vents directly to the turbine building. Steam from a line break in either room 237 or 238 would enter the turbine building at elevation 585’. Since the AFW pump rooms are ventilated from the turbine building, the unaffected AFW pump room temperature will approach the turbine building temperature. However, this temperature will not adversely affect the operation of the opposite AFW train. The ventilation system in the AFW room associated with the crack or break is assumed to fail.

Water flooding would be insignificant and would drain into the detergent waste tank flood room or into the clean waste tank through the 14-inch flood drain in the AFW rooms.

Room 401

Steam flooding from room 401 directly enters rooms 400,404 and 300. In the model, the only initial flow path out of the fuel handling area is a small flowpath to room 405 from corridor 404. Room 405 then is vented to the atmosphere to simulate operation of the Radwaste Ventilation System. This increases the temperature of room 405 until the flow path is isolated by closure of fire dampers.

The mass of water accumulated in the rooms is minor and can be easily handled by the floor drainage system.

Rooms 500 and 501

Door 517 in room 501, which opens to the turbine building heater bay, limits steam flooding from rooms 500, 501, and 515 to other auxiliary building areas. Manually opening the AVV on the affected side would divert steam from the break and lessen the severity of the blowdown to the rooms, if available.

A pressure door prevents steam flooding of the control room. The mass of water accumulated is minor and can be easily handled by the floor drainage system.

Rooms 601 and 602

Steam flooding from either the break in 601 or 602 is essentially limited to these two rooms only. The equipment in the room with the break is assumed to fail. Therefore, the AVV for the affected steam generator is not available to mitigate the blowdown for breaks of the 6” line. The reach rods for the AVV which are located in rooms 500 and 501 would be available following the 6” break but are not taken credit for in the analysis. The essential equipment in the room adjacent to the critical crack or break is qualified. The AVV of the adjacent train is qualified for the elevated temperature resulting from the crack or break.

Shutdown:

Once the break is detected, and is determined to be non-isolable, manual action would trip the reactor (a loss of offsite power is postulated to occur coincident with the reactor trip), initiate SFRCS and isolate the affected steam generator. Automatic initiation of low pressure SFRCS in the affected steam generator would occur following the manual actions to complete the lineup. Manually opening the AVV would divert flow from the break and lessen the severity of the blowdown to the rooms. The reactor is maintained in a hot standby condition with feedwater being supplied by at least one auxiliary feedwater pump to the unaffected steam generator. Cooldown will eventually proceed using the unaffected steam generator.

3.6.2.7.1.6 Main Feedwater System

Description:

The two 18” main feedwater lines in the auxiliary building enter from the Turbine Building on elevation 585’ and end at the containment penetrations on that elevation. The line to SG1 enters the auxiliary building in room 313, crosses into room 304 and immediately turns upward to elevation 603’ into room 404, traverses room 404, then turns downward into room 303 on elevation 585’ where the containment penetration is located. The line to SG2 is located wholly in room 314.

The normal operating conditions of the main feedwater system are approximately 1000 psia and 460 degrees F as the water enters the auxiliary building. For a description of the feedwater system, see subsection 10.4.7

The steam generator side of a MFW break may be non-isolable. In the original analysis, the feedwater pump side of the break was fed until the main feedwater stop valve closes. (The stop valve has a slower closing time than the main feedwater control valve, which was assumed to be disabled by a single failure.) This break is quickly detected by the SFRCS, which sends closure signals to the main feedwater stop, block, and control valves. Because this transient results in an automatic reactor trip, a concurrent loss of offsite power is assumed.

In the present analysis, a single failure of an EDG has been considered. This failure could prevent proper automatic valve realignment upon receipt of the SFRCS low pressure trip. For a break on steam generator 2 (with a failure of EDG 2), this would result in continued feeding of the affected steam generator by auxiliary feedwater. Therefore, continued AFW flow to the affected steam generator was assumed to continue until terminated by operator action at 10 minutes after the break initiation. The continued feeding was added to previous blowdown by maintaining steam flow out of the break at the maximum AFW flow rate for the 10 minute transient duration. Steam blowdown enthalpy for the continued feed portion was maintained at the enthalpy of the last data without continued feed. This represented a depressurized steam generator but does not take credit for any subsequent RCS cooldown which would be caused by the continued feed flow.

Mass and energy releases due to postulated breaks were calculated in reference 68 using the RELAP5/MOD2 computer program.

Proximity to Required Safety Related Equipment:

Safety related equipment required to mitigate the consequences of MFW line breaks in the auxiliary building are as presented on figures 3.6-63 and 3.6-64. Cables which are required to mitigate the consequences of these events are qualified for the resultant environments within the affected areas.

Postulated Design Basis Break Locations:

MFW line breaks are postulated in accordance with the section 3.6 criteria in rooms 303 and 314. The compartment response from any critical cracks in rooms 303 or 314 will be bounded by the response due to a MFW line break. Critical cracks only are postulated at the most adverse location on the MFW piping in rooms 304, 313, and 404 (reference 53 and 60). However, resulting pressure effects due to crack(s) in rooms 304 and 310/313 are bounded by MFW breaks in room 314. Pressure effects due to MFW cracks in room 404 are bounded by MS to AFW breaks in room 401.

The only postulated MFW breaks inside the auxiliary building which are utilized for analysis purposes are located between the last MFW isolation valve and the shield building. The resultant compartment temperatures and pressures from these locations are the most severe because the break is non-isolable. Figure 3.6-15 is included to indicate selected pipe rupture locations in accordance with Reference 40. The break locations are designated by the prefix MF.

Compartment Pressurization:

Postulated main feedwater line breaks involve a large number of compartments with steam/air flows, starting with the initial blowdown during the first few seconds. Doors and blowout panels of most immediately adjoining compartments will open. Significant flow paths were selected from the auxiliary building model as presented in figure 3.6-56. Doors and blowout panels will open according to their set pressure. This information is contained in reference 60.

Compartment pressurization from postulated critical cracks is mild compared to full breaks. The energy releases associated with main feedwater line cracks are sufficiently small that little or no pressure overshoot will occur as the first flow path away from the crack compartment is established by the rupture of blowout panels or doors. Thus, the compartment models for cracks can be substantially simplified.

Room 303 and room 314 each contain six large blowout panels. In room 314, all panels relieve to the turbine building heater bay area. In room 303, all panels relieve to the environment. The setpoints are sufficiently high to prevent undesired lifting during the initial heatup following a Large Break LOCA accident (as described in chapter 6.2). In addition, the setpoint is sufficiently low to protect the structural integrity of the auxiliary building and containment vessel. The panels in these rooms provide the major venting path for all MFW line breaks.

Peak compartment pressures for breaks in room 303 and 314 are presented in table 3.6-6, and occur within 10 seconds. Maximum compartment pressure is well above the setpoint of the blowout panels. The compartment pressure transient for both breaks is essentially complete within one minute, when the initial steam generator inventory has been largely depleted.

Breaks in Room 303 (Mechanical Penetration Room #3):

Steam will enter rooms 105, 113, and 115 via a pipe chase connecting rooms 303, 208, and 105. Large ducting which is part of the radwaste (RW) ventilation system is routed within room 208 in and near the ventilation duct chase. For MFW line breaks in room 303, room 208 pressure follows closely behind room 303 pressure. In general, flowpaths that interconnect rooms by the RW ventilation system are not considered because of large resistance and inertia associated with the ductwork and actuation of the fire dampers (where applicable).

Breaks in Room 314 (Mechanical Penetration Room #4):

The compartment model for a MFWLB in room 314 is substantially similar to the model presented for a MFWLB in room 303. Steam will enter rooms 105, 113, and 115 through a pipe chase connecting rooms 314, 236, and 115.

The CTMT purge exhaust is routed from room 314, through room 427, to room 515. (However, this path is ignored due to large ductwork, register resistances, and inertia. The response in room 515 is bounded by the Main Steam to Auxiliary Feedwater Pump turbine line break in room 501.)

Cracks in Room 313 (Boric Acid Mixing Area)

The compartment model for room 313 includes the adjoining corridor (room 310) and also includes a flow path into adjacent room 312 (Spent Fuel Pool Pump room) through an open ventilation grill in the North wall of room 312. Pressure build-up within these rooms is limited by the failure of a door and a resulting flow path to atmosphere. The flow resulting from a 3.78 in^2

critical crack in the 18-inch MFW piping is below the sensitivity of the SFRCS system, and so the reactor is considered to continue to operate with no action taken for 30 minutes. Long before this time, room 313 will reach the failure pressure for the release path to atmosphere.

Breaks in the Turbine Building:

Breaks within the turbine building are described in Section 3.6.2.7.1.16. To prevent jets from potential breaks in the heater bay area from adversely impacting essential equipment in room 328, the original openings into room 328 have been sealed closed.

Environmental Effects:

Compartment temperature responses were calculated using the same or substantially similar models as described previously for compartment pressures. Equipment and cables which are required to mitigate the consequences of these breaks have been environmentally qualified for the resultant environments. Peak compartment temperatures are presented in table 3.6-11. Additional considerations for resultant water flooding and specific compartment details are as follows:

Room 303 (Mechanical Penetration room #3):

Details of this break are provided on figure 3.6-63. Immediate operation of the equipment in rooms 105 or 208, which are connected to room 303 via an open pipe chases, is not required after this break. Adequate time is available for manual lineup of any affected valves in the Decay heat Removal System.

The six inch Auxiliary Feedwater line to Steam Generator 1-1 has been shielded from jet impingement forces. Both full pipe area and critical crack area jets were considered. A protective shield has been provided at a tray carrying essential cables located below the main feedwater line for protection against a critical crack area jet. Thus no essential structures are affected.

Water would enter the pipe chases to room numbers 208 and 105. Therefore, a curb has been provided around the pipe chases in room number 303. Water will also enter room 304. Water flooding which enters room 304 will proceed through the double doors into the room 300 area or may flow into the room 310/313 area. Water levels are not of concern in these areas.

Room 314 (Mechanical Penetration room #4):

Details of this break are provided on figure 3.6-64. Immediate operation of the equipment in rooms 115, 113 and 236, which are connected to room 314 via an open pipe chase, is not required after this break. Adequate time is available for manual lineup of any affected valves in the Decay heat Removal System.

The six inch Auxiliary Feedwater line to Steam Generator 1-2 has been shielded from jet impingement forces. Both full pipe area and critical cracks were considered. No essential structures are affected.

Water would enter the pipe chases to room numbers 236 and 115. Therefore, a curb has been provided around the pipe chases in room number 314. Water will also enter room 304. Water flooding which enters room 304 will proceed through the double doors into the room 300 area or may flow into the room 310/313 area. Water levels are not of concern in these areas.

Room 313 (Boric Acid Mixing Area):

Temperature effects due to a critical crack in room 313, and adjoining rooms 310 and 312, are limited to a peak temperature of 212°F due to the quick return of these areas to atmospheric pressure. While flooding effects from this crack are far less significant, as previously noted for the above break evaluations, water levels are not a concern in these areas.

Breaks in the Turbine Building

Breaks within the turbine building are described in USAR Section 3.6.2.7.1.16. Curbs are provided around the ventilation openings to rooms 237 and 238 to prevent potential water flooding in the turbine building from entering the rooms.

Shutdown:

Following a main feedwater line break, differential pressure (high) instrumentation on the main feedwater line initiates SFRCS closure of the main feedwater control valve, startup control valve, and main feedwater stop valve on both lines. In addition, the auxiliary feedwater pumps are started, feeding their own steam generators, and alarms are actuated in the control room. The main turbine and reactor trip, and both main steam isolation valves will close. A loss of offsite power is assumed to occur at this time. Within a few seconds following the break, the SFRCS will receive a low pressure trip from the affected steam generator, and will realign both auxiliary feedwater pumps to the unaffected steam generator. This realignment may require operator action to isolate the affected steam generator if a single failure consisting of a valve failure or an emergency diesel generator occurs. The reactor is maintained in hot standby conditions, with feedwater being supplied by at least one auxiliary feedwater pump, to the unaffected steam generator. Cooldown will eventually proceed using the unaffected steam generator. Following dryout of the affected steam generator, auxiliary building temperatures will drop rapidly. This will ensure that a makeup source from the normal makeup system will be available as required.

3.6.2.7.1.7 Reactor Coolant Letdown System

During normal station operation, reactor coolant is removed through the letdown system for purification. Normal operating conditions outside the containment and prior to pressure breakdown is 2200 psia and 120°F. For a description of this system, see Subsection 9.3.4. The physical arrangement of this system in the Auxiliary Building is shown in Figure 3.6-4.

Proximity to Required Safety-Related Equipment:

The high energy portion of this system is considered that portion between the containment vessel penetration and pressure breakdown orifice in room number 208.

Postulated Design Basis Break Locations:

Figure 3.6-16 is included to indicate piping for which pipe ruptures were analyzed in accordance with Reference 40. Critical cracks only are postulated at the most adverse location.

Compartment Pressurization:

This room vents into the containment annulus, room 105 and room 303. A critical crack area of 0.2 in² was calculated for the 2-1/2 inch letdown line. Based on orifice formulae, a flow coefficient of 1, and a constant pressure of 2200 psia, a break flow of 46.6 lb/sec was calculated. Temperature switches sensing a cooler discharge temperature of about 160°F initiate closure of valves to isolate the break.

Based on a conservative analysis which assumed no heat transfer and isentropic blowdown for a duration of 40 seconds, a maximum compartment pressure of 14.44 psia is obtained. Compartment pressurization for room 208 due to a Main Feedwater line break in room 303 is a more severe accident. See Table 3.6-6.

Environmental Effects:

A maximum room temperature of 160°F occurs due to this break. However, the room temperature for room 208 due to a Main Feedwater line break in room 303 is bounding as shown in Table 3.6-11. Steam flooding and jet impingements do not affect any essential equipment in this room or interconnected rooms. Flooding of about 400 gallons (including about 150 gallons between the isolation valve and break) may occur. This volume can be easily handled by the floor drainage system which transmits this water to an auxiliary building sump (duplex pump) on El. 545.

Shutdown:

Two high temperature switches are provided to close isolation valves inside the Containment Vessel and actuate an alarm in the control room. Each switch actuates closure of a redundant isolation valve in a letdown cooler loop. Therefore, the results of this break do not prevent a safe shutdown.

3.6.2.7.1.8 Auxiliary Feedwater System

Description:

During normal station operation, only that portion of the auxiliary feedwater line from the steam generators to the check valves at the containment isolation valves are pressurized at 935 psia. However, under hot standby conditions the system could be used in an emergency. When the auxiliary feedwater system is in use, the operating conditions upstream of the check valves will be 1100 psig and 90°F.

The Motor Driven Feedwater Pump (MDFP) design provides the capability to discharge to the auxiliary feedwater discharge piping. An evaluation was done to allow aligning the MDFP suction to the main feedwater system while the discharge is aligned to the auxiliary feedwater system during Mode 4. Limitations were established to ensure water temperature is limited to less than 200°F.

For a description of this system see Subsections 9.2.6 and 9.2.7. The physical arrangement of the system is shown in Figures 3.6-3 and 3.6-4.

Proximity to Required Safety-Related Equipment:

The high energy portion of the auxiliary feedwater system outside containment is located in rooms 303, 304, 314, 236, 235, 124, 238, 237, and the containment annulus.

Postulated Design Basis Break Locations:

Figure 3.6-17 is included to indicate selected pipe rupture locations inside containment only. The break locations are designated by the prefix AF. Critical cracks at the most adverse locations are postulated in accordance with Table 3.6-1 and Reference 40, outside containment as described in the following paragraphs. On normally depressurized piping, these cracks are only postulated during long-term (after 24 hours) usage of this system, subsequent to actions necessary for safe shutdown.

Environmental Effects:

Rooms 303, 304, and 314:

The effects of water or steam flooding are more severe for the main feedwater break.

Room Numbers 237 and 238:

Based on orifice formulae, a critical crack area of 0.62 in^2 in the 6-inch pump discharge line, a flow coefficient of 1, and a constant pump discharge pressure of 1100 psia, and a break flow of 780 gpm were calculated. A 14-inch drain pipe into the clean waste receiver tank room number 124 is provided in each room. With the break flow indicated, the flood level would not exceed about 2.5 feet. The normal room floor drains would further reduce this level. The wall and pressure door separating these rooms would prevent flooding in one room from entering the adjacent room. Jet impingements would only affect equipment associated with the pipe crack.

Room Number 124:

The 6-inch discharge line from auxiliary feed pump 1-1 passes through this room. This is a flood room with a capacity of about 50,000 gallons (excluding the tank volume). Room number 123 is also connected to this room, giving a total flood volume of 100,000 gallons. The operator has adequate time to isolate the crack (see discussion on shutdown).

This line is in close proximity to a tray carrying essential cables feeding auxiliary feed pump/turbine unit 1-2. Therefore, a shield has been provided to protect this tray from any jet effects. No other equipment or structures are affected by the jet force.

Containment Annulus:

The annulus response due to the critical crack is bounded by that from a Steam Generator Blowdown Line break in the annulus, as discussed in Subsection 3.6.2.7.1.15. The station drainage system is adequate to accommodate the total accumulation of AFW and condensed steam released from the critical crack.

Room Number 235:

The 6-inch discharge line from auxiliary feed pump 1-1 passes through this room. A 4’ x 4’ opening in the wall between rooms 235 and 124 has been provided. With door 212 (between

rooms 235 and 227) closed, the maximum flood level in Room 235 due to this break is approximately 3.5 feet. Door 212 is capable of withstanding the hydrostatic pressures resulting from this flood. Whether open or closed, water leakage that might occur through this door can be easily removed by both the floor drains in room 227 and by a grating covered cutout in the room 227 floor, which will allow drainage to flow into areas 110 and 112 on the 545' elevation. No essential equipment needed for this event will be affected by water in these areas. Jet impingement forces do not affect essential structures.

Room Number 236:

A 6-inch discharge line from each auxiliary feed pump passes through this room. A crack flow of 780 gpm from either line would flow through the floor drainage system to an auxiliary building sump on El. 545 and also through the pipe chase into room number 115 on El. 545. The essential MCC in room number 236 is not affected due to rapid drainage of water into room number 115. No essential equipment or structures are affected by jet impingements.

The submersible duplex sump pumps in room number 115 have a total capacity of greater than 100 gpm, which is inadequate for this break flow. The sump pump running lights and a high sump alarm are activated in the control room. Flooding is contained by a 10-foot flood wall; however, the services of the H.P. injection pump, decay heat removal pump, and containment spray pump are lost. Under the operating conditions of this accident, this equipment is not needed. In addition, the redundant pumps are not affected. After isolation of the break, this room is pumped out and the required maintenance is performed to bring these pumps back into service.

Shutdown:

A critical crack could occur during hot shutdown with the AFW system operating. During this time, steam generator level is maintained automatically or manually. Steam generator secondary side feedwater flow is in the range of about 150 gpm, decreasing with time.

Under automatic control, a pipe rupture would cause the auxiliary feedwater pump flow control valve to open in order to maintain steam generator level. In addition, control room alarms would occur for flooding in the following rooms:

a. Room numbers 237 and 238 — flood level switches.

b. Room number 124 — floor sump high level.

c. Room number 235 — floor sump high level in room number 124.

d. Room number 236 — sump pump running lights and floor sump high level in room number 115.

The operator has adequate time to investigate the incident, stop the pump, and continue hot shutdown/cooldown on the redundant auxiliary feedpump turbine unit. Under normal station operation, Auxiliary Feedwater lines (downstream of the check valves) in rooms 303, 314, and the annulus are at Steam Generator pressure and ambient temperature. In the event of a critical crack in the Auxiliary Feedwater line between the Steam Generator and the Auxiliary Feedwater isolation valves, a high temperature alarm due to blowdown out the crack from the Steam Generator will be received in the Control Room. After verification that a steam leak exists, the reactor will be tripped, and Auxiliary Feedwater initiated via manual initiation of

SFRCS. After confirmation of which is the affected side, the operators will take actions to ensure that all feedwater flow has been terminated and the MSIV closed on the affected steam generator, and will then open the Atmospheric Vent Valve (AVV) to vent steam to the atmosphere. The resultant steam generator blowdown will reduce, and eventually terminate, critical crack leakage into the annulus.

3.6.2.7.1.9 Auxiliary Steam and Condensate System

Description:

The auxiliary steam system originates outside the auxiliary building and consists of headers with various nominal pressures ranging from 235 psig to 5 psig. The Auxiliary Steam System may be supplied with steam from several sources, including Main Steam, Extraction Steam, and the Auxiliary Boiler. Functional drawings of the system are provided in Figures 10.1-2 and 10.1-2A. Breaks or cracks are postulated in the auxiliary building and in essential areas of the intake structure. Analysis of compartment response to high energy line breaks and cracks conservatively assumes that the Main Steam System is supplying steam to the Auxiliary Steam System without benefit of desuperheating. Therefore, the enthalpy of steam exiting from all breaks is assumed to be equal to or less than that of main steam at normal operating conditions. Auxiliary steam is supplied as required to auxiliary building loads consisting of boric acid evaporators, and the borated water storage tank heat exchanger. Auxiliary steam piping to other auxiliary building loads is normally isolated inside the turbine building.

Proximity to Required Safety Related Equipment:

Required safety related equipment to mitigate certain postulated Auxiliary Steam System breaks and cracks includes High Pressure Injection (for primary inventory control), Auxiliary Feedwater, Service Water, essential Component Cooling Water, and the Decay Heat Removal System (to maintain plant conditions for breaks occurring in lower modes). Power and control cables are qualified for the environments through which they pass. Rooms located in proximity to postulated breaks (or cracks) and which contain required safety related equipment are rooms 314, 304, 303, 236, 227, 209, 208, 115, 113, and 105. Required safety related equipment is also located in the Diesel Fire Pump room and the Service Water pump room. The latter two rooms are affected by postulated breaks or cracks in the intake structure.

Postulated Design Basis Break Locations:

All piping in the mechanical penetration rooms (numbers 314,303,208, 105, 113, and 115) has been upgraded by obtaining material certification and 100 percent NDT of all welds, where applicable, and installing seismic supports. Other piping in the vicinity of essential equipment has also been upgraded. For this piping, critical cracks are postulated at the most adverse locations in accordance with reference 40. Cracks are explicitly analyzed in rooms 208,303,314, and the Diesel Fire Pump room. Full area pipe breaks are considered to occur anywhere in Seismic Class II piping. In rooms where both breaks and critical cracks are postulated, breaks provide the largest blowdown energy. Breaks are explicitly analyzed in room 209,304,234,235, and the Service Water Pump room. In other areas, break or crack flow is sufficiently small that room temperatures or pressures are readily bounded by cracks or breaks in other systems.

Compartment Pressurization:

The PCFLUD models used for auxiliary steam system breaks and cracks in the auxiliary building are adapted from Figure 3.6-56. Descriptions for specific auxiliary steam system break and crack locations are provided under the topic, Environmental Effects. Because all auxiliary steam system breaks and cracks involve relatively low mass flow rates, pressure response in all essential areas is either insignificant or is bounded by the pressurization produced by breaks in other systems. Compartment pressures are also mitigated by equalizing flows in the ventilation ductwork. Therefore, ventilation ducts (including selected fire dampers), have been modeled where they are considered to be significant.

Environmental Effects:

Compartment temperature responses were calculated using the same or substantially similar models as described previously for compartment pressures. Equipment and cables which are required to mitigate the consequences of these breaks have been environmentally qualified for the resultant environments. Peak Compartment temperatures are presented in table 3.6-11. Although numerous indirect indications of the break/crack might be available in the control room, there would be no direct indication. Therefore, break flow is arbitrarily analyzed to continue until manually isolated at 30 minutes following the break. Additional considerations for resultant water flooding and specific compartment details are as follows.

Auxiliary Steam System Cracks in Rooms 208, 303, 314 (Inside EVS Negative pressure area):

Only critical cracks are postulated in rooms 208, 303, and 314. The basic model which was used for these cracks encompasses most rooms in the emergency ventilation system negative pressure area. In room 314, the largest postulated critical crack is a 0.94 sq. in. crack on the 10 inch 50 psig header. In rooms 208 and 303, the largest postulated critical cracks are 0.43 sq. in. cracks on the 6 inch 15 psig header. The mass blowdown rate from these cracks is very small compared to postulated breaks on other systems which would influence these areas. Therefore, no required equipment is adversely affected. Rooms 303 and 314 each contain large blowout panels, however, the blowout panels will not lift due to these cracks. Humidity could be conveyed to other interconnected compartments, but significant steam flow to other compartments will not occur. Jet impingement will not affect required equipment.

Auxiliary Steam Line Break in Boric Acid Evaporator rooms 234 and 235:

Upstream of the boric acid evaporators, the 6 inch piping is Seismic Class 1 (subject only to postulated critical cracks). A bounding full area break is postulated on non-seismic piping. However, break flow is limited by a 2 1/2 inch orifice located upstream of the evaporators. Closure of the boric acid evaporator steam inlet valves, desuperheating water supplied at the evaporators, and heat removal by the evaporators was conservatively not credited. Room 234 has a flow path to room 227 via a door, and room 235 communicates through a 16 sq. ft. opening to room 124. Therefore, a break in room 235 provides the bounding temperature for room 124, and a break in room 234 provides the bounding temperature for room 227. There is no required equipment in rooms 123, 124, 234, or 235.

The floor drainage system passes water to the sump in the clean waste receiver tank room number 124. A duplex sump pump transmits this water to the miscellaneous waste drain tank for disposal.

Auxiliary Steam System Breaks in Corridor numbers 209 and 304:

The limiting postulated auxiliary steam system break in room 209 is a 4.79 square inch break at the exit of the BWST heat exchanger, while for room 304 it is a critical crack on the auxiliary steam header. Heat removal by a cooler is not credited to reduce blowdown enthalpy.

Before the branch steam supply line to the Primary Water Storage Tank (PWST) heat exchanger was removed, a full area 2.95 square inch break was postulated on the non-seismic condensate drain piping from the PWST heat exchanger in room 304. This break, which bounds the postulated critical crack in room 304, is conservatively retained in the analysis. Capability to bring the plant to a stable, hot standby condition will not be impacted. If necessary, cooldown will proceed following isolation of the auxiliary steam critical crack. If the plant is in a condition where the decay heat removal system is already in service, operation of the decay heat removal system will not be affected.

The postulated full area break on the non-seismic condensate drain piping from the BWST heat exchanger provides the bounding temperature for room 209. Jet impingement from postulated cracks on the 6 inch BWST heat exchanger could damage a motor control center, however, capability to bring the plant to a stable, hot standby condition will not be impacted. If necessary, cooldown will proceed following isolation of the auxiliary steam break. If the plant is in a condition where the decay heat removal system is already in service, operation of the decay heat removal system will not be affected. Starters for the ECCS room cooler fans, which might be required, are located in an area which is unaffected by this postulated steam/condensate line break. No other essential starters associated with MCCs in room 209 are required to operate after this accident.

Room numbers 105, 113, and 115:

A 3-inch, 10 psig (nominal) condensate return line from the BWST heater passes through room numbers 105 and 113 to a drain tank. In addition, a 4-inch, 10 psig (nominal) condensate return line from the boric acid evaporators passes through room number 115 to the same drain tank. Critical crack areas of 0.17 sq. in. and 0.24 sq. in., respectively, are postulated. Steam traps located upstream of the postulated cracks will limit effluent flow and energy. Therefore, an explicit compartment response analysis of this event was not performed. Jet impingements do not affect essential equipment due to their small magnitude. These rooms are interconnected with all other mechanical penetration rooms and the containment annulus, in which hounding breaks are located. Continued plant operation or normal shutdown will not be directly impacted. The floor drainage system passes water to any one of three sumps. Duplex pumps at each sump transmit this water to the miscellaneous waste drain tank for disposal.

Pressurization in Other Auxiliary Building Areas (rooms 112, 116, 117, 122, 119, 123, 124, and adjoining areas):

Condensate return lines serve the BWST heat exchanger and boric acid evaporator packages. These lines normally contain saturated water, however, all piping in these rooms is non-seismic. Therefore, full area breaks are postulated. (The waste evaporator and degasifier are no longer normally used, therefore, the steam supply piping is normally isolated and breaks or cracks are not postulated.) Steam flooding and jet impingements from postulated breaks will not affect essential equipment or significantly pressurize the associated areas. Continued plant operation or shutdown/cooldown will not be directly impacted.

Two sumps with duplex pumps transmit water to the miscellaneous waste drain tank for disposal.

Intake Structure (Service Water Pump and Diesel Fire Pump Rooms):

A bounding full area break is postulated on the non-seismic 1½ inch diameter, 5 psig (nominal) line associated with the unit heater inside the Service Water Pump room. A seismic Class I wall is in place between the diesel driven fire pump and the Service Water pump rooms. Automatic ventilation control switches to start the essentially powered ventilation system are located in the service water pump room. Therefore, ventilation system operation is credited only for the break inside the Service Water Pump Room. Pressurization is negligible due to the small break flow rates and large ventilation openings.

Since a long term continuous steam flow from a postulated break on the 1 ½ inch steam line in the Service Water Pump room is not desirable for Service Water Pump operation, an isolation valve is installed in this line outside the service water pump area. An ambient air temperature switch is provided to sense an abnormally high temperature and activate closure of the isolation valve. However, in the analysis, this interlock is not credited due to ample flow from one train of the redundant ventilation system. Water flooding from a postulated break on the condensate line would be limited by closure of the steam line isolation valve or would also be easily accommodated by an 8-inch drain line to a sump in the valve room. Jet impingement forces are of small magnitude and do not affect essential equipment.

The Diesel Fire Pump area has a 3-inch, 5 psig (nominal) steam line and 2-inch condensate return line passing through this area. All required equipment is located in the service water pump area except for an MCC which supplies one train of service water room ventilation. Failure of any of the other motor control center loads will not prevent safe shutdown of the reactor. Except for the ventilation system, no immediate operation of any of the associated motor control center loads is required. There is sufficient time for an operator to manually operate any necessary equipment after isolation of the crack.

Shutdown:

Breaks or Cracks within the Auxiliary Building:

Manual isolation from unaffected areas is available for all breaks. An interlock has been provided on the 12-inch, 50 psig header to automatically isolate the auxiliary building headers upon sensing low header pressure. The interlock valve is located in the turbine building. Postulated break flows may be below the sensitivity of the interlock. Therefore, this valve provides additional assurance, but is not credited in the analysis. Floor sumps, which may eventually flood due to sump pump failure, are provided with high level switches for control room alarm. A small area pipe rupture or critical crack will be detected by abnormally high steam consumption, fire alarms, abnormal evaporator package operation, noise, humidity, visual observation, etc. Temporary loss of the auxiliary steam supply to the auxiliary building does not directly upset normal station operation.

During mode 1, auxiliary steam line breaks are not sufficiently large to directly affect plant operation by imposition of additional steam load. Because all potentially affected non-essential equipment has not been qualified, a reactor trip could occur. A reactor trip with assumed loss of offsite power is also assumed for breaks on non-seismic piping where the postulated break occurs due to a seismic event. Ventilation systems with non-essential power are therefore not credited to operate. For break/cracks occurring during modes 1, 2, or 3, both steam generators

will continue to be available to reach and maintain hot standby, if necessary. The decay heat system will function if required to reduce system temperature further. During modes 2, 3, 4, 5, or 6, lower auxiliary steam enthalpy would reduce resultant compartment temperatures compared to mode 1 breaks or cracks. If in service at the time of the break, operation of the decay heat system will not be affected.

It is conservatively assumed that breaks or cracks will not be isolated for 30 minutes, regardless of operating mode. This assumed duration provides sufficient time for operator discovery, diagnosis, and mitigation. Following isolation of the auxiliary steam break or crack, compartment temperatures will quickly fall and allow access to affected areas.

Intake Structure:

Postulated cracks in the diesel fire pump room will not affect plant operation. For postulated breaks or cracks in the Service Water Pump Room, one essential HVAC train can maintain acceptable temperature. Breaks in other intake structure areas will not affect the ability of the plant to continue operation or shutdown, as required.

3.6.2.7.1.10 Reactor Coolant Makeup System

Description:

During normal station operation, purified water is added to the Reactor Coolant System by means of the makeup pumps. Normal process conditions are 2800 psia and 120°F. For a description of this system, see Subsection 9.3.4. A functional drawing of the Makeup and Purification System is shown in Figure 9.3-16. The physical arrangement is shown in Figure 3.6-4.

Proximity to Required Safety-Related Equipment:

The high energy portion of the system is located in Rooms 225 and 236.

Postulated Design Basis Break Locations:

All piping in this system has been upgraded to Seismic Class I. Critical cracks are postulated at the most adverse locations in accordance with Reference 40.

Environmental Effects:

Room Number 225:

A crack of the 21/2 -inch pump discharge line would be the largest line rupture. A critical crack area of 0.2 in² was calculated. A crack flow rate of 325 gpm was established by evaluating the flow characteristics of the orifice with respect to the operating characteristics of the makeup pumps. (Reference 102)

Water would drain from room 225 through two 16 ft² openings into rooms 105 and 113 through the floor drainage system. The duplex sump pumps located in rooms 105 and 113 have a total capacity of greater than 100 gpm which is not sufficient to prevent flooding. Assuming that all flow enters either room 105 or 113, approximately three minutes after the crack occurs the operator will have received two alarms, a high/low pump flow alarm depending on crack location and a high sump level alarm, as well as the sump pump running indication lights. Operator

action should be initiated after receipt of the second alarm. Assuming that 10 minutes is sufficient time for the operator to identify the problem and take action and only one sump pump is operating, the flood water height in rooms 105 and 113 thirteen minutes after the break (3 minutes to activate the alarms and 10 minutes for operator action) would be less than 5”. Therefore, no essential equipment would be affected by flooding. In addition, the small jet impingement forces would not affect any essential structures. (Reference 102)

Room Number 236:

A crack of the 2 1/2-inch pump discharge line in this room would follow the same scenario as indicated for room number 225, except as follows:

a. Drainage would be through a pipe chase into room 115 at El. 545.

b. The flood water height in room 115 thirteen minutes after the break would be less than 5”.

No essential equipment or structures would be affected by flooding or jet impingements.

Shutdown:

The following signals and/or actions would result due to the pipe rupture:

1. Control room alarm on high level from floor sump level

2. Control room alarm on high or low pump flow depending on pipe rupture location

3. Control room alarm on low makeup tank level and automatic alignment of the makeup pump suction to the BWST

4. Control room alarm on RCP low seal water flow

5. Control room indication lights of running sump pumps

The above mentioned signals alert the operator of a Makeup System failure and a flooding condition. Safe shutdown procedures would be initiated. Reactor Coolant pump operation could continue with Component Cooling Water providing seal cooling. Should Component Cooling Water not be available, the Reactor Coolant Pumps will be stopped. Plant cooldown will then be accomplished using natural circulation.

3.6.2.7.1.11 Low Pressure Injection System

Description:

This system is not in use during normal station operation. During the decay heat removal mode of cold shutdown, breaks are not postulated due to the short time duration that process conditions exceed the criteria of Reference 40. After a LOCA this system is used on a long-time basis in the recirculation mode with fluid temperatures greater than 200°F. However as stated in response to NRC question 15.2.11, pipe failures in ECCS piping are not postulated following a LOCA.

Proximity to Required Safety-Related Equipment:

The high energy portion of the system is located in rooms 105, 113 and 115.

Postulated Design Basis Break Locations:

As stated above, piping failures are not postulated in the ECCS piping outside containment. A pump seal failure 24 hours after a LOCA assumed to cause a leak rate of 120 gpm.

Environmental Effects:

Water flooding due to a seal failure in room numbers 105 or 115 does not affect safe shutdown.

All three ECCS rooms (numbers 105, 113, and 115) are interconnected above the 10-foot flood walls. ECCS room sump pump capability is discussed in 3.6.2.7.1.14.

Shutdown:

The sump pump running indication lights and the high level alarm from the ECCS room sump alert the operator of a flooding condition. Safe shutdown is continued on the redundant Decay Heat Removal System.

3.6.2.7.1.12 Reactor Coolant Pump Seal Supply Water System

Description:

Seal water supply to the reactor coolant pumps is supplied by the makeup system (see Subsection 3.6.2.7.1.10).

Proximity to Required Safety-Related Equipment:

The high energy portion of the system is located in rooms 208, 209, 225 and 303.

All piping in this system has been upgraded to Seismic Class I. Critical cracks are postulated at the most adverse locations in accordance with Reference 40.

Environmental Effects:

Room numbers 225, 208,209 and 303

This piping is 2 inches and smaller. The effects of flooding are bounded by the Makeup System piping failures in room 225 as discussed in 3.6.2.7.1.10. Jet impingement forces are of small magnitude and do not affect any essential equipment or structures. Due to the low temperature of this system, breaks or cracks in other plant systems will result in a more severe environment in these rooms (see Table 3.6-11).

Shutdown:

See Subsection 3.6.2.7.1.10.

3.6.2.7.1.13 Hot Water Heating System

Description:

This system supplies hot water to various unit heaters in the Auxiliary Building during the winter months. Normal operating conditions are 45 psia and 250°F. A system description is given in Subsection 9.4.2. A functional drawing is shown in Figure 9.4-7.

Proximity to Required Safety-Related Equipment:

All hot water heaters in the containment annulus and Rooms 318 and 319 have been replaced by electric heaters. This measure was taken to avoid the possibility of heaters or piping falling on containment penetrations or impinging in the Emergency Diesel Generator Sets in Rooms 318 and 319. This also eliminates piping and valving in the penetration rooms which could have caused steam and water flooding due to a break. The electric heaters have been carefully located to preclude their falling on any penetrations. The hot water heater in Room 603 was redesigned as a two loop low pressure system within the room.

Other portions of this system in the Auxiliary Building are in rooms 500, 501, 515, 601 and 602.

Postulated Design Basis Break Locations:

Full area pipe breaks are considered to occur anywhere in any orientation.

Environmental Effects:

Room numbers 601 and 602:

A postulated main steam line rupture is the worst accident. Therefore, no analysis was made.

Room numbers 500, 501 and 515:

Unit heaters in these interconnected rooms are supplied by a 6-inch hot water supply line. A line rupture would cause the primary heating water pump to runout to approximately 880 gpm and a loss-of-system pressure. Neglecting flashing at the pump suction and discharge lines and assuming no damage to the pump impeller, the pump continues to run at this flow until most of the system inventory is depleted. Based on the entire system inventory of about 3000 gallons, flow would terminate in about 3.4 minutes. A high flow switch on the pump discharge has been provided to prevent the system makeup water supply valve from opening. A control room alarm has also been provided. Jet impingements and steam flooding do not affect essential equipment. Water flooding would be minor and easily handled by the floor drainage system. Essential equipment in these rooms is not required after this accident.

Shutdown:

A postulated rupture in this system does not affect normal station operation.

3.6.2.7.1.14 Containment Spray System

Description:

This system is not in use during normal station operation. However, after a LOCA, this system is used on a long-time basis, first taking cold water from the Borated Water Storage Tank and then from the Emergency Sump recirculation line. The postulation of passive piping failures (i.e., pipe breaks or cracks) subsequent to a pipe rupture or LOCA are not assumed to occur as stated in the General Protection Criteria of Subsection 3.6.2.1. In the long-term operation (i.e., beyond initial twenty-four-hour period) of accident mitigating piping systems, a credible passive failure (such as a pump seal failure) is considered. As such, piping failures are not considered in the spray system piping. For a description of this system, see Subsections 6.2.1.3.2 and 6.2.2, and Figures 6.3-1 and 6.2-11.

Proximity to Required Safety-Related Equipment:

The high energy portion of the system is located in rooms 105, 113, 115, 208, 236, 303 and 314.

Postulated Design Basis Break Locations:

Pipe breaks are not postulated in the spray system piping.

Environmental Effects:

Piping failures are not considered in spray system piping outside containment. Twenty four hours after a LOCA, a pump seal failure is assumed causing a leak rate of 120 gpm.

Flooding between ECCS rooms from the station drainage system is prevented by 10-foot walls between the rooms. ECCS room leakage will be transferred by the ECCS room sump pumps to the miscellaneous waste drain tank.

The ECCS room sump pump is a duplex type. There are two sump pumps in each sump. Each is Seismic Class I and Q-listed. Pumps 1A, 2A, 3A, 1B, 2B, and 3B have a capacity of greater than 50 gpm (Reference 102). A float switch and electrical alternator are provided in each sump to automatically start, stop, and alternate each of the pumps and start the second pump when one in operation is unable to maintain the sump level. In addition to the above, a sump level switch with associated alarm and qualified pump running lights are provided for each of the ECCS room sumps on the main control board. The frequent or steady sump pump running light indication in the control room will alert the operator of the leak. The operator will remote manually (from the control room) isolate the train in which the leak has occurred and the redundant ECCS train in the other room will provide adequate core cooling. (Reference 102)

Shutdown:

The sump pump running indication lights and high level alarm from the ECCS room sump alert the operator of a flooding condition. Safe shutdown is continued on the redundant train.

3.6.2.7.1.15 Steam Generator Blowdown System

Description:

The two 4” Steam Generator Blowdown (SGBD) lines from the containment building enter the Auxiliary Building in room 236. Both lines are routed from room 236 through room 314 and then into the Turbine Building. Single containment isolation valves are located on each line inside room 236. When in use, the steam generator blowdown system normal operating conditions are 930 psig and 536°F. For a description of the blowdown system, see section 10.4.8.

Following a SGBD line break, abnormally high room temperatures in Rooms 236 and/or 314 would alert the operator. Also, drainage from either a rupture or crack in either of these two rooms would activate sump pump running indication lights in the main control room for the sump in room 115. This break is below the sensitivity level of the SFRCS so the plant would continue to operate until the operators take manual action. Once detected, required operator action would be to attempt to isolate the leak. If unable to isolate the leak, the operator would then trip the plant. A loss of offsite power is assumed to occur at this time if reactor power is above 15 percent.

The operator would continue by stopping all main and auxiliary feedwater to the affected steam generator to allow it to boil dry, and by opening the steam generator atmospheric vent valve (AVV) associated with the affected steam generator (if possible) to divert as much steam as possible from the unisolable break. The analysis assumed that the break is detected and all required operator actions are completed within 10 minutes after break initiation.

SGBD line breaks in room 314 are analyzed assuming an initial power level of 14.7 percent or below. This is because the isolation valves located in rooms 236 are normally closed above 14.7 percent power, depressurizing the downstream piping. For analysis at or below 14.7 percent power, the valves might be open. Therefore, a single failure of the isolation valve on the affected steam generator is assumed for these breaks.

SGBD line breaks in the annulus and in room 236 are analyzed for the bounding break upstream of the isolation valve. For these breaks, an initial reactor power of 102 percent of 2772 MWt is assumed, which equates to 100.37% of 2817 MWt. Following a manual reactor trip and subsequent isolation of feedwater to the affected steam generator, the affected steam generator will depressurize due to mass loss through the break. An assumed single failure of the atmospheric vent valve prevents the operator from limiting the energy release into the auxiliary building. This effect is most pronounced after the steam generator liquid has been expelled and the higher enthalpy steam is released. In addition, loss of offsite power following the reactor trip is assumed. This increases the reactor coolant system hot leg temperature and will affect the energy content of the steam release prior to full depressurization of the steam generator.

Proximity to Required Safety Related Equipment:

Safety related equipment required to mitigate SGBD line breaks are as presented on figures 3.6-65 through 3.6-70. Cables which are require mitigate the consequences of these events are qualified for the resultant environments within the affected areas.

Postulated Design Basis Break Locations:

All piping and valves have been analyzed and meet the seismic requirements of section 3.7 of the USAR and Reg. Guide 1.29 for normal, upset, and faulted conditions. High energy line break considerations meet the requirements of Standard Review Plan 3.6.1 and 3.6.2 including Branch Technical Position MEB 3-1. Break locations are postulated for each flow path at 1) all terminal ends and 2) all points where stress levels exceed .8 (1.2 Sh + Sa). Critical cracks are bounded by breaks. SGBD line breaks are postulated in rooms 236, 314, and the annulus. The locations of the postulated breaks are as presented in Figures 3.6-35, 3.6-36, 3.6-37, 3.6-38 and 3.6-39.

In addition to evaluating the consequences of breaks at required locations, environmental consequences of non-isolable breaks in Room 236 have been analyzed. The break locations are depicted in Figures 3.6-66 and 3.6-69. These analyses were performed to assure that all possible system configurations are analyzed to provide operational flexibility.

Compartment Pressurization:

SGBD line breaks involve all of the compartments within the Emergency Ventilation System negative pressure area. Significant flow paths were selected from the auxiliary building model as presented in figure 3.6-56. This information is contained in reference 61.

Rooms 303 and 314 each contain six large blowout panels. In room 314, all panels relieve to the turbine building heater bay area. In room 303, all six panels relieve to the environment. Because the pressurization rate is relatively slow, significant pressure differences between compartments do not develop. Therefore, the first blowout panel to lift could be located in either room 303 or in room 314. The first panel to lift will prevent further pressure increases and will prevent lifting of other panels or failure of doors. For each postulated break location, rupture of a blowout panel in the most adverse room (303 or 314) was evaluated with respect the environmental qualification of equipment.

Peak compartment pressures are presented in table 3.6-6. Peak pressures for all SGBD line breaks are bounded by main feedwater line breaks and/or main steam to AFWPT breaks.

Environmental Effects:

Compartment temperature responses were calculated using the same or substantially similar models as described previously for compartment pressures. All equipment and cables which are required to mitigate the consequences of the transients have been environmentally qualified for the resultant environments. Additional considerations for specific compartments are as follows:

Containment Annulus:

Due to large metal and concrete surface areas, the annulus construction is well suited for condensation of steam. For a SGBD line break in the annulus, the annulus was modeled as two separate volumes. This prevented conditions in the local area of steam release from being masked by averaging with the more remote regions. The immediate region of the steam release was modeled to be only large enough to include the flow paths into rooms 314 and 236. Thus, the predicted steam temperature in this area was maximized. Functional drawings for postulated breaks in the annulus are provided in figures 3.6-65 and 3.6-68.

Room 236 (Mechanical Penetration room #2):

Steam will enter all parts of the EVS negative pressure area. The most significant amounts of steam will enter rooms 314 and 115 through a pipe chase connecting rooms 314, 236, and 115. Blowout panels are modeled to lift (in separate cases) in either room 314 or 303. Immediate operation of the equipment on the 545' elevation is not required after this event. Equipment and cables needed to mitigate the break are qualified for the resultant environments. Adequate time is available for manual lineup of any affected valves in the Decay Heat Removal system before it is needed. Breaks are postulated both upstream and downstream of the containment isolation valves as shown on the functional drawings included as figures 3.6-66, 3.6.67, and 3.6-69.

Room 314 (Mechanical Penetration room #4):

Steam will enter all parts of the EVS negative pressure area. The most significant amounts of steam will enter rooms 236 and 115 through a pipe chase connecting rooms 314, 236, and 115. A blowout panel is conservatively assumed to lift in room 303, which maximizes steam flow through adjacent compartments. A functional drawing for the postulated break in room 314 is provided in figure 3.6-70. Effects of breaks in room 236 will bound effects of the postulated break in room 314 for all compartments except room 314. The rupture of the main feedwater line in this room would also be a more severe accident.

Shutdown:

Following dryout of the affected steam generator, auxiliary building temperatures will drop rapidly. This will ensure that a makeup source from the normal makeup system will be available as required. The reactor is maintained in hot standby conditions, with feedwater being supplied by at least one auxiliary feedwater pump to the unaffected steam generator. Cooldown will eventually proceed using the unaffected steam generator.

3.6.2.7.1.16 Environmental Effects Due to HELB in the Turbine Building

Analyses were performed to predict the pressure and temperature conditions that would result at essential equipment locations following pipe breaks in the following lines in the Davis-Besse Unit I Turbine Building:

Main Steam (MS)
Main Feedwater (MF)
Extraction Steam (ES)

The purpose of these analyses was to determine the environmental qualification parameters for equipment required to function in order to mitigate these Turbine Building High Energy Line Breaks. Other high energy systems in the Turbine Building were not evaluated because their mass and energy release are bounded by the above systems. While there may be additional safety related components exposed to the Turbine Building HELB environments, they are not required to function to mitigate the HELB.

Proximity to Required Safety Related Equipment:

The areas of essential equipment evaluated for potential effect by HELBs in the Turbine Building include:

Component Cooling Water Pump and Heat Exchanger Room
Control Room HVAC and Emergency Ventilation System Equipment Room
Low Voltage Switchgear Rooms E and F
Battery Rooms A and B
High Voltage Switchgear Rooms A and B
Auxiliary Feedwater Pump Rooms

Compartment Pressurization:

A Main Steam Line Break (MSLB) was found to cause the highest mass and energy release in the Turbine Building. This would cause the largest pressure and temperature excursion in the building and in adjoining Auxiliary Building structures. The blowdown utilized in the analysis is the same as that used for a Main Steam Line Break upstream of a Main Steam Isolation Valve. This models the worst case single failure.

Reference 100 documents the Turbine Building HELB transient analysis. The predicted pressure remains below the pressure required to cause structural failure of doors and barriers that protect the Auxiliary Building structures and components. The resulting pressure in the rooms of interest consequently remain low, as the restricted flow paths through ventilation openings limit the flow of steam into the rooms.

Environmental Effects:

The temperatures in the rooms of interest also remain close to the initial room temperature. The temperature within each room remains below the maximum acceptable value for all the commercial grade equipment that must function in the room, except for the Auxiliary Feedwater (AFW) Pump Rooms (Room 237 and 238). Consequently, those two rooms have additional environmental qualification requirements. In those rooms, the equipment required to function during a MSLB has been environmentally qualified to ensure acceptable accident mitigation.

Water flooding is not of concern in the rooms listed above. The doors and walls between the Auxiliary Building rooms and the Turbine Building prevent the entry of water. Curbs have been installed around the ventilation openings to the AFW Pump Rooms to prevent entry of floodwater. In addition, drainage from the AFW Pump rooms is adequate to prevent sufficient accumulation of water to affect the AFW pumps. Consequently, flooding hazards do not exist for the included essential equipment.

There are no significant radiation sources associated with the analyzed HELBs. Consequently, no radiation dose analyses are required.

The relative humidity reached in each of the evaluated rooms is within the design capability of the equipment in the room.

Shutdown:

Following a High Energy Line Break in the Turbine Building, the Steam and Feed Rupture Control System (SFRCS) will trip on either low Steam Generator pressure or high reverse

differential pressure, depending on break location. The mass and energy release for breaks below the sensitivity of SFRCS are bounded by the analyzed breaks. The MSLB, which causes the highest temperatures and pressures, is assumed to continue until the Steam Generator is empty due to assumed failure of its Main Steam Isolation Valve to close. This bounds smaller breaks where operator action will eventually cause isolation.

The resulting environment in each of the adjoining Auxiliary Building structures containing safe shutdown equipment has been shown to be below the maximum capability of the installed equipment. Consequently, all essential equipment will remain available to support safe shutdown of the facility.

3.6.2.7.2 Seismic Class II Fluid Systems

Seismic Class II fluid systems outside the Containment Vessel, which fall in the following categories, were subjected to an analysis to determine the environmental effects of a postulated rupture on safe shutdown:

a. Systems with fluid conditions equal to or below 275 psig and 200°F which are within Seismic Class I structures.

b. High volume water systems in Seismic Class II structures.

These systems are listed in Table 3.6-7.

All piping which penetrates the negative pressure areas (mechanical penetration rooms 105, 113, 115, 208, 225, 236, 314, and 303, shown on Figures 3.6-3, 3.6-4, and 3.6-5) has been seismically supported. Where deemed necessary to prevent damage to safety-related systems, portions of piping systems in the auxiliary building and intake structure have been seismically supported. In addition, material certification and 100 percent NDT of all welds, where applicable, are provided. This upgrading of Seismic Class II piping has been performed to preclude the effects of falling pipe, flooding, or loss of the negative pressure area. The affected systems are indicated in the following sections.

3.6.2.7.2.1 Fire Water System (Reference Figure 9.5-1)

Auxiliary Building:

Approximately 75 percent of this system has been seismically supported. This action was taken to prevent falling pipe or flooding from affecting safety-related equipment.

Intake structure:

The discharge piping for the diesel-driven fire water pump, from the check valve to the external wall, is seismically supported. This action is required to prevent the fire water system from feeding a break in this piping and consequently flooding the service pump room.

3.6.2.7.2.2 Demineralized Water System (Reference Figure 9.2-4a)

Portions of this system in the Auxiliary Building have been upgraded to Seismic Class I to prevent the loss of safety-related systems due to flooding or falling pipe. In addition, an automatic isolation valve is provided (outside the auxiliary building) on the discharge of the transfer pumps which closes on low system pressure.

Demineralized water supply line to the PWST (4”-HCC-40) has been repurposed as the normal fill line for the emergency feedwater storage tank (EWST) with isolation valves DW525, DW526, DW527 and EF24 normally closed and the pipe downstream of DW526 normally drained to prevent flooding.

3.6.2.7.2.3 Primary Water System

Portions of this system in the auxiliary building have been upgraded to Seismic Class I to prevent the loss of safety-related systems due to flooding or falling pipe. In addition, the primary water storage tank has been removed, and the primary water transfer pumps and associated piping has been abandoned in place by installing a blank flange in the seismic I piping downstream of the primary water transfer pump discharge isolation valve and upstream of the distribution header. The distribution header is now supplied with demineralized water through a one inch supply valve (PW 15).

3.6.2.7.2.4 Domestic Water System (Reference Figure 9.2-5)

The Domestic water supply from Carroll Township enters the Service Water tunnel about halfway between the Auxiliary Building and the Intake Structure. The line is non-seismic low energy and is subject to a line break.

The flooding rate would be as high as 850 gpm for a line break at the point where the 6 inch line enters the Service Water Tunnel. The volume of water is considered as infinite because the offsite Carroll Township water tower would be refilled as soon as it reached its low level setpoint. Analysis of this leak using the physical room data determined that the flood water would rise at a rate of approximately 1.25 feet per hour. The level would increase until it reached about 3.5 feet when the flow out of a ventilation louver to the water treatment building would equal the 850 gpm flood rate. In addition the operators would be alerted by the pressure switches on the Domestic Water to the Turbine Building or the supply to the Auxiliary Building or by the sump level alarms from the Valve Room. However, no Operator action is necessary to limit further water level increases. No safety related equipment will be impacted. The lowest equipment in the tunnel is SW 1395 and SW 1399 which are 52 inches above the floor. The above analysis does not include the sump pumps operating which could remove up to 400 gpm of the leakage until they flood out at about 1 foot in the room. The water leakage into the Water Treatment Building will not impact any safety related equipment.

Domestic Water Passes through or is located in some essential areas of the Auxiliary Building. Here a rupture would cause some minor flooding. In order to mitigate the effects of a pipe break in this area, the following provisions are provided:

a. No piping is located in the penetrations rooms.

b. Low pressure switches and automatic isolation valves are located outside the Auxiliary Building in the main headers entering the building. A rupture inside the auxiliary building would initiate switch actuation, alarming the operator and closing the valve.

If the Domestic Water System wash down valves are left open, the original design of the floor drainage system provides adequate drainage capability for each wash down station. No safety-related equipment would be affected by a Domestic Water system rupture.

A low pressure switch (PSL 5521B) in Room 603 was provided to alarm the operator and close an isolation valve (DM5796) in the Domestic Water supply to Chilled Water in the event of a rupture. This isolation valve is located in the Service Water Tunnel.

3.6.2.7.2.5 Roof and Floor Drainage System (Reference Figure 9.3-5)

Roof and floor drain piping in the following areas of the Auxiliary Building has been seismically supported to prevent flooding or the falling of pipe on safety-related systems:

a. Roof drains and down spouts in the diesel generator rooms, fuel handling area, mechanical equipment room, main steam line rooms, and HVAC equipment rooms

b. Control room

c. East side electrical penetration room

d. Low voltage switchgear room

e. Battery room

f. El. 603 east-west corridor

g. El. 585 east-west corridor

3.6.2.7.2.6 Borated Water Recirculation System (Reference Figure 9.1-5)

The system from the borated water tank discharge line to the recirculation pump, the recirculation water heater, and return line to the tank have been upgraded to Seismic Class I. In addition, a pressure switch in the discharge of the heater senses low system pressure, closes an isolation valve in the suction line to the recirculation pump, trips the pump, and alarms the operator.

3.6.2.7.2.7 Cooling Tower Makeup System (Reference Figure 10.4-4)

The Cooling Tower Makeup system, in the intake structure service pump room, has been upgraded to Seismic Class I from the makeup pump anchorage to the external wall. For added assurance against flooding due to a pipe rupture, a low pressure switch is provided on each pump discharge to trip the pump and close an automatic isolation valve.

3.6.2.7.2.8 Cooling Water System (Reference Figure 9.2-2)

Piping in non-essential portions of this system has been upgraded to Seismic Class I in the Auxiliary Building in order to prevent flooding or falling pipe from affecting other safety-related systems.

3.6.2.7.2.9 Ventilation/Air Conditioning System

An analysis of all ducting in the Auxiliary Building was undertaken to determine the effect of failing duct in safety-related areas since falling ducting could cause damage in the following areas:

a. Control room

b. Control cabinet room

c. Office

d. Cable spreading room

Seismic supports have been provided for ducting in all of the above areas.

3.6.2.7.2.10 Chilled Water System

This system is located in the mechanical equipment room above the control room in the auxiliary building. Ducting penetrates the floor into the control room. An analysis was made to determine the effect of a major rupture in this system. It was determined that the water level in the room would be about 0.5 inches if outleakage through the floor drain system and leakage past the ducts into the control room were neglected. Therefore, 6-inch curbs were provided around the duct penetrations.

3.6.2.7.2.11 Spent Fuel Pool Recirculation System (Reference Figure 9.1-5)

A portion of the spent fuel pool recirculation piping from the spent fuel pool to the spent fuel pool pumps has been upgraded to Seismic Class I in order to preclude the effects of flooding and falling pipe in a safety-related area.

3.6.2.7.2.12 Startup Feed Pump System (Reference Figure 10.4-12)

Prior to the installation of the motor driven feed pump (see Section 10.4.7.2), the startup feed pump was used for steam generator warmup and normal station shutdown. The startup feed pump is available for use as a backup to the motor driven feed pump in modes 4, 5, and 6. It can also be used in modes 1, 2, or 3 in the event that the main feed pumps, auxiliary feed pumps and motor driven feed pump are unavailable. It is located in the same room as one of the auxiliary feed pumps in the auxiliary building. A rupture of the suction pipe from the condensate storage tank or a rupture of the pump seal and lube oil cooler piping from the turbine plant cooling water system could cause flooding in the auxiliary feed pump room.

To support its availability as a backup to the MDFP in modes 4, 5, or 6 the following system configuration was required so that the startup feedwater pump did not create a flooding hazard to the auxiliary feed pumps:

a. The suction piping from the Auxiliary Building wall to the normally closed pump suction isolation valve was upgraded to Seismic Class I.

b. Normally closed isolation valves are installed in the cooling water supply and return piping to the coolers. These valves are located in the Turbine Building.

3.6.2.7.2.13 Circulating Water System (Reference Figure 10.4-4)

During normal station operation, four circulating water pumps, each with a capacity of 120,000 gpm, are in operation. Assuming a complete rupture of the main condenser circulating water expansion joint (108 inch I.D. pipe) on the inlet side of the condenser, two circulating pumps on

one train would run out. In addition, water from the cooling tower and piping flows back through the rupture. The condenser pit (El. 567) floods. At about El. 570, the motor driven feedwater pump auxiliary oil pump would be flooded rendering the MDFP out of service. At an elevation of approximately 571 feet, the Anticipatory Reactor Trip System (ARTS) switches which detect trips of Main Feedwater Pump Turbines will be flooded. ARTS was not designed and is not required to mitigate the flooding, but if ARTS would actuate, there would be no adverse consequences. The main feed pump turbine hydraulic and lube oil pump motors are flooded at El. 573. This results in a trip of the main feed pump turbines. At about El. 575 the condensate pump motors are flooded, causing the trip of these pumps. There is no other major equipment of consequence below El. 585. There are no flow paths for water to escape below El. 585. See Figures 3.6-20, 3.6-21, and 3.6-22.

Assuming a complete rupture of the expansion joints, the pumps on the affected line run out to about 149,000 gpm each where cavitation occurs due to inadequate NPSH. A flow of 298,000 gpm flows out of the break from the pump end. In addition, neglecting inertia, a flow of 159,000 gpm from components above the break drains into the pit from the condenser end. The total volume of water from the cooling tower to the condenser, which can drain into the pit, is approximately 282,000 gallons. The condenser pit to elevation 585 has a total capacity to hold approximately 1.48 X 10^6 gallons. The east side pit is connected to the west side pit by two open doorways and several Additional flow paths. These flow paths consist primarily of space between the high and low pressure condenser bottoms and the building floor and between the condenser walls and surrounding structures. The total volume in the east side pit is 542,000 gallons.

In order to detect early signs of flooding in the condenser pit, high level alarm switches have been provided in the condenser East and West pit sumps to alert the operators to a high condenser pit sump level by an annunciator located in the control room. In order to prevent flooding above the 585 foot level and then into areas which contain essential equipment, additional level switches are installed along the wall of the East condenser pit. These level switches will provide an automatic trip of all circulating water pumps and closure of their respective discharge isolation valves, thereby isolating the break from the circulating water canal. The circuit includes a short time delay to avoid spurious trips. Closure of the discharge valves is not required to prevent flooding of the 585 foot level of the turbine building. Assuming that the pumps trip but one or more discharge valves do not close, the level produced by water flowing back from the cooling tower and from the canal will equalize at approximately the 582 foot level, which is below the level of the turbine building floor.

3.6.2.7.2.14 Condensate System (Reference Figure 10.4-11)

This system is housed completely in the Turbine Building. The condensate pumps are located at El. 562 feet 6 inches, with portions of the system extending to El. 679. See Figures 3.6-20, 3.6-21, and 3.6-22. The maximum condensate design flow is 22,000 gpm with the operation of three pumps at full capacity. The termination of the Condensate System, by definition, is the deaerator heaters and storage tanks. They are located in the heater bay adjacent to the Auxiliary Building.

The potential for failures in this system are minimized by designing the piping system for pressures well above any pressures possible in the system. The L.P. heaters are designed for maximum pump shutoff head, while the deaerators are designed for maximum turbine extraction steam pressure.

Failures in the Condensate System can be readily detected by equipment performance alarms

in the control room indicating the malfunction to the operator. The type of alarm depends on the severity and. location of the failure and allows the operator the option of either isolating the failed component and reducing load or complete shutdown of the station. However, as with any gross failure in this system, the alarms on the steam generator would ultimately indicate decreased flow, loss of pressure, and low liquid level resulting in a possible reactor trip by the reactor protection system. The types of alarms are dependent on the location of the pipe rupture or equipment failure and are summarized below:

a. Failures in the system between the condenser and the condensate pumps would result in gradual flooding of the condenser pit setting off the condenser pit sump high-level alarms. In addition, the condenser hotwell low-level alarms are registered in the control room.

b. Failures in the equipment or piping between the condensate pumps and the deaerators would result in a high deaerator pressure and also low storage tank level alarm(s) in the control room.

c. A control valve in the discharge of the condensate pumps maintains a condensate flow not exceeding 26,800 gpm if the rupture occurs downstream of this valve.

The entire system can be brought to rest, with the termination of all flow, within several seconds from the time the condensate pumps are tripped. Static head would be the only driving force for water escaping through leaks at lower elevations.

Since water on El. 585 could leak into the essential areas noted in Subsection 3.6.2.7.2.13, 8-inch curbing has been provided preventing water entrance into the following rooms:

1. Auxiliary feed pump room

2. High voltage switchgear rooms

Water which flows into the Service Water tunnel or component cooling water equipment room would not cause levels dangerous to any essential equipment.

In the unlikely event of a rupture of a deaerator storage tank, a roof is provided above the auxiliary feed pump room stairwell to prevent water from the tank from draining directly into the room. Due to the large volume of the turbine building, steam from flashing water has no effect in essential areas.

Therefore, any postulated failures in the condensate system can be detected and controlled quickly, and in no way affect the operation of required essential equipment.

3.6.2.7.2.15 Feedwater System (Reference Figure 10.4-12)

The Seismic Class II portion of this system is completely in the Turbine Building. By definition, the feedwater system begins at the deaerators. The main feed pumps are located at El. 567. Normal total feedwater flow is 25,620 gpm (two pumps). See Figures 3.6-20, 3.6-21, and 3.6-22.

The potential for failure in this system is minimized by designing the piping system for pressures above any pressures possible in the system. The H.P. heaters are designed for the maximum pump shutoff head.

Failures in the feedwater system can be readily detected by equipment performance alarms in the control room, indicating the malfunction to the operator. The type of alarm depends on the severity and location of the failure and allows the operator the option of either isolating the failed component and reducing load or complete shutdown of the station. However, with a gross failure, alarms on the steam generator would ultimately indicate decreased flow, loss of pressure, and low liquid level resulting in a possible reactor trip by the Reactor Protection System. The type of alarms are dependent on the location of the pipe rupture or equipment failure and are summarized below:

a. Failure in the suction pipe to the booster feed pumps would result in a low pump suction pressure alarm.

b. Failures in the equipment or piping between the main feed pump discharge and the steam generator(s) would result in pressure differential alarm(s) and closure of the feedwater control valves. Steam generator alarms would occur on low flow, low pressure, and low water level.

The entire system can be brought to rest, with the termination of all flow, within several seconds from the time the condensate and steam generator feed pumps are tripped. Static head would be the only driving force for water escaping through leaks at lower elevations.

Protection from flooding above El. 585 has been provided as indicated in Subsection 3.6.2.7.2.14. In addition, a wall has been provided to prevent jets from entering the component cooling equipment room. Due to the large volume of the turbine building, steam flooding has no effect in essential areas.

3.6.2.7.2.16 Service Water System (Reference Figure 9.2-1)

The Seismic Class II portion of the service water system supplies cooling water for the turbine building cooling water system. The piping is located at El. 585, entirely in the turbine building. One service water pump with a normal flow of 10,000 gpm supplies cooling water needs.

A rupture of the service water pipe upstream of the cooler temperature control valves would cause the pump to runout. A low-pressure switch in the pump discharge line closes an isolation valve and alarms the operator in the control room. A second low-pressure switch in the piping to the coolers could normally open the isolation valve from the circulating water system alternate cooling water supply. However, a reverse flow signal from a pressure differential switch across a check valve in the cooler return line negates opening of this valve, thereby isolating the break.

If the above circuitry failed to operate or the break was downstream of the cooler temperature control valves, flooding would be initially undetected. However, the rate of flooding would be much less than a circulating water system rupture.

Water from the break would flow on El. 585 into both the circulating pump house and condenser pit. The high level alarm at El. 567 would alarm the operator. There is adequate time for operator action to isolate the break long before any essential areas could be affected.

3.6.2.7.2.17 Turbine Plant Cooling Water System (Reference Figure 3.6-24)

The Cooling Water System in the Turbine Building removes heat from the turbine generator accessories and air compressors. Two cooling water pumps with a total flow of 8000 gpm normally are in operation. A 20,000 gallon high level cooling water tank is located in the heater bay at El. 657 and a low level cooling water tank (concrete) at El. 585.

A rupture between the pump and high level tank would result in low level alarms in both tanks and rising temperatures in all cooler equipment. This system, except for the supply pipe to the high level tank has a redundant loop. A rupture in either of the return lines from the high level to the low level tank would result in low level alarms in both tanks. Water from the break would drain to El. 567. Operator action to isolate the break would be taken long before flooding could affect any essential areas.

3.6.2.7.2.18 Chlorination Water Treatment System (Reference Figure 10.4-4)

Two screen wash pumps are located in the intake structure in the traveling screen room (see Figure 3.6-18). A 10-inch discharge header passes through the service water pump room to supply water for the cooling tower and intake water chlorination. This pipe was upgraded to Seismic Class I in order to prevent a rupture which could cause flooding of this room.

3.6.2.7.2.19 Station Makeup Water Treatment System

Two water treatment system makeup pumps are located in the intake structure in the traveling screen room (see Figure 3.6-18). A 6-inch discharge header passes through the service water pump room to supply water to the station makeup water-system. This pipe was upgraded to Seismic Class I in order to prevent a rupture which could cause flooding in this room.

TABLE 3.6-1

Criteria for Identification of Systems Outside Containment

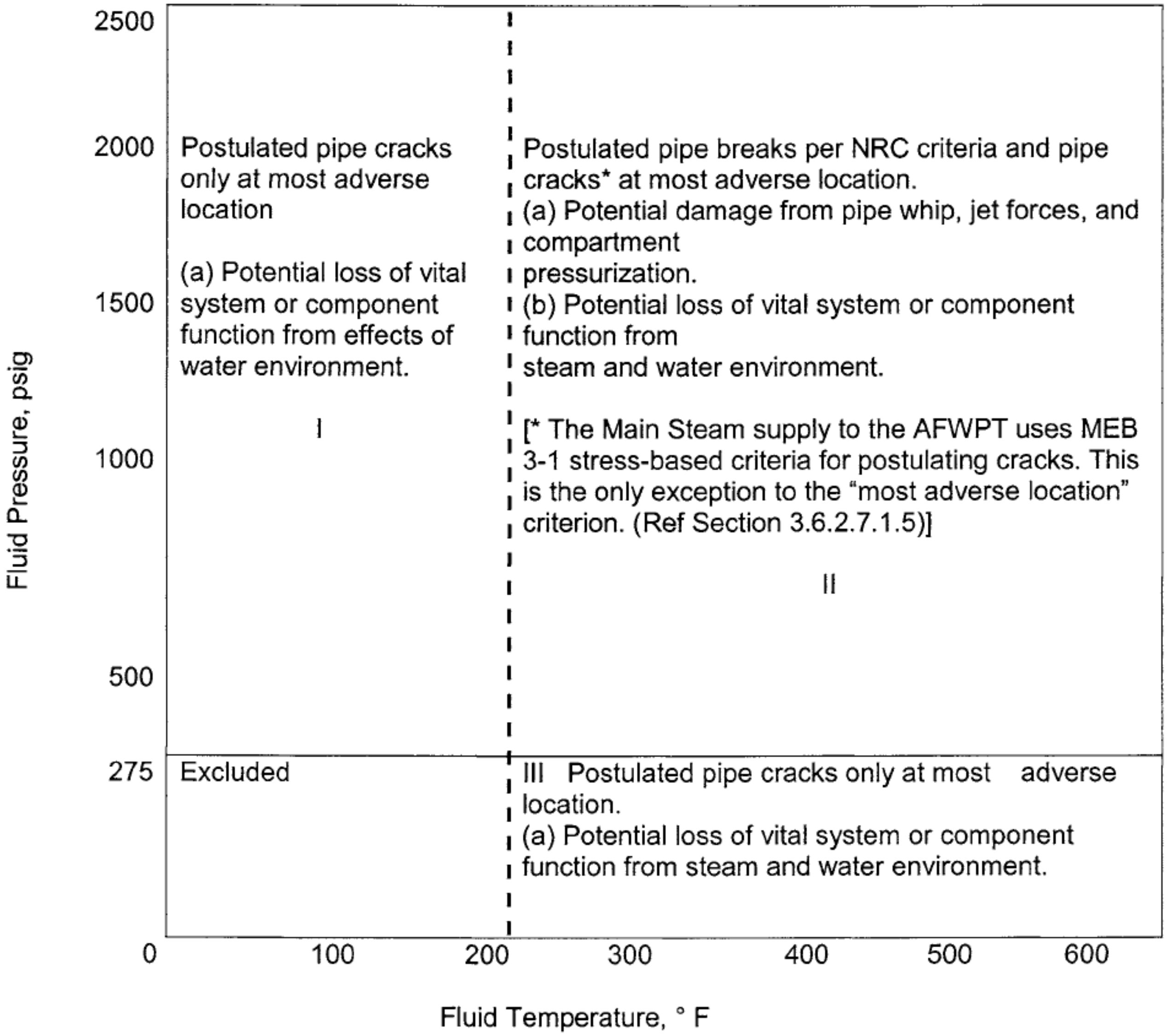


TABLE 3.6-2

Postulated Break Point Stresses (1)

System	Postulated Break Point	Stress (psi)

Note 1: Pipe break locations are shown on the applicable USAR Figure. For current stress values refer to the appropriate stress analysis.

TABLE 3.6-3

Pipe Whip and Jet Impingement Protection

Lines	Protection Categories*						
	A	B	C	D	E	F	Figures
High-pressure safety injection lines			X	X	X		3.6-28
Low-pressure safety injection lines	X	X	X	X	X		3.6-26
Core flooding lines	X	X	X	X	X		3.6-26
Makeup line	X	X					3.6-28
Cooling water lines to containment air coolers			X	X			-------
Reactor coolant letdown lines	X	X		X		X	3.6-16 & 3.6-27
Decay-heat removal lines				X	X		3.6-31
Main steam lines	X	X	X	X	X	X	3.6-10, 3.6-11, 3.6-12, 3.6-12A, 3.6-13 & 3.6-14
Main Feedwater lines	X	X	X	X	X	X	3.6-15
Auxiliary feedpump discharge lines	X	X		X	X		3.6-17
Primary cooling system lines up to and including the second isolation valves		X	X	X			3.6-30, 3.6-31, 3.6-32, 3.6-33, 3.6-34 & 3.6-28
Containment spray lines			X	X			3.6-25
Reactor coolant pump seal supply lines		X		X	X		3.6-29
Steam generator blowndown lines	X	X		X	X	X	3.6-35, 3.6-36, 3.6-37, 3.6-38, 3.6-39, 3.6-40, 3.6-41, 3.6-42, 3.6-43, 3.6-44, 3.6-45, 3.6-46, 3.6-47, 3.6-48, 3.6-49, 3.6-50, and 3.6-51
Reactor head vent**	X						3.6-56A

* PROTECTION CATEGORIES

A Lines that are restrained from damaging the primary coolant system

B Lines that are restrained from damaging the containment vessel

C Lines that are protected from damage by ruptured primary system piping (Note: Per the criteria in USAR Section 3.6.2, dynamic effects from a postulated rupture in the RCS can be excluded.)

TABLE 3.6-3 (Continued)

Pipe Whip and Jet Impingement Protection

* PROTECTION CATEGORIES

D Lines that are protected from damage by main steam and feedwater lines

E Lines that are protected from damage by or restrained from damaging their parallel redundant lines

F Lines that are restrained from damaging Safety-related structures or systems outside of containment vessel

**Whip restraint R-16 was removed from the reactor head vent line at its attachment point on OTSG 1-2.

TABLE 3.6-4

Typical Summary Values of Dimensionless Thrust Multiplication Factors

		Cold T_0 = 70°	Sub-cooled $0<(T_s - T_0)<60°$	Saturated $T_0 = T_s$
L/D - ->0	Sharp edged orifice	1.2	1.2	1.2
	Round-edged orifice	1.92	Value approaches 1.9 as L/D—>0	Value approaches 1.9 as L/D—>0
1<L/D<3	Sharp entrance	1.34	1.2	1.15
	Round entrance	1.84	1.45	1.25
L/D>3	Sharp and round entrances	Round and sharp entrances by conventional fluid mechanics calculations not described in technical note.	Not given in terms of upstream pressure Po. Reference is made to Moody's paper, method of calculation described; Values approximately 1.0	

TABLE 3.6-5

High Energy Piping Systems

Piping System	Failure Criteria	Compartment Pressurization	Steam Flooding	Water Flooding
System above 275 psig and 200° F				
Main Steam	II	X	X	X
Main feedwater	II	X	X	X
Systems above 275 psig or 200° F				
Reactor coolant letdown	I	X	X	X
Auxiliary feedwater	I			X
Auxiliary steam	III(*)		X	X
Reactor coolant makeup	I			X
L.P. injection	(1)			
Reactor coolant pump seal supply	I			X
Hot water heating	(*)		X	X
Containment spray	(2)			
Steam generator blowdown lines	II	X	X	X

(*) Pipe breaks for Seismic Class II piping

(1) The Low Pressure Injection system piping is exempt from the postulation of piping failures as discussed in Subsection 3.6.2.7.1.11.

(2) The Containment Spray system piping is exempt from the postulation of piping failures as discussed in Subsection 3.6.2.7.1.14.

See Table 3.6-6 for a listing of all rooms which can be pressurized. This table indicates room volumes, structural capacity, and the results of the pressurization analysis.

TABLE 3.6-6

Summary of Compartment Pressurization (1)(3)

Room No.	Free Volume	Structural Capacity psig	System Analyzed	Max Pressure Due to Break/Crack psig	Time for Peak Pressure sec
			El 643'0"		
601	69,326	6.19	36" Main steam	6.0	<1.0
601A	5,000	>21	36" Main steam	10.9	<1.0
602	31,205	7.0	36" Main steam	6.8	<1.0
602A	3,200	>21	36" Main steam	8.3	<1.0
			El. 623'0"		
500	61,300	3.6	6" Main steam to aux feed pump turbine	<0.3	1.0
501	52,600	2.5	6" Main steam to aux feed pump turbine	<0.3	1.0
515	57,100	2.9	6" Main steam to aux feed pump turbine	<0.3	1.0
			El. 603'0"		
400	17,600	9.3	6" Main steam to aux feed pump turbine	0.3	2.7
401	15,300	17.7	6" Main steam to aux feed pump turbine	0.8	7.4
404	11,300	5.0	6" Main steam to aux feed pump turbine	0.3	2.7
			El 585'0"		
300 300 B 301	397,000	14.5 18 >20	Main feed water and 6" main steam to aux. feed pump turbine	0.3	1.9
303	27,900	4	Main feed water	3.2	0.8
304	16,300	>20	Main feed water	0.9	0.9
312	11,200	4.4	Main feed water	0.9	1.5
313	25,900	2.7	Main feed water	0.8	1.9
314	40,400	6.5	Main feed water	3.9	1.1
Containment Vessel Annulus	642,000	1.93[2] (vessel)	Main feed water	0.8	7.1

TABLE 3.6-6 (Continued)

Room No.	Free Volume	Structural Capacity psig	System Analyzed	Max Pressure Due to Break/Crack psig	Time for Peak Pressure sec
			El 565'0"		
124	23,500	5.1	Main feedwater	0.7	2.8
208	23,900	17.7	Main feedwater	2.1	1.2
235	4,610	14.5	Main feedwater	0.7	1.6
236	17,400	4.0	Main feedwater	1.6	1.3
237	7,520	2.5	6" Main steam to aux feed pump turbine	0.6	53
238	10,100	2.5	6" Main steam to aux feed pump turbine	0.5	52

Summary of Compartment Pressurization (1)(3)

(1) Note: The assumptions and parameters used for the environmental qualification (EQ) of essential equipment/components that are affected by the break are provided in Tables 3.6-10 and 3.6-11.

(2) This value represents the theoretical critical buckling differential pressure of the containment vessel which is higher than the design valves given in Section 3.8.2.1.4d.

(3) High Energy Line Breaks in the Turbine Building were evaluated in Reference 100. The pressure in the Turbine Building has a peak of 15.7 psia. This pressure is exerted on adjoining Auxiliary Building Structures and barriers. All barriers have been evaluated and found capable of withstanding this load. See Section 3.6.2.7.1.16.

TABLE 3.6-7

Seismic Class II Fluid Systems

Piping System	Water Flooding	Falling Pipe
Category 'a'		
Fire Water	X	X
Demineralized water	X	X
Primary water[1]	X	X
Domestic water	X	X
Roof and floor drainage	X	X
Borated water recirculation	X	X
Cooling tower makeup	X	
Component cooling water	X	X
Ventilation/air conditioning		X
Chilled water	X	
Spent fuel pool recirculation	X	X
Startup feed pump	X	
Chlorination water	X	
Station makeup water treatment	X	
Category 'b'		
Circulating water	X	
Condensate	X	
Feedwater	X	
Service water	X	
Turbine plant cooling water	X	

(1) Note: Portions of the primary water system are abandoned in place and flooding is no longer possible from these points

TABLE 3.6-8

Dynamic Amplification Factors

A. Dynamic amplification factors derived for Arkansas Power and Light, Unit 1. NSSS: Babcock & Wilcox Company.

μ*	0.10 cps	0.50 cps	1.0 cps	5.0 cps	10.0 cps	50.0 cps
3	0.48	0.75	0.86	1.08	1.13	1.20
5	0.39	0.66	0.76	0.96	1.02	1.10

B. Dynamic amplification factors derived for Point Beach Nuclear Plant, Unit No. 1. NSSS: Westinghouse

Natural Frequency of Structure

μ	0.10 cps	0.50 cps	1.0 cps	5.0 cps	10.0 cps	50.0 cps
3	0.49	0.76	0.85	1.08	1.13	1.20
5	0.39	0.67	0.75	0.95	1.02	1.10

C. Dynamic amplification factors derived for Pilgrim. NSSS: General Electric

Natural Frequency of Structure

μ	0.10 cps	0.50 cps	1.0 cps	5.0 cps	10.0 cps	50.0 cps
3	0.81	1.11	1.15	1.21	1.21	1.21
5	0.67	0.99	1.05	1.10	1.11	1.11

* μ = ductility ratio = maximum deflection/yield deflection

TABLE 3.6-9

Safety Margin Evaluation* for Longitudinal Breaks

Piping System	Material	Yield Stress @ Oper. Temp (KSI)	Min Ultimate Tensile Strength (KSI)	Design Margin
INSIDE CONTAINMENT				
36" cold leg	A-106 Gr. C	19.7	70.0	3.55
2 ½ " pressurizer spray line	A-376 TP 316SS	20.0	75.0	3.75
2 ½ " high pressure injection line	SA-376 TP 316SS	20.0	75.0	3.75
OUTSIDE CONTAINMENT				
36" main steam line	A-155 Gr. KCF 70 Class I	18.7	70.0	3.74
6" main steam line	SA-106 Gr. B	17.0	60.0	3.52
18" main feedwater line	A-106 Gr. B	18.9	60.0	3.17
4" steam generator blowdown lines	SA-106 Gr. B	17.3	60.0	3.46

* Pipe whip restraint spacings are adjusted so as not to exceed yield stress of pipe at operating temperature.

TABLE 3.6-10

Analysis Input & Assumptions

INPUT OR ASSUMPTION	SOURCE
Pipe breaks are postulated as described in “Postulated Design Basis Break Locations.”	USAR Section 3.6.2.7
Double ended guillotine type pipe breaks are considered.	USNRC Standard Review Plan Section 3.6.1.
Worst case single failure in the active component of a required safety system is assumed when analyzing break effects.	USAR Section 3.6.2.1.k
For a break in the 2 1/2” MSAFPT line or the SGBD line operators will detect the break and initiate corrective actions within 10 minutes of break occurrence.	Ref. 61 & 62 (See Note 1)
Auxiliary and Turbine Building rooms begin at ≥ 104F except rooms 100, 105, 113 & 115, which begin at ≥ 95 F.	Ref. 60, 61, 62, 63 (See Note 2)
Loss of offsite power is assumed at the time of any reactor trip which occurs due to the postulated break.	Ref. 60, 61, 62
Concrete heat sinks are credited. Forced convection and/or radiative heat transfer are credited in some cases.	Ref. 60, 61, 62, 63, 67
Condensing heat transfer coefficients based on the UCHIDA correlation are used. 8 percent re-evaporation of condensate from heat sink surfaces is credited.	Ref. 60, 61, 62, 63, 67
Fire dampers will close to mitigate transmission of steam when justified by sufficiently low flow rates and sufficient high temperatures.	Ref. 60, 61, 62, 63

TABLE 3.6-10 (Continued)

Analysis Input & Assumptions

Ventilation system fan operation is not credited to reduce temperatures regardless of power supply.	Ref. 60, 61, 62, 63

NOTE 1 Upon initiation of a HELB, a transmitter will send an alarm to the control room operators thus giving them 10 minutes to identify the problem and to take action.

NOTE 2 The thermostat setting in the ECCS rooms have been adjusted such that the temperature in rooms 100, 105, 113 and 115 will be maintained below 95F.

TABLE 3.6-11

Results Summary – Peak Temperatures in Each Room
Break or Crack Type and Location

Affected Rooms	Annulus SGBD(b)	Rm 209 AS (b)	Rm 234 AS (b)	Rm 235 AS (b)	Rm 236 SGBD(b)	Rm 237 MS (b)	Rm 238 MS (b)	Rm 303 MFW (b)	Rm 304 AS (b)	Rm 314 SGBD(b)	Rm 314 MFW (b)	Rm 400 MS (c)	Rm 401 MS (b)	Rm 404 MS (c)	Rm 500 MS (b)	Rm 501 MS (b)
100	-	-	-	-	-	-	-	125	-	-	-	-	-	-	-	-
105	153	-	-	108	211	-	-	211	211	211	181	-	-	-	-	-
113	153	-	-	109	211	-	-	209	211	211	187	-	-	-	-	-
115	187	-	-	145	224	-	-	190	211	211	209	-	-	-	-	-
123	-	-	-	111	-	116	114	-	-	-	-	-	-	-	-	-
124	-	-	-	123	-	116	114	-	-	-	117	-	-	-	-	-
125	-	-	-	111	-	116	114	-	-	-	-	-	-	-	-	-
126	-	-	-	111	-	116	114	-	-	-	-	-	-	-	-	-
208	136	-	-	108	209	-	-	212	211	206	146	-	-	-	-	-
209	-	219	154	-	-	-	-	130	191	-	111	-	-	-	-	-
221	-	209	227	-	-	-	-	176	179	-	122	-	-	-	-	-
225	153	-	-	111	211	-	-	209	211	211	138	-	-	-	-	-
227	-	118	291	-	-	-	-	190	134	-	210	-	-	-	-	-
234	-	-	443	-	-	-	-	-	-	-	-	-	-	-	-	-
235	-	-	-	384	-	116	114	-	-	-	146	-	-	-	-	-
236	241	-	-	257	300	-	-	152	211	211	212	-	-	-	-	-
237	-	-	-	-	-	446	156	-	-	-	-	-	-	-	-	-
238	-	-	-	-	-	155	440	-	-	-	-	-	-	-	-	-
300	-	-	-	-	-	-	-	153	211	-	161	138	307	128	-	-
303	135	-	-	115	199	-	-	283	211	190	159	-	-	-	-	-
304	-	111	110	-	-	-	-	211	211	-	211	-	-	-	-	-
310	-	111	178	-	-	-	-	180	111	-	206	-	-	-	-	-
312	-	111	164	-	-	-	-	114	111	-	111	-	-	-	-	
313	-	111	178	-	-	-	-	180	111	-	206	-	-	-	-	-
314	215	-	-	142	211	-	-	119	211	211	264	-	-	-	-	-
400	-	-	-	-	-	-	-	-	-	-	-	311	233	237	-	-
401	-	-	-	-	-	-	-	-	-	-	-	110	425	107	-	-
404	-	-	-	-	-	-	-	-	-	-	-	190	198	314	-	-
405	-	-	-	-	-	-	-	-	-	-	-	112	115	137	-	-
500	-	-	-	-	-	-	-	-	-	-	-	-	-	-	370	192
501	-	-	-	-	-	-	-	-	-	-	-	-	-	-	324	377
515	-	-	-	-	-	-	-	-	-	-	-	-	-	-	207	181
Annulus	375/156	-	-	110	175	-	-	122	154	176	124	-	-	-	-	-
T/B	-	-	-	115	-	170	169	-	180	-	176	-	-	-	-	-

TABLE 3.6-11 (Continued)

Results Summary – Peak Temperatures in Each Room
Break or Crack Type and Location

Break Type and Other Locations

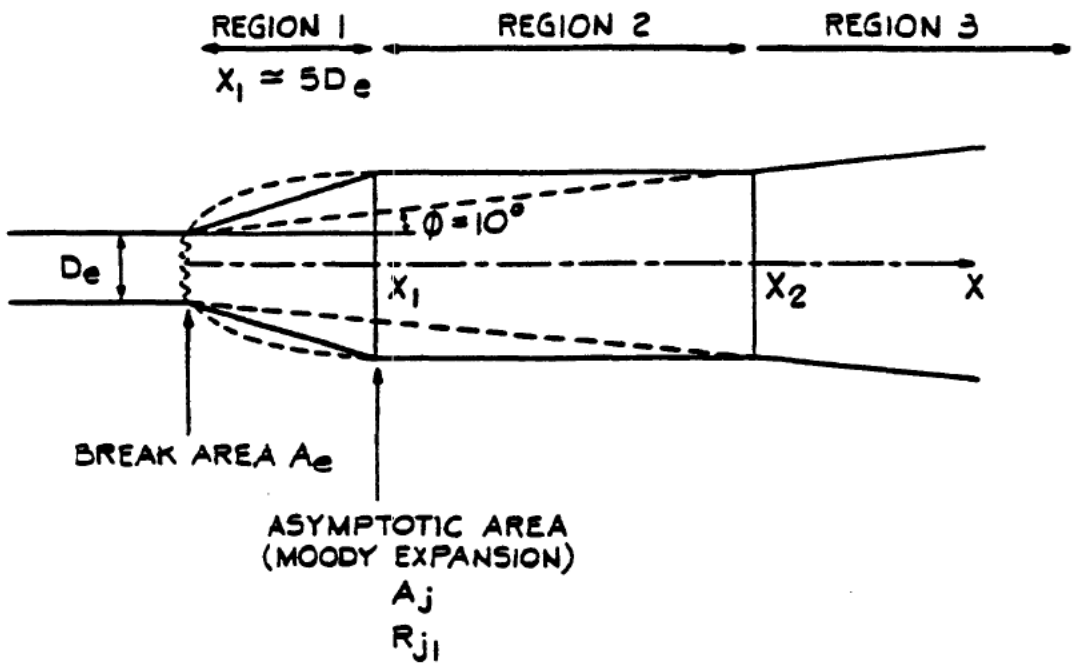
Affected Rooms	Rm 601 2 ½" MS	RM 601 6" MS	RM 601 36" MS	RM 602 6" MS	RM 602 36" MS
601 (East)	358	424	421	331	377
601 (West)	380	439	396	217	345
601A	147	145	387	150	334
602	328	264	357	451	409
602A	323	154	355	440	385

Notes: (c)=crack (b)=break
MS = Main Steam to AFPT
MFW = 18" Main Feedwater Line
AS = Auxiliary Steam
SGBD = 4" SG Blowdown Line ;

Lesser Cracks and Location

Affected Rooms	Rm 208 AS (c)	Rm 303 AS (c)	Rm 314 AS (c)	Rm 313 MFW (c)
208	127	105	106	-
236	-	-	113	-
303	-	136	112	-
314	-	-	167	-
Annulus	-	-	110	-
310	-	-	-	211
312	-	-	-	205
313	-	-	-	211

Affected Room	Rm 404 MFW (c)
300	211
400	211
401	105
404	211
405	167



DAVIS-BESSE NUCLEAR POWER STATION
FLUID JET GEOMETRY
FIGURE 3.6-1

REVISION 0
JULY 1982

Removed in Accordance with RIS 2015-17

DAVIS-BESSE NUCLEAR POWER STATION
CONTAINMENT AND AUXILIARY BUILDINGS-
PLAN ELEV. 643'-0"
M 120 Q
FIGURE 3.6-2

Removed in Accordance with RIS 2015-17

DAVIS-BESSE NUCLEAR POWER STATION
CONTAINMENT AND AUXILIARY BUILDINGS-

PLAN EL.585'-0"

M-123
FIGURE 3.6-3

08 03-22 16 DFN=H/PLDSGN/M123.DGN/TIF

Removed in Accordance with RIS 2015-17

DAVIS-BESSE NUCLEAR POWER STATION

CONTAINMENT AND AUXILIARY BUILDINGS-
PLAN ELEV. 565'-0"
M-124
FIGURE 3.6-4

DB 08 09 18 DFN=/J/MECH/M124.DGN/TIF

[redacted]

N

-BESSE NUCLEAR POWER STATION
NMENT AND AUXILIARY BUILDINGS-
PLAN EL.545'-0"
M-125
FIGURE 3.6-5

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OCTOBER 2016

DB 06-06-16 DFN=H/PLDSGN/M125.DGN/TIF

Removed in Accordance with RIS 2015-17

DAVIS-BESSE NUCLEAR POWER STATION
CONTAINMENT AND AUXILIARY BUILDINGS-
PLAN ELEV. 623'-0"
M-121
FIGURE 3.6-6

DB 11-14-17 DFN=/H/PLDSGN/M121.DGN/TIF

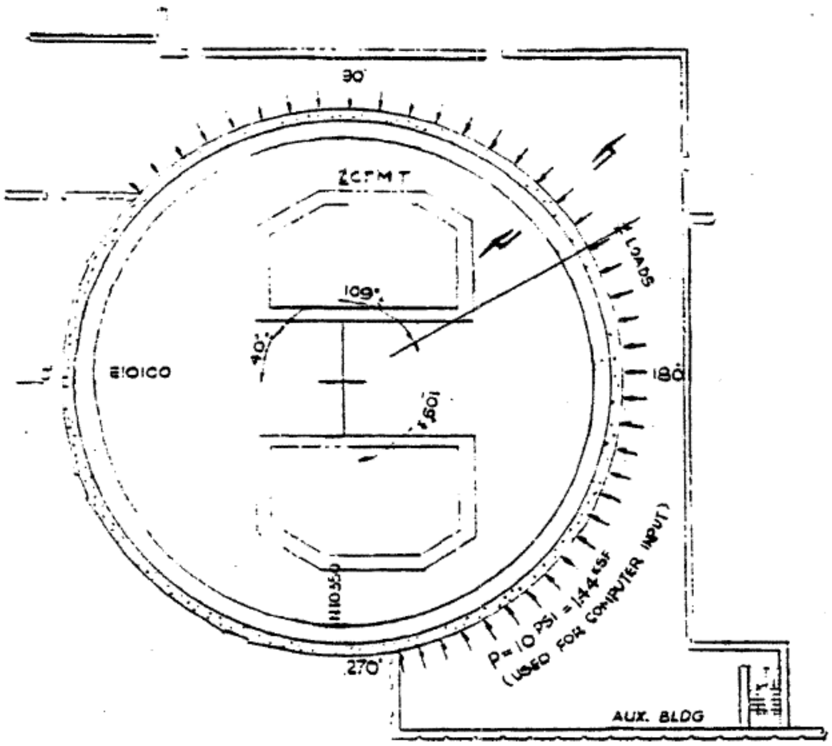
[redacted]RS SEE DRAWING M-103.

[redacted]SE NUCLEAR POWER STATION

[redacted]T AND AUXILIARY BUILDINGS
[redacted]LAN EL. 603'-0"
M-122
Q
FIGURE 3.6-7

REVISION 30
OCTOBER 2014

DB 07-17-14 DFN=J/RASDGN/M122.DGN/CIT



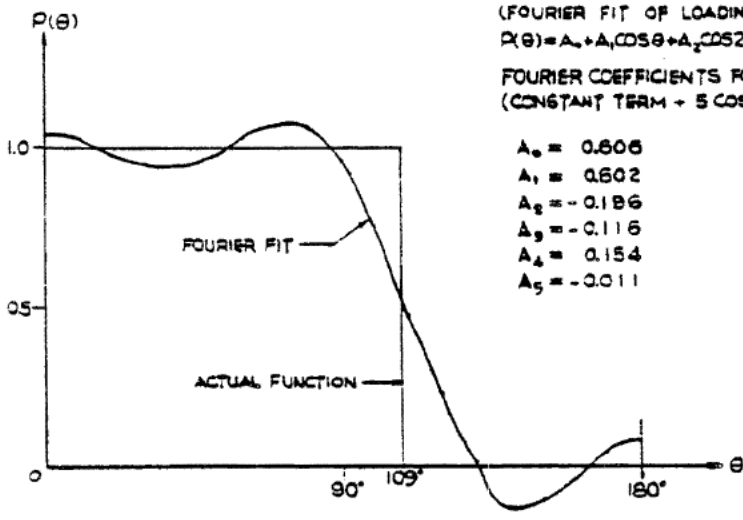
CONTAINMENT SHIELD BLDG CONCRETE SHELL WALL MAX. AREA OF PRESSURIZATION DUE TO POSTULATED PIPE RUPTURES

FOURIER SERIES EXPANSION:
(FOURIER FIT OF LOADING CONDITION AS SHOWN)

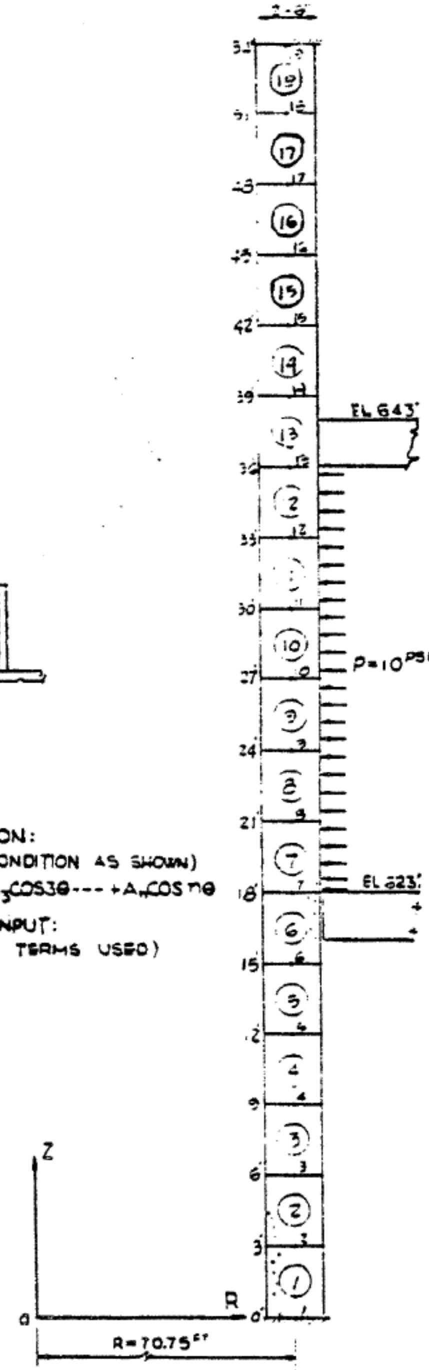
$P(\theta)=A_0+A_1\cos\theta+A_2\cos 2\theta+A_3\cos 3\theta --- +A_n\cos n\theta$

FOURIER COEFFICIENTS FOR INPUT:
(CONSTANT TERM + 5 COSINE TERMS USED)

$A_0 = 0.606$
$A_1 = 0.602$
$A_2 = -0.186$
$A_3 = -0.116$
$A_4 = 0.154$
$A_5 = -0.011$

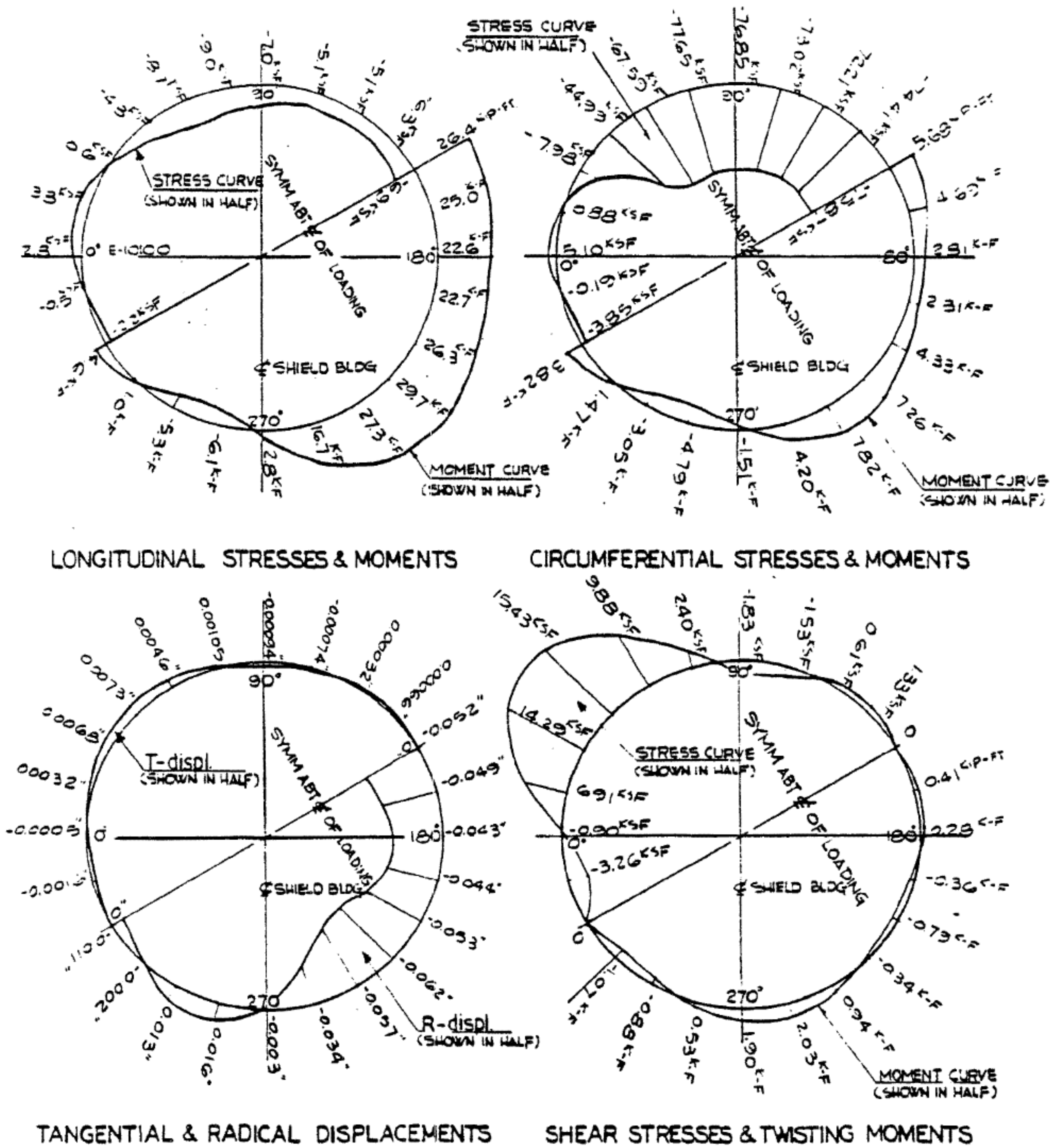


ACCURACY OF FOURIER FIT
(FOR UNSYMMETRICAL LOADING ON SHELL)



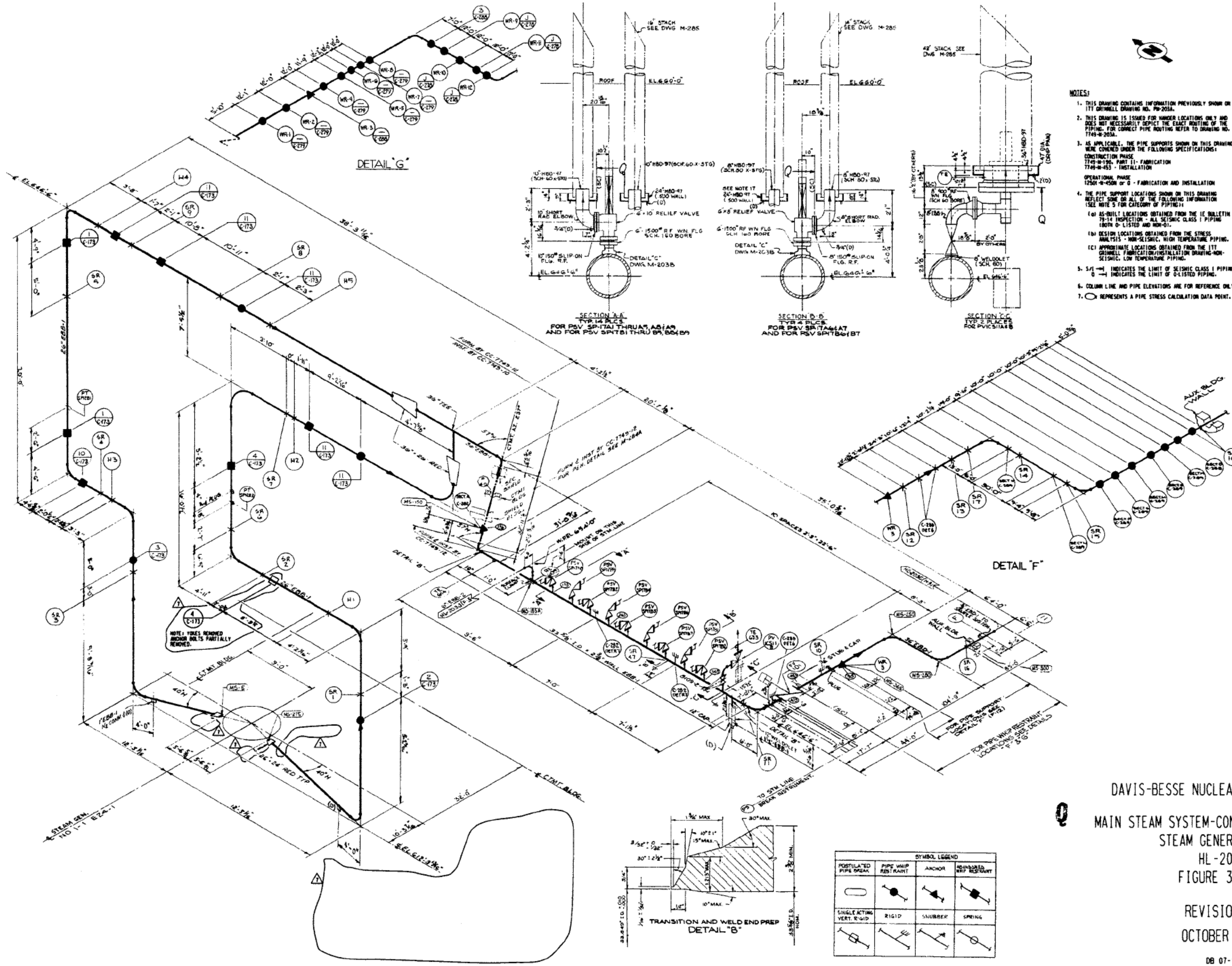
SECTION OF SHELL WALL
(FINITE ELEMENTS & NODAL PTS)

DAVIS-BESSE NUCLEAR POWER STATION
SHIELD BUILDING PRESSURIZATION
COMPUTER INPUT INFORMATION
FIGURE 3.6-8



DAVIS-BESSE NUCLEAR POWER STATION
SHIELD BUILDING SHELL STRESS AND DISPLACEMENTS
FIGURE 3.6-9

REVISION 0
JULY 1982



NOTES:

1. THIS DRAWING CONTAINS INFORMATION PREVIOUSLY SHOWN ON ITT GRINNELL DRAWING NO. PM-203A.
2. THIS DRAWING IS ISSUED FOR HANGER LOCATIONS ONLY AND DOES NOT NECESSARILY DEPICT THE EXACT ROUTING OF THE PIPING. FOR CORRECT PIPE ROUTING REFER TO DRAWING NO. 7749-M-203A.
3. AS APPLICABLE, THE PIPE SUPPORTS SHOWN ON THIS DRAWING WERE COVERED UNDER THE FOLLOWING SPECIFICATIONS:

   CONSTRUCTION PHASE
   7749-M-190, PART II - FABRICATION
   7749-M-453 - INSTALLATION

   OPERATIONAL PHASE
   12501-M-450N or Q - FABRICATION AND INSTALLATION
4. THE PIPE SUPPORT LOCATIONS SHOWN ON THIS DRAWING REFLECT SOME OR ALL OF THE FOLLOWING INFORMATION (SEE NOTE 5 FOR CATEGORY OF PIPING):
   (a) AS-BUILT LOCATIONS OBTAINED FROM THE IE BULLETIN 79-14 INSPECTION - ALL SEISMIC CLASS I PIPING (BOTH Q- LISTED AND NON-Q).
   (b) DESIGN LOCATIONS OBTAINED FROM THE STRESS ANALYSIS - NON-SEISMIC, HIGH TEMPERATURE PIPING.
   (C) APPROXIMATE LOCATIONS OBTAINED FROM THE ITT GRINNELL FABRICATION/INSTALLATION DRAWING-NON-SEISMIC, LOW TEMPERATURE PIPING.
5. S/I —| INDICATES THE LIMIT OF SEISMIC CLASS I PIPING.
   Q —| INDICATES THE LIMIT OF Q-LISTED PIPING.
6. COLUMN LINE AND PIPE ELEVATIONS ARE FOR REFERENCE ONLY.
7. ◯: REPRESENTS A PIPE STRESS CALCULATION DATA POINT.

DAVIS-BESSE NUCLEAR POWER STATION

Q

MAIN STEAM SYSTEM-CONTAINMENT BUILDING
STEAM GENERATOR 1-1
HL-203A
FIGURE 3.6-10

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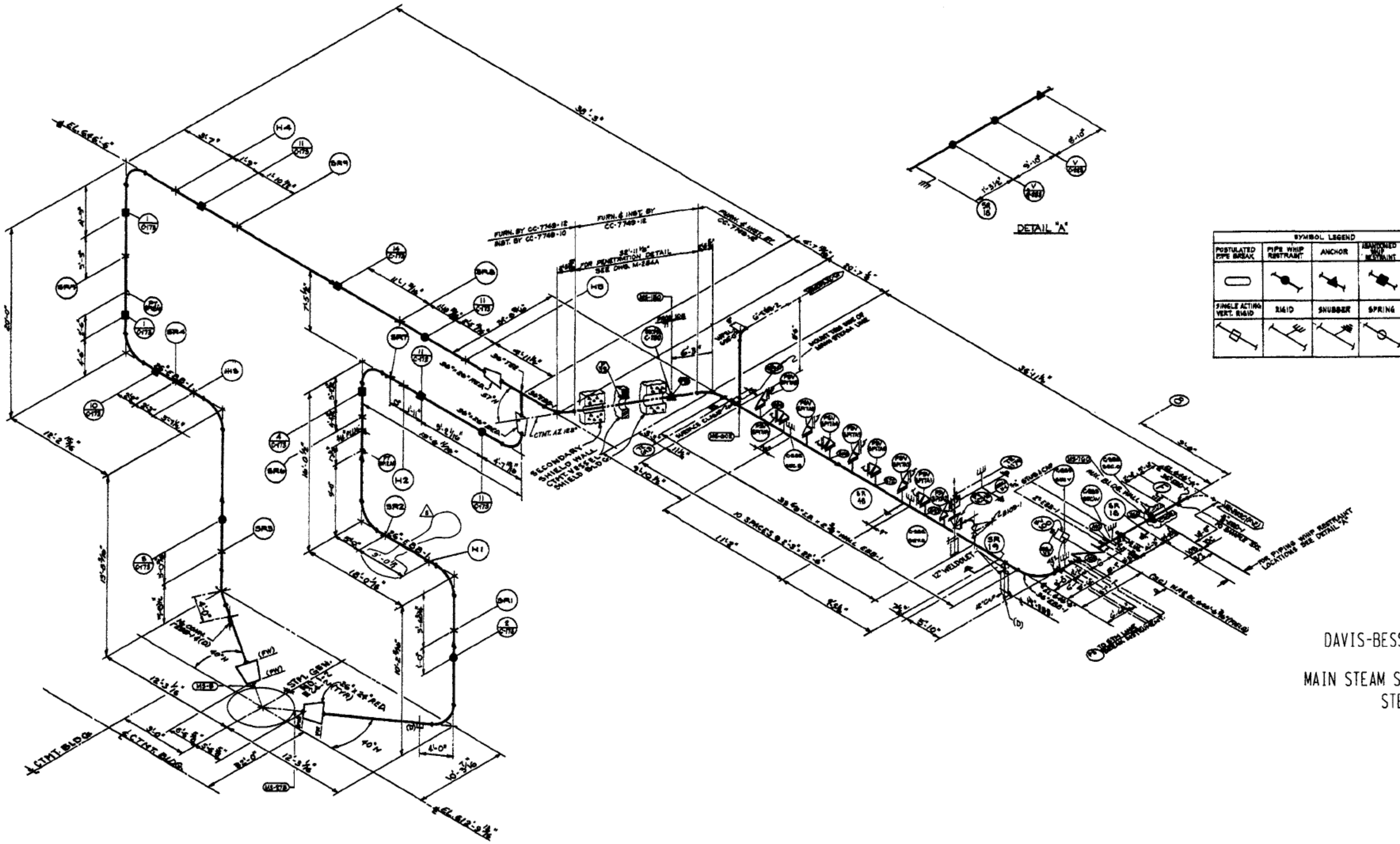
DB 07-17-14 DFN=H/PLDSGN/HL203B.DGN/TIF

NOTES:

1. THIS DRAWING CONTAINS INFORMATION PREVIOUSLY SHOWN ON ITT GRINNELL DRAWING NO. PW-203B.
2. THIS DRAWING IS ISSUED FOR HANGER LOCATIONS ONLY AND DOES NOT NECESSARILY DEPICT THE EXACT ROUTING OF THE PIPING. FOR CORRECT PIPE ROUTING REFER TO DRAWING NO. 7749-M-203B.
3. AS APPLICABLE, THE PIPE SUPPORTS SHOWN ON THIS DRAWING WERE COVERED UNDER THE FOLLOWING SPECIFICATIONS:

   CONSTRUCTION PHASE
   7749-M-190, PART II - FABRICATION
   7749-M-453 - INSTALLATION

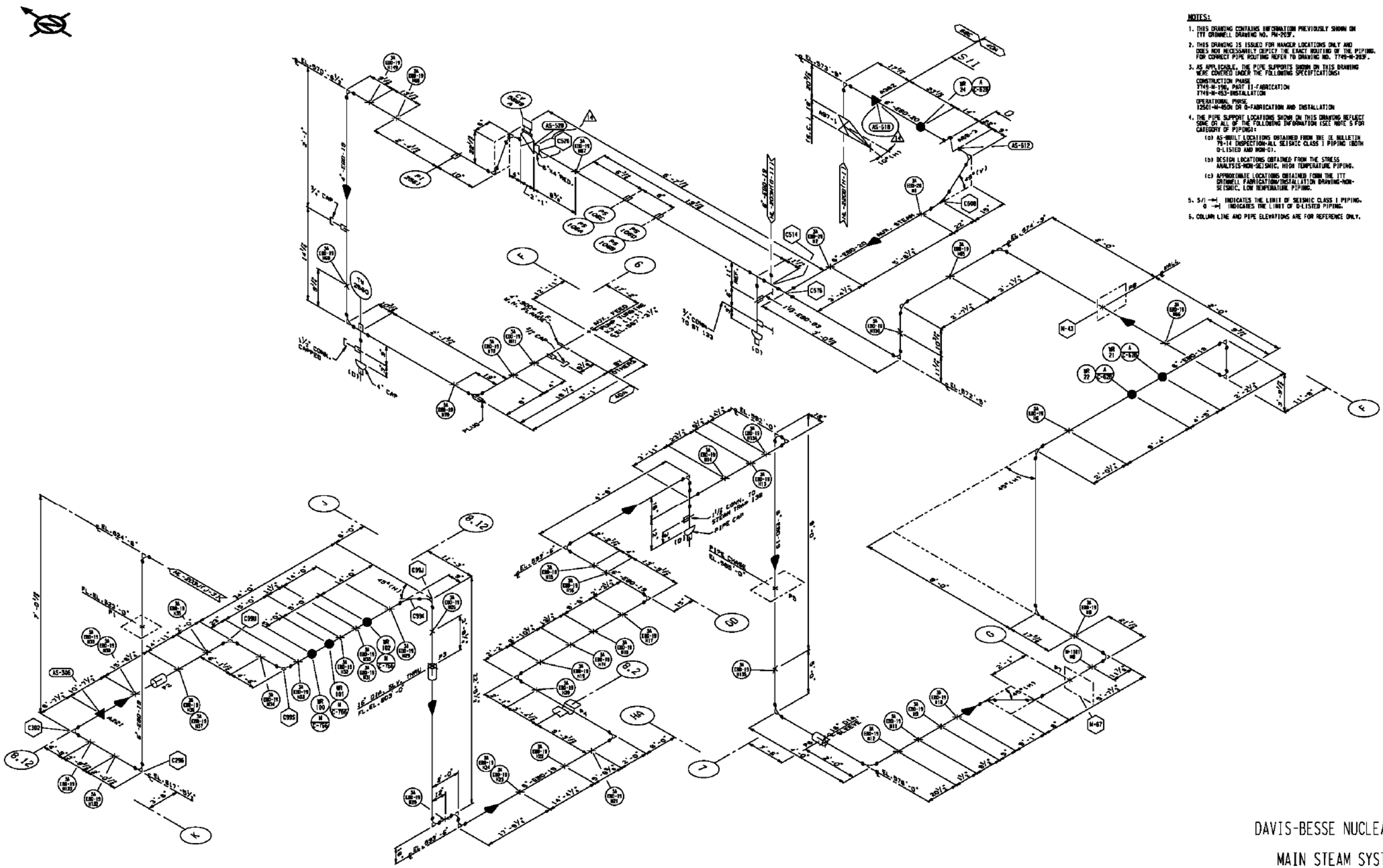
   OPERATIONAL PHASE
   12501-M-450N or Q - FABRICATION AND INSTALLATION
4. THE PIPE SUPPORT LOCATIONS SHOWN ON THIS DRAWING REFLECT SOME OR ALL OF THE FOLLOWING INFORMATION (SEE NOTE 5 FOR CATEGORY OF PIPING):
   (a) AS-BUILT LOCATIONS OBTAINED FROM THE IE BULLETIN 79-14 INSPECTION - ALL SEISMIC CLASS I PIPING (BOTH Q- LISTED AND NON-Q).
   (b) DESIGN LOCATIONS OBTAINED FROM THE STRESS ANALYSIS - NON-SEISMIC, HIGH TEMPERATURE PIPING.
   (C) APPROXIMATE LOCATIONS OBTAINED FROM THE ITT GRINNELL FABRICATION/INSTALLATION DRAWING-NON-SEISMIC, LOW TEMPERATURE PIPING.
5. S/I —| INDICATES THE LIMIT OF SEISMIC CLASS I PIPING.
   Q —| INDICATES THE LIMIT OF Q-LISTED PIPING.
6. COLUMN LINE AND PIPE ELEVATIONS ARE FOR REFERENCE ONLY.
7. ◯ REPRESENTS A PIPE STRESS CALCULATION DATA POINT.



DAVIS-BESSE NUCLEAR POWER STATION

MAIN STEAM SYSTEM-CONTAINMENT BUILDING
STEAM GENERATOR 1-2
HL-203B
FIGURE 3.6-11

REVISION 30
OCTOBER 2014

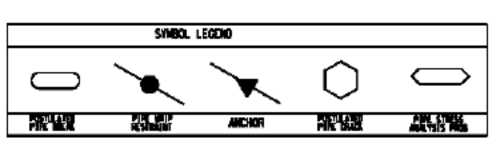


NOTES:

1. THIS DRAWING CONTAINS INFORMATION PREVIOUSLY SHOWN ON ITT GRINNELL DRAWING NO. PM-203F.
2. THIS DRAWING IS ISSUED FOR HANGER LOCATIONS ONLY AND DOES NOT NECESSARILY DEPICT THE EXACT ROUTING OF THE PIPING. FOR CORRECT PIPE ROUTING REFER TO DRAWING NO. 7749-M-203F.
3. AS APPLICABLE, THE PIPE SUPPORTS SHOWN ON THIS DRAWING WERE COVERED UNDER THE FOLLOWING SPECIFICATIONS:

   CONSTRUCTION PHASE
   7749-M-190, PART II-FABRICATION
   7749-M-453-INSTALLATION

   OPERATIONAL PHASE
   12501-M-450N OR Q-FABRICATION AND INSTALLATION
4. THE PIPE SUPPORT LOCATIONS SHOWN ON THIS DRAWING REFLECT SOME OR ALL OF THE FOLLOWING INFORMATION (SEE NOTE 5 FOR CATEGORY OF PIPING):
   (a) AS-BUILT LOCATIONS OBTAINED FROM THE IE BULLETIN 79-14 INSPECTION-ALL SEISMIC CLASS I PIPING (BOTH Q-LISTED AND NON-Q).
   (b) DESIGN LOCATIONS OBTAINED FROM THE STRESS ANALYSIS-NON-SEISMIC, HIGH TEMPERATURE PIPING.
   (c) APPROXIMATE LOCATIONS OBTAINED FORM THE ITT GRINNELL FABRICATION/INSTALLATION DRAWING-NON-SEISMIC, LOW TEMPERATURE PIPING.
5. S/I —| INDICATES THE LIMIT OF SEISMIC CLASS I PIPING.
   Q —| INDICATES THE LIMIT OF Q-LISTED PIPING.
6. COLUMN LINE AND PIPE ELEVATIONS ARE FOR REFERENCE ONLY.



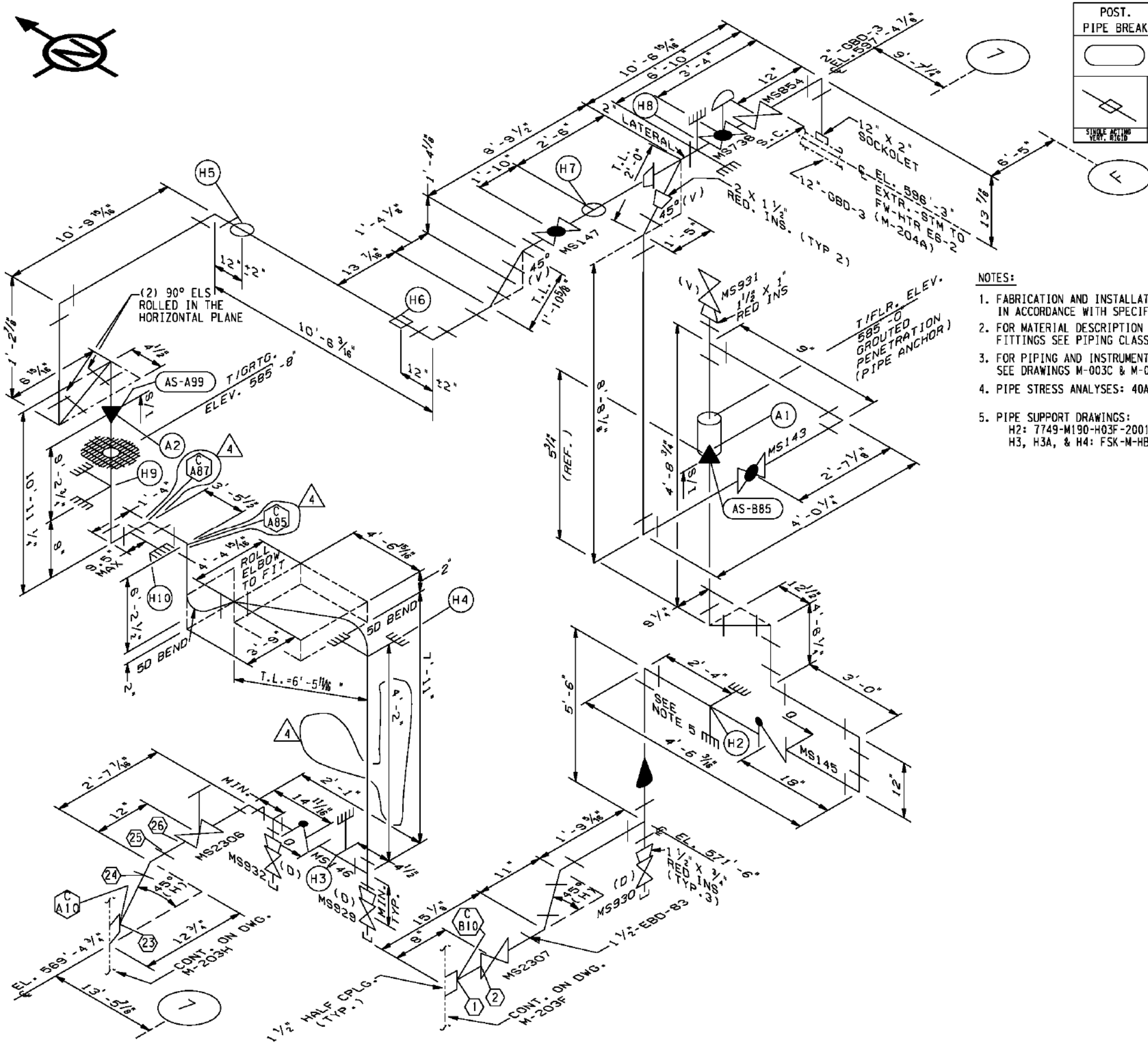
DAVIS-BESSE NUCLEAR POWER STATION

MAIN STEAM SYSTEM-SUPPLY TO
AUXILIARY FEEDWATER PUMP TURBINE 1-1
HL-203F
FIGURE 3.6-12

Q

REVISION 29
DECEMBER 2012

DB 08-14-12 DFN:H:/PLDSGN/HL203F.DGN



POST. PIPE BREAK	PIPE WHIP RESTRAINT	ANCHOR	POST. PIPE CRACK
SINGLE ACTING VERT. RIGID	RIGID	SNUBBER	SPRING

NOTES:

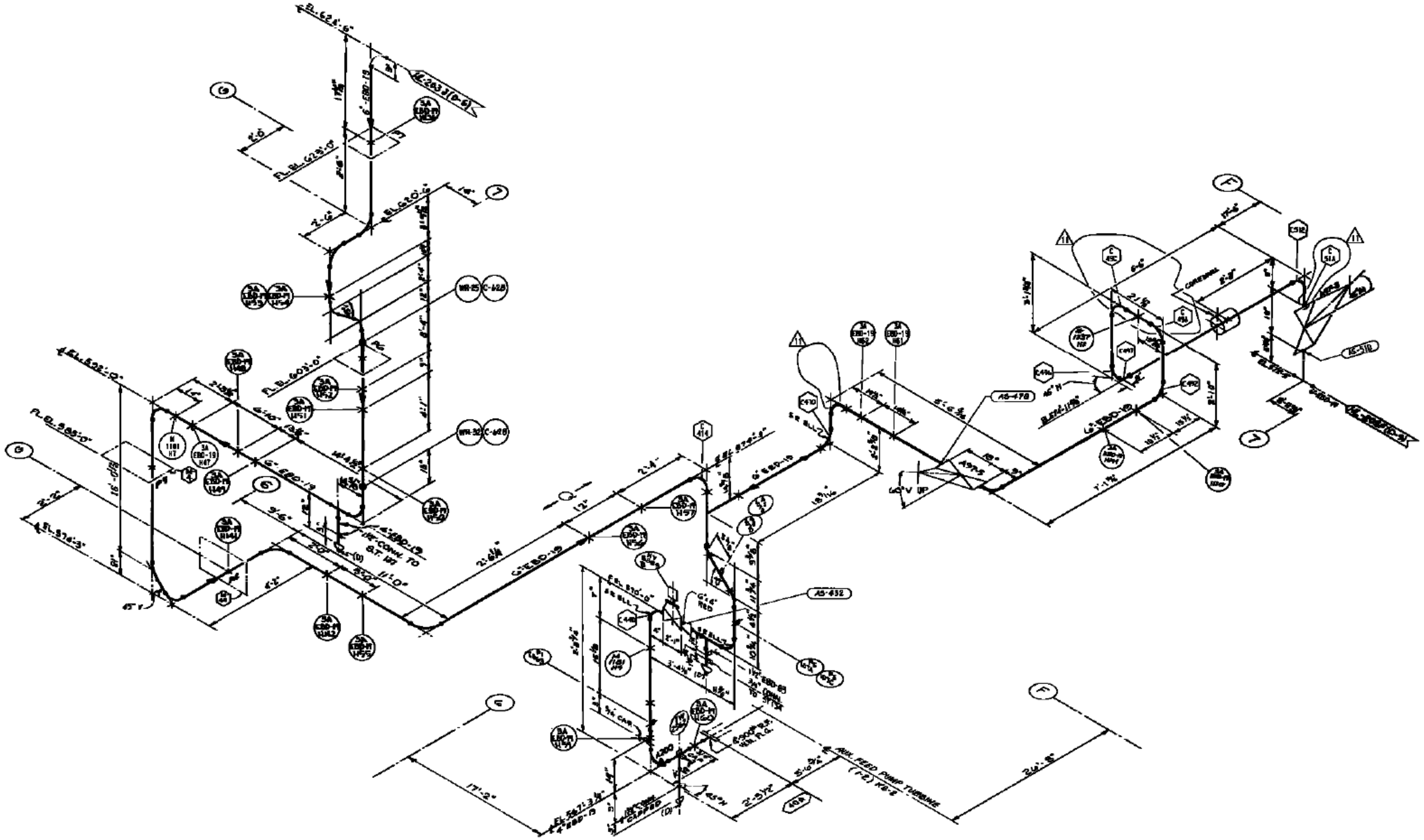
1. FABRICATION AND INSTALLATION TO BE DONE IN ACCORDANCE WITH SPECIFICATION M-453Q & M-453.
2. FOR MATERIAL DESCRIPTION OF PIPING, VALVES, AND FITTINGS SEE PIPING CLASS SHEETS M-601.
3. FOR PIPING AND INSTRUMENTATIONS DIAGRAMS SEE DRAWINGS M-003C & M-004B.
4. PIPE STRESS ANALYSES: 40A.
5. PIPE SUPPORT DRAWINGS:
   H2: 7749-M190-H03F-2001 (W/SUPPORT 3A-EBD-20-H2)
   H3, H3A, & H4: FSK-M-HBD-430-4-H

DAVIS-BESSE NUCLEAR POWER STATION

AFPT MAIN STEAM
MINIMUM FLOW LINE
FSK-M-EBD-83-1
FIGURE 3.6-12A

REVISION 29
DECEMBER 2012

08-13-12 DFN=H:/PLDSGN/EBD831.DGN



NOTES:

1. THIS DRAWING CONTAINS INFORMATION PREVIOUSLY SHOWN ON ITT GRINNELL DRAWING NO. PM-203H.
2. THIS DRAWING IS ISSUED FOR HANGER LOCATION ONLY AND DOES NOT NECESSARILY DEPICT THE EXACT ROUTING OF THE PIPING. FOR CORRECT PIPE ROUTING REFER TO DRAWING NO. 7749-M-203H.
3. AS APPLICABLE, THE PIPE SUPPORTS SHOWN ON THIS DRAWING WERE COVERED UNDER THE FOLLOWING SPECIFICATIONS:

   CONSTRUCTION PHASE
   7749-M-190, PART II-FABRICATION
   7749-M-453-INSTALLATION

   OPERATIONAL PHASE
   12501-M-450N or Q-FABRICATION AND INSTALLATION
4. THE PIPE SUPPORT LOCATIONS SHOWN ON THIS DRAWING REFLECT SOME OR ALL OF THE FOLLOWING INFORMATION (SEE NOTE 5 FOR CATEGORY OF PIPING):
   (a) AS-BUILT LOCATIONS OBTAINED FROM THE IE BULLETIN 79-14 INSPECTION-ALL SEISMIC CLASS 1 PIPING (BOTH Q-LISTED AND NON-Q).
   (b) DESIGN LOCATIONS OBTAINED FROM THE STRESS ANALYSIS-NON-SEISMIC, HIGH TEMPERATURE PIPING.
   (c) APPROXIMATE LOCATIONS OBTAINED FROM THE ITT GRINNELL FABRICATION/INSTALLATION DRAWING-NON-SEISMIC, LOW TEMPERATURE PIPING.
5. S/I INDICATES THE LIMIT OF SEISMIC CLASS I PIPING.

   Q INDICATES THE LIMIT OF Q-LISTED PIPING.
6. COLUMN LINE AND PIPE ELEVATIONS ARE FOR REFERENCE ONLY.

SYMBOL LEGEND

POSTULATED PIPE BREAK	PIPE WHIP RESTRAINT	ANCHOR	POSTULATED PIPE CRACK	PIPE STRESS ANALYSIS PROB.

DAVIS-BESSE NUCLEAR POWER STATION

MAIN STEAM SYSTEM
SUPPLY TO AUXILIARY FEEDWATER
PUMP TURBINE 1-2 AND EXHAUST
HL-203H

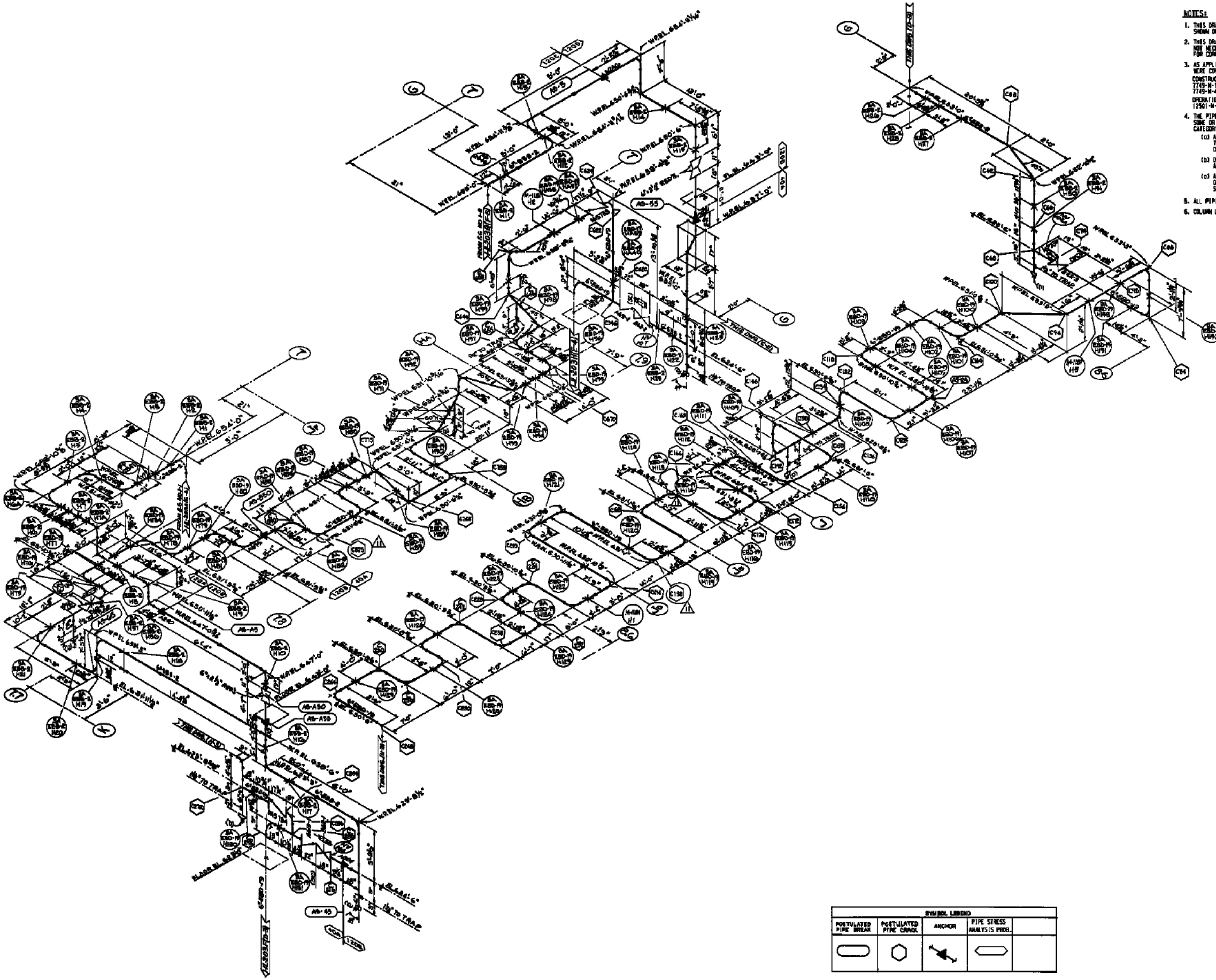
Q

FIGURE 3.6-13

REVISION 29

DECEMBER 2012

DB 08-14-12 DFN=/J/RASDGN/HL203H.DGN/CIT



NOTES:

1. THIS DRAWING CONTAINS PARTIAL INFORMATION PREVIOUSLY SHOWN ON ITT GRINNELL DRAWING NO. PM-203J.
2. THIS DRAWING IS ISSUED FOR HANGER LOCATIONS ONLY AND DOES NOT NECESSARILY DEPICT THE EXACT ROUTING OF THE PIPING. FOR CORRECT PIPE ROUTING REFER TO DRAWING NO. 7749-M-203J.
3. AS APPLICABLE, THE PIPE SUPPORTS SHOWN ON THIS DRAWING WERE COVERED UNDER THE FOLLOWING SPECIFICATIONS:

   CONSTRUCTION PHASE
   7749-M-190, PART II-FABRICATION
   7749-M-453-INSTALLATION

   OPERATIONAL PHASE
   12501-M-450M OR Q-FABRICATION AND INSTALLATION
4. THE PIPE SUPPORT LOCATIONS SHOWN ON THIS DRAWING REFLECT SOME OR ALL OF THE FOLLOWING INFORMATION (SEE NOTE 5 FOR CATEGORY OF PIPING):
   (a) AS-BUILT LOCATIONS OBTAINED FROM THE IE BULLETIN 79-14 INSPECTION-ALL SEISMIC CLASS I PIPING (BOTH Q-LISTED AND NON-Q).
   (b) DESIGN LOCATIONS OBTAINED FROM THE STRESS ANALYSIS-NON-SEISMIC, HIGH TEMPERATURE PIPING.
   (c) APPROXIMATE LOCATIONS OBTAINED FROM THE ITT GRINNELL FABRICATION/INSTALLATION DRAWING-NON-SEISMIC, LOW TEMPERATURE PIPING.
5. ALL PIPING SHOWN ON THIS DRAWING IS SEISMIC CLASS I.
6. COLUMN LINE AND PIPE ELEVATIONS ARE FOR REFERENCE ONLY.

SYMBOL LEGEND				
POSTULATED PIPE BREAK	POSTULATED PIPE CRACK	ANCHOR	PIPE STRESS ANALYSIS PROB.	

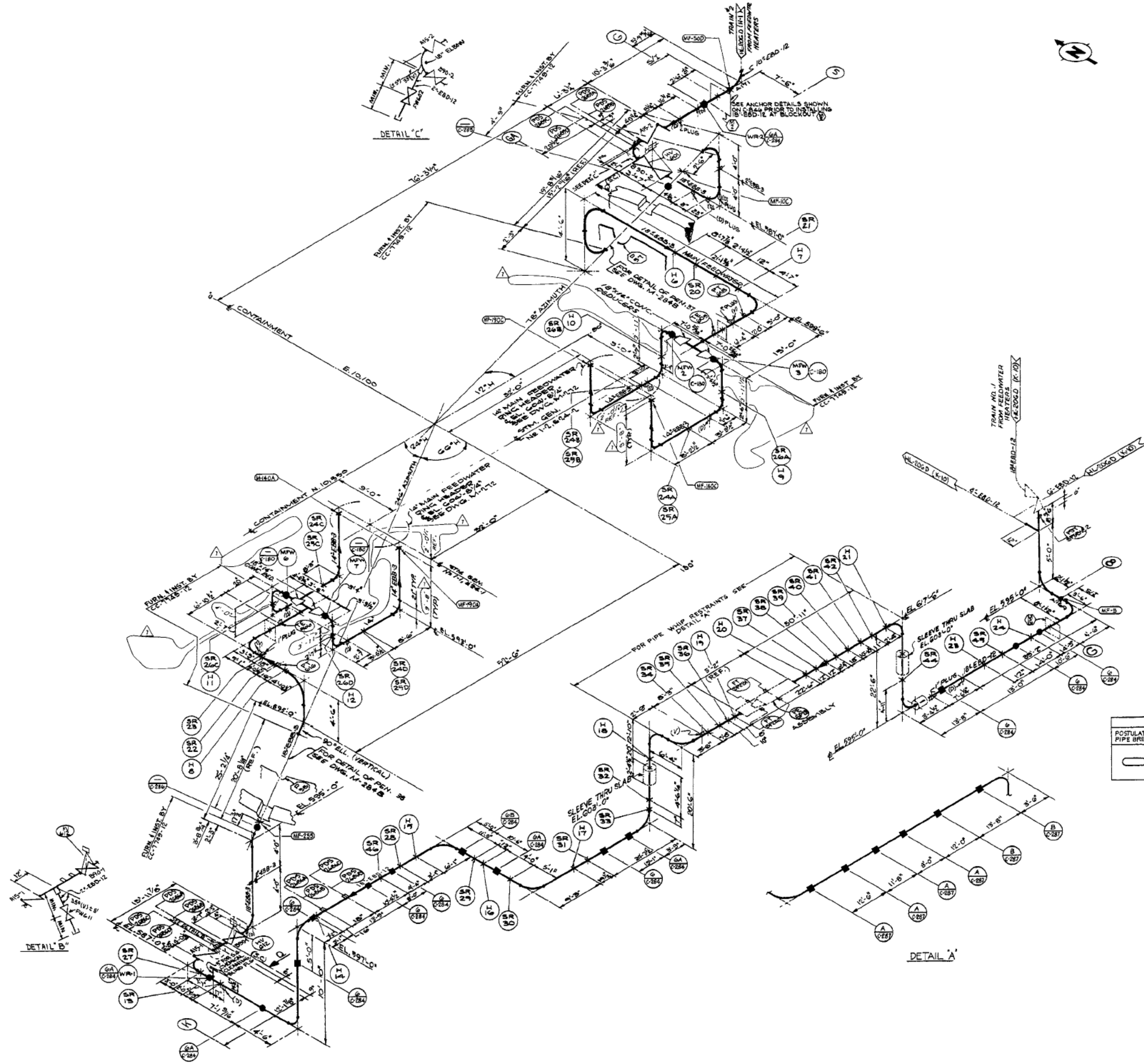
DAVIS-BESSE NUCLEAR POWER STATION

MAIN STEAM SYSTEM
MAIN STEAM CROSSOVER
HL-203J
FIGURE 3.6-14

REVISION 29

DECEMBER 2012

DB 08-14-12 DFN=/J/RASDGN/HL203J.DGN/TIF



NOTES:

1. THIS DRAWING CONTAINS INFORMATION PREVIOUSLY SHOWN ON ITT GRINNELL DRAWING NO. PM-207C.
2. THIS DRAWING IS ISSUED FOR HANGER LOCATIONS ONLY AND DOES NOT NECESSARILY DEPICT THE EXACT ROUTING OF THE PIPING. FOR CORRECT PIPE ROUTING REFER TO DRAWING NO. 7749-M-207C.
3. AS APPLICABLE, THE PIPE SUPPORTS SHOWN ON THIS DRAWING WERE COVERED UNDER THE FOLLOWING SPECIFICATIONS:

   CONSTRUCTION PHASE
   7749-M-190, PART II-FABRICATION
   7749-M-453-INSTALLATION

   OPERATIONAL PHASE
   12501-M-450N OR Q-FABRICATION AND INSTALLATION
4. THE PIPE SUPPORT LOCATIONS SHOWN ON THIS DRAWING REFLECT SOME OR ALL OF THE FOLLOWING INFORMATION (SEE NOTE 5 FOR CATEGORY OF PIPING):
   (a) AS-BUILT LOCATIONS OBTAINED FROM THE IE BULLETIN 79-14 INSPECTION-ALL SEISMIC CLASS I PIPING (BOTH Q-LISTED AND NON-Q).
   (b) DESIGN LOCATIONS OBTAINED FROM THE STRESS ANALYSIS-NON-SEISMIC, HIGH TEMPERATURE PIPING.
   (c) APPROXIMATE LOCATIONS OBTAINED FROM THE ITT GRINNELL FABRICATION/INSTALLATION DRAWING-NON-SEISMIC, LOW TEMPERATURE PIPING.
5. S/I ⊣ INDICATES THE LIMIT OF SEISMIC CLASS I PIPING
   Q ⊣ INDICATES THE LIMIT OF Q-LISTED PIPING.
6. COLUMN LINE AND PIPE ELEVATIONS ARE FOR REFERENCE ONLY.
7. MAIN FEEDWATER WHIP RESTRAINTS NO. MFW1, MFW4, MFW5, AND MFW8 HAVE BEEN REMOVED UNDER ECP 12-0063. 7

SYMBOL LEGEND				
POSTULATED PIPE BREAK	PIPE WHIP RESTRAINT	ANCHOR	ABANDONED WHIP RESTRAINT	

DAVIS-BESSE NUCLEAR POWER STATION

MAIN FEEDWATER SYSTEM AUXILIARY AND CONTAINMENT BUILDING

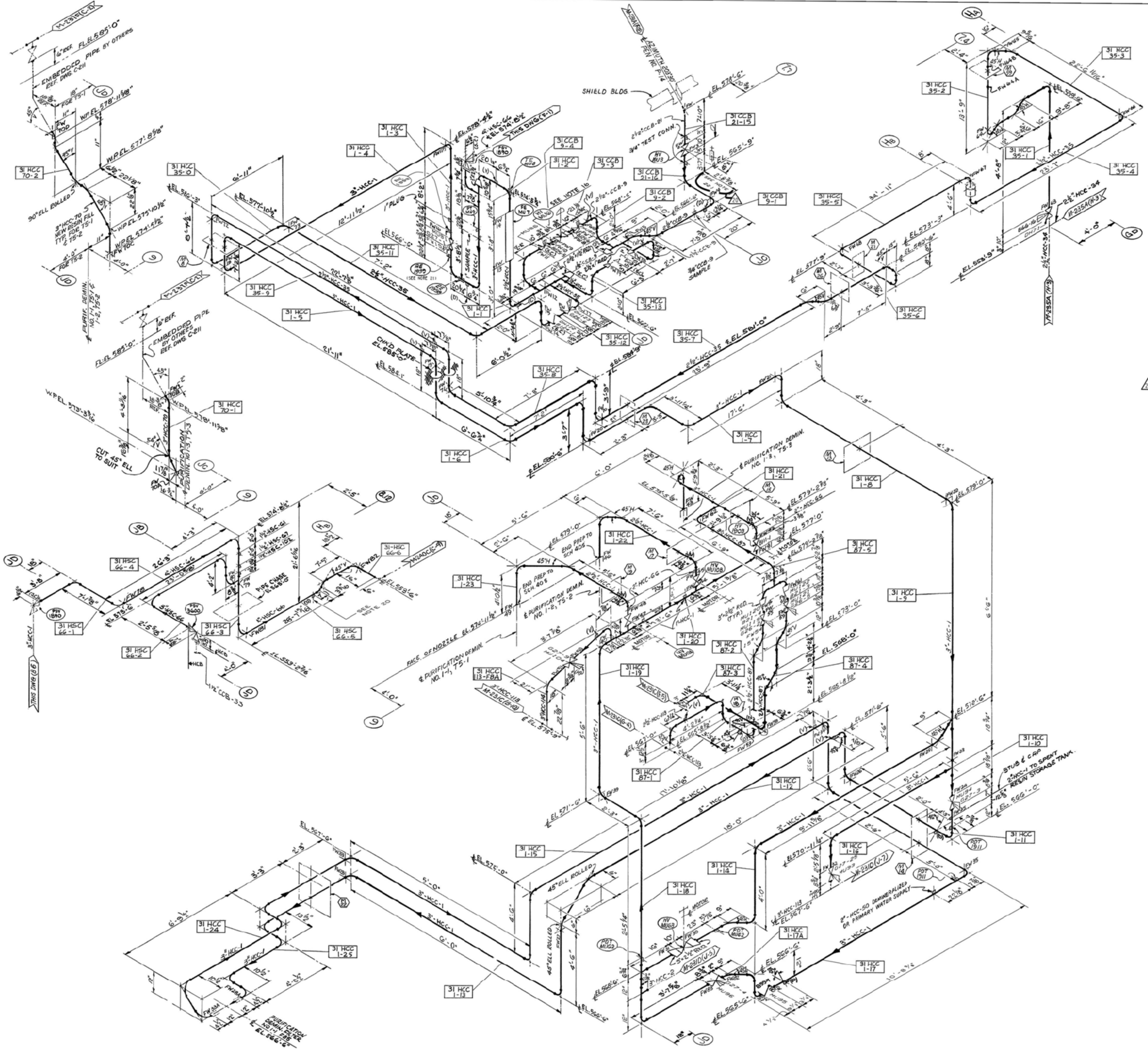
HL-207C

FIGURE 3.6-15

REVISION 30

OCTOBER 2014

DB 10-01-14 DFN=H/PLDSGN/HL207C.DGN/TIF



NOTES:

1. ALL PIPING TO BE FABRICATED AND ERECTED IN ACCORDANCE WITH CONSTRUCTION DOCUMENT 7749-12.
2. FOR MATERIAL DESCRIPTION OF PIPE, VALVES AND FITTINGS, SEE PIPING CLASS SHEETS M-601 OF SPEC. 7749-M-190.
3. FOR PIPING AND INSTRUMENT DIAGRAM SEE DWG. M-031 AND M-033.
4. FOR EQUIPMENT LIST SEE M-600.
5. FOR INSTRUMENT CONNECTION DETAIL SEE DRAWINGS M-550 AND M-551.
6. NO ALLOWANCE HAS BEEN MADE FOR WELDED JOINTS OR GASKETS AND ALL PIPING IS DIMENSIONED IN THE COLD ERECTED POSITION UNLESS OTHERWISE NOTED ON THE DRAWING.
7. FOR PIPING SYSTEMS COMPOSITE DRAWING(S) THIS AREA SEE M-260A, M-260B, M-261B, M-262B, M-267B, AND M-268A.
8. FOR MATERIAL DESCRIPTION OF INSULATION SEE SPEC. 7749-M-197 AND 7749-M-198.
9. PIPE BENDS, UNLESS OTHERWISE NOTED, HAVE A MINIMUM CENTER LINE RADIUS OF FIVE (5) TIMES THE NOMINAL PIPE DIAMETER.
10. FOR PIPING ISOMETRIC AND COMPOSITE DWG. INDEX SEE DWG. M-200.
11. FOR PIPING AND INSTRUMENT SYMBOLS SEE DWG. M-001 AND M-002.
12. FIELD WELD JOINTS WHERE SPECIFICALLY REQUIRED ARE INDICATED (FW). VENT AND DRAIN CONNECTIONS ARE INDICATED (V) AND (D). SPEC. CHANGE INDICATED (SC).
13. LIMIT OF "Q-LISTED" MATERIAL IS DENOTED Q
14. UNLESS OTHERWISE NOTED, ALL VENT CONN'S. ARE ¾" AND ALL DRAIN CONN'S. ARE 1"
15. THIS ISO. TO BE WORKED WITH DRAWINGS M-231A, M-235A, M-231C AND M-231D.
16. 2½" × 2" REDUCERS SUPPLIED WITH HV-MU6.
17. S/I DENOTES LIMITS OF SEISMIC CLASS I PIPING (OTHER THAN "Q-LISTED").
18. THIS DRAWING CONTAINS INFORMATION PREVIOUSLY SHOWN ON ITT GRINNELL DRAWING NO. PM-231B
19. DELETED (20)
20. PERMANENT SHIELDING INSTALLED ON VALVE RC158. SEE CALC 69A.
21. THE BORONOMETER HAS BEEN ABANDONED-IN-PLACE PER ECP 16-0507.
22. FOR MECHANICAL PENETRATION ROOM #1 RADIATION SHIELDING DETAILS, SEE VENDOR MANUAL NUMBER G-M-200 AND DRAWING NUMBERS G-M-200, G-M-201 SH.1, G-M-201 SH.2, G-M-202, G-M-203, G-M-204, G-M-205, G-M-206, G-M-207, G-M-208, G-M-210, G-M-211, G-M-212, G-M-213, G-M-214, G-M-215, G-M-216, G-M-217, G-M-218, G-M-219, G-M-220, G-M-221, G-M-222, G-M-223, G-M-224 AND G-M-225.

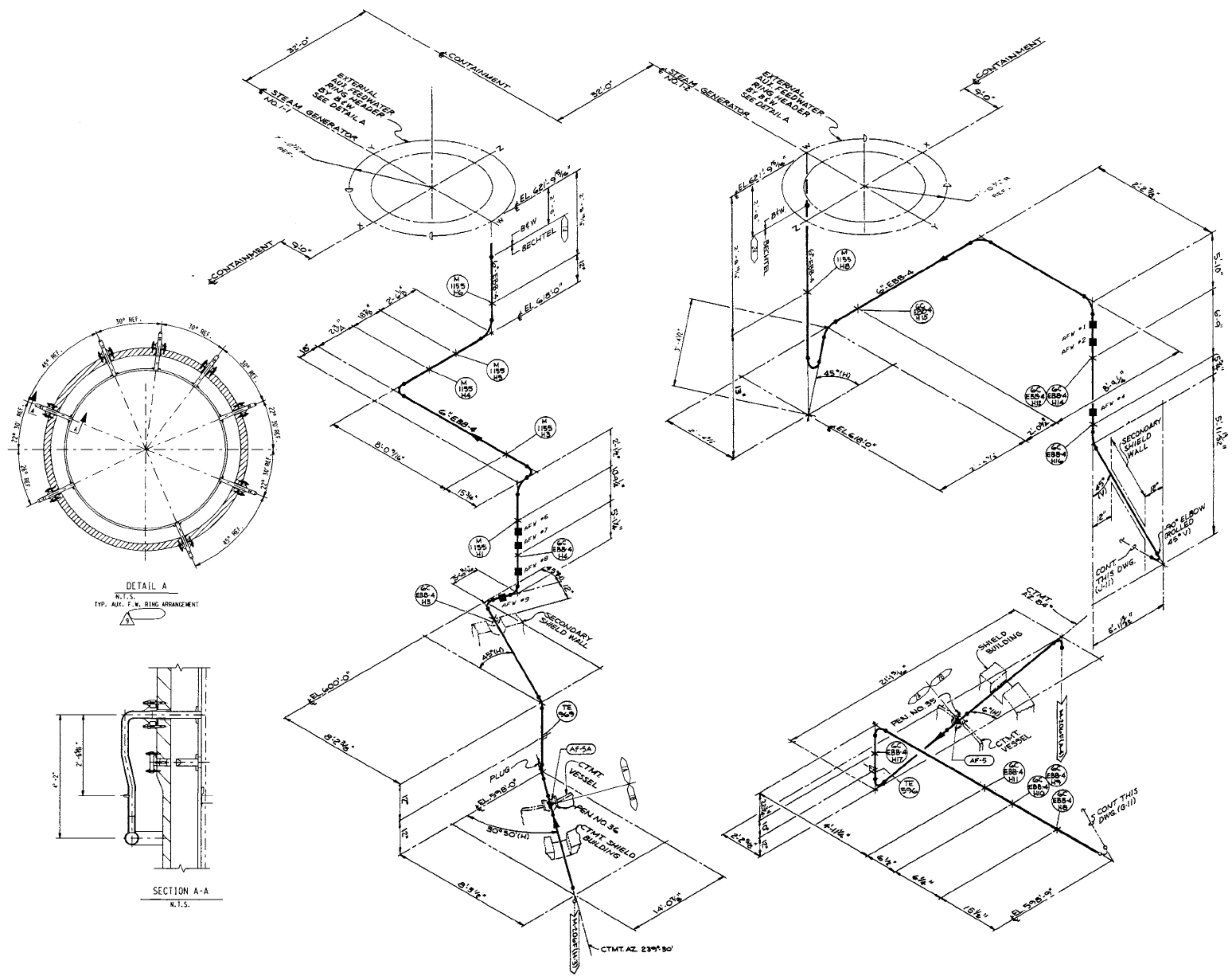
LINE NO.	NORMAL		MAX.	
	PSIG	°F	PSIG	°F
CCB-9	2,185	120	2,300	300
CCB-21	2,185	120	2,300	600
HCC-1 & 35	150	120	150	200
HSC-66	15	250	150	300
HCC-70 & 87	150	120	150	200
HCC-114	150	120	150	200

DAVIS-BESSE NUCLEAR POWER STATION

MAKE-UP AND PURIFICATION SYSTEM
AUXILIARY BUILDING
M-231B
FIGURE 3.6-16

REVISION 33
SEPTEMBER 2020

DB 08-14-20 DFN=H/PLDSGN/M231B.DGN/TIF



NOTES:

3. AS APPLICABLE, THE PIPE SUPPORTS SHOWN ON THIS DRAWING WERE COVERED UNDER THE FOLLOWING SPECIFICATIONS:
   CONSTRUCTION PHASE
   7749-M-190, PART II-FABRICATION
   7749-M-453-INSTALLATION
   OPERATIONAL PHASE
   12501-M-450N OR Q-FABRICATION AND INSTALLATION
4. THE PIPE SUPPORT LOCATIONS SHOWN ON THIS DRAWING REFLECT SOME OR ALL OF THE FOLLOWING INFORMATION (SEE NOTE 5 FOR CATEGORY OF PIPING):
   (a) AS-BUILT LOCATIONS OBTAINED FROM THE IE BULLETIN 79-14 INSPECTION-ALL SEISMIC CLASS I PIPING (BOTH Q-LISTED AND NON-Q).
   (b) DESIGN LOCATIONS OBTAINED FROM THE STRESS ANALYSIS-NON-SEISMIC, HIGH TEMPERATURE PIPING.
   (c) APPROXIMATE LOCATIONS OBTAINED FROM THE ITT GRINNELL FABRICATION/INSTALLATION DRAWING-NON-SEISMIC, LOW TEMPERATURE PIPING.
5. ALL PIPING SHOWN ON THIS DRAWING IS SEISMIC CLASS I.
6. PIPE BREAK LOCATIONS FOR STEAM GENERATOR RING HEADER ARE AS FOLLOWS:
   SG 1-1: AF-A20 THRU AH-H20
   SG 1-2: AF-A5 THRU AF-H5
7. COLUMN LINE AND PIPE ELEVATIONS ARE FOR REFERENCE ONLY.
8. DELETED
9. DUE TO REROUTING/REDESIGN OF THE AUXILIARY FEEDWATER (AFW) PIPING AND THE ELIMINATION OF ARBITRARY INTERMEDIATE BREAKS, NUMEROUS AFW WHIP RESTRAINTS HAVE BEEN ABANDONED OR IN THE CASE OF AFW #3 & #5, DISMANTLED. ABANDONED WHIP RESTRAINTS ARE NOTED ON THE DRAWING WITH THE EXCEPTION OF THE TWO DISMANTLED RESTRAINTS.

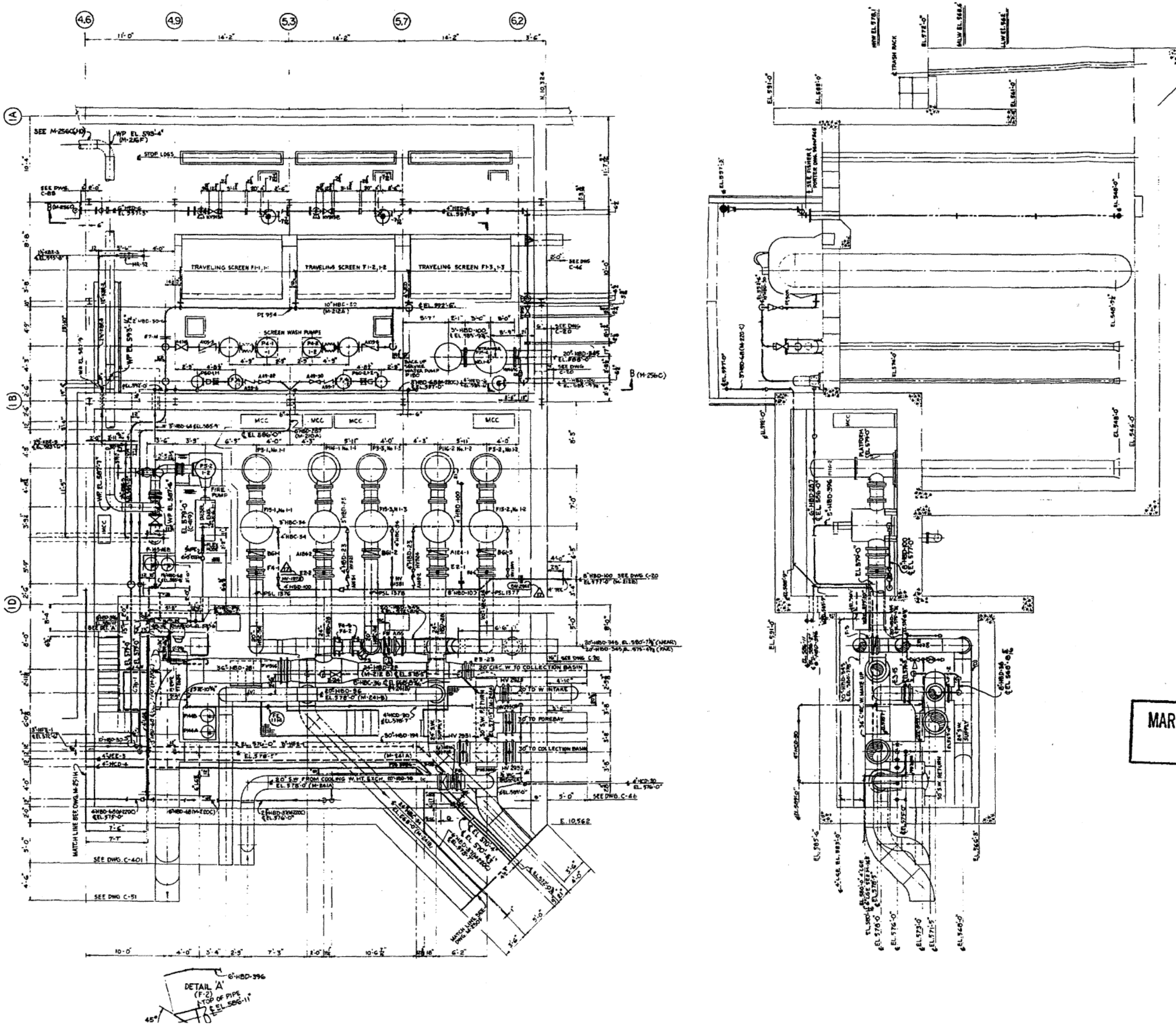
SYMBOL LEGEND				
POSTULATED PIPE BREAK	PIPE WHIP RESTRAINT	ANCHOR	ABANDONED WHIP RESTRAINT	PIPE STRESS ANALYSIS PROB.

DAVIS-BESSE NUCLEAR POWER STATION

AUXILIARY FEEDWATER SYSTEM AUXILIARY FEEDWATER PUMP DISCHARGE TO STEAM GENERATORS
HL-1155
FIGURE 3.6-17

REVISION 30
OCTOBER 2014

DB 10-08-14 DFN=H/PLDSGN/HL1155.DGN/TIF



GENERAL NOTE:

1. PIPING SYSTEM COMPOSITE DRAWINGS HAVE BEEN PREPARED TO SHOW ALL 2½" AND LARGER PIPING SYSTEMS IN EACH AREA. UNLESS OTHERWISE NOTED ON DRAWING. THESE SYSTEM LAYOUTS ARE ONLY TO BE USED AS REFERENCE TO PERFORM OTHER ENGINEERING AND CONSTRUCTION TASKS.

2. THE FOLLOWING PIPING SYSTEMS SHALL BE FABRICATED AND INSTALLED PER THIS DRAWING. NO ISOMETRIC OF THIS PIPING WILL BE AVAILABLE, UNLESS PROVIDED BY THE FABRICATOR.

SYSTEM	LINE NOS.
DEMINERALIZED WATER	4" HCD-4 (H-2)
DOMESTIC WATER	4" JEE-3 (H-2)
SUMP PUMP DISCHARGE	4" LGE (G-2)
	6" LGE (G-3)
CHLORINATION	6" HED-6 (E-3)
	4" HED-6 (C-6)
DIESEL FIRE PUMP EXHAUST	6" HBD-396 (F-3)
NEUTRALIZING TK. DISCH	3" HFE-1 (G-3)

SYSTEMS STATUS LEDGER

SYSTEM	NO.	ISSUED FAB.	ISSUED CONST.
SERVICE WATER	M-241A	10 NOV 72	10/17/72
	M-241B	29 OCT 72	10/31/72
CIRC. WATER	M-212B	1/31/72	10/24/72
FIRE PROTECTION	M-216F	4-9-73	4-9-73
WTR. TREATMENT	M-210A	2-16-73	2-16-73
SCREEN WASH & CHL.	M-212A	3-7-73	3-7-73
DOMESTIC WTR.	THIS DWG.	5-16-73	5-16-73
AUX. STEAM	M-220C	5-16-73	5-16-73
DEMINERALIZED WTR.	THIS DWG.	5-16-73	5-16-73
SUMP PUMP DISCH.	THIS DWG.	8-29-73	8-29-73
CHLORINATION	THIS DWG.	8-29-73	8-29-73

MARGINAL QUALITY DOCUMENT
BEST COPY AVAILABLE

DAVIS-BESSE NUCLEAR POWER STATION

INTAKE STRUCTURE

FIGURE 3.6-18

REVISION 24
JUNE 2004

[redacted]R POWER STATION

ILDING
PLANS
3
.6-20

N 30
2014

I13.DGN/TIF

[redacted]

SECTION 'B-B'
LOOKING NORTH

DAVIS-BESSE NUCLEAR POWER STATION
TURBINE BUILDING
SECTION B-B
(M-114)
FIGURE 3.6-21

REVISION 0
JULY 1982

[redacted]

SECTION 'C-C
LOOKING EAST

REFERENCE DWG.
A-500 PERSONNEL SHOP FACILITY

DAVIS-BESSE NUCLEAR POWER STATION

TURBINE BUILDING SECTION C-C

(M-115)
FIGURE 3.6-22
REVISION 29
DECEMBER 2012

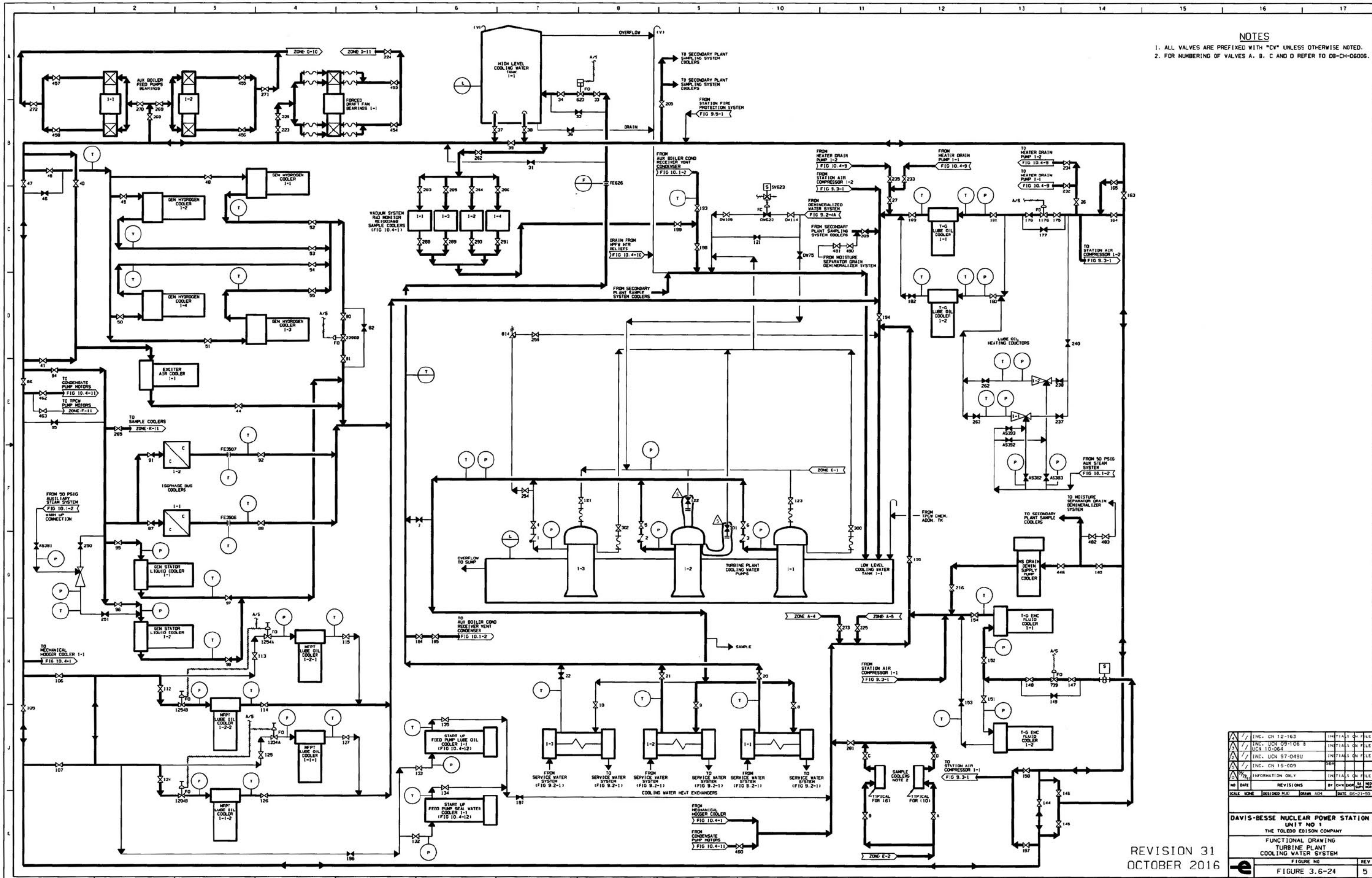
DB 04-02-12 DFN= H/PLDSGN/M115.DGN/TIF

[redacted]

DAVIS-BESSE NUCLEAR POWER STATION
TURBINE BUILDING PLAN EL.585'-0"
M-112
FIGURE 3.6-23

REVISION 31
OCTOBER 2016

DB 09-06-16 DFN=H/PLDSGN/M112.DGN/TIF



REVISION 31
OCTOBER 2016

NO	DATE	REVISIONS	BY	CH'K	ENGR	SA SUPV	NED MGR
4	/ /	INC. CN 12-163	INITIALS ON FILE				
3	/ /	INC. UCN 09-106 & UCN 10-064	INITIALS ON FILE				
2	/ /	INC. UCN 97-049U	INITIALS ON FILE				
5	/ /	INC. CN 15-009	SBW				
0	09/20/91	INFORMATION ONLY	INITIALS ON FILE				

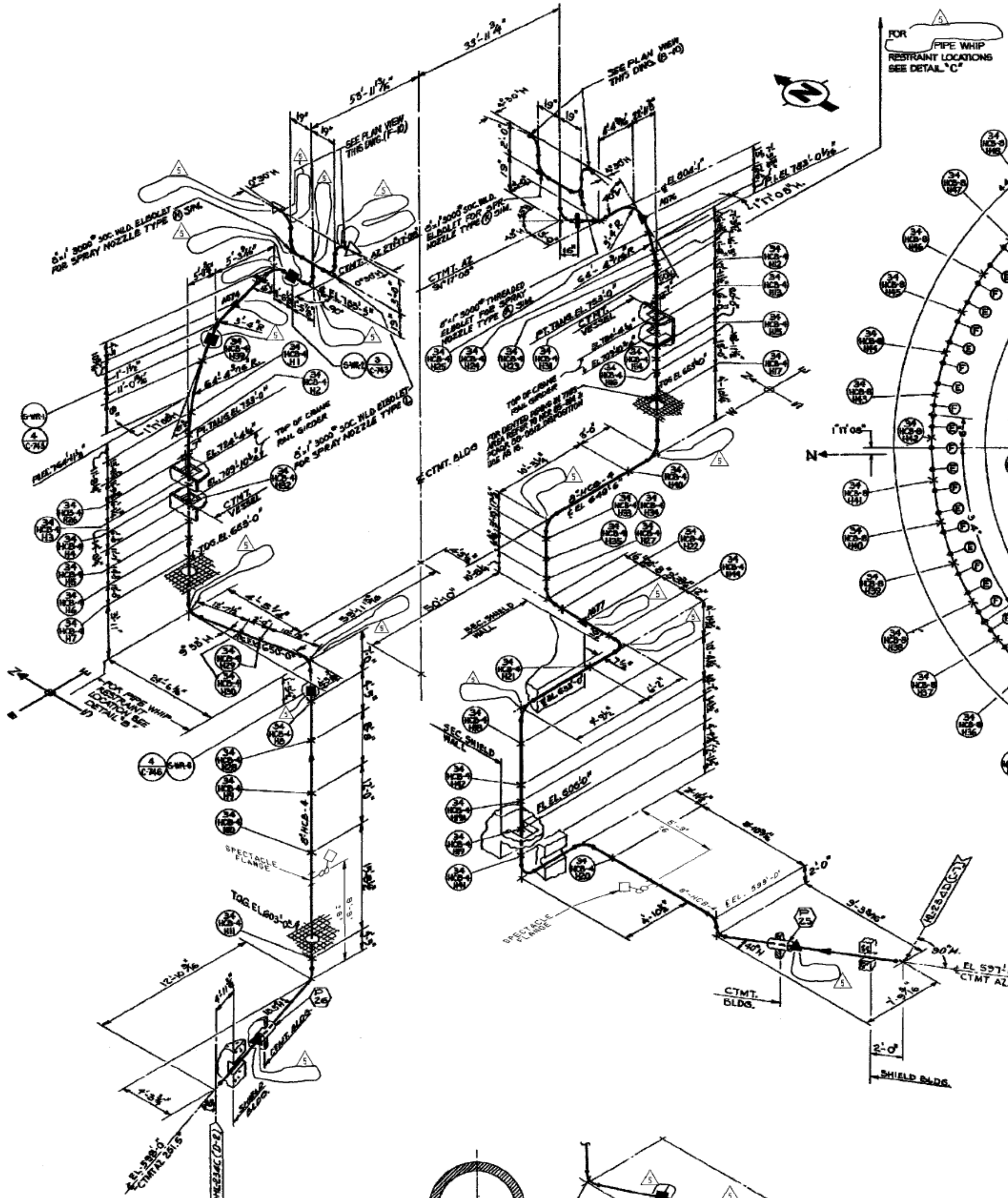
SCALE NONE | DESIGNED MJD | DRAWN ACH | DATE 06-21-90

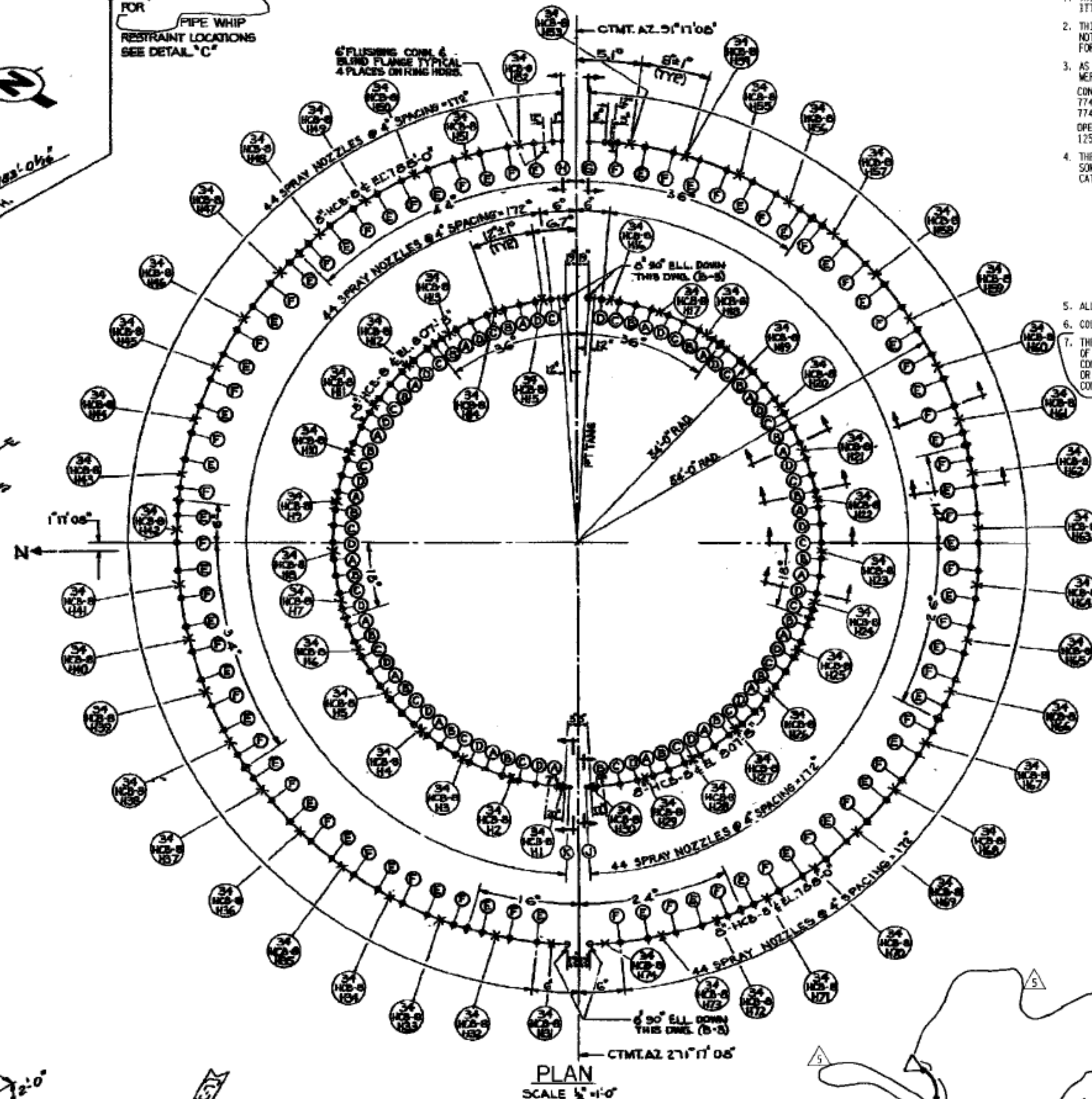
DAVIS-BESSE NUCLEAR POWER STATION
UNIT NO 1
THE TOLEDO EDISON COMPANY

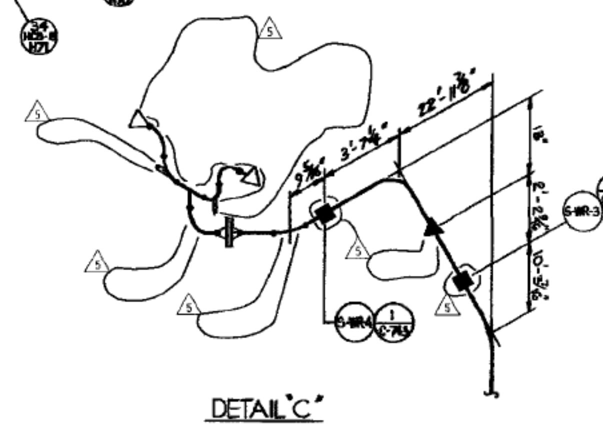
FUNCTIONAL DRAWING
TURBINE PLANT
COOLING WATER SYSTEM

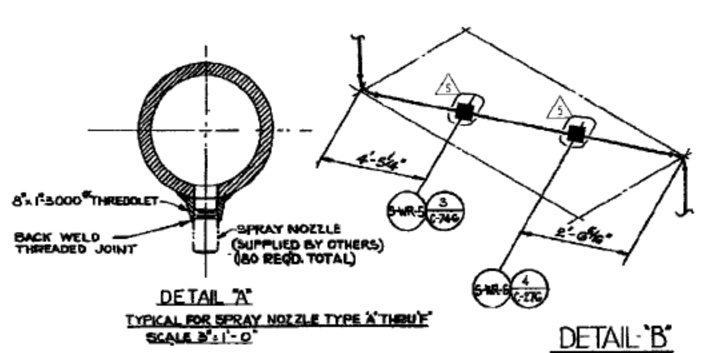
FIGURE NO: FIGURE 3.6-24 | REV: 5

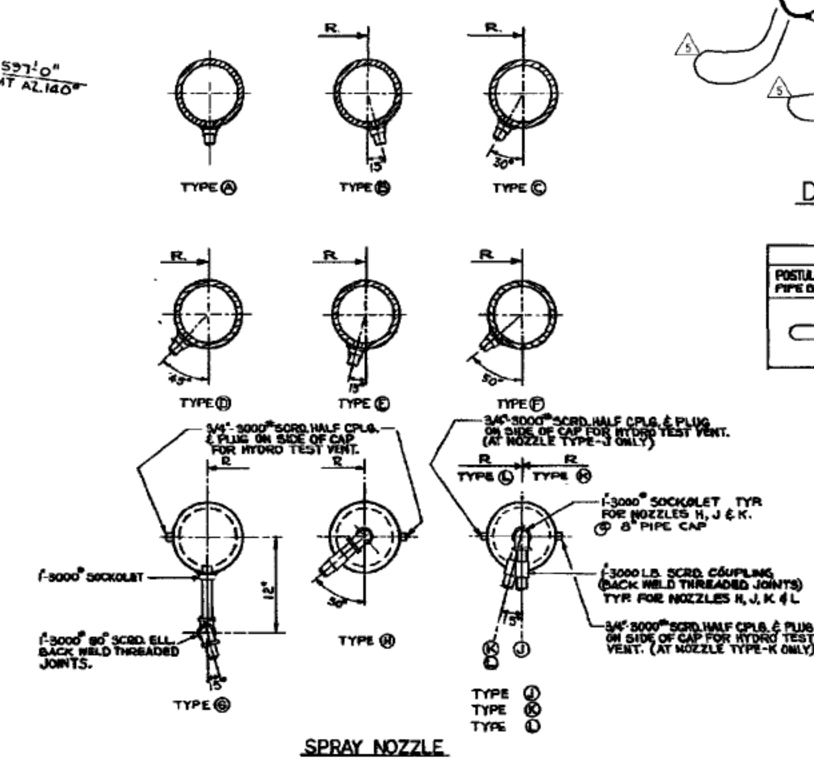
DFN-6/USAR/UFIG3624.DGN
DB 03-17-16











SYMBOL LEGEND			
POSTULATED PIPE BREAK	PIPE WHIP RESTRAINT	ANCHOR	ABANDONED WHIP RESTRAINT

NOTES:

1. THIS DRAWING CONTAINS INFORMATION PREVIOUSLY SHOWN ON ITT GRINNELL DRAWING NO. PM-234B.
2. THIS DRAWING IS ISSUED FOR HANGER LOCATIONS ONLY AND DOES NOT NECESSARILY DEPICT THE EXACT ROUTING OF THE PIPING. FOR CORRECT PIPE ROUTING REFER TO DRAWING NO. 7749-M-234B.
3. AS APPLICABLE, THE PIPE SUPPORTS SHOWN ON THIS DRAWING WERE COVERED UNDER THE FOLLOWING SPECIFICATIONS:

   CONSTRUCTION PHASE
   7749-M-190, PART II-FABRICATION
   7749-M-453-INSTALLATION

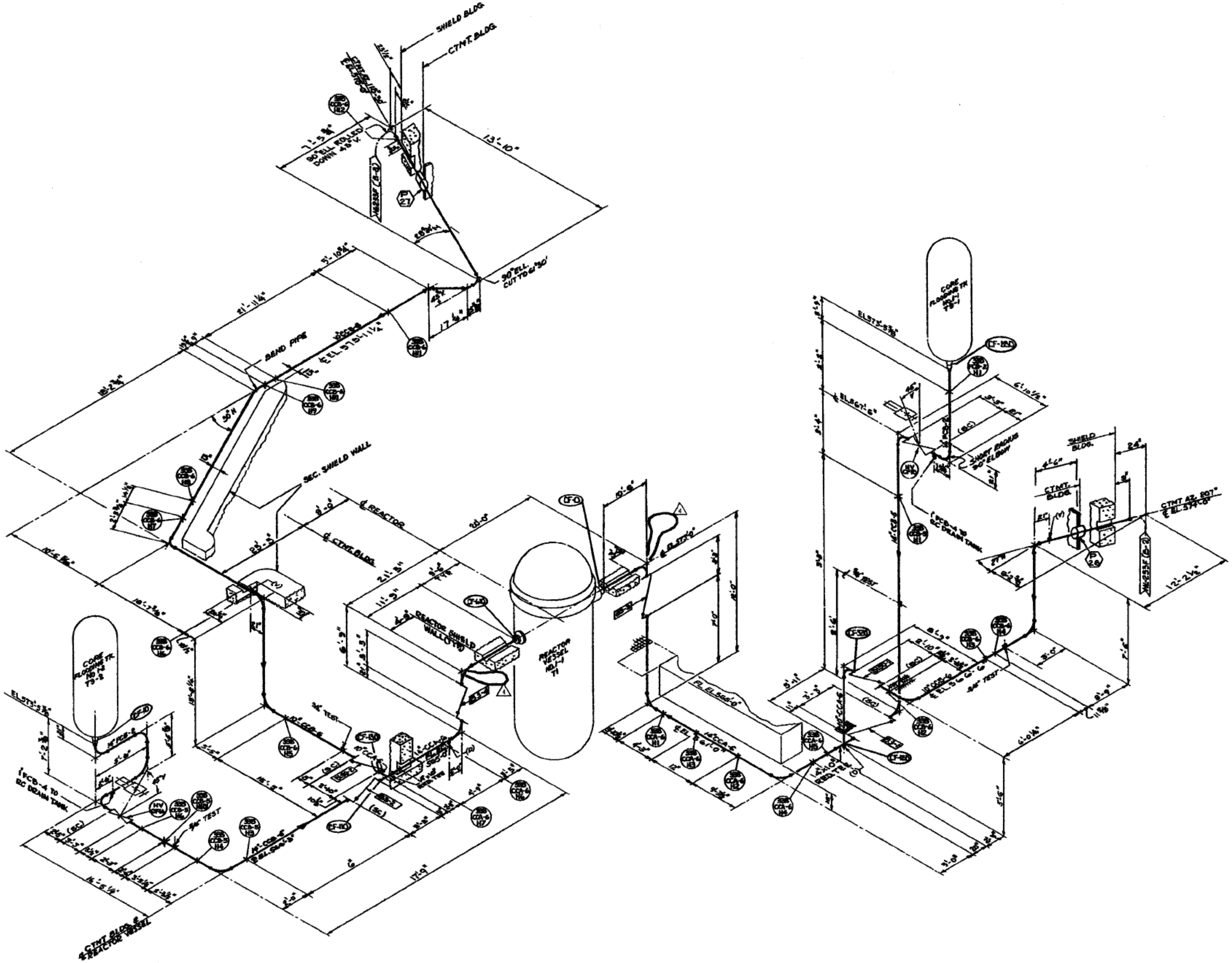
   OPERATIONAL PHASE
   12501-M-450N OR Q-FABRICATION AND INSTALLATION
4. THE PIPE SUPPORT LOCATIONS SHOWN ON THIS DRAWING REFLECT SOME OR ALL OF THE FOLLOWING INFORMATION (SEE NOTE 5 FOR CATEGORY OF PIPING):
   (a) AS-BUILT LOCATIONS OBTAINED FROM THE IE BULLETIN 79-14 INSPECTION-ALL SEISMIC CLASS I PIPING (BOTH Q-LISTED AND NON-Q).
   (b) DESIGN LOCATIONS OBTAINED FROM THE STRESS ANALYSIS-NON-SEISMIC, HIGH TEMPERATURE PIPING.
   (c) APPROXIMATE LOCATIONS OBTAINED FROM THE ITT GRINNELL FABRICATION/INSTALLATION DRAWING-NON-SEISMIC, LOW TEMPERATURE PIPING.
5. ALL PIPING SHOWN ON THIS DRAWING IS SEISMIC CLASS I.
6. COLUMN LINE AND PIPE ELEVATIONS ARE FOR REFERENCE ONLY.
7. THE CONTAINMENT SPRAY SYSTEM IS EXEMPT FROM THE POSTULATION OF HIGH-ENERGY PIPE BREAKS, AS IT ONLY SEES HIGH-ENERGY CONDITIONS IN RESPONSE TO A LOCA EVENT. TWO CONCURRENT, OR CLOSELY SPACED IN TIME, PIPE RUPTURE EVENTS ARE NOT CONSIDERED CREDIBLE. (REF. ECP 06-0016-00)

DAVIS-BESSE NUCLEAR POWER STATION

CONTAINMENT SPRAY SYSTEM
CONTAINMENT BUILDING
HL-234B
FIGURE 3.6-25

REVISION 26
JUNE 2008

DB 03-12-98 DFN-H/PLDSGN/HL231B.DGN/TIF



NOTES:

1. THIS DRAWING PARTIAL CONTAINS INFORMATION PREVIOUSLY SHOWN ON ITT GRINNELL DRAWING NO. PM-234A.
2. THIS DRAWING IS ISSUED FOR HANGER LOCATIONS ONLY AND DOES NOT NECESSARILY DEPICT THE EXACT ROUTING OF THE PIPING. FOR CORRECT PIPE ROUTING REFER TO DRAWING NO. 7749-M-234A.
3. AS APPLICABLE, THE PIPE SUPPORTS SHOWN ON THIS DRAWING WERE COVERED UNDER THE FOLLOWING SPECIFICATIONS:

   CONSTRUCTION PHASE
   7749-M-190, PART II-FABRICATION
   7749-M-453-INSTALLATION

   OPERATIONAL PHASE
   12501-M-450N OR Q-FABRICATION AND INSTALLATION
4. THE PIPE SUPPORT LOCATIONS SHOWN ON THIS DRAWING REFLECT SOME OR ALL OF THE FOLLOWING INFORMATION (SEE NOTE 5 FOR CATEGORY OF PIPING):
   (a) AS-BUILT LOCATIONS OBTAINED FROM THE IE BULLETIN 79-14 INSPECTION-ALL SEISMIC CLASS I PIPING (BOTH Q-LISTED AND NON-Q).
   (b) DESIGN LOCATIONS OBTAINED FROM THE STRESS ANALYSIS-NON-SEISMIC, HIGH TEMPERATURE PIPING.
   (c) APPROXIMATE LOCATIONS OBTAINED FROM THE ITT GRINNELL FABRICATION/INSTALLATION DRAWING-NON-SEISMIC, LOW TEMPERATURE PIPING.
5. ALL PIPING SHOWN ON THIS DRAWING IS SEISMIC CLASS I.
6. COLUMN LINE AND PIPE ELEVATIONS ARE FOR REFERENCE ONLY.

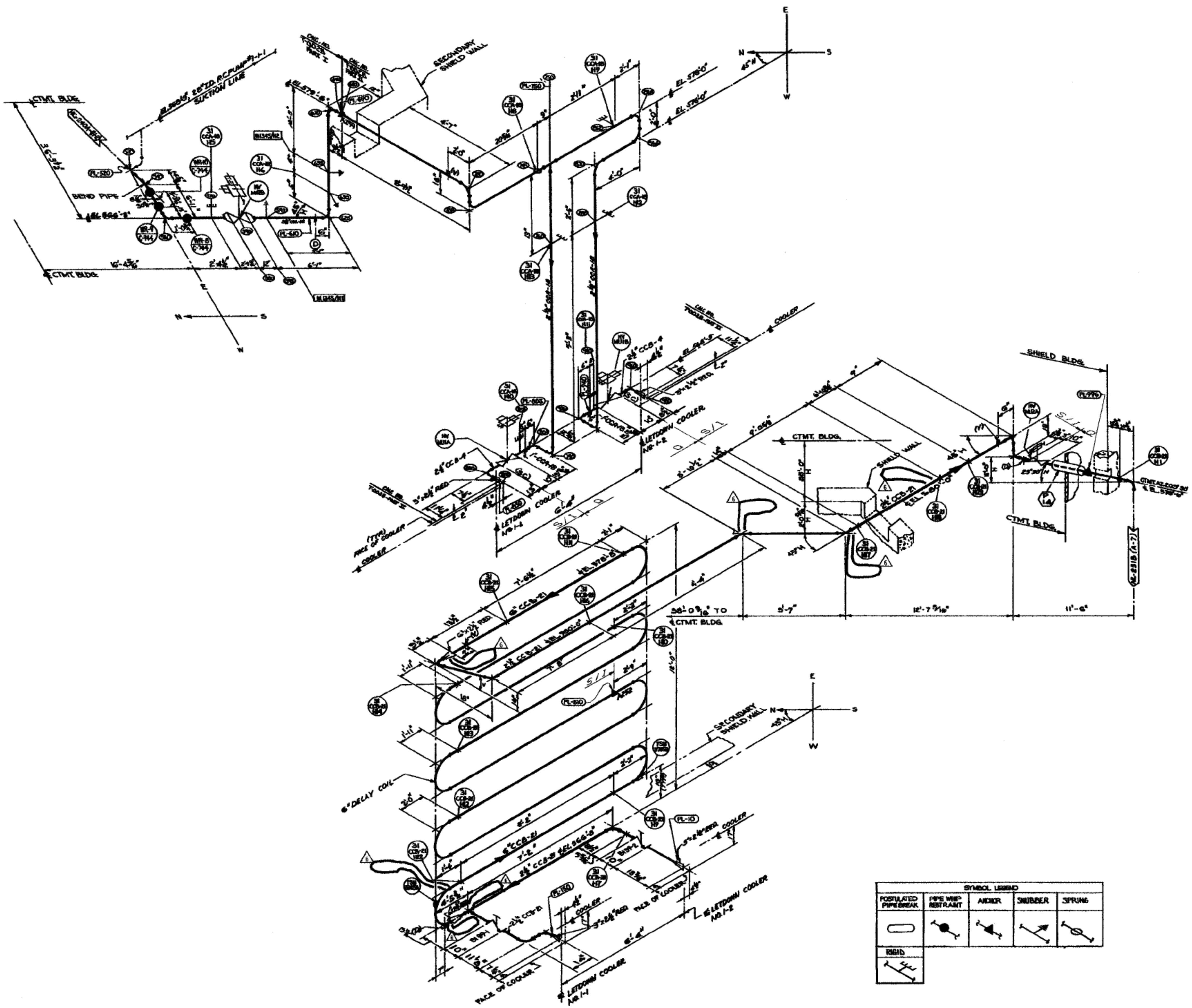
DAVIS-BESSE NUCLEAR POWER STATION

LOW PRESSURE INJECTION
CORE FLOODING SYSTEM
HL-234A
FIGURE 3.6-26

SYMBOL LEGEND				
POSTULATED PIPE BREAK	PIPE WHIP RESTRAINT	ANCHOR	WHIP REST /PIPE SUPP	

REVISION 26
JUNE 2008

DB 03-12-08 DFN=H/PLDSGN/HL234A.DGN/TIF

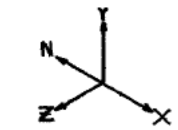


NOTES:

1. THIS DRAWING CONTAINS INFORMATION PREVIOUSLY SHOWN ON ITT GRINNELL DRAWING NO. PM-231A.
2. THIS DRAWING IS ISSUED FOR HANGER LOCATIONS ONLY AND DOES NOT NECESSARILY DEPICT THE EXACT ROUTING OF THE PIPING. FOR CORRECT PIPE ROUTING REFER TO DRAWING NO. 7749-M-231A.
3. AS APPLICABLE, THE PIPE SUPPORTS SHOWN ON THIS DRAWING WERE COVERED UNDER THE FOLLOWING SPECIFICATIONS:

   CONSTRUCTION PHASE
   7749-M-190, PART II- FABRICATION
   7749-M-453 - INSTALLATION

   OPERATIONAL PHASE
   12501-M-450N or Q - FABRICATION AND INSTALLATION
4. THE PIPE SUPPORT LOCATIONS SHOWN ON THIS DRAWING REFLECT SOME OR ALL OF THE FOLLOWING INFORMATION (SEE NOTE 5 FOR CATEGORY OF PIPING):
   (a) AS-BUILT LOCATIONS OBTAINED FROM THE IE BULLETIN 79-14 INSPECTION - ALL SEISMIC CLASS I PIPING (BOTH Q- LISTED AND NON-Q).
   (b) DESIGN LOCATIONS OBTAINED FROM THE STRESS ANALYSIS - NON-SEISMIC, HIGH TEMPERATURE PIPING.
   (c) APPROXIMATE LOCATIONS OBTAINED FROM THE ITT GRINNELL FABRICATION/INSTALLATION DRAWING-NON-SEISMIC, LOW TEMPERATURE PIPING.
5. S/I INDICATES THE LIMIT OF SEISMIC CLASS I PIPING. Q INDICATES THE LIMIT OF Q-LISTED PIPING.
6. COLUMN LINE AND PIPE ELEVATIONS ARE FOR REFERENCE ONLY.
7. ◯ : DENOTES A PIPE STRESS ANALYSIS CALCULATION DATA POINT.
8. PIPE STRESS CALCULATIONS REFERENCED ON THIS ISOMETRIC: T-002B-PART I, T-002B-PART II.
9. COORDINATE SYSTEM IS AS FOLLOWS:

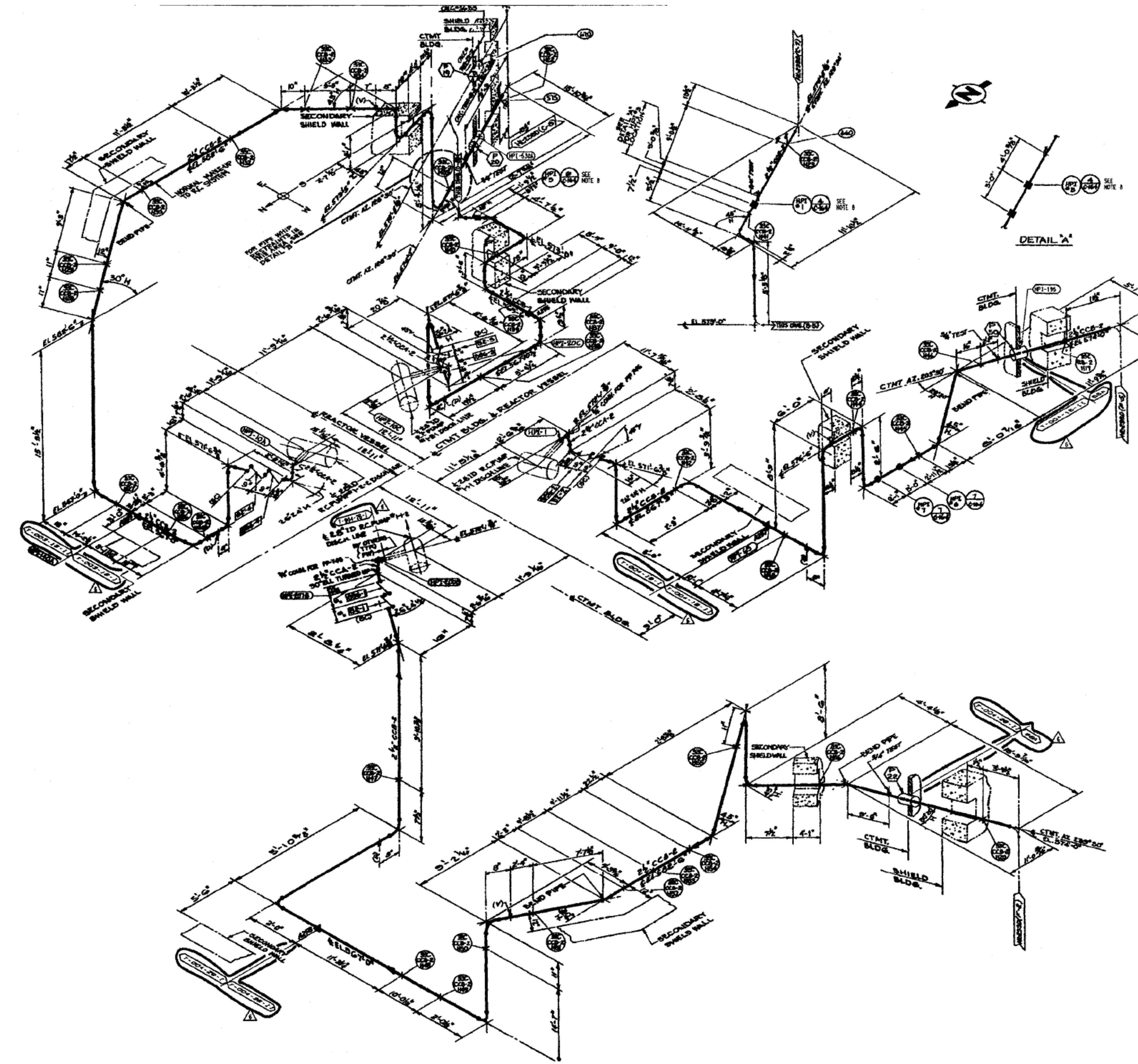


DAVIS-BESSE NUCLEAR POWER STATION

PURIFICATION R.C. LETDOWN SYSTEM
CONTAINMENT BUILDING
HL-231A
FIGURE 3.6-27

REVISION 26

JUNE 2008



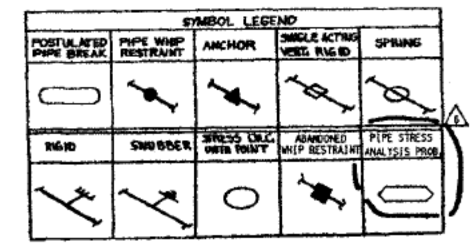
DETAIL "A"

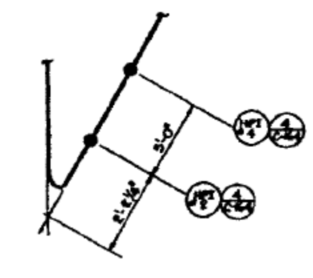
NOTES:

1. THIS DRAWING PARTIAL CONTAINS INFORMATION PREVIOUSLY SHOWN ON ITT GRINNELL DRAWING NO. PW-233E.
2. THIS DRAWING IS ISSUED FOR HANGER LOCATIONS ONLY AND DOES NOT NECESSARILY DEPICT THE EXACT ROUTING OF THE PIPING. FOR CORRECT PIPE ROUTING REFER TO DRAWING NO. 7749-M-233E.
3. AS APPLICABLE, THE PIPE SUPPORTS SHOWN ON THIS DRAWING WERE COVERED UNDER THE FOLLOWING SPECIFICATIONS:

   CONSTRUCTION PHASE
   7749-M-190, PART II-FABRICATION
   7749-M-453-INSTALLATION

   OPERATIONAL PHASE
   12501-M-450N OR Q-FABRICATION AND INSTALLATION
4. THE PIPE SUPPORT LOCATIONS SHOWN ON THIS DRAWING REFLECT SOME OR ALL OF THE FOLLOWING INFORMATION (SEE NOTE 5 FOR CATEGORY OF PIPING):
   (a) AS-BUILT LOCATIONS OBTAINED FROM THE IE BULLETIN 79-14 INSPECTION-ALL SEISMIC CLASS I PIPING (BOTH Q-LISTED AND NON-Q).
   (b) DESIGN LOCATIONS OBTAINED FROM THE STRESS ANALYSIS-NON-SEISMIC, HIGH TEMPERATURE PIPING.
   (c) APPROXIMATE LOCATIONS OBTAINED FROM THE ITT GRINNELL FABRICATION/INSTALLATION DRAWING-NON-SEISMIC, LOW TEMPERATURE PIPING.
5. ALL PIPING SHOWN ON THIS DRAWING IS SEISMIC CLASS I & Q LISTED.
6. COLUMN LINE AND PIPE ELEVATIONS ARE FOR REFERENCE ONLY.
7. STRESS ANALYSIS CALCULATIONS LOCATED ON THIS DRAWING:
   T-003-1B 56B2
   T-003-2B 56B3
8. ABANDONED WHIP RESTRAINTS HIGH PRESSURE INJECTION (HPI) #1, #5 AND #6 NO LONGER PROVIDE A WHIP RESTRAINT FUNCTION, HOWEVER, THEY CONTINUE TO PROVIDE A VERTICAL AND LATERAL PIPE SUPPORT FUNCTION. (REF. ECP 06-0016-00)





DETAIL "B"

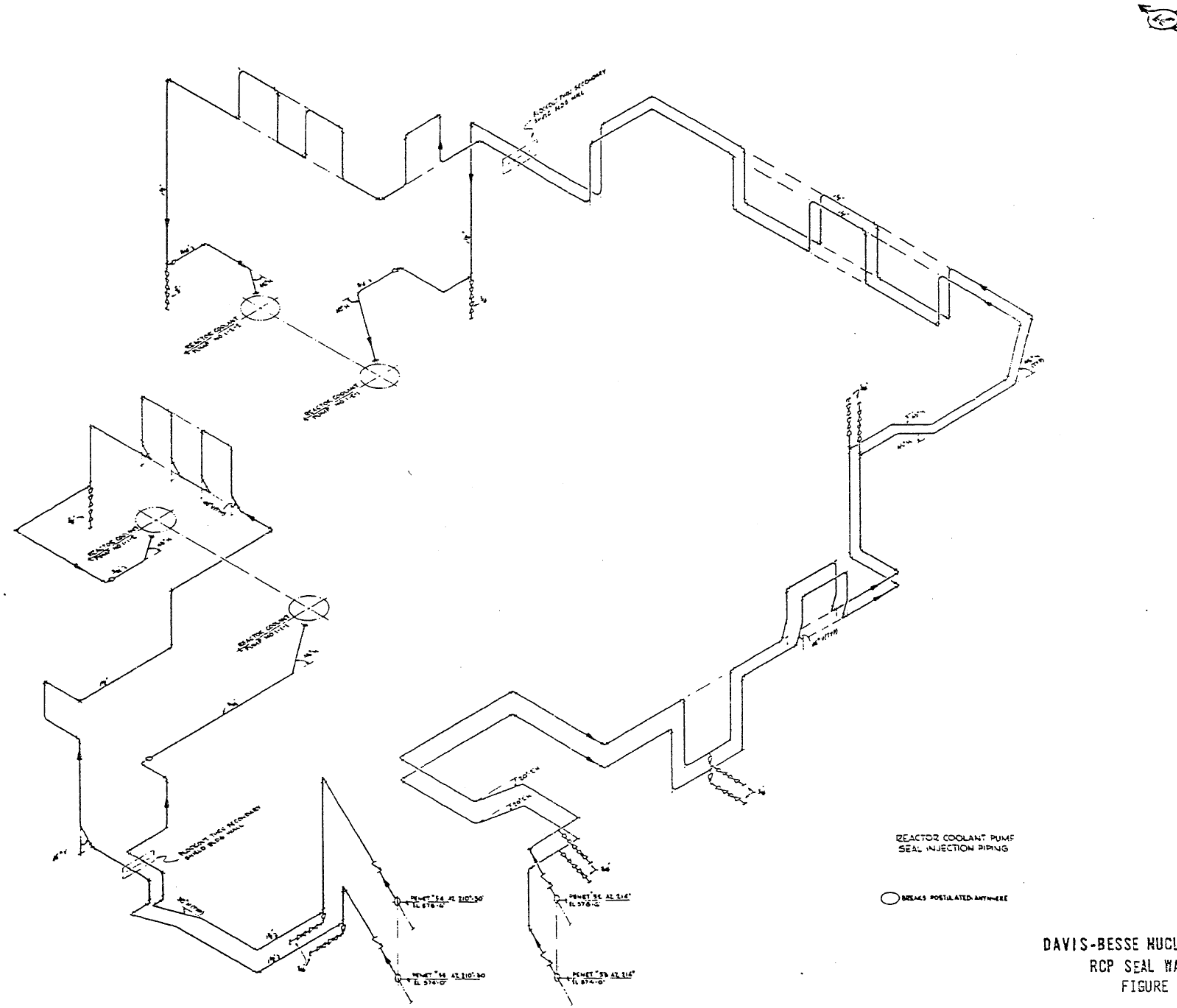
DAVIS-BESSE NUCLEAR POWER STATION

HIGH PRESSURE INJECTION SYSTEM-
CONTAINMENT BUILDING
HL-233E
FIGURE 3.6-28

REVISION 27

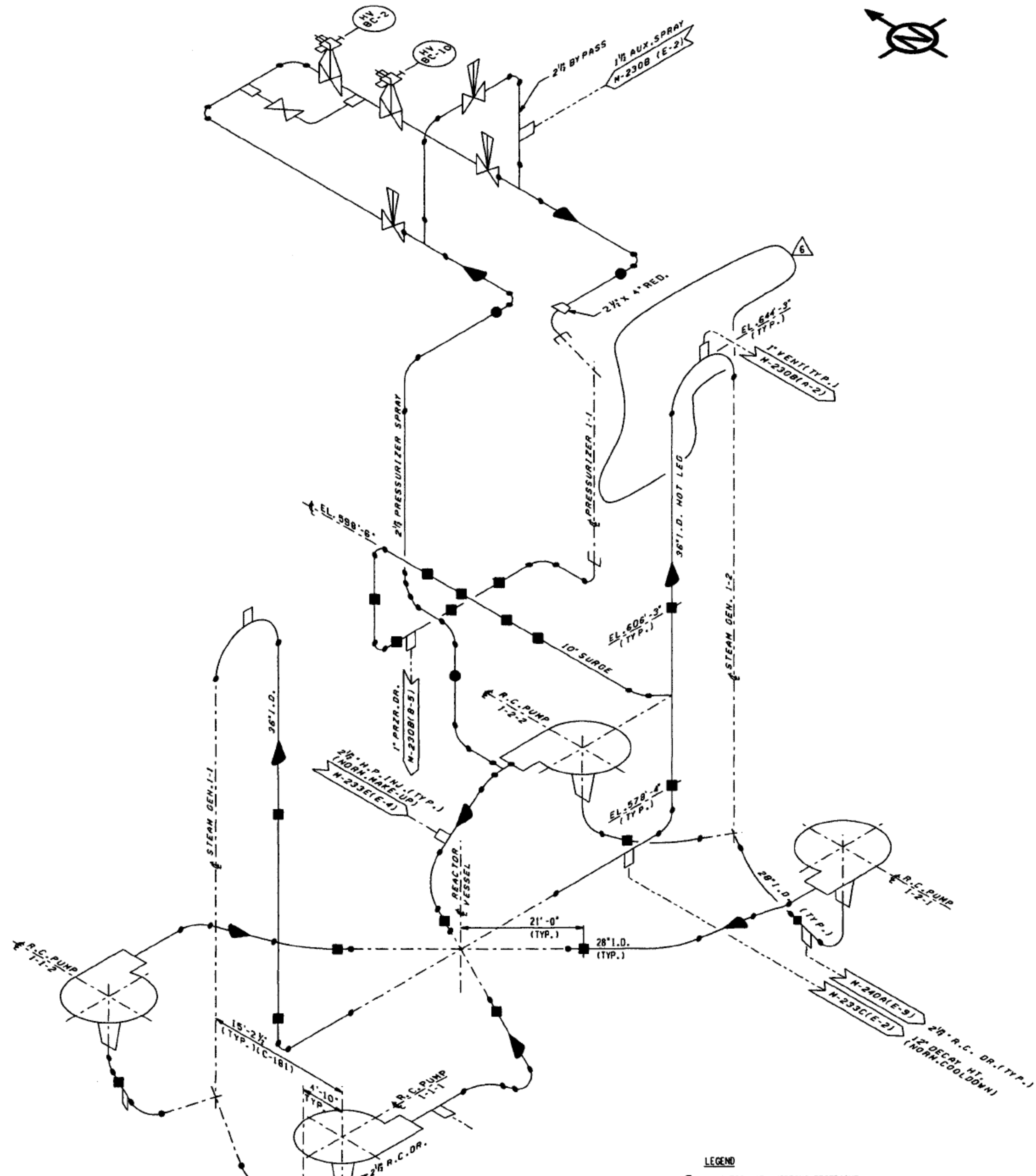
JUNE 2010

DB 03-22-10 DFN=H/PLDSGN/HL233E.DGN/TIF



DAVIS-BESSE NUCLEAR POWER STATION
RCP SEAL WATER SUPPLY
FIGURE 3.6-29

REVISION 0
JULY 1982



LEGEND

● LOCATION OF WHIPPING RESTRAINT.

■ ABANDONED WHIP RESTRAINTS.

NOTES:

1. PIPE BREAKS ARE POSTULATED TO OCCUR AT ANY LOCATION FOR THE PRESSURIZER SURGE LINE, PRESSURIZER SPRAY LINE, AND PRESSURIZER SPRAY BYPASS LINE.
2. NO BREAKS ARE POSTULATED IN THE PRIMARY LOOP PIPING (i.e., HOT AND COLD LEGS) AS THESE PIPES UTILIZE A "LEAK-BEFORE-BREAK" METHODOLOGY (SEE SECTION 3.6.2.2.1).
3. ALL PRIMARY LOOP WHIP RESTRAINTS ARE ABANDONED. IN ADDITION TO THE EIGHTEEN RESTRAINTS SHOWN ON THIS DRAWING, THERE ARE THREE LOCATED ON EACH REACTOR COOLANT PUMP AND ONE NEAR EACH REACTOR VESSEL NOZZLES WITHIN THE PRIMARY SHIELD WALL, FOR A TOTAL OF THIRTY-SIX ORIGINAL RESTRAINTS. THE TWO WIRE ROPE WHIP RESTRAINTS ON EACH RCP MAY BE PERMANENTLY REMOVED TO FACILITATE MAINTENANCE OF THE PUMPS, IF DESIRED.

THIS DWG. WAS REDRAWN ON CADD AND SUPERSEDES REV.0

NO.	DATE	REVISIONS	BY	CH'K	ENGR.	ENGR. SUPV.	GEN. SUPV.
5		INC. CN 11-017	INITIALS ON FILE				
4		INC. CN 08-069	INITIALS ON FILE				
3		INC. UNC 05-015U	INITIALS ON FILE				
2		INC. UCN 97-127U	INITIALS ON FILE				
6		INC. CN 12-166	SBW				

SCALE NONE | DESIGNED | DRAWN JOR | DATE 1-25-95

DAVIS-BESSE NUCLEAR POWER STATION
UNIT NO. 1
THE TOLEDO EDISON COMPANY

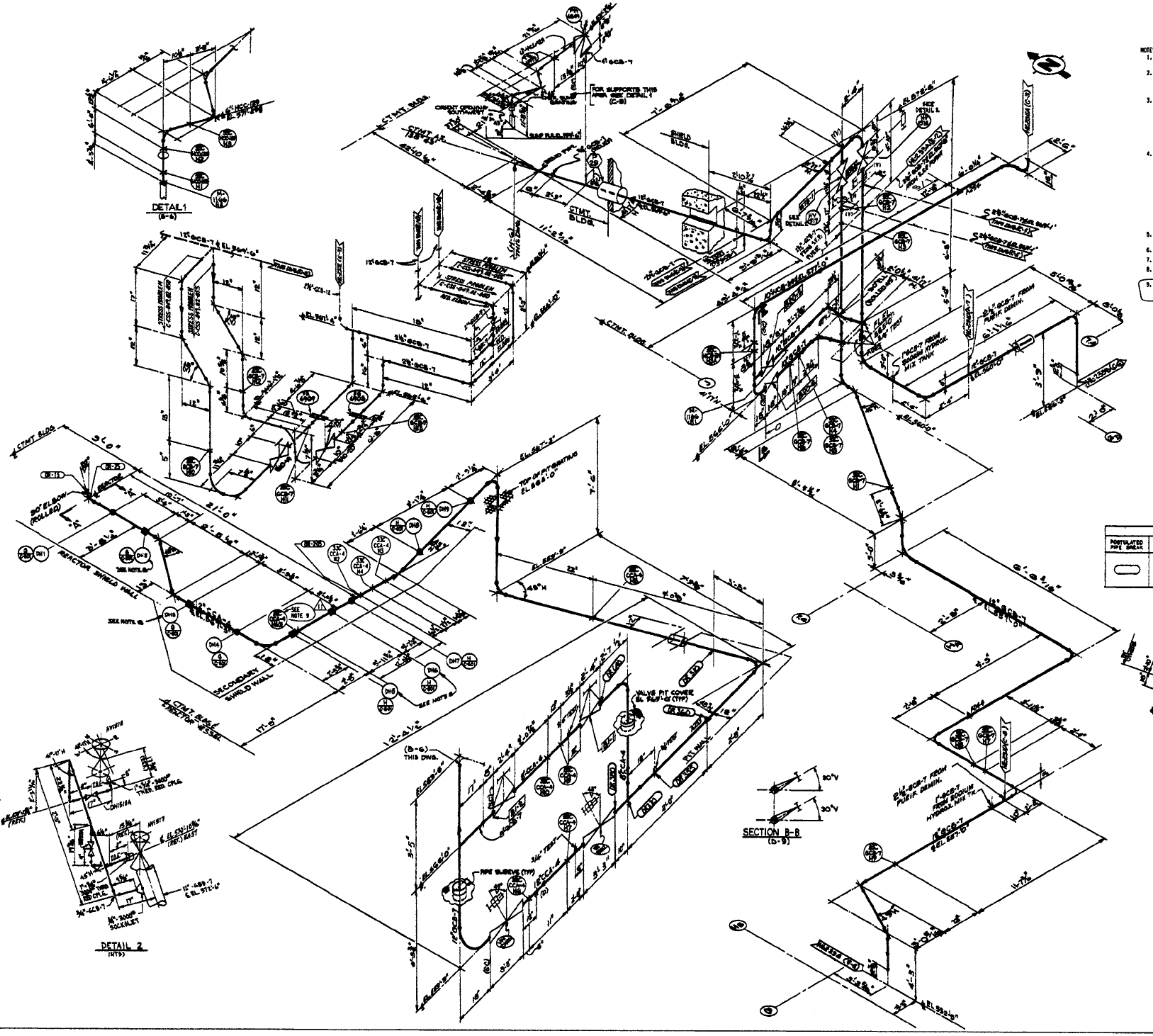
FUNCTIONAL DRAWING
REACTOR BUILDING
PRIMARY SYSTEM

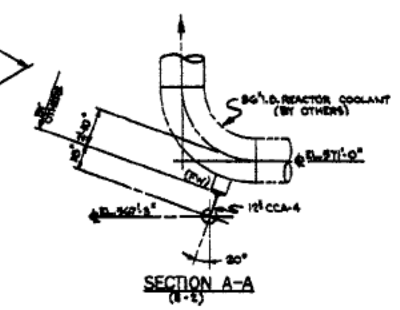
REVISION 30
OCTOBER 2014

DRAWING NO. FIGURE 3.6-30 | REV. 6

DB 07-17-14

DFN=G/USAR/UFIG3630.DGN





NOTES:

1. THIS DRAWING CONTAINS INFORMATION PREVIOUSLY SHOWN ON ITT GRINNELL DRAWING NO. PH-233C.
2. THIS DRAWING IS ISSUED FOR HANGER LOCATIONS ONLY AND DOES NOT NECESSARILY DEPICT THE EXACT ROUTING OF THE PIPING. FOR CORRECT PIPE ROUTING REFER TO DRAWING NO. 7749-M-233C.
3. AS APPLICABLE, THE PIPE SUPPORTS SHOWN ON THIS DRAWING WERE COVERED UNDER THE FOLLOWING SPECIFICATIONS:

   CONSTRUCTION PHASE
   7749-M-190, PART II - FABRICATION
   7749-M-453 - INSTALLATION

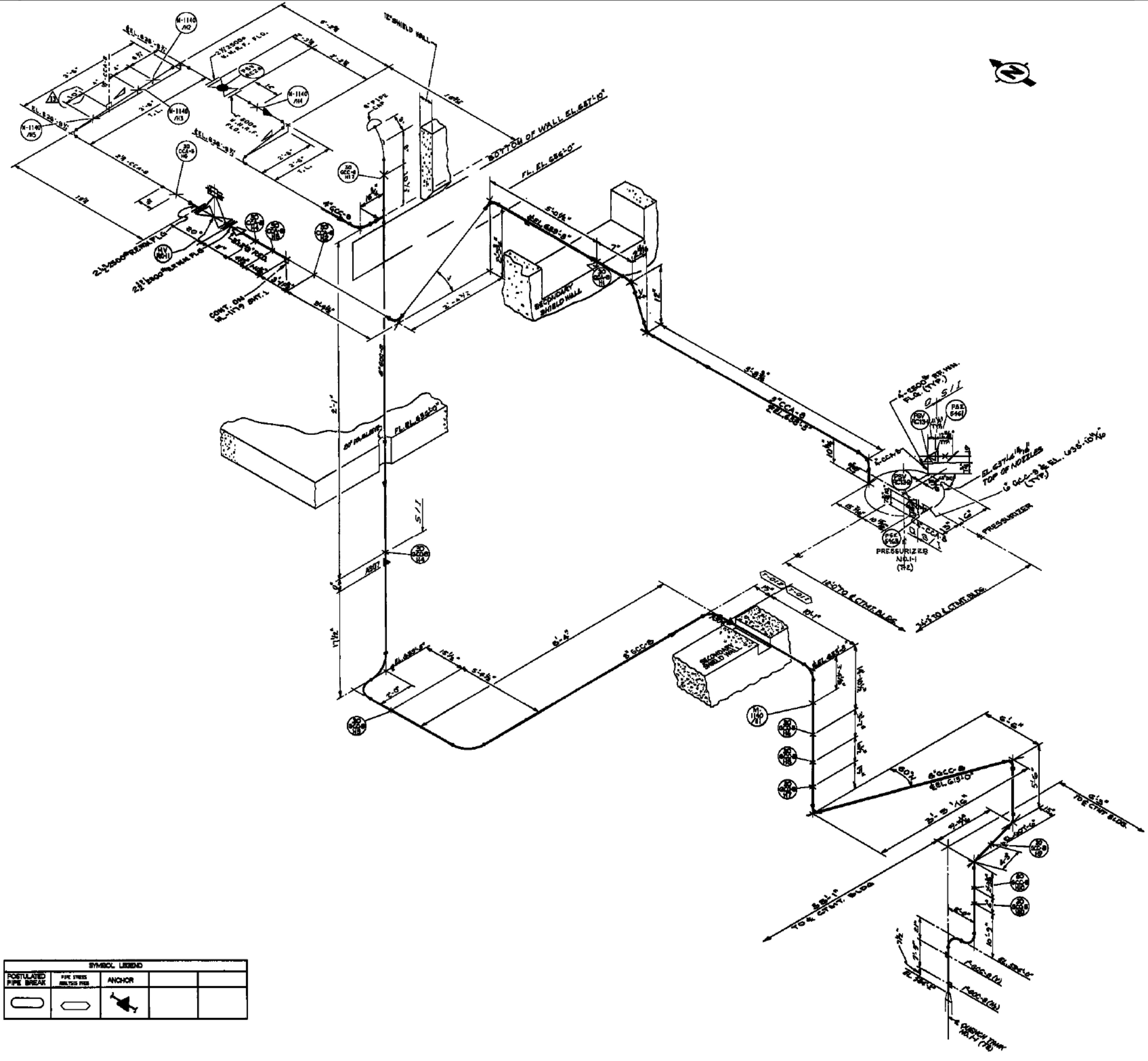
   OPERATIONAL PHASE
   12501-M-450N or Q - FABRICATION AND INSTALLATION
4. THE PIPE SUPPORT LOCATIONS SHOWN ON THIS DRAWING REFLECT SOME OR ALL OF THE FOLLOWING INFORMATION (SEE NOTE 5 FOR CATEGORY OF PIPING):
   - (a) AS-BUILT LOCATIONS OBTAINED FROM THE IE BULLETIN 79-14 INSPECTION - ALL SEISMIC CLASS I PIPING (BOTH Q- LISTED AND NON-Q).
   - (b) DESIGN LOCATIONS OBTAINED FROM THE STRESS ANALYSIS - NON-SEISMIC, HIGH TEMPERATURE PIPING.
   - (C) APPROXIMATE LOCATIONS OBTAINED FROM THE ITT GRINNELL FABRICATION/INSTALLATION DRAWING-NON-SEISMIC, LOW TEMPERATURE PIPING.
5. S/I —| INDICATES THE LIMIT OF SEISMIC CLASS I PIPING.
   Q —| INDICATES THE LIMIT OF Q-LISTED PIPING.
6. DELETE
7. COLUMN LINE AND PIPE ELEVATIONS ARE FOR REFERENCE ONLY.
8. DECAY HEAT WHIP RESTRAINTS NO. DH2, DH3, DH5, AND DH6 HAVE BEEN ABANDONED IN PLACE UNDER DCR 90-0071.
9. FOR SUPPORT 33C-CCA-4-H601 DETAILS, SEE DRAWING C-601, ELEVATION H. (REF. ECP 13-0327)

DAVIS-BESSE NUCLEAR POWER STATION

DECAY HEAT REMOVAL SYSTEM
HL-233C
FIGURE 3.6-31

REVISION 30
OCTOBER 2014

DB 07-16-14 DFN=H/PLDSGN/HL233C.DGN/TIF



NOTES:

1. THIS DRAWING CONTAINS INFORMATION PREVIOUSLY SHOWN ON ITT GRINNELL DRAWING NO. PH-230A.
2. THIS DRAWING IS ISSUED FOR HANGER LOCATIONS ONLY AND DOES NOT NECESSARILY DEPICT THE EXACT ROUTING OF THE PIPING. FOR CORRECT PIPE ROUTING REFER TO DRAWING NO. 7749-M-230A.
3. AS APPLICABLE, THE PIPE SUPPORTS SHOWN ON THIS DRAWING WERE COVERED UNDER THE FOLLOWING SPECIFICATIONS:

   CONSTRUCTION PHASE
   7749-M-190, PART II - FABRICATION
   7749-M-453 - INSTALLATION

   OPERATIONAL PHASE
   12501-M-450N or Q - FABRICATION AND INSTALLATION
4. THE PIPE SUPPORT LOCATIONS SHOWN ON THIS DRAWING REFLECT SOME OR ALL OF THE FOLLOWING INFORMATION (SEE NOTE 5 FOR CATEGORY OF PIPING):
   (a) AS-BUILT LOCATIONS OBTAINED FROM THE IE BULLETIN 79-14 INSPECTION - ALL SEISMIC CLASS I PIPING (BOTH Q- LISTED AND NON-Q).
   (b) DESIGN LOCATIONS OBTAINED FROM THE STRESS ANALYSIS - NON-SEISMIC, HIGH TEMPERATURE PIPING.
   (C) APPROXIMATE LOCATIONS OBTAINED FROM THE ITT GRINNELL FABRICATION/INSTALLATION DRAWING-NON-SEISMIC, LOW TEMPERATURE PIPING.
5. S/I INDICATES THE LIMIT OF SEISMIC CLASS I PIPING.
   Q INDICATES THE LIMIT OF Q-LISTED PIPING.
6. PIPE BREAK LOCATIONS HAVE BEEN POSTULATED TO OCCUR EVERYWHERE ON LINE CCA-8.
7. COLUMN LINE AND PIPE ELEVATIONS ARE FOR REFERENCE ONLY.
8. NO BREAKS ARE POSTULATED IN LINE CCC-8 DUE TO THE SHORT PERIOD UNDER WHICH THIS LINE IS AT HIGH-ENERGY CONDITIONS.

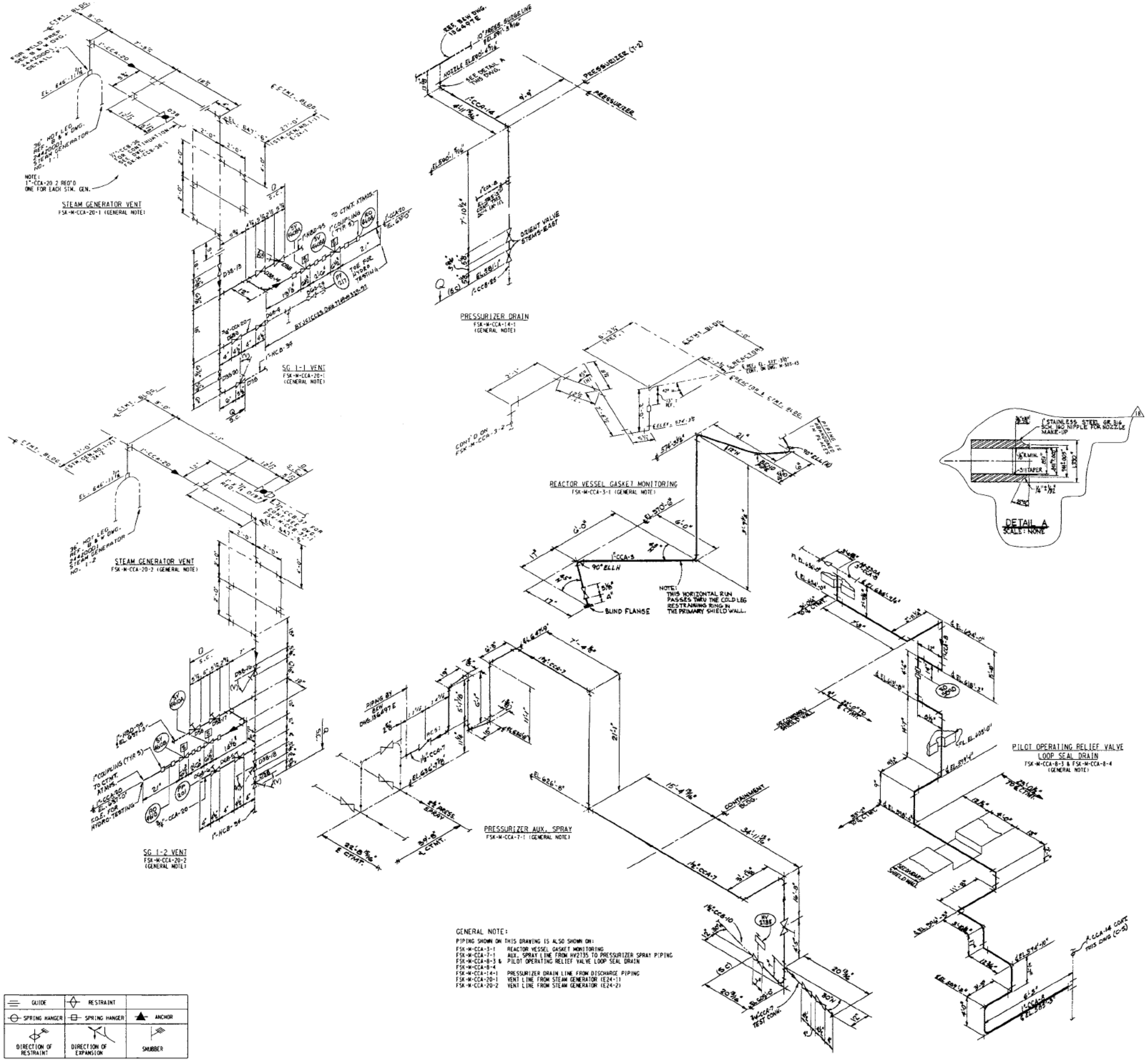
DAVIS-BESSE NUCLEAR POWER STATION

PRESSURIZER RELIEF SYSTEM
CONTAINMENT BUILDING
HL-230A
FIGURE 3.6-32

REVISION 29

DECEMBER 2012

DB 10-05-12 DFN=H/PLDSGN/HL230A.DGN/TIF



GENERAL NOTE:

PIPING SHOWN ON THIS DRAWING IS ALSO SHOWN ON:

FSK-M-CCA-3-1	REACTOR VESSEL GASKET MONITORING
FSK-M-CCA-7-1	AUX. SPRAY LINE FROM HV2735 TO PRESSURIZER SPRAY PIPING
FSK-M-CCA-8-3 & FSK-M-CCA-8-4	PILOT OPERATING RELIEF VALVE LOOP SEAL DRAIN
FSK-M-CCA-14-1	PRESSURIZER DRAIN LINE FROM DISCHARGE PIPING
FSK-M-CCA-20-1	VENT LINE FROM STEAM GENERATOR (E24-1)
FSK-M-CCA-20-2	VENT LINE FROM STEAM GENERATOR (E24-2)

NOTES:

1. ALL PIPING TO BE FABRICATED AND ERECTED IN ACCORDANCE WITH CONSTRUCTION DOCUMENT 7749-12.
2. FOR MATERIAL DESCRIPTION OF PIPE, VALVES AND FITTINGS, SEE PIPING CLASS SHEETS M-601 OF SPEC. 7749-M-190.
3. FOR PIPING AND INSTRUMENT DIAGRAM SEE DWG. M-030 AND M-033.
4. FOR EQUIPMENT LIST SEE M-600.
5. FOR INSTRUMENT CONNECTION DETAIL SEE DRAWINGS M-550 AND M-551.
6. NO ALLOWANCE HAS BEEN MADE FOR WELDED JOINTS OR GASKETS AND ALL PIPING IS DIMENSIONED IN THE COLD ERECTED POSITION UNLESS OTHERWISE NOTED ON THE DRAWING.
7. FOR PIPING SYSTEMS COMPOSITE DRAWING(S) THIS AREA SEE M-270 THRU M-275.
8. FOR MATERIAL DESCRIPTION OF INSULATION SEE SPEC. 7749-M-197 AND 7749-M-198.
9. PIPE BENDS, UNLESS OTHERWISE NOTED, HAVE A MINIMUM CENTER LINE RADIUS OF FIVE (5) TIMES THE NOMINAL PIPE DIAMETER.
10. FOR PIPING ISOMETRIC AND COMPOSITE DWG. INDEX SEE DWG. M-200.
11. FOR PIPING AND INSTRUMENT SYMBOLS SEE DWG. M-001 AND M-002.
12. FIELD WELD JOINTS WHERE SPECIFICALLY REQUIRED ARE INDICATED (FW), VENT AND DRAIN CONNECTIONS ARE INDICATED (V) AND (D). SPEC. CHANGE INDICATED (SC). FOR FIELD WELD IDENTIFICATION NUMBERS SEE INDIVUIAL "FSK" DRAWINGS.
13. UNLESS OTHERWISE NOTED, ALL VENT CONNS ARE 3/4" AND ALL DRAIN CONNS ARE 1".
14. THIS ISO. TO BE WORKED WITH DWG. M-230C.
15. ALL PIPING ON THIS DWG. IS "D" LISTED.
16. ORIENT VALVE STEMS UPRIGHT, IN THE VERTICAL PLANE, UNLESS NOTED.
17. DELETED

OPERATING CONDITIONS

LINE NO.	NORMAL		MAXIMUM	
	PSIG	°F	PSIG	°F
CCA-3	ATM	200	2185	600
CCA-7	300	100	400	608
CCA-14	2185	650	2185	650
CCA-20	2185	608	2185	608

PA-4							
PA-3							
PA-2							
PA-1							
NO	DATE	REVISIONS	BY	CK			

PIPING ANALYSIS DATA

MATERIAL						
OPER TEMP °F	SEE TABLE ABOVE	OPER PRESS PSIG				SEE TABLE ABOVE
MOD of ELAS E PSI COLD / HOT						
POISSONS RATIO						
EXP COEFF MIL IN °F						
LINE DATA O D (IN)						
LINE NO						
WALL THICK. (IN)						
SPECIAL REMARKS						

DAVIS-BESSE NUCLEAR POWER STATION

NUCLEAR PIPING

ASME SECTION III CLASS 1

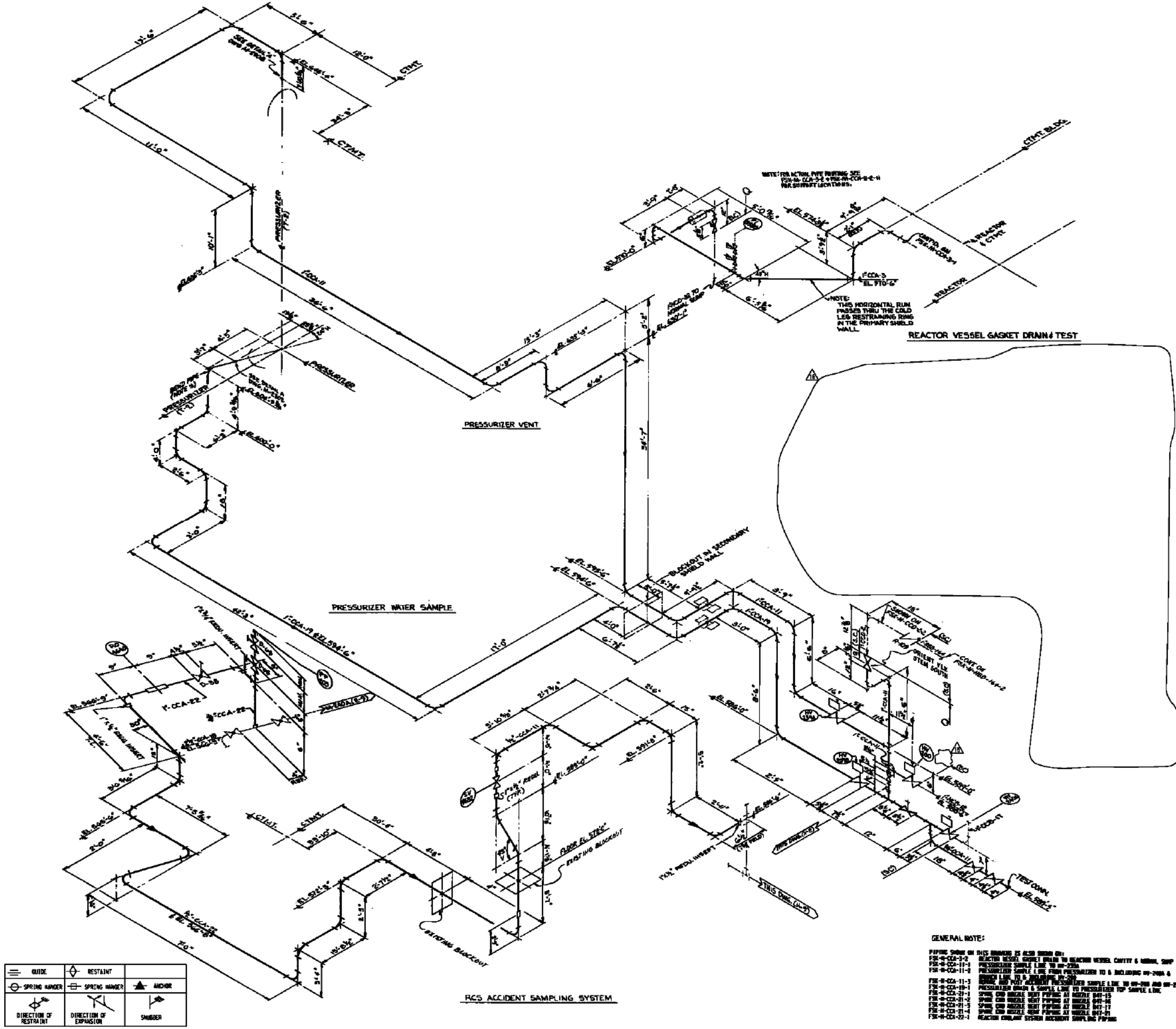
M-230B

FIGURE 3.6-33

REVISION 30

OCTOBER 2014

DB 10-07-14 DFN=/J/RASDGN/M230B.DGN/TIF



NOTES:

1. ALL PIPING TO BE FABRICATED AND ERECTED IN ACCORDANCE WITH CONSTRUCTION DOCUMENT 7749-12.
2. FOR MATERIAL DESCRIPTION OF PIPE, VALVES, AND FITTING, SEE PIPING CLASS SHEETS M-601 OF SPEC. 7749-M-190.
3. FOR PIPING AND INSTRUMENT DIAGRAM SEE DWG. M-030 & M-033.
4. FOR EQUIPMENT LIST SEE M-600.
5. FOR INSTRUMENT CONNECTION DETAIL SEE DRAWINGS M-550 AND M-551.
6. NO ALLOWANCE HAS BEEN MADE FOR WELDED JOINTS OR GASKETS AND ALL PIPING IS DIMENSIONED IN THE COLD ERECTED POSITION UNLESS OTHERWISE NOTED ON THE DRAWING.
7. FOR PIPING SYSTEM COMPOSITE DRAWING(S) THIS AREA SEE M-270 THRU M-276.
8. FOR MATERIAL DESCRIPTION OF INSULATION SEE SPEC. 7749-M-197 AND 7749-M-198.
9. PIPE BENDS, UNLESS OTHERWISE NOTED, HAVE A MINIMUM CENTER-LINE RADIUS OF FIVE (5) TIMES THE NOMINAL PIPE DIAMETER.
10. FOR PIPING ISOMETRIC AND COMPOSITE DWG. INDEX SEE DWG. M-200.
11. FOR PIPING AND INSTRUMENT SYMBOLS SEE DWG. M-001 AND M-002.
12. FIELD WELD JOINTS WHERE SPECIFICALLY REQUIRED ARE INDICATED (FW). VENT & DRAIN CONNECTIONS ARE INDICATED (V) & (D). SPEC. CHANGES ARE INDICATED (SC). FOR FIELD WELD IDENTIFICATION NUMBERS SEE INDIVIDUAL "FSK" DRAWINGS.
13. Q—| INDICATES THE LIMIT OF Q-LISTED PIPING.
14. THIS ISO. TO BE WORKED WITH DWG. M-230B.
15. OREINT VALVE STEMS UPRIGHT, IN THE VERTICAL PLANE, UNLESS NOTED.
16. FOR HANGER LOCATIONS SEE DRAWING HL-1143

OPERATING CONDITIONS

LINE NO.	NORMAL		MAXIMUM	
	PSIG	°F	PSIG	°F
CCA-3	ATM	200	[illegible]	600
CCA-11	2185	640	[illegible]	640
CCA-19	2185	650	[illegible]	650
CCA-21	2155	650	2750	650

NO.	DATE	REVISIONS		BY			

PIPING ANALYSIS DATA

MATERIAL:

OPER. TEMP. °F — SEE TABLE ABOVE — OPER. PRESS PSIG — SEE TABLE ABOVE

MOD. of ELAS. E. PSI COLD / HOT

POISSONS RATIO

EXP. COEFF. MIL. IN. °F

LINE DATA: O.D. (IN.) / LINE NO. / WALL THICK. (IN.)

SPECIAL REMARKS:

GENERAL NOTE:

PIPING SHOWN ON THIS DRAWING IS ALSO SHOWN ON:

FSK-M-CCA-3-2	REACTOR VESSEL GASKET DRAIN TO REACTOR VESSEL CAVITY & NORMAL SUMP
FSK-M-CCA-11-1	PRESSURIZER SAMPLE LINE TO HV-239A
FSK-M-CCA-11-2	PRESSURIZER SAMPLE LINE FROM PRESSURIZER TO & INCLUDING HV-240A & BRANCH LINE TO & INCLUDING HV-240
FSK-M-CCA-11-3	NORMAL AND POST ACCIDENT PRESSURIZER SAMPLE LINE TO HV-240 AND HV-240A
FSK-M-CCA-19-1	PRESSURIZER DRAIN & SAMPLE LINE TO PRESSURIZER TOP SAMPLE LINE
FSK-M-CCA-21-1	SPARE CRD NOZZLE VENT PIPING AT NOZZLE 047-15
FSK-M-CCA-21-2	SPARE CRD NOZZLE VENT PIPING AT NOZZLE 047-16
FSK-M-CCA-21-3	SPARE CRD NOZZLE VENT PIPING AT NOZZLE 047-17
FSK-M-CCA-21-4	SPARE CRD NOZZLE VENT PIPING AT NOZZLE 047-21
FSK-M-CCA-22-1	REACTOR COOLANT SYSTEM ACCIDENT SAMPLING PIPING

DAVIS-BESSE NUCLEAR POWER STATION

ASME SECTION III CLASS 1

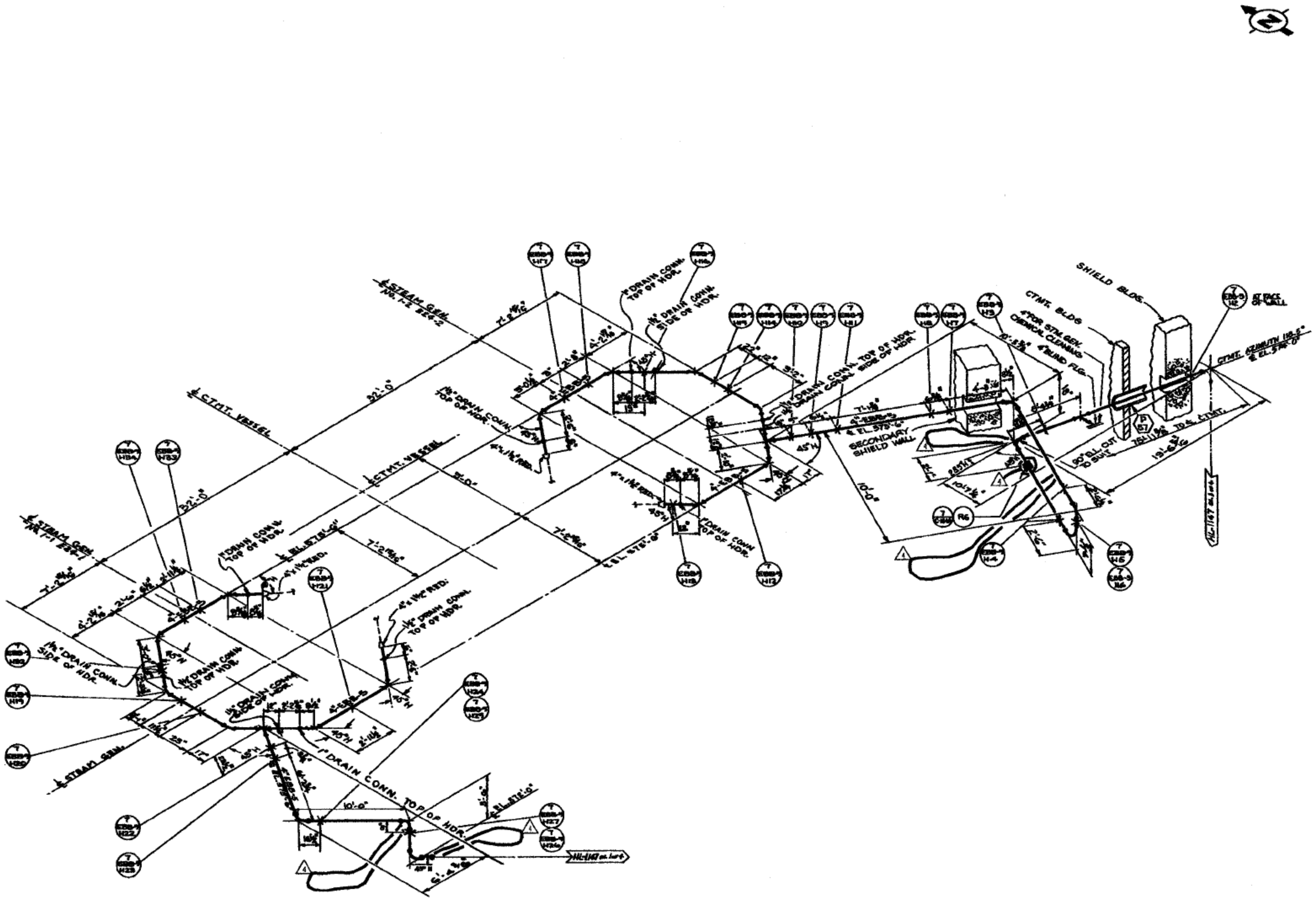
M-230C

FIGURE 3.6-34

REVISION 29

DECEMBER 2012

DB 08-02-12 DFN=/J/RASDON/M230C.DGN/CIT



NOTES:

1. THIS DRAWING PARTIAL CONTAINS INFORMATION PREVIOUSLY SHOWN ON ITT GRINNELL DRAWING NO. PH-207A.
2. THIS DRAWING IS ISSUED FOR HANGER LOCATIONS ONLY AND DOES NOT NECESSARILY DEPICT THE EXACT ROUTING OF THE PIPING. FOR CORRECT PIPE ROUTING REFER TO DRAWING NO. 7749-N-207A.
3. AS APPLICABLE, THE PIPE SUPPORTS SHOWN ON THIS DRAWING WERE COVERED UNDER THE FOLLOWING SPECIFICATIONS:

   CONSTRUCTION PHASE
   7749-M-190, PART II-FABRICATION
   7749-M-453-INSTALLATION

   OPERATIONAL PHASE
   12501-M-450N OR Q-FABRICATION AND INSTALLATION
4. THE PIPE SUPPORT LOCATIONS SHOWN ON THIS DRAWING REFLECT SOME OR ALL OF THE FOLLOWING INFORMATION (SEE NOTE 5 FOR CATEGORY OF PIPING):
   (a) AS-BUILT LOCATIONS OBTAINED FROM THE IE BULLETIN 79-14 INSPECTION-ALL SEISMIC CLASS I PIPING (BOTH Q-LISTED AND NON-Q).
   (b) DESIGN LOCATIONS OBTAINED FROM THE STRESS ANALYSIS-NON-SEISMIC, HIGH TEMPERATURE PIPING.
   (c) APPROXIMATE LOCATIONS OBTAINED FROM THE ITT GRINNELL FABRICATION/INSTALLATION DRAWING-NON-SEISMIC, LOW TEMPERATURE PIPING.
5. ALL PIPING SHOWN ON THIS DRAWING IS SEISMIC CLASS 1.
6. COLUMN LINE AND PIPE ELEVATIONS ARE FOR REFERENCE ONLY.

SYMBOL LEGEND

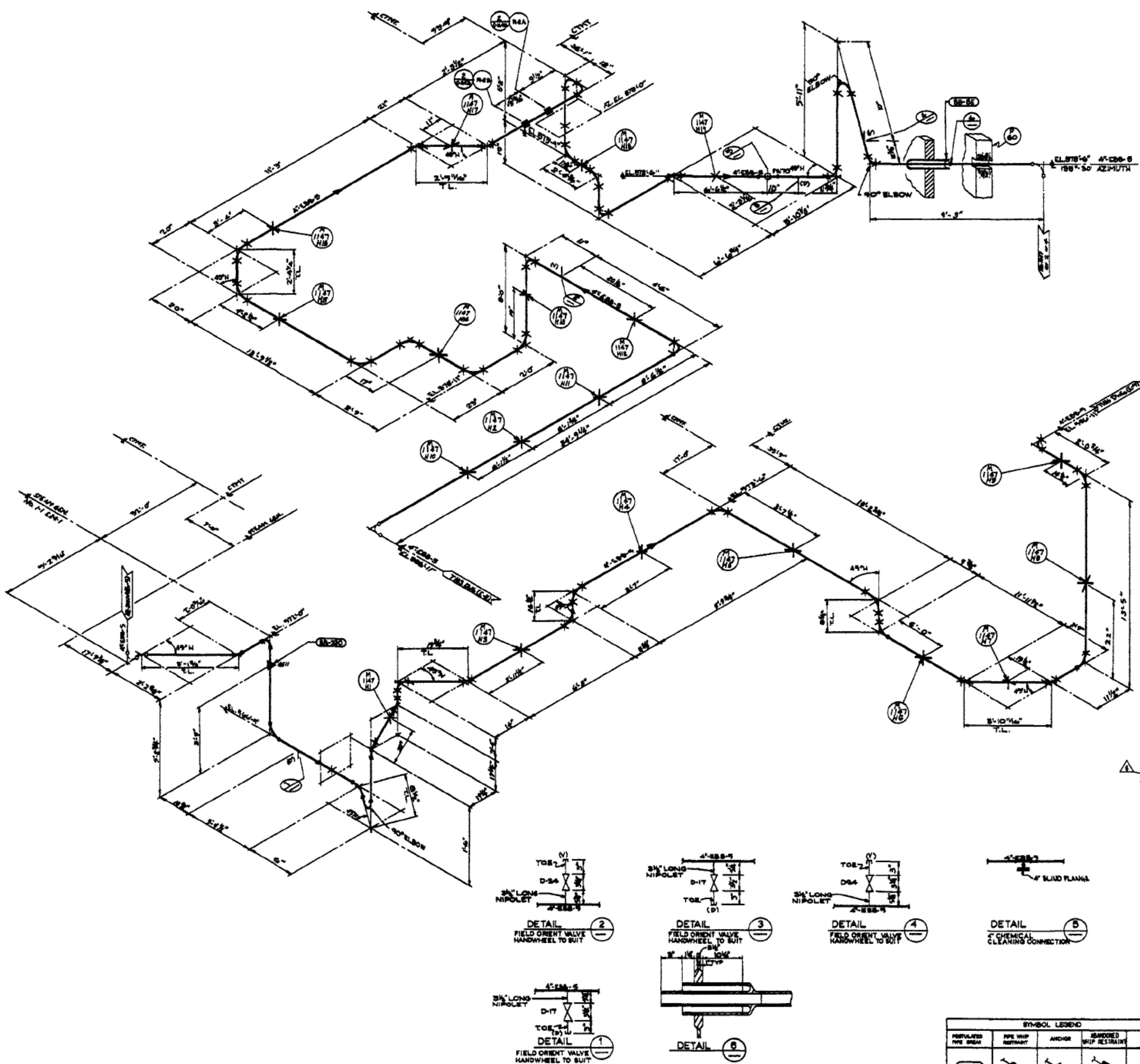
POSTULATED PIPE BREAK	PIPE WHIP RESTRAINT	ABANDONED WHIP RESTRAINT
		4

DAVIS-BESSE NUCLEAR POWER STATION

STEAM GENERATOR DRAIN SYSTEM-
CONTAINMENT BUILDING
HL-207A
FIGURE 3.6-35

REVISION 26
JUNE 2008

06 03-11-08 DFN=H/PLDSGN/HL207A.DGN/TIF



NOTES:

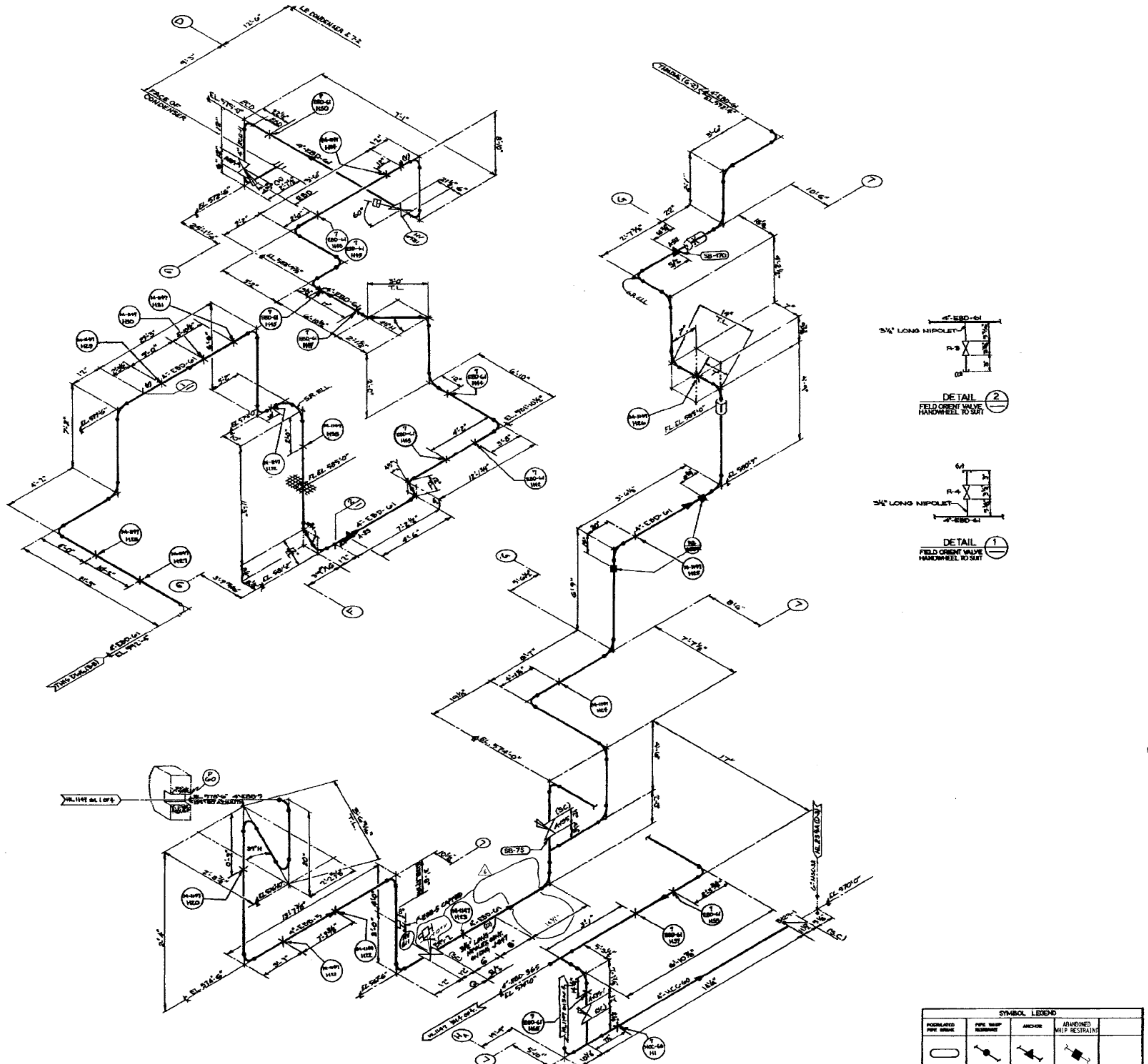
1. THIS DRAWING IS ONLY FOR HANGER LOCATIONS IT MAY NOT NECESSARILY DEPICT THE EXACT ROUTING OF THE PIPING. SEE LAYOUT DRAWING FOR PIPE ROUTING. M-207E
2. ALL PIPING ON THIS DRAWING IS SEISMIC CATEGORY I AND Q-LISTED.
3. COLUMN LINE AND PIPE ELEVATIONS ARE FOR REFERENCE ONLY.

SYMBOL LEGEND
POSTULATED PIPE BREAK
PIPE WHIP RESTRAINT
ANCHOR
ABANDONED WHIP RESTRAINT

DAVIS-BESSE NUCLEAR POWER STATION
STEAM GENERATOR BLOWDOWN SYSTEM-CONTAINMET
BUILDING STEAM GENERATOR 1-1 DRAIN SYSTEM
HL-1147 SH. 1
FIGURE 3.6-36

REVISION 30
OCTOBER 2014

DB 02-08-14 DFN=H/PLDSGN/HL1147SH1.DGN/IF



NOTES:

1. THIS DRAWING IS ONLY FOR HANGER LOCATIONS IT MAY NOT NECESSARILY DEPICT THE EXACT ROUTING OF THE PIPING. SEE LAYOUT DRAWING FOR CORRECT PIPE ROUTING M-207F.
2. COLUMN LINE AND PIPE ELEVATIONS ARE FOR REFERENCE ONLY.
3. S/I INDICATES THE LIMIT OF SEISMIC CLASS 1 PIPING.
   Q INDICATES THE LIMIT OF Q-LISTED PIPING.

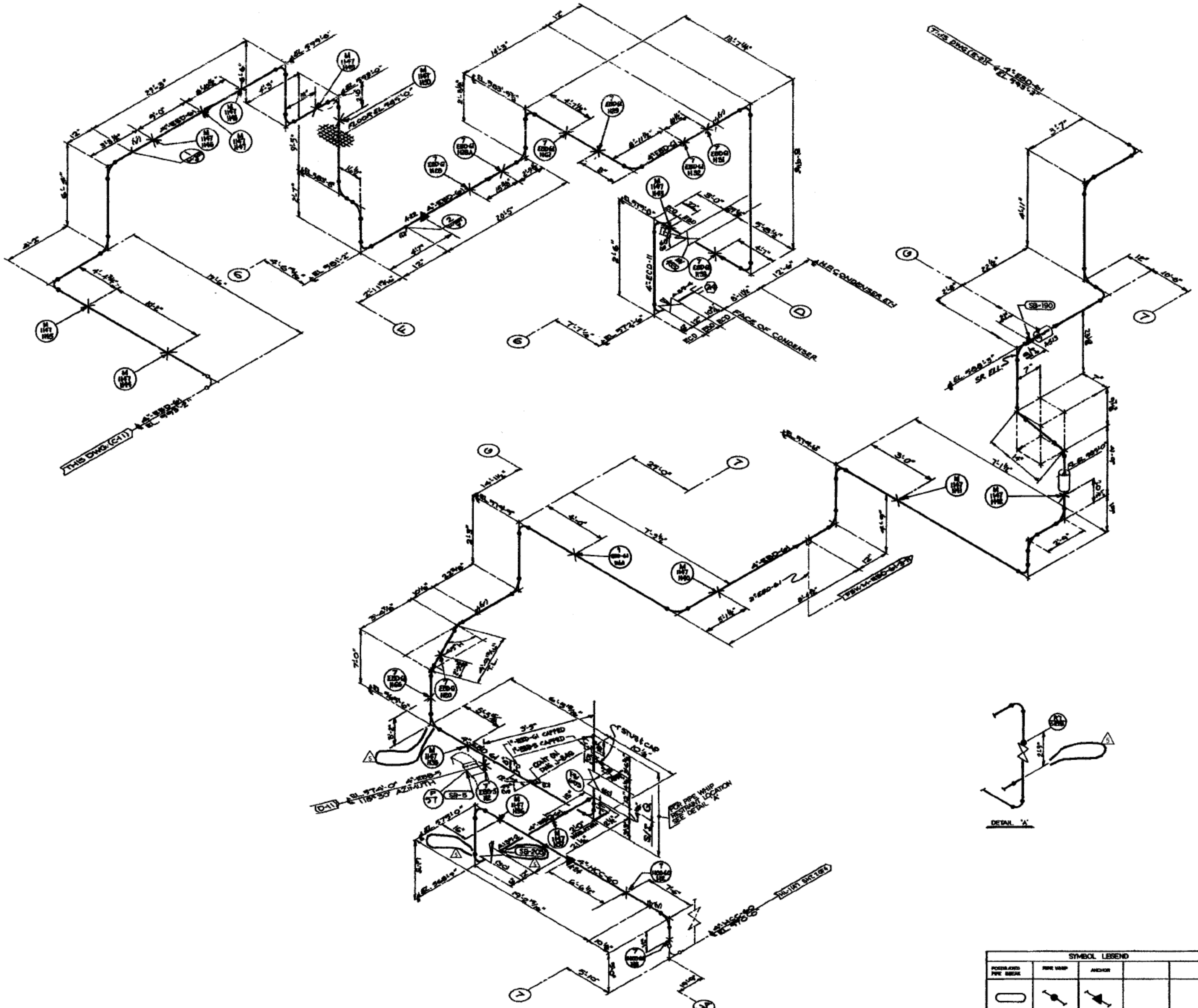
DAVIS-BESSE NUCLEAR POWER STATION

STEAM GENERATOR BLOWDOWN SYSTEM
AUXILIARY AND TURBINE BUILDING
STEAM GENERATOR 1-1 DRAIN SYSTEM
HL-1147 SHEET 2
FIGURE 3.6-37

REVISION 26

JUNE 2008

DB 03-12-08 DFN=H/PLDSGN/HL1147SH2.DGN/TIF



NOTES:

1. THIS DRAWING IS ONLY FOR HANGER LOCATIONS IT MAY NOT NECESSARILY DEPICT THE EXACT ROUTING OF THE PIPING. SEE LAYOUT DRAWING FOR CORRECT PIPE ROUTING M-207G.
2. COLUMN LINE AND PIPE ELEVATION ARE FOR REFERENCE ONLY.
3. S/I ⟶| INDICATES THE LIMIT OF SEISMIC CLASS I PIPING.
   Q ⟶| INCICATES THE LIMIT OF Q-LISTED PIPING.

SYMBOL LEGEND				
POSTULATED PIPE BREAK	PIPE WHIP	ANCHOR		

DAVIS-BESSE NUCLEAR POWER STATION

STEAM GENERATOR BLOWDOWN SYSTEM
AUXILIARY AND TURBINE BUILDING
STEAM GENERATOR 1-2 DRAIN SYSTEM
HL-1147 SHEET 3
FIGURE 3.6-38

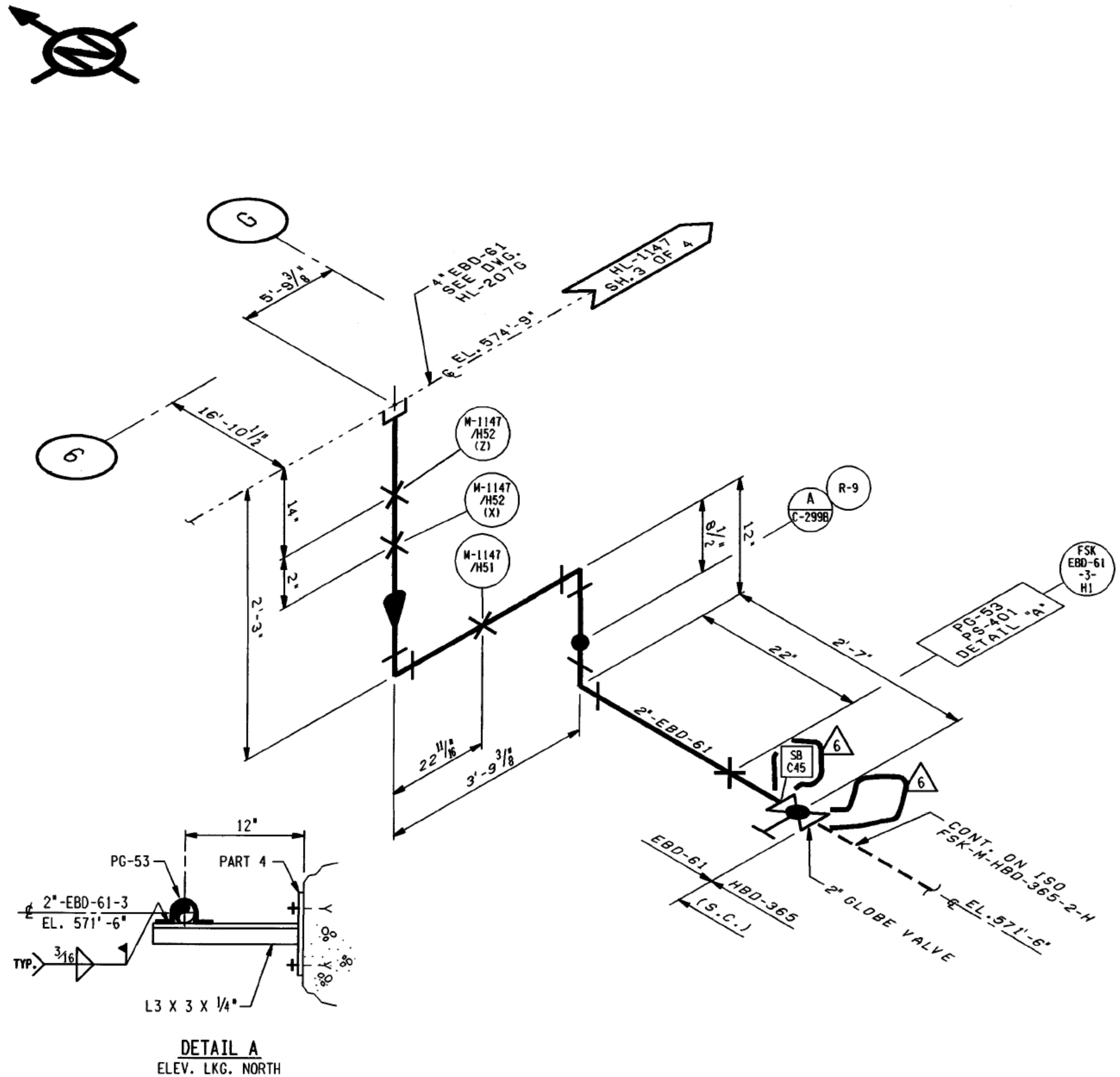
REVISION 26

JUNE 2008

DB 03-12-08 DFN-H/PLDSGN/HL1147SH3.DGN/TIF

NOTES:

1. ⬡ INDICATES WELD IDENTIFICATION NUMBER
2. + INDICATES 3" ALLOWANCE FOR FIELD FIT-UP.
3. FABRICATE AND INSTALL PER FSK-M-97 SHEETS 1-6, FSK-M-104, FSK-M-139 AND SPECIFICATIONS 12501-M-450Q.
4. PAINT AS REQUIRED PER SPECIFICATION 12501-A-24N.
5. THIS DRAWING IS FOR HANGER LOCATION ONLY. IT MAY NOT NECESSARY DEPICT THE EXACT ROUTING OF THE PIPING. SEE DRAWING FSK-M-EBD-61-3 FOR CORRECT PIPE ROUTING.



DETAIL A
ELEV. LKG. NORTH

SYMBOL LEGEND	
POSTULATED PIPE BREAK	PIPE WHIP RESTRAINT
□	

DAVIS-BESSE NUCLEAR POWER STATION

SECONDARY SIDE DRAIN PIPING FROM 4" EBD-61 TO STEAM GENERATOR WET LAYUP PUMP (P182-2)
FSK-M-EBD-61-3-H
FIGURE 3.6-39

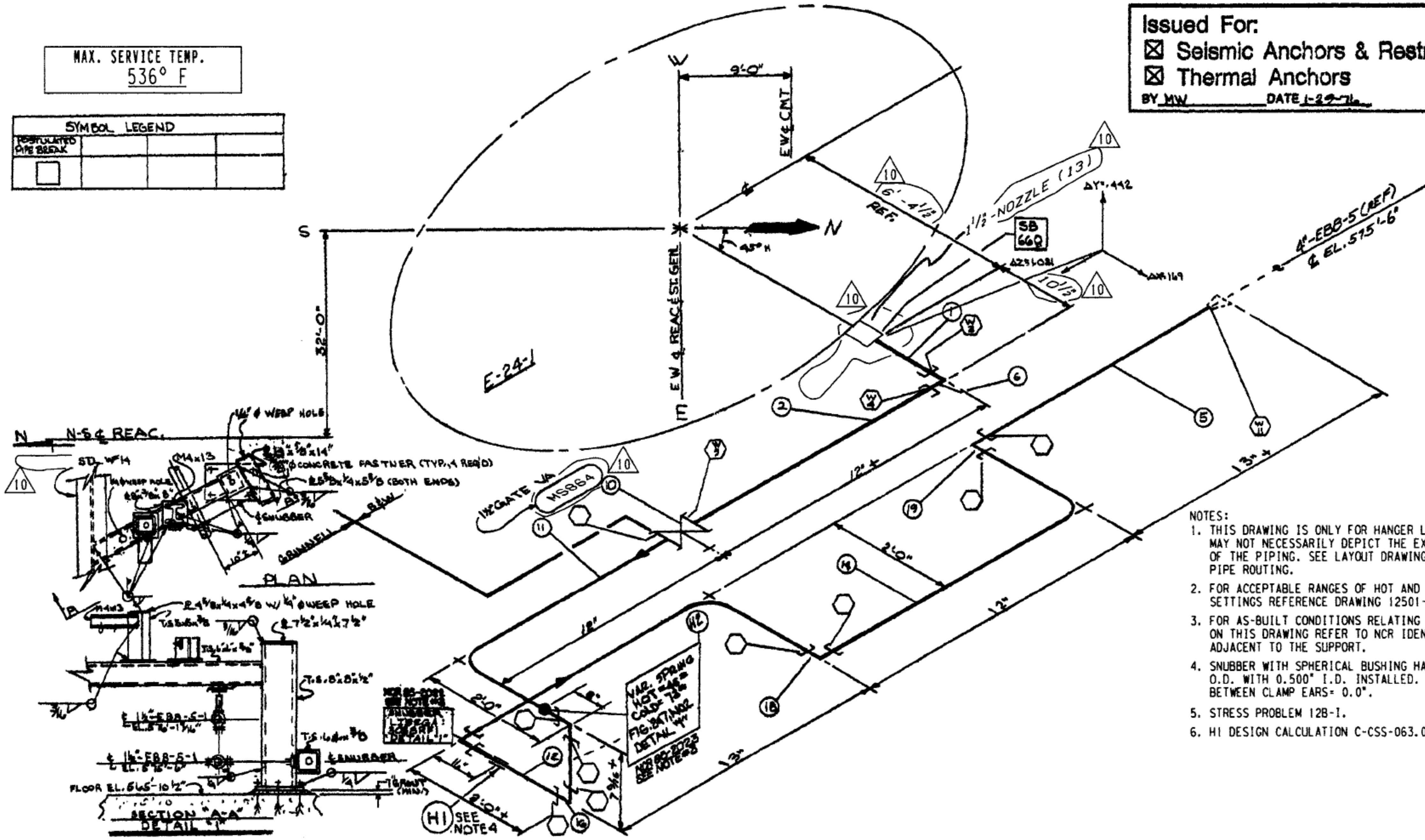
REVISION 26
JUNE 2008

03-11-08 DFN=H:/PLDSGN/MEBD613H.DGN

MAX. SERVICE TEMP.
536° F

SYMBOL LEGEND			
POSTULATED PIPE BREAK			
☐			

Issued For:
☒ Seismic Anchors & Restraints
☒ Thermal Anchors
BY MW DATE 1-29-76



NOTES:

1. THIS DRAWING IS ONLY FOR HANGER LOCATION IT MAY NOT NECESSARILY DEPICT THE EXACT ROUTING OF THE PIPING. SEE LAYOUT DRAWING FOR CORRECT PIPE ROUTING.
2. FOR ACCEPTABLE RANGES OF HOT AND COLD PISTON SETTINGS REFERENCE DRAWING 12501-M-618.
3. FOR AS-BUILT CONDITIONS RELATING TO SUPPORTS ON THIS DRAWING REFER TO NCR IDENTIFIED ADJACENT TO THE SUPPORT.
4. SNUBBER WITH SPHERICAL BUSHING HAS SLEEVE 0.624" O.D. WITH 0.500" I.D. INSTALLED. FREE PLAY BETWEEN CLAMP EARS= 0.0".
5. STRESS PROBLEM 12B-I.
6. HI DESIGN CALCULATION C-CSS-063.01-109.

P&I.D. NO.: M-007 (7)
PIPING DWG.: M-207A
AREA No. 9 ELEV. 575'-576'
UNIT No.: 1
BLDG: CONTAINMENT
PIPING REV. (5)

LINE SPEC: EBB
APP. CODE: ASME SECT. III
CLASS: NUCLEAR II
Q.C. SYSTEM NO.: 31255

NOTE:
DISMANTLE DIAPHRAGM VALVES BEFORE WELDING.
1/16" MIN. GAP REQUIRED BETWEEN SOCKET WELD JOINTS.

DAVIS-BESSE NUCLEAR POWER STATION

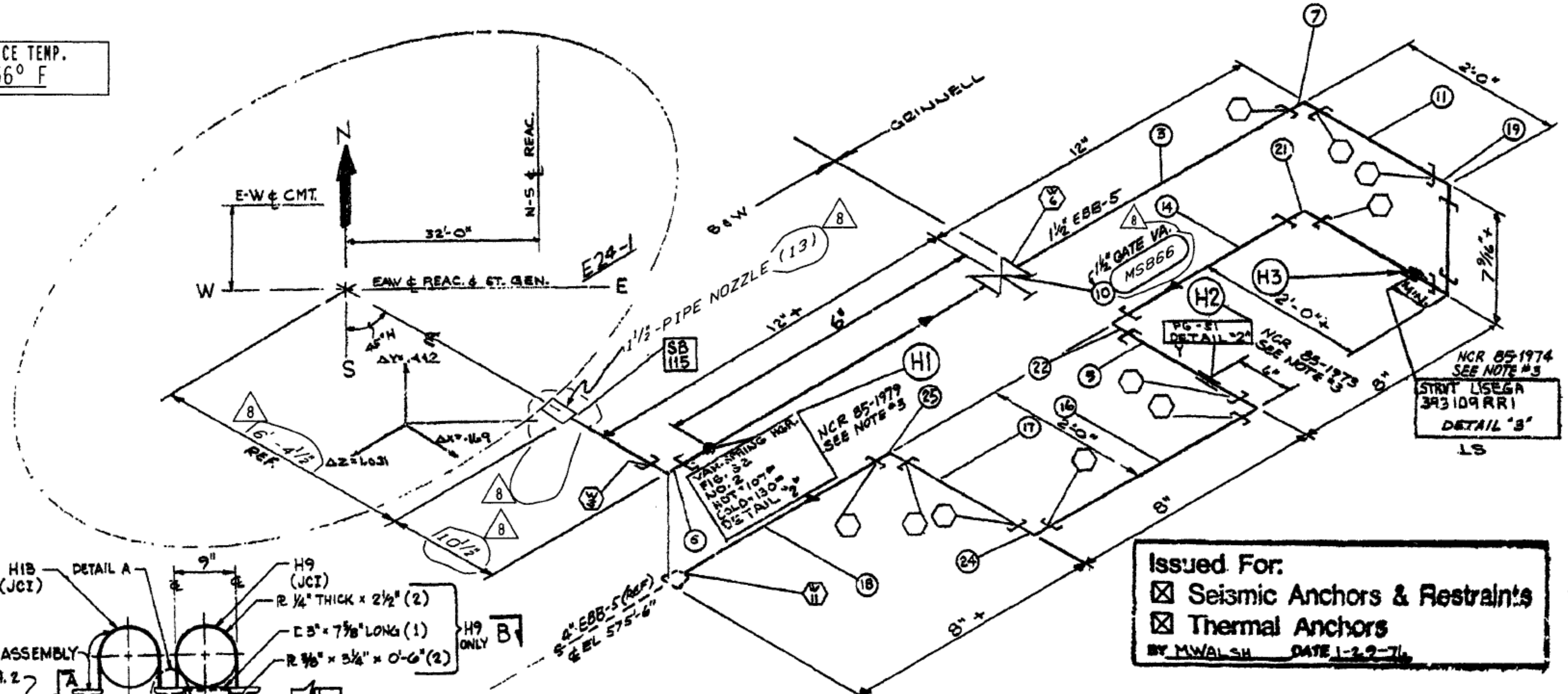
Q

SECONDARY SIDE DRAIN PIPING
FROM STEAM GENERATOR E24-1
NOZZLE (13) TO 4" DRAIN HEADER
FSK-M-EBB-5-1-H
FIGURE 3.6-40

REVISION 30
OCTOBER 2014

DB 07-22-14 DFN: H/PLDSGN/EBB51H.DGN/TIF

MAX. SERVICE TEMP.
536° F



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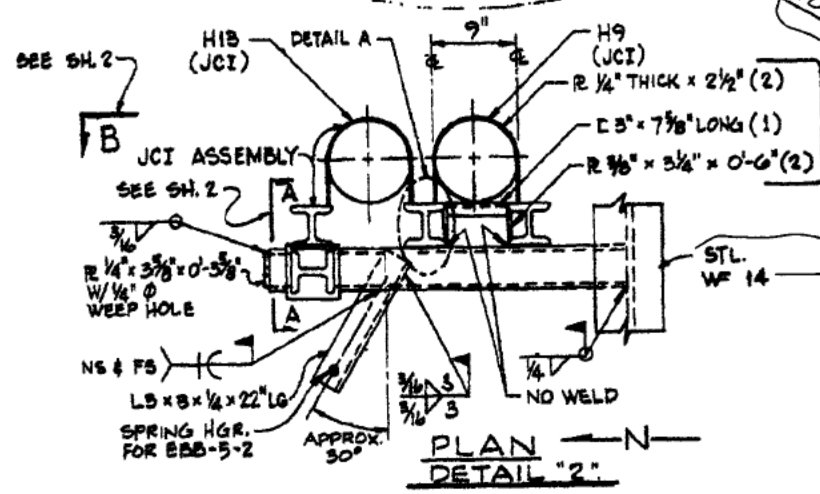
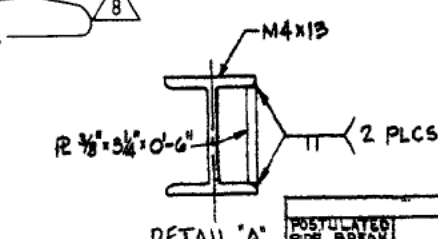
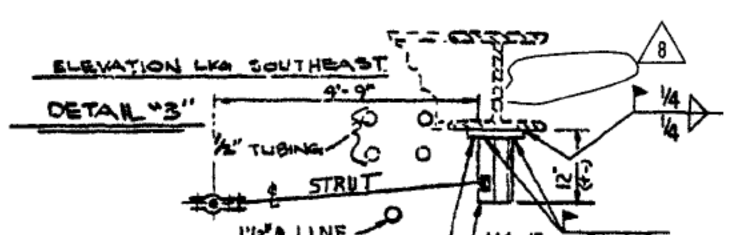
☒ Thermal Anchors

BY MWALSH DATE 1-29-76







NOTES:

1. THIS DRAWING IS ONLY FOR HANGER LOCATION IT MAY NOT NECESSARILY DEPICT THE EXACT ROUTING OF THE PIPING. SEE LAYOUT DRAWING FOR CORRECT PIPE ROUTING.
2. STRESS ANALYSIS PROBLEM 12B-1.
3. FOR AS-BUILT CONDITIONS RELATING TO SUPPORTS ON THIS DRAWING REFER TO NCR IDENTIFIED ADJACENT TO THE SUPPORT.

SYMBOL LEGEND				
POSTULATED PIPE BREAK				
□				

P&I.D. NO.: M-007 (7)
PIPING DWG.: M-207A
AREA No.: 9 ELEV 575' TO 576'
UNIT No.: 1
BLDG.: CONTAINMENT
PIPING REV. (5)
HANGER TYPE
DWG NO.:

LINE SPEC. EBB
APP CODE: ASME SECT. III
CLASS: NUCLEAR III
D C SYSTEM NO.. 31255

NOTE:
DISMANTLE DIAPHRAGM VALVES BEFORE WELDING.
1/16" MIN. GAP REQUIRED BETWEEN SOCKET WELD JOINTS.

DAVIS-BESSE NUCLEAR POWER STATION

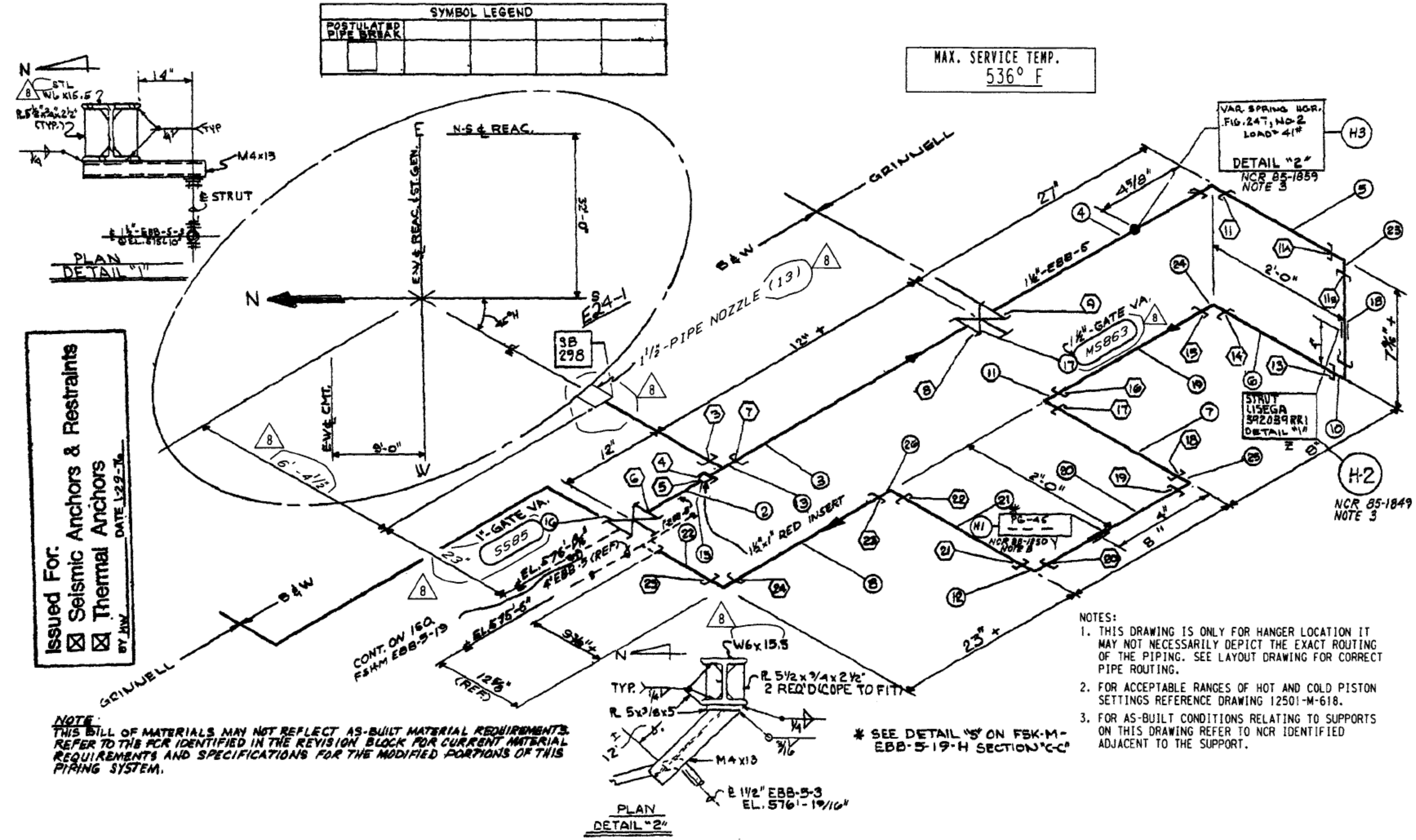
Q

SECONDARY SIDE DRAIN PIPING
FROM STEAM GENERATOR E24-1
NOZZLE (13) TO DRAIN HEADER
FSK-M-EBB-5-2-H SH. 1

FIGURE 3.6-41

REVISION 30
OCTOBER 2014

DB 07-22-14 DFN: H/PLDSGN/EBB52HS1.DGN/TIF



NOTE:
THIS BILL OF MATERIALS MAY NOT REFLECT AS-BUILT MATERIAL REQUIREMENTS. REFER TO THE FCR IDENTIFIED IN THE REVISION BLOCK FOR CURRENT MATERIAL REQUIREMENTS AND SPECIFICATIONS FOR THE MODIFIED PORTIONS OF THIS PIPING SYSTEM.

* SEE DETAIL "S" ON FSK-M-EBB-5-19-H SECTION "C-C"

NOTES:
1. THIS DRAWING IS ONLY FOR HANGER LOCATION IT MAY NOT NECESSARILY DEPICT THE EXACT ROUTING OF THE PIPING. SEE LAYOUT DRAWING FOR CORRECT PIPE ROUTING.
2. FOR ACCEPTABLE RANGES OF HOT AND COLD PISTON SETTINGS REFERENCE DRAWING 12501-M-618.
3. FOR AS-BUILT CONDITIONS RELATING TO SUPPORTS ON THIS DRAWING REFER TO NCR IDENTIFIED ADJACENT TO THE SUPPORT.

REFERENCE

P&I.D. NO. M-007 (6)
PIPING DWG. M-207A
AREA No 9 ELEV. 575'-576'
UNIT No. 1
BLDG CONTAINMENT
PIPING REV. (6) SPDCN 2016
HANGER TYPE:
DWG. NO.

LINE SPEC.: EBB
APP. CODE: ASME SECT III
CLASS: NUCLEAR II
Q.C. SYSTEM NO.: 31266

NOTE:
DISMANTLE DIAPHRAGM VALVES BEFORE WELDING.
1/16" MIN. GAP REQUIRED BETWEEN SOCKET WELD JOINTS.

DAVIS-BESSE NUCLEAR POWER STATION

SECONDARY SIDE DRAIN PIPING
FROM STEAM GENERATOR E24-1
NOZZLE (13) TO 4" DRAIN HEADER
FSK-M-EBB-5-3-H
FIGURE 3.6-42

Q

REVISION 30
OCTOBER 2014

DB 07-22-14 DFN: H/PLDSGN/EBB53H.DGN/TIF

MAX. SERVICE TEMP.
536° F

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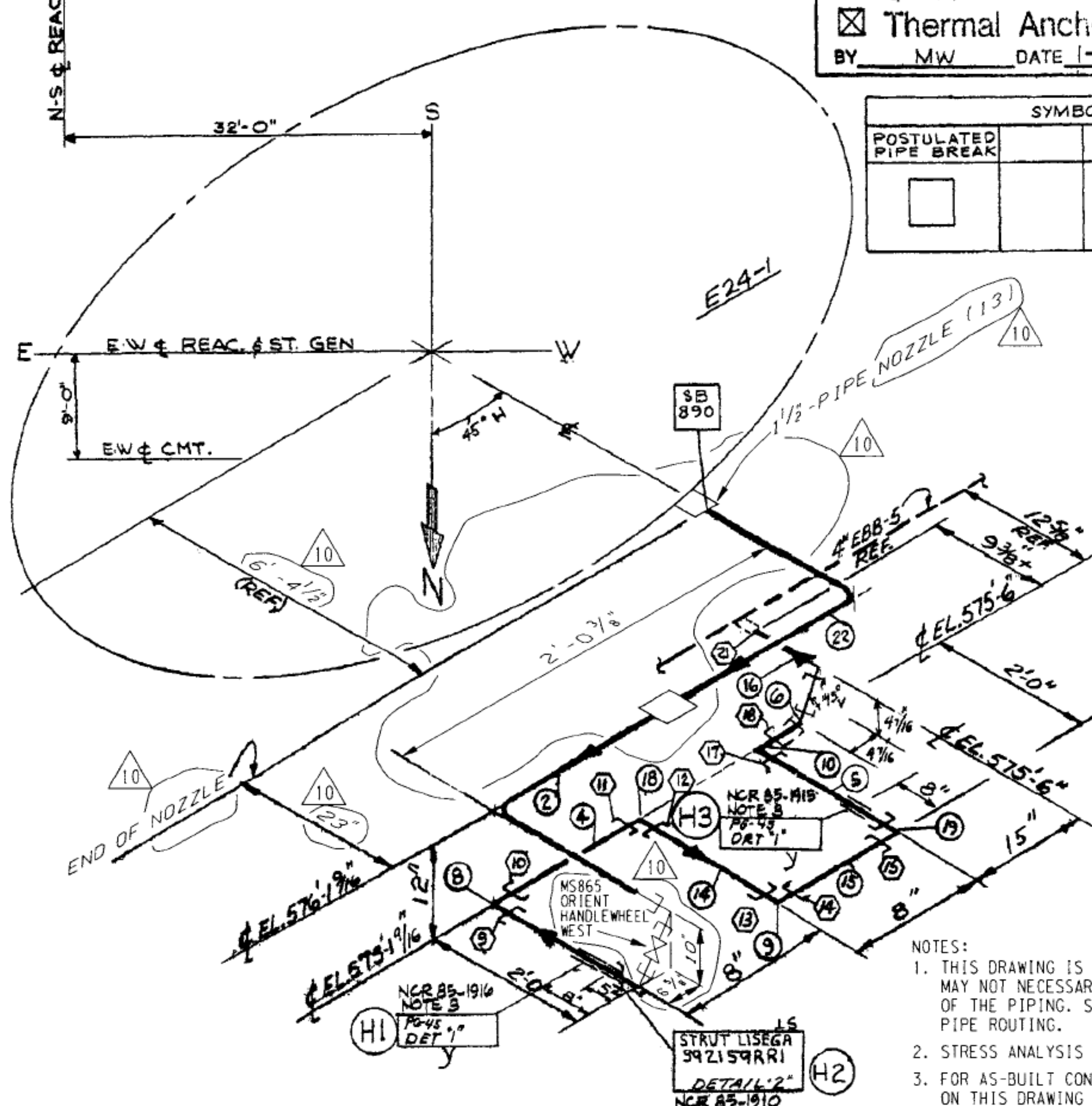
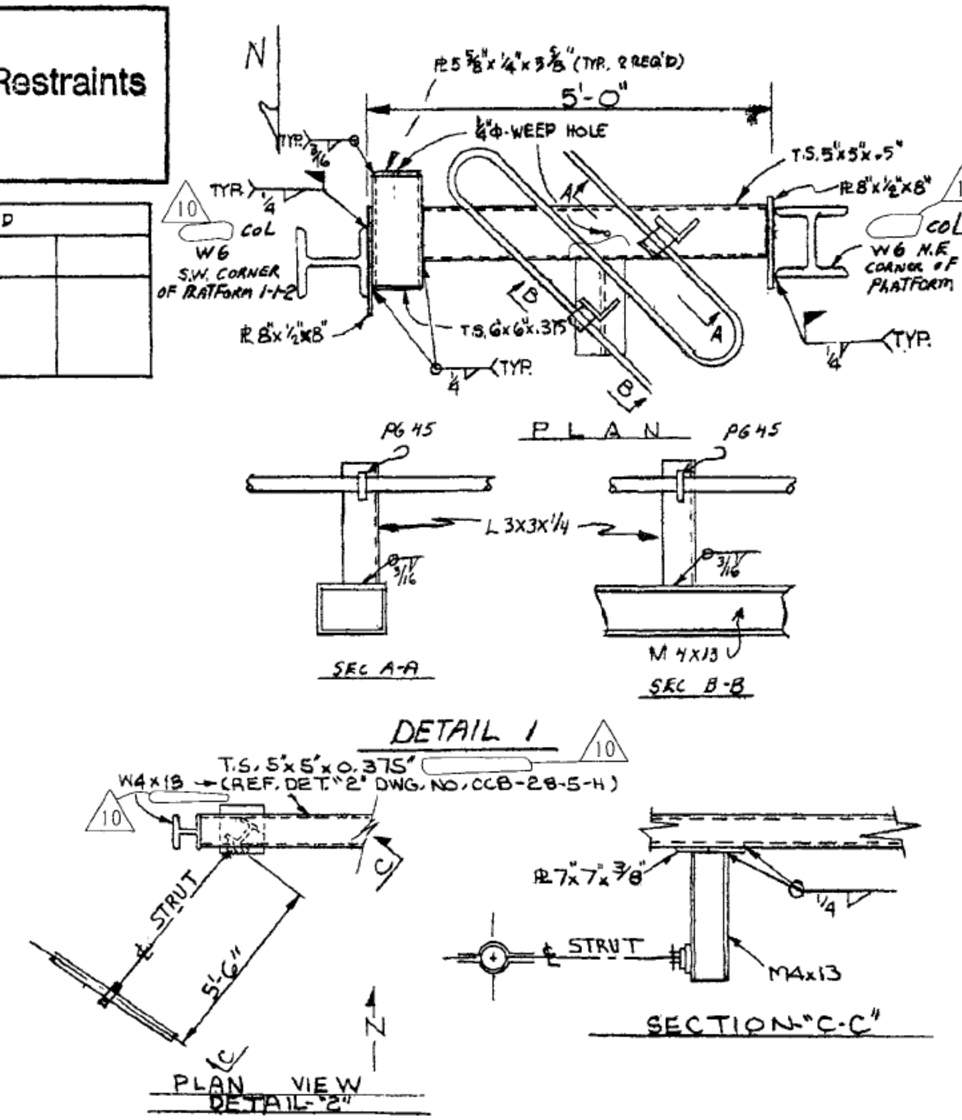
☒ Thermal Anchors

BY MW DATE 1-29-76

SYMBOL LEGEND				
POSTULATED PIPE BREAK				
□				





NOTES:

1. THIS DRAWING IS ONLY FOR HANGER LOCATION IT MAY NOT NECESSARILY DEPICT THE EXACT ROUTING OF THE PIPING. SEE LAYOUT DRAWING FOR CORRECT PIPE ROUTING.
2. STRESS ANALYSIS PROBLEM 12B-1.
3. FOR AS-BUILT CONDITIONS RELATING TO SUPPORTS ON THIS DRAWING REFER TO NCR IDENTIFIED ADJACENT TO THE SUPPORT.

START UP NO. 63A	COST CODE: 2276	INSULATION CLASS. THICK:
WELDING QC/QA REQ'MENTS		REFERENCE
WELDING PROCEDURE PRE WELD-INSP WELDING CONTROL ROD. HELIC. ARC ROD STICK ROD. PRE-HEAT F° N.D.T.: DYE PENE □ X-RAY □ MAX. INTER: INTER PURGE:		P&ID NO M-007 (16) PIPING DWG. M-207A AREA No 9 ELEV 575' TO 576' UNIT No ONE. BLDG: CONTAINMENT 2454 PIPING (7) SPDCN # 2073, 2342
		HANGER TYPE: DWG. NO..

LINE SPEC.: EBB APP. CODE: ASME SECT. III CLASS: NUCLEAR II Q.C. SYSTEM NO.: 3.1255	NOTE: DISMANTLE DIAPHRAGM VALVES BEFORE WELDING. 1/16" MIN. GAP REQUIRED BETWEEN SOCKET WELD JOINTS.

ACCEPTANCE DATES	PIPE SYS.	BY	HANGERS	BY.

DAVIS-BESSE NUCLEAR POWER STATION

Q

SECONDARY SIDE DRAIN PIPING FROM STEAM GENERATOR E24-1 NOZZLE (13) TO 4" DRAIN HEADER

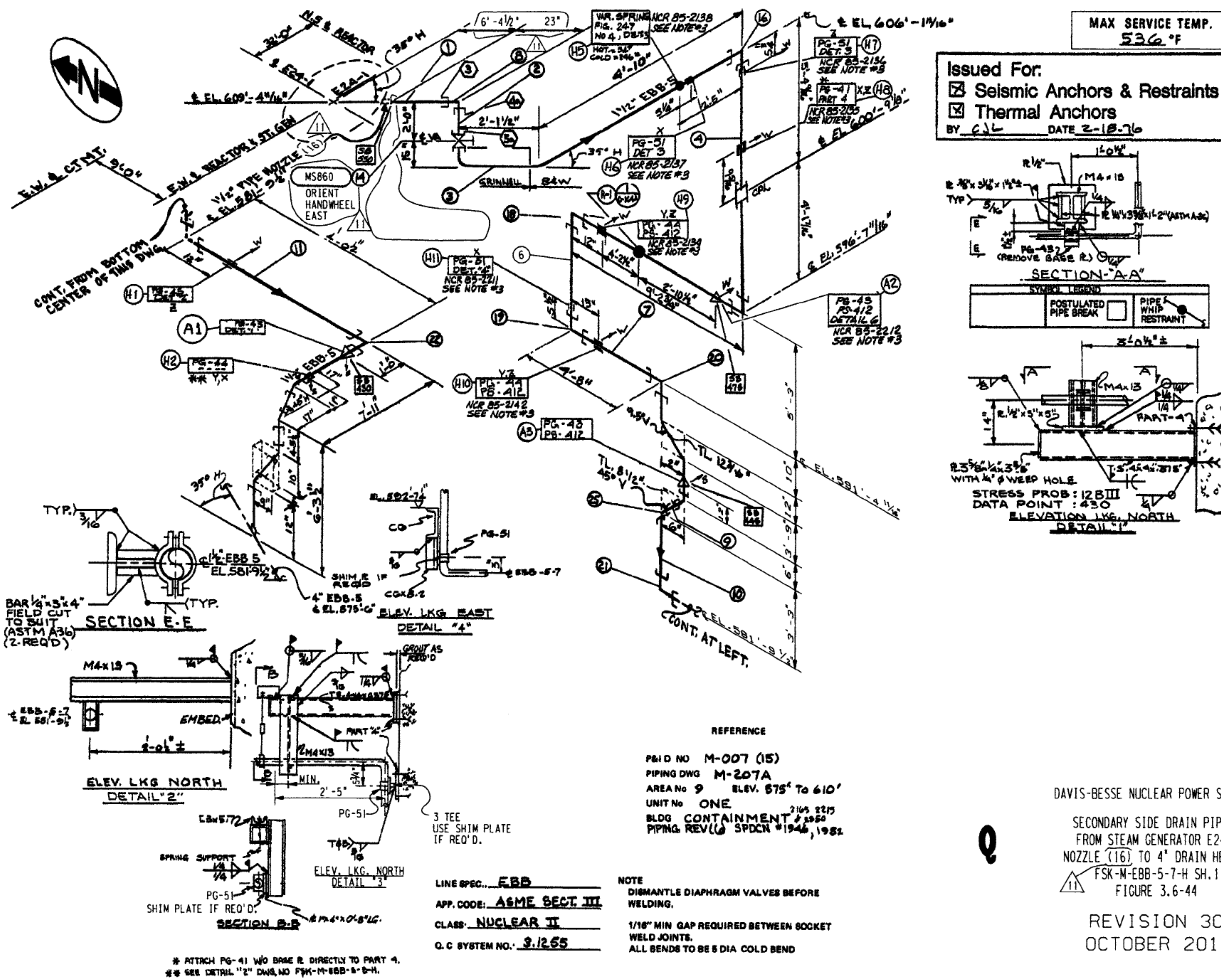
FSK-M-EBB-5-4-H

FIGURE 3.6-43

REVISION 30

OCTOBER 2014

DB 10-08-14 DFN: H/PLDSGN/EBB54H.DGN/TIF



MAX SERVICE TEMP. 536 °F

Issued For:
☒ Seismic Anchors & Restraints
☒ Thermal Anchors
BY CJL DATE 2-18-76

REFERENCE

P&ID NO M-007 (15)
PIPING DWG M-207A
AREA No 9 ELEV. 575' To 610'
UNIT No ONE
BLDG CONTAINMENT
PIPING REV(6) SPDCN #1946, 1982

LINE SPEC.: EBB
APP. CODE: ASME SECT. III
CLASS: NUCLEAR II
Q.C SYSTEM NO.: 3.1265

NOTE
DISMANTLE DIAPHRAGM VALVES BEFORE WELDING.
1/16" MIN GAP REQUIRED BETWEEN SOCKET WELD JOINTS.
ALL BENDS TO BE 5 DIA COLD BEND

* ATTACH PG-41 W/O BASE PL DIRECTLY TO PART 4.
** SEE DETAIL "2" DWG NO FSK-M-EBB-5-8-H.

Q

DAVIS-BESSE NUCLEAR POWER STATION

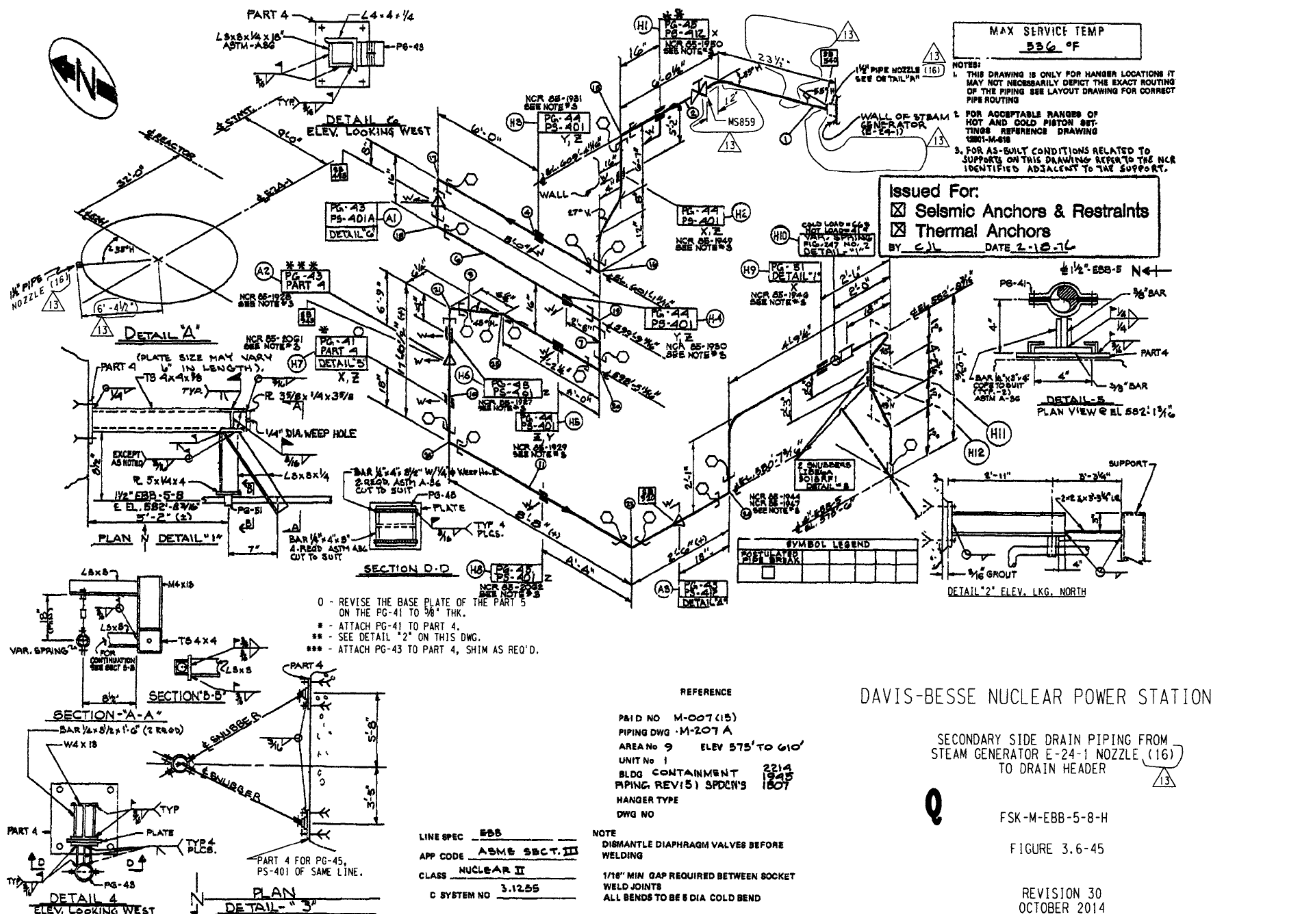
SECONDARY SIDE DRAIN PIPING FROM STEAM GENERATOR E24-1 NOZZLE (16) TO 4" DRAIN HEADER

FSK-M-EBB-5-7-H SH. 1

FIGURE 3.6-44

REVISION 30
OCTOBER 2014

DB 07-22-14 DFN: H/PLDSGN/EBB57H.DGN/TIF



DAVIS-BESSE NUCLEAR POWER STATION

SECONDARY SIDE DRAIN PIPING FROM STEAM GENERATOR E-24-1 NOZZLE (16) TO DRAIN HEADER

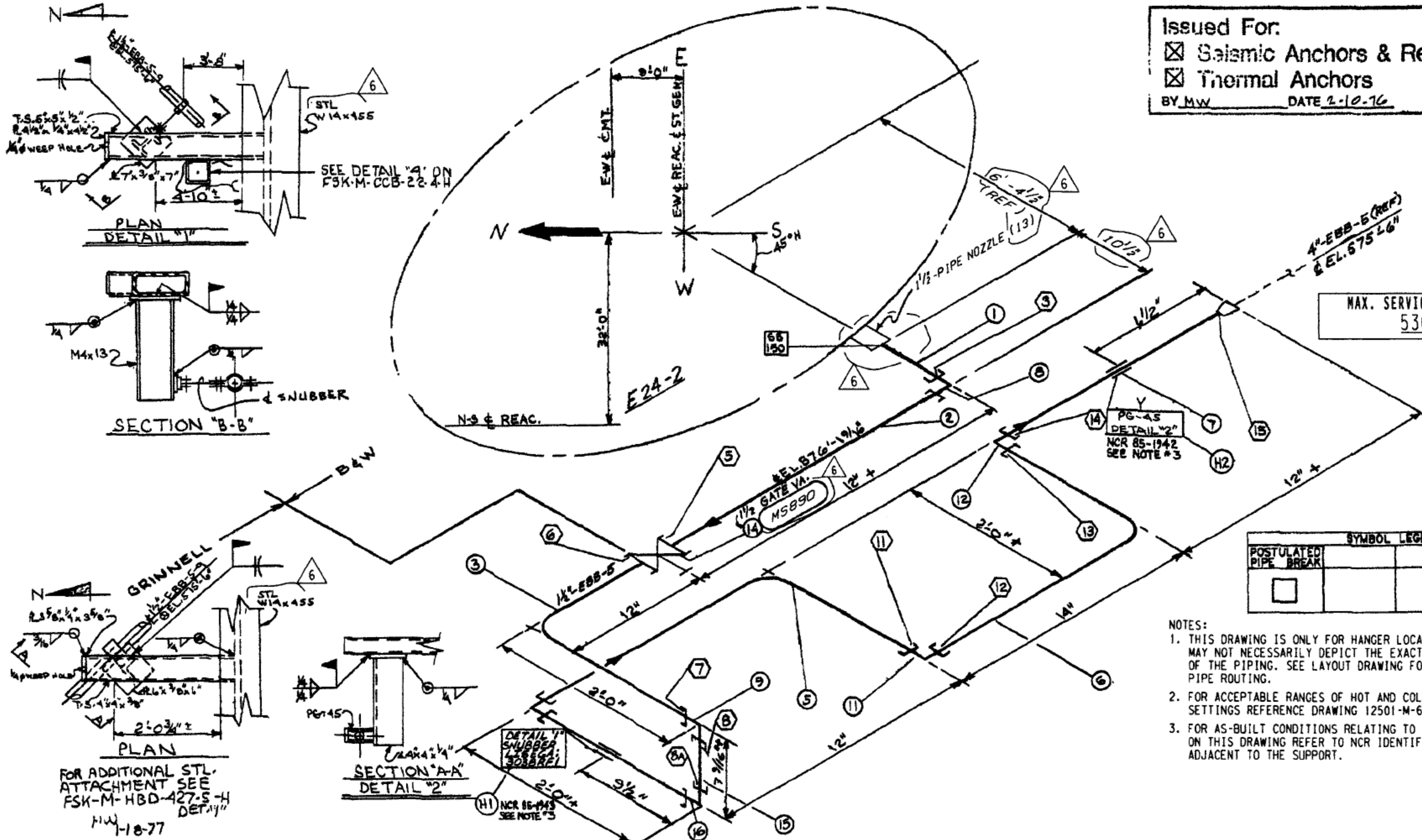
Q

FSK-M-EBB-5-8-H

FIGURE 3.6-45

REVISION 30
OCTOBER 2014

DB 07-22-14 DFN: H/PLDSGN/EBB58H.DGN/TIF



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☒ Thermal Anchors
BY MW DATE 2-10-76

MAX. SERVICE TEMP.
536° F

SYMBOL LEGEND			
POSTULATED PIPE BREAK			
☐			

NOTES:
1. THIS DRAWING IS ONLY FOR HANGER LOCATION IT MAY NOT NECESSARILY DEPICT THE EXACT ROUTING OF THE PIPING. SEE LAYOUT DRAWING FOR CORRECT PIPE ROUTING.
2. FOR ACCEPTABLE RANGES OF HOT AND COLD PISTON SETTINGS REFERENCE DRAWING 12501-M-618.
3. FOR AS-BUILT CONDITIONS RELATING TO SUPPORTS ON THIS DRAWING REFER TO NCR IDENTIFIED ADJACENT TO THE SUPPORT.

This bill of materials may not reflect as-built material requirements. Refer to the FCR identified in the revision block for current material requirements and specifications for the modified portions of this piping system.

START UP NO. 63 | COST CODE: 2276 | INSULATION CLASS: | THICK:

REFERENCE
P&I.D. NO.: M-007 (7)
PIPING DWG.: M-207A
AREA No. 9 ELEV. 575' TO 576'
UNIT No.: 1
BLDG.: CONTAINMENT
PIPING REV: (4)
HANGER TYPE:
DWG. NO.:

LINE SPEC.: EBB
APP. CODE: ASME SECT. III
CLASS: NUCLEAR II
Q.C. SYSTEM NO.: 31255

NOTE:
DISMANTLE DIAPHRAGM VALVES BEFORE WELDING.
1/16" MIN. GAP REQUIRED BETWEEN SOCKET WELD JOINTS.

ACCEPTANCE DATES | PIPE SYS. | BY: | HANGERS | BY:

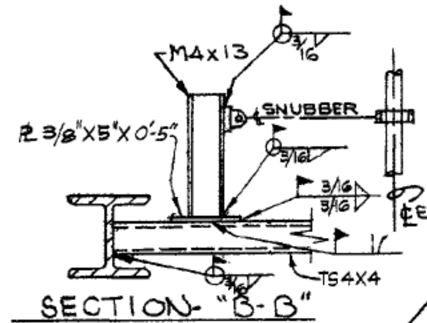
DAVIS-BESSE NUCLEAR POWER STATION

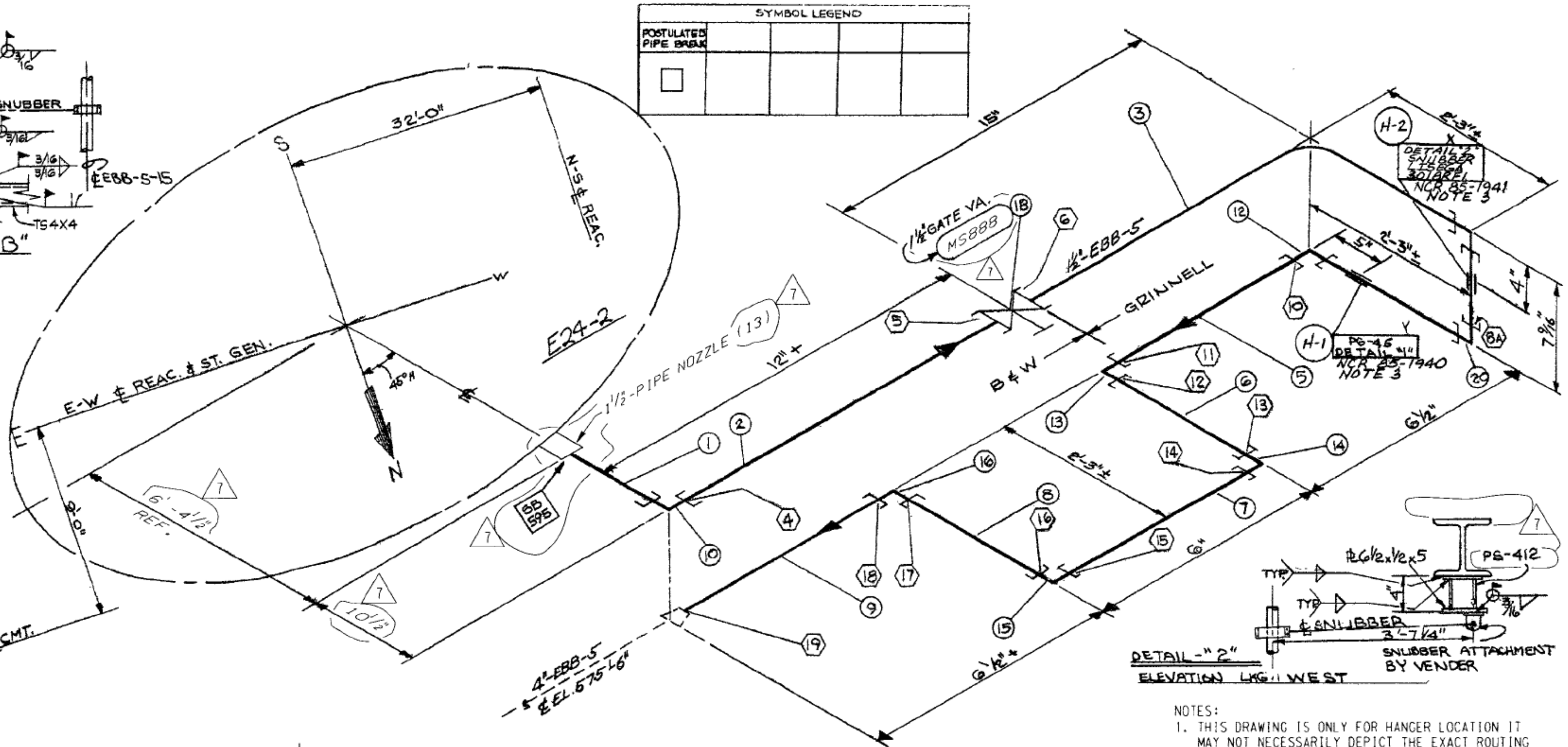
Q

SECONDARY SIDE DRAIN PIPING FROM STEAM GENERATOR E24-2 NOZZLE (13) TO DRAIN HEADER
FSK-M-EBB-5-9-H
FIGURE 3.6-46

REVISION 30
OCTOBER 2014

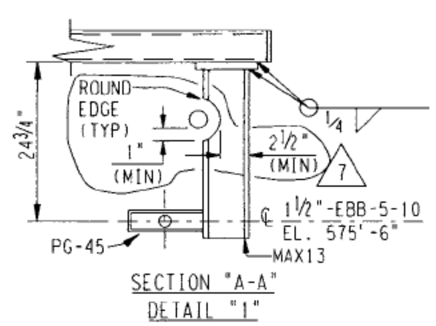
DB 07-22-14 DFN: H/PLDSGN/EBB59H.DGN/TIF





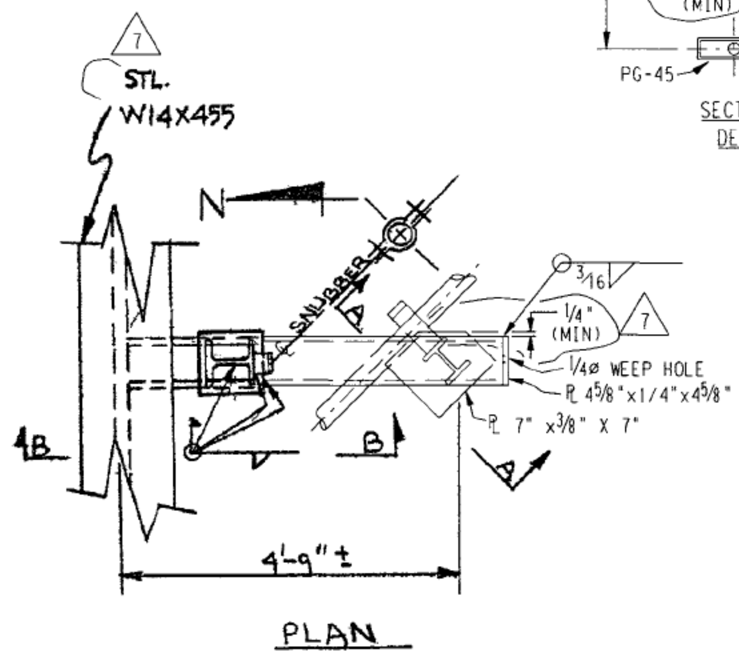
NOTES:

1. THIS DRAWING IS ONLY FOR HANGER LOCATION IT MAY NOT NECESSARILY DEPICT THE EXACT ROUTING OF THE PIPING. SEE LAYOUT DRAWING FOR CORRECT PIPE ROUTING.
2. FOR ACCEPTABLE RANGES OF HOT AND COLD PISTON SETTINGS REFERENCE DRAWING 12501-M-618.
3. FOR AS-BUILT CONDITIONS RELATING TO SUPPORTS ON THIS DRAWING REFER TO NCR IDENTIFIED ADJACENT TO THE SUPPORT.



MAX. SERVICE TEMP.
536° F

Issued For:
☒ Seismic Anchors & Restraints
☒ Thermal Anchors
BY CJL DATE 2-10-76



START UP NO. 63	COST CODE: 2276	INSULATION CLASS:	THICK:

WELDING QC/QA REQ'MENTS	REFERENCE
WELDING PROCEDURE	P&I.D. NO.: M-007 (7)
PRE WELD INSP.	PIPING DWG.: M-207A
WELDING CONTROL ROD:	AREA No 9 ELEV. 575' TO 576'
HELIC ARC ROD.	UNIT No.: 1
STICK ROD	BLDG.: CONTAINMENT
PRE-HEAT °F:	PIPING REV. (4)
N.D.T.: DYE PENE ☐ X-RAY ☐	HANGER TYPE:
MAX INTER: INTER PURGE:	DWG. NO.:

LINE SPEC.: EBB
APP. CODE: ASME SECT. III
CLASS: NUCLEAR II
Q.C. SYSTEM NO.: 31255

NOTE:
DISMANTLE DIAPHRAGM VALVES BEFORE WELDING.
1/16" MIN. GAP REQUIRED BETWEEN SOCKET WELD JOINTS.

ACCEPTANCE DATES	PIPE SYS	BY.	HANGERS	BY.

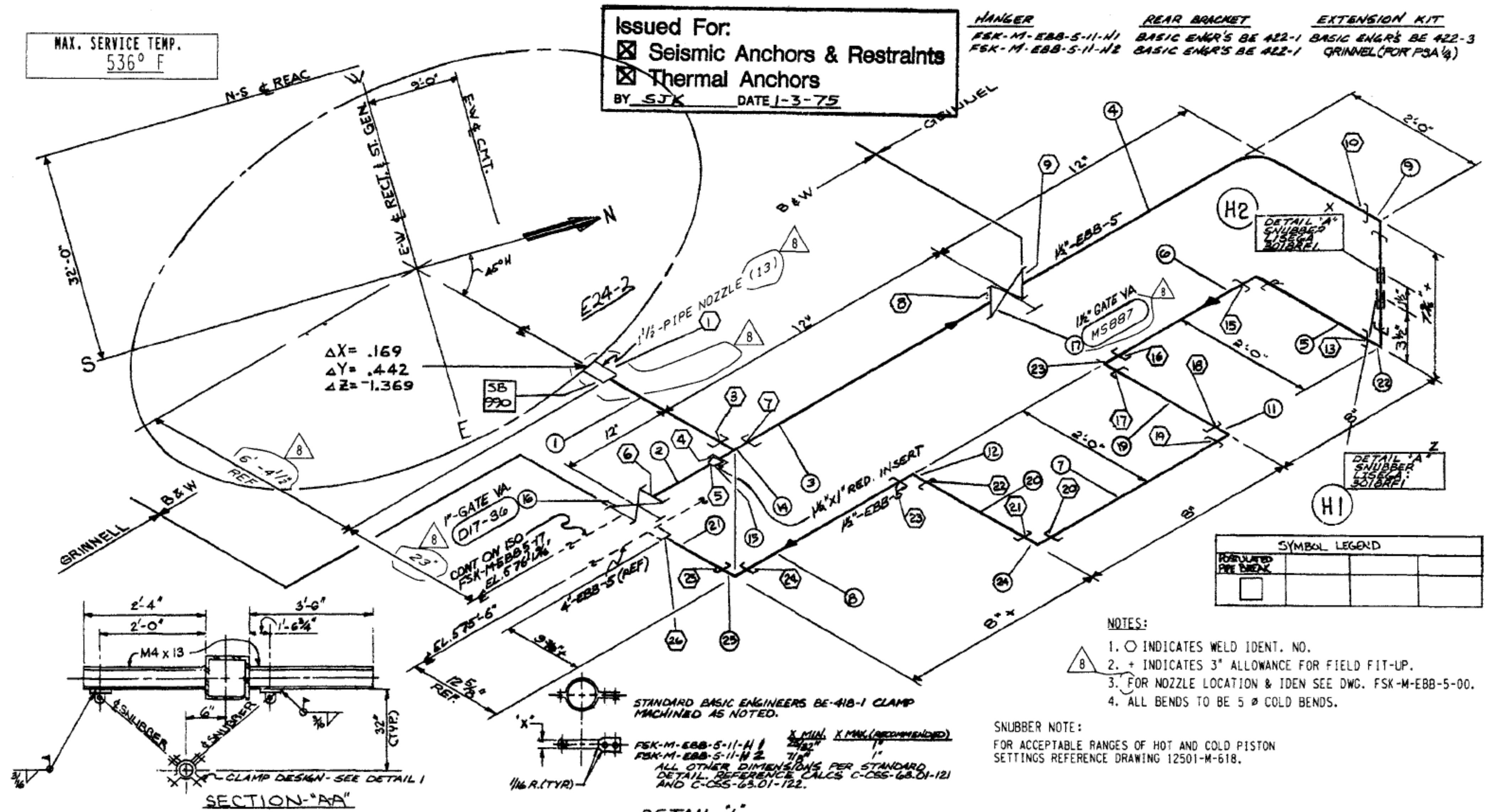
DAVIS-BESSE NUCLEAR POWER STATION

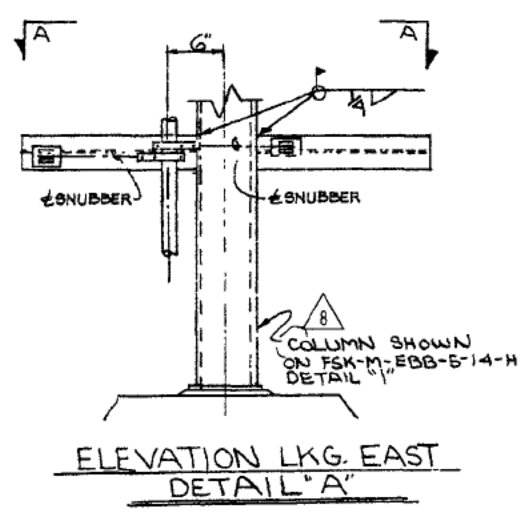
Q

SECONDARY SIDE DRAIN PIPING
FROM STEAM GENERATOR E24-2
NOZZLE (13) TO DRAIN HEADER
FSK-M-EBB-5-10-H
FIGURE 3.6-47

REVISION 30
OCTOBER 2014

DB 10-08-14 DFN: H/PLDSGN/EBB510H.DGN/TIF





START UP NO 50	COST CODE 2276	INSULATION CLASS	THICK
WELDING QC/QA REQ'MENTS		REFERENCE	
WELDING PROCEDURE		P&ID NO: M-007 (7)	
PRE WELD-INSP		PIPING DWG M-207A	
WELDING CONTROL ROD		AREA No 9 ELEV 575' TO 576'	
HELIC ARC ROD		UNIT No 1	
STICK ROD		BLDG CONTAINMENT	
PRE-HEAT F°		PIPING REV (4)	
NDT DYE PENE ☐ X-RAY ☐		HANGER TYPE	
MAX INTER INTER PURGE		DWG NO	
LINE SPEC.: EBB		NOTE. DISMANTLE DIAPHRAGM VALVES BEFORE WELDING.	
APP. CODE: ASME SECT III		1/16" MIN. GAP REQUIRED BETWEEN SOCKET WELD JOINTS.	
CLASS. NUCLEAR II			
Q. C. SYSTEM NO. 31255			
ACCEPTANCE DATES	PIPE SYS BY	HANGERS BY	

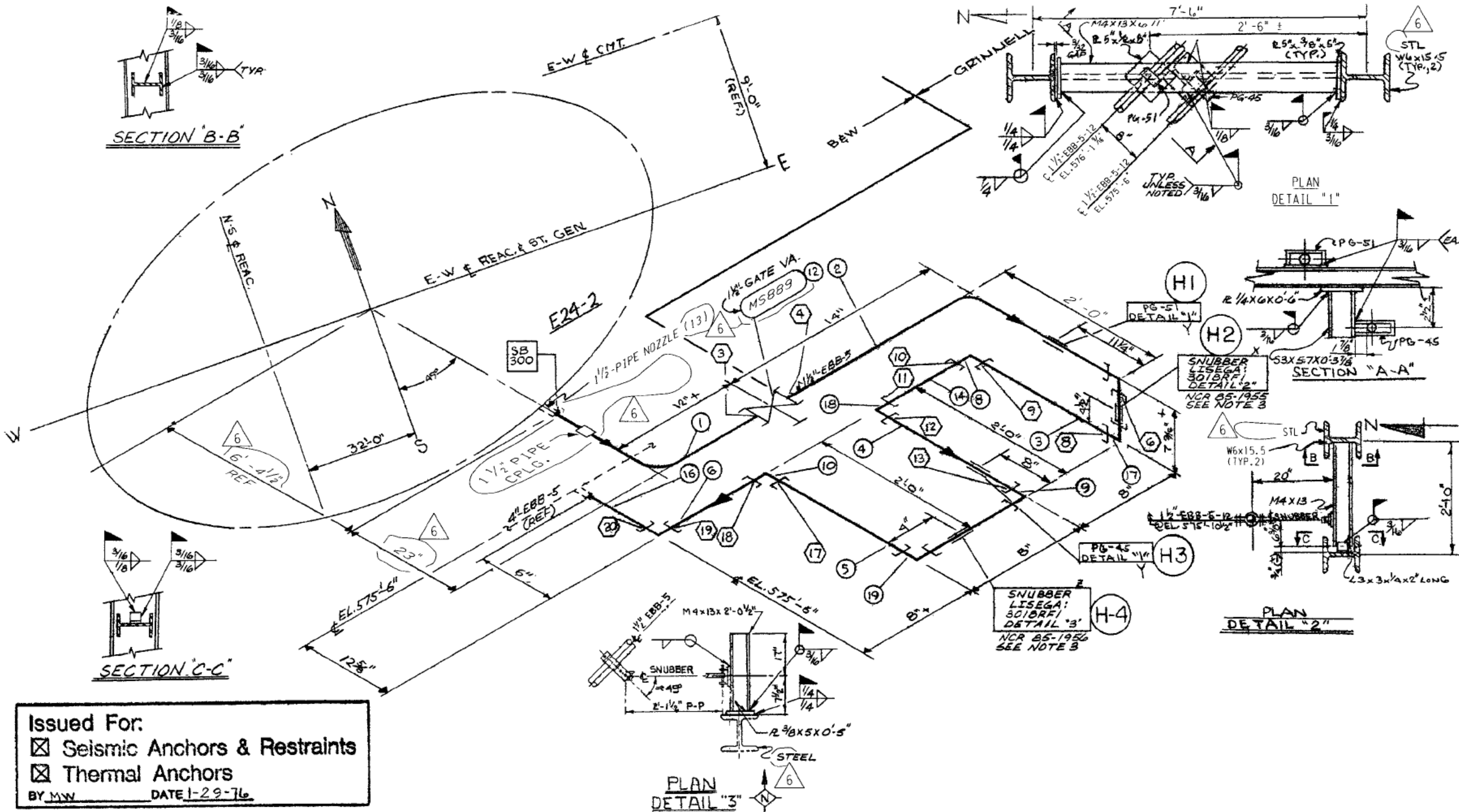
Q

DAVIS-BESSE NUCLEAR POWER STATION

SECONDARY SIDE DRAIN PIPING
FROM STEAM GENERATOR E24-2
NOZZLE (13) TO DRAIN HEADER
FSK-M-EBB-5-11-H
FIGURE 3.6-48

REVISION 30
OCTOBER 2014

DB 07-22-14 DFN: H/PLDSGN/EBB511H.DGN/TIF



Issued For:

☒ Seismic Anchors & Restraints

☒ Thermal Anchors

BY MW DATE 1-29-76

SYMBOL LEGEND				
POSTULATED PIPE BREAK				
☐				

MAX. SERVICE TEMP.
536° F

NOTES:

1. THIS DRAWING IS ONLY FOR HANGER LOCATION IT MAY NOT NECESSARILY DEPICT THE EXACT ROUTING OF THE PIPING. SEE LAYOUT DRAWING FOR CORRECT PIPE ROUTING.
2. FOR ACCEPTABLE RANGES OF HOT AND COLD PISTON SETTINGS REFERENCE DRAWING 12501-M-618.
3. FOR AS-BUILT CONDITIONS RELATING TO SUPPORTS ON THIS DRAWING REFER TO NCR IDENTIFIED ADJACENT TO THE SUPPORT.

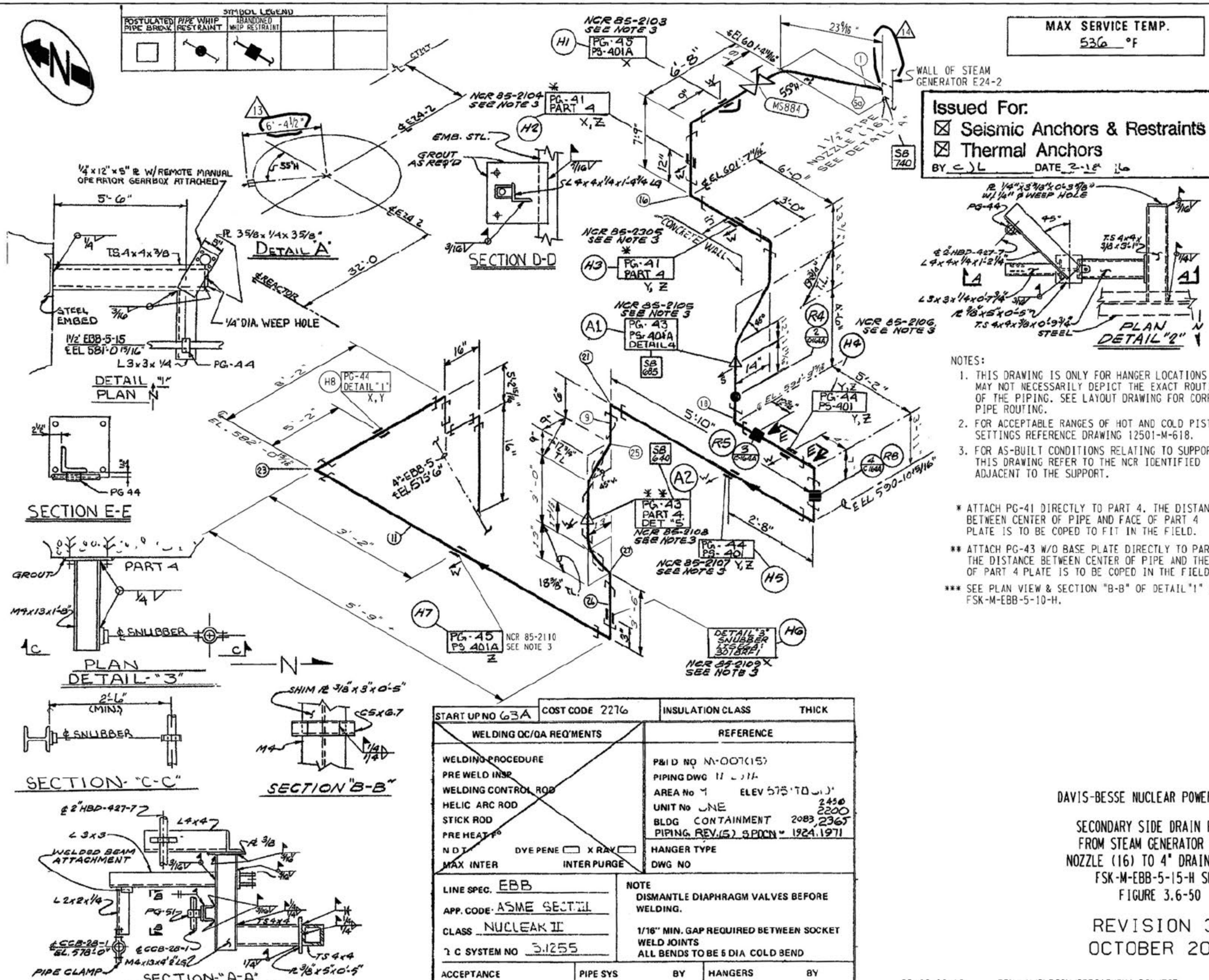
START UP NO. 63	COST CODE	INSULATION CLASS.	THICK
WELDING QC/QA REQ'MENTS		REFERENCE	
WELDING PROCEDURE		P&I.D. NO. M-007 (9)	
PRE WELD INSP.		PIPING DWG. M-207A	
WELDING CONTROL ROD		AREA No. 9	ELEV. 575' TO 576'
HELIC ARC ROD		UNIT No. 1	
STICK ROD		BLDG. CONTAINMENT	
PRE-HEAT F°:		PIPING REV. (3) SPDCN 1999	
N.D.T.	DYE PENE. X-RAY	HANGER TYPE	
MAX. INTER:	INTER PURGE	DWG. NO.	
LINE SPEC. EBB		NOTE	
APP. CODE: ASME SECT. III		DISMANTLE DIAPHRAGM VALVES BEFORE WELDING.	
CLASS: NUCLEAR II		1/16" MIN. GAP REQUIRED BETWEEN SOCKET WELD JOINTS.	
Q.C. SYSTEM NO. 31255			
ACCEPTANCE DATES	PIPE SYS	BY	HANGERS

DAVIS-BESSE NUCLEAR POWER STATION

Q SECONDARY SIDE DRAIN PIPING FROM STEAM GENERATOR E24-2 NOZZLE (13) TO DRAIN HEADER
FSK-M-EBB-5-12-H
FIGURE 3.6-49

REVISION 30
OCTOBER 2014

DB 10-08-14 DFN: H/PLDSGN/EBB512H.DGN/TIF



MAX SERVICE TEMP.
536 °F

Issued For:
☒ Seismic Anchors & Restraints
☒ Thermal Anchors
BY C)L DATE 2-18-16

NOTES:

1. THIS DRAWING IS ONLY FOR HANGER LOCATIONS IT MAY NOT NECESSARILY DEPICT THE EXACT ROUTING OF THE PIPING. SEE LAYOUT DRAWING FOR CORRECT PIPE ROUTING.
2. FOR ACCEPTABLE RANGES OF HOT AND COLD PISTON SETTINGS REFERENCE DRAWING 12501-M-618.
3. FOR AS-BUILT CONDITIONS RELATING TO SUPPORTS ON THIS DRAWING REFER TO THE NCR IDENTIFIED ADJACENT TO THE SUPPORT.

* ATTACH PG-41 DIRECTLY TO PART 4. THE DISTANCE BETWEEN CENTER OF PIPE AND FACE OF PART 4 PLATE IS TO BE COPED TO FIT IN THE FIELD.

** ATTACH PG-43 W/O BASE PLATE DIRECTLY TO PART 4. THE DISTANCE BETWEEN CENTER OF PIPE AND THE FACE OF PART 4 PLATE IS TO BE COPED IN THE FIELD.

*** SEE PLAN VIEW & SECTION "B-B" OF DETAIL "1" ON FSK-M-EBB-5-10-H.

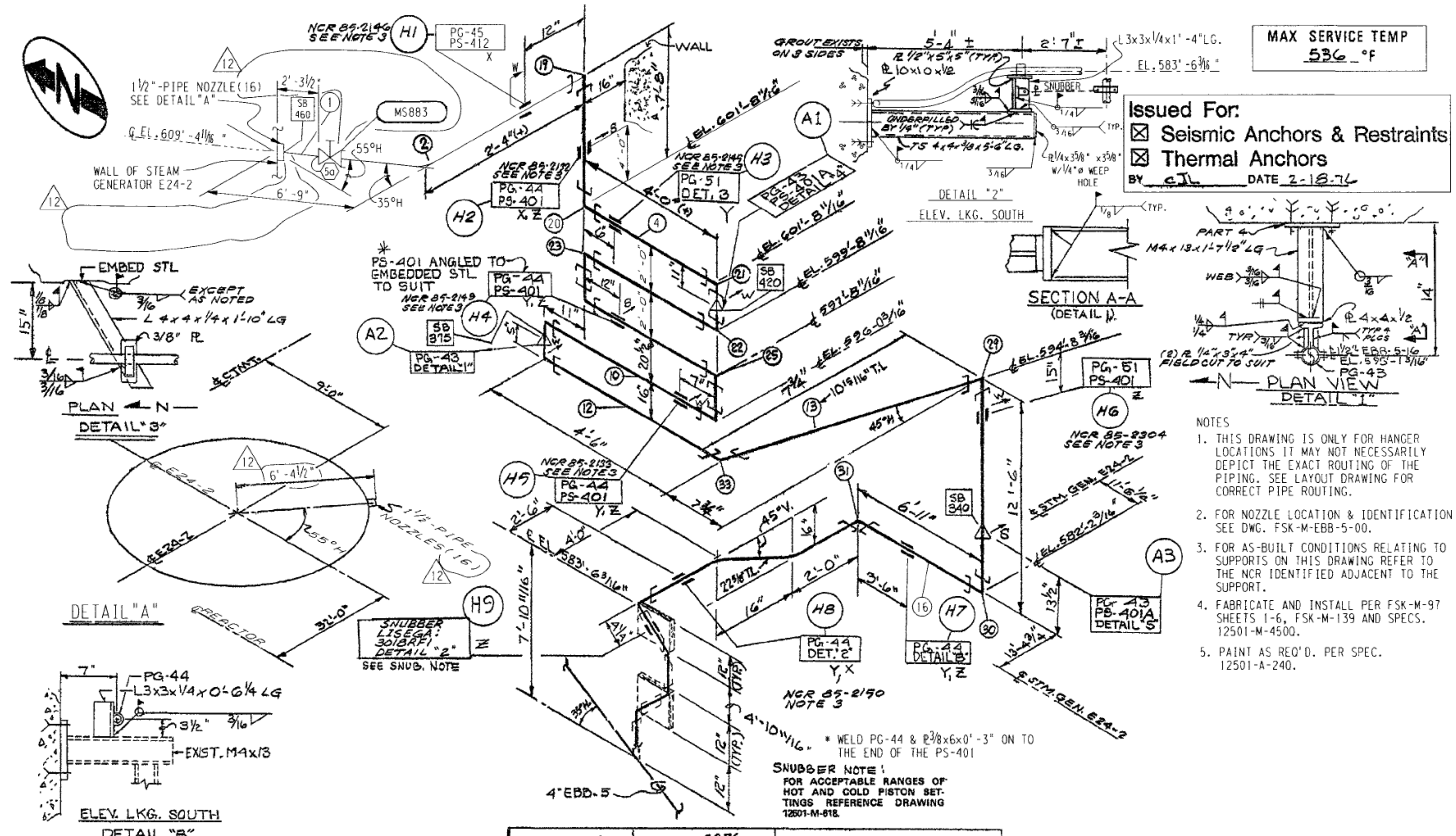
START UP NO 63A	COST CODE 2276	INSULATION CLASS	THICK
WELDING QC/QA REQ'MENTS		REFERENCE	
WELDING PROCEDURE / PRE WELD INSP / WELDING CONTROL ROD / HELIC ARC ROD / STICK ROD / PRE HEAT F° / NDT DYE PENE ☐ X RAY ☐ / MAX INTER INTER PURGE		P&ID NO M-007(15) / PIPING DWG / AREA No 4 ELEV 575'TO / UNIT No ONE / BLDG CONTAINMENT 2083, 2450, 2200, 2365 / PIPING REV.(5) SPDCN 1924, 1971	
		HANGER TYPE / DWG NO	
LINE SPEC. EBB / APP. CODE ASME SECT III / CLASS NUCLEAR II / Q C SYSTEM NO 3.1255		NOTE: DISMANTLE DIAPHRAGM VALVES BEFORE WELDING. 1/16" MIN. GAP REQUIRED BETWEEN SOCKET WELD JOINTS. ALL BENDS TO BE 5 DIA COLD BEND	
ACCEPTANCE DATES	PIPE SYS BY	HANGERS BY	

DAVIS-BESSE NUCLEAR POWER STATION

SECONDARY SIDE DRAIN PIPING
FROM STEAM GENERATOR E24-2
NOZZLE (16) TO 4" DRAIN HEADER
FSK-M-EBB-5-15-H SH.1
FIGURE 3.6-50

REVISION 31
OCTOBER 2016

DB 06-06-16 DFN: H/PLDSGN/EBB515HSH1.DGN/TIF



NOTES

1. THIS DRAWING IS ONLY FOR HANGER LOCATIONS IT MAY NOT NECESSARILY DEPICT THE EXACT ROUTING OF THE PIPING. SEE LAYOUT DRAWING FOR CORRECT PIPE ROUTING.
2. FOR NOZZLE LOCATION & IDENTIFICATION SEE DWG. FSK-M-EBB-5-00.
3. FOR AS-BUILT CONDITIONS RELATING TO SUPPORTS ON THIS DRAWING REFER TO THE NCR IDENTIFIED ADJACENT TO THE SUPPORT.
4. FABRICATE AND INSTALL PER FSK-M-97 SHEETS 1-6, FSK-M-139 AND SPECS. 12501-M-450Q.
5. PAINT AS REQ'D. PER SPEC. 12501-A-240.

* WELD PG-44 & ℞3/8x6x0'-3" ON TO THE END OF THE PS-401

SNUBBER NOTE:
FOR ACCEPTABLE RANGES OF HOT AND COLD PISTON SETTINGS REFERENCE DRAWING 12501-M-618.

START UP NO. 63	COST CODE 2276	INSULATION CLASS: THICK:
WELDING QC/QA REQ'MENTS		REFERENCE
WELDING PROCEDURE PRE WELD-INSP.: WELDING CONTROL ROD. HELIC. ARC ROD: STICK ROD: PRE-HEAT °F N D T.: DYE PENE ☐ X-RAY ☐ MAX. INTER: INTER PURGE:		P&I D. NO: M-007 (15) PIPING DWG M-207A AREA No. 9 ELEV 575' TO 610' UNIT No. 1 BLDG: CONTAINMENT 2361 PIPING REV. (5) SPEC N 1972 HANGER TYPE. DWG. NO.
LINE SPEC.: EBB APP. CODE: ASME SECT. III CLASS: NUCLEAR II Q.C. SYSTEM NO.: 3.1255		NOTE: DISMANTLE DIAPHRAGM VALVES BEFORE WELDING. 1/16" MIN. GAP REQUIRED BETWEEN SOCKET WELD JOINTS, ALL BENDS TO BE 5 DIA. COLD BEND.
ACCEPTANCE DATES	PIPE SYS BY.	HANGERS BY.

SYMBOL LEGEND				
POSTULATED PIPE BREAK				
☐				

Q

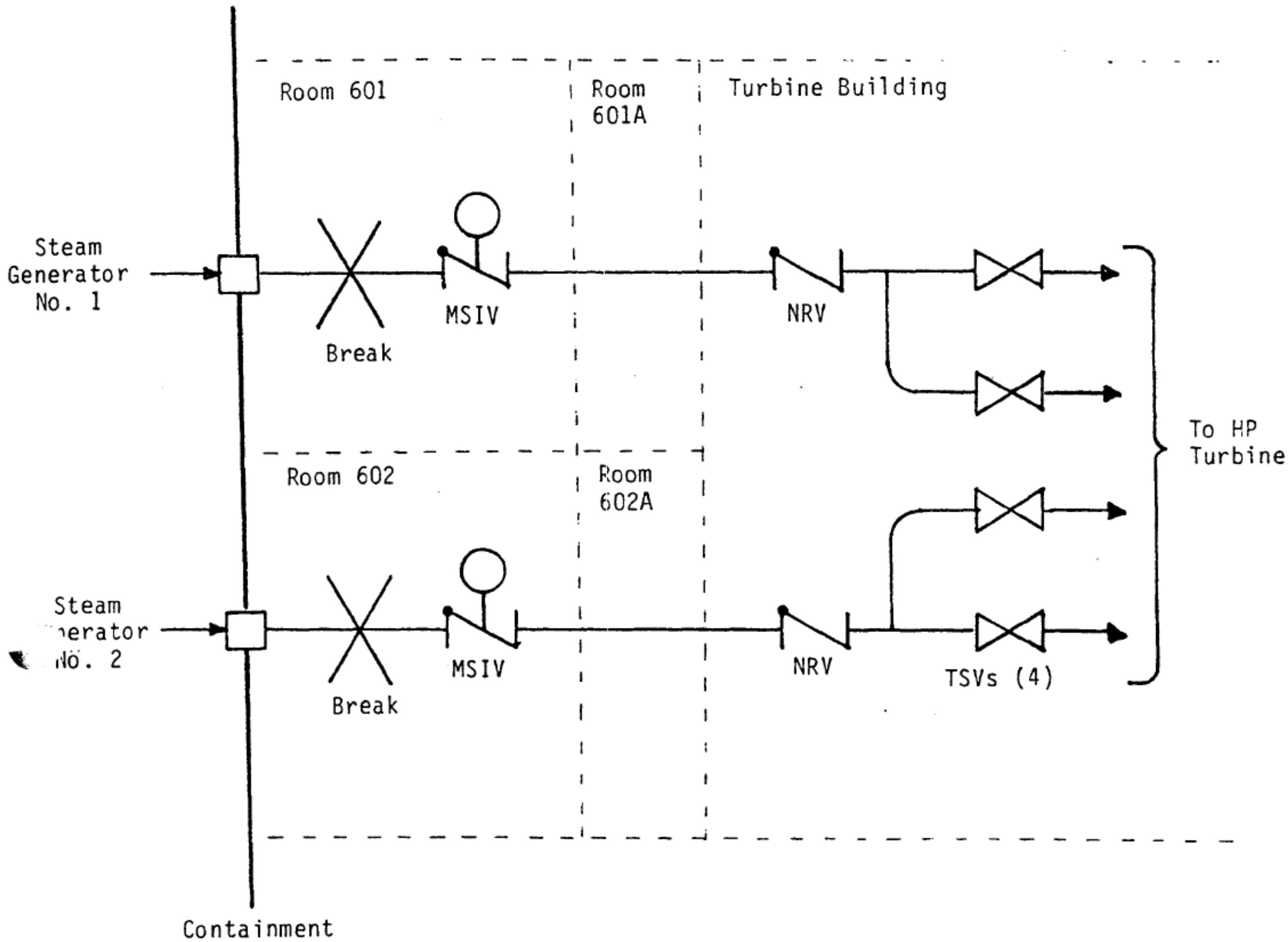
DAVIS-BESSE NUCLEAR POWER STATION

SECONDARY SIDE DRAIN PIPING FROM STEAM GENERATOR E24-2 NOZZLE (16) TO DRAIN HEADER
FSK-M-EBB-5-16-H, SH. 1

FIGURE 3.6-51

REVISION 30
OCTOBER 2014

DB 10-08-14 DFN: H/PLDSGN/EBB516HSH1.DGN/TIF



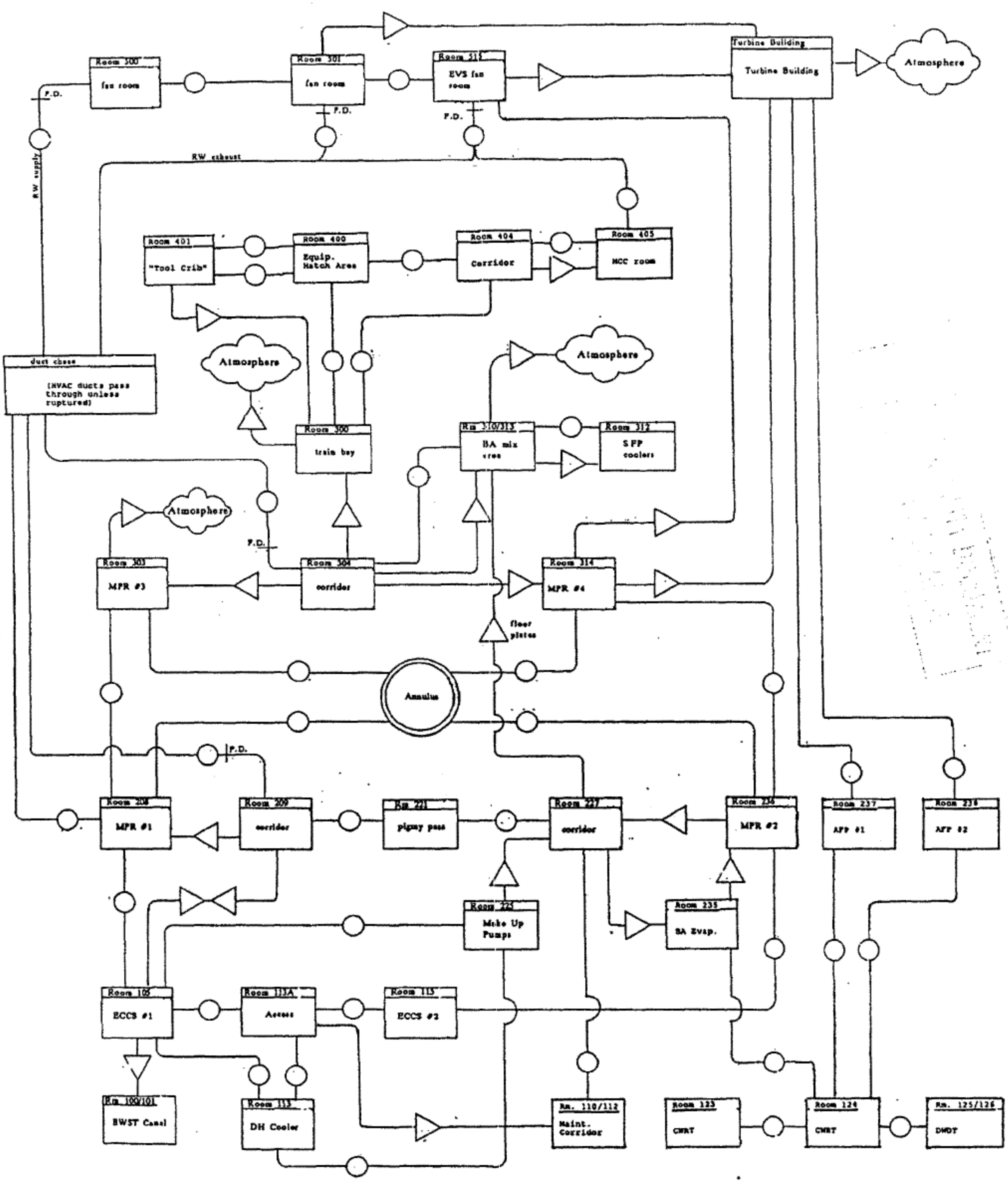
DAVIS-BESSE NUCLEAR POWER STATION

MAIN STEAM LINES FLOW AND ROUTING SCHEMATIC

FIGURE 3.6-53

REVISION 5

JULY 1987



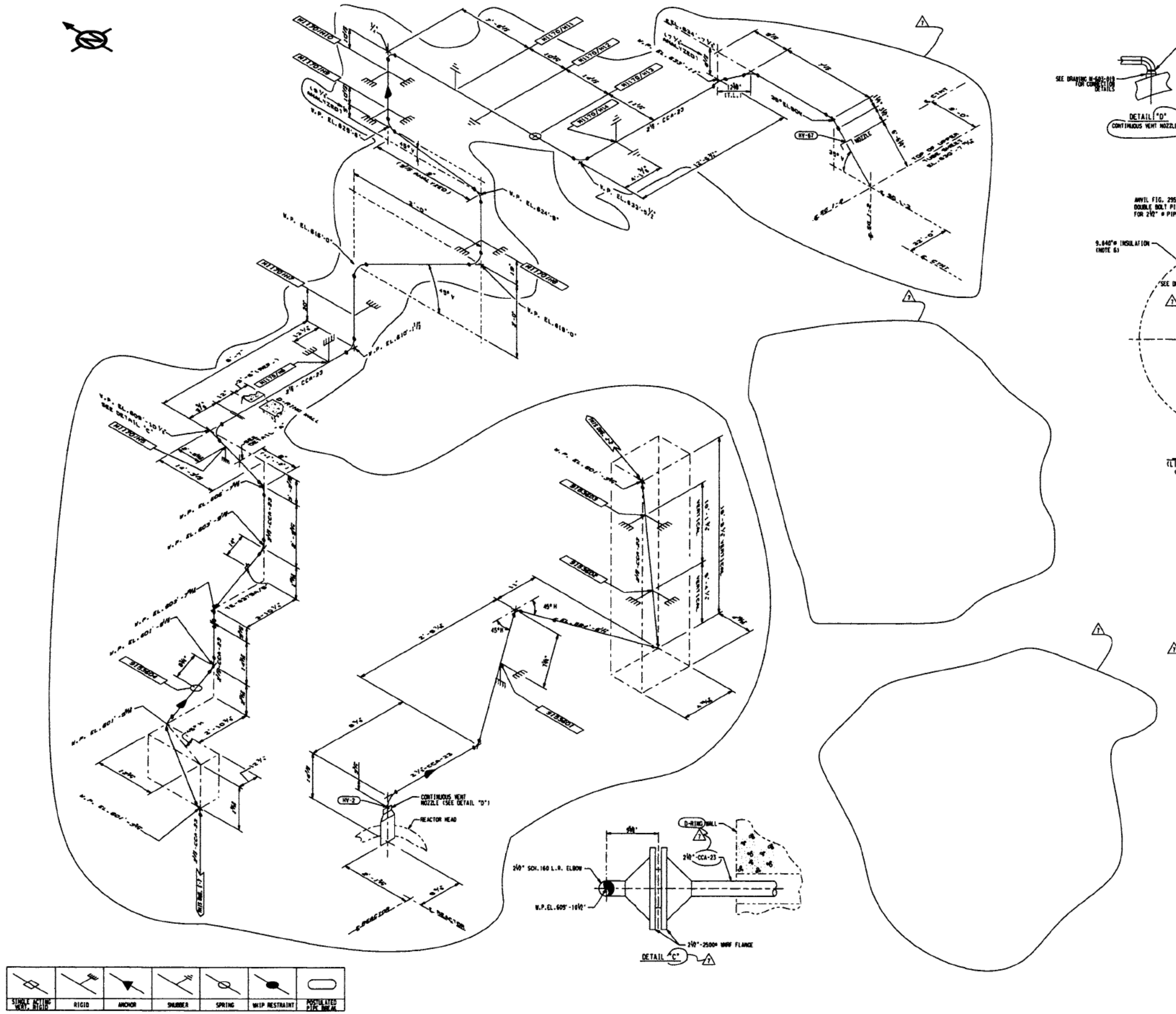
This diagram represents an overall connection diagram from which specific break models are taken. For specific details and HVAC modelling refer to applicable calculations.

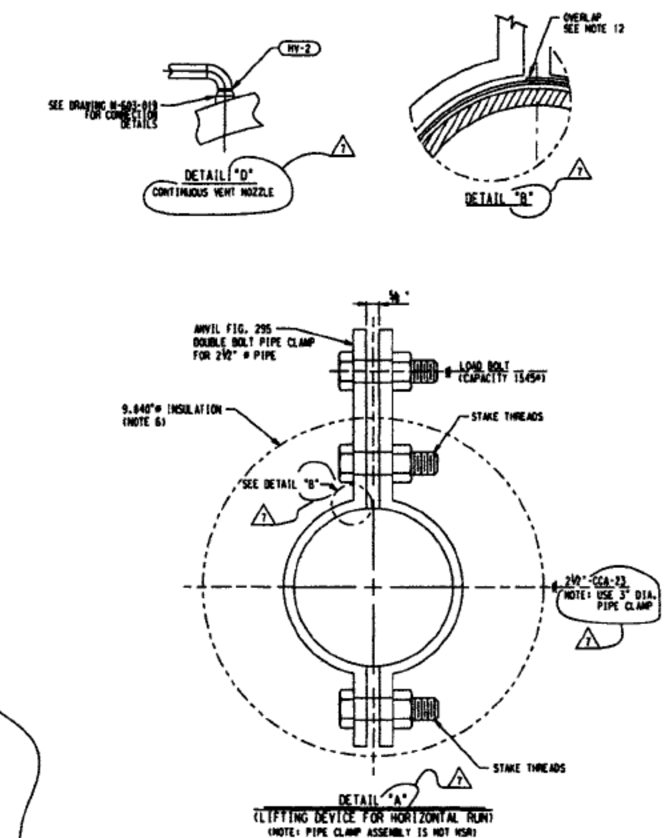
◯ - Open Flow Areas

◁ - Doors or Blowout Panels

DAVIS-BESSE NUCLEAR POWER STATION
AUXILIARY BUILDING FLOW MODEL
FIGURE 3.6-56

REVISION 18
NOVEMBER 1993





NOTES:

1. THIS ISOMETRIC IS FOR HANGER LOCATIONS ONLY AND REFLECTS THE DESIGN LOCATION OF HANGERS.
2. FOR PIPING SYSTEM ROUTING REFER TO 12501-M-238D. (TES D-9032)
3. THIS ISOMETRIC WAS GENERATED IN CONJUNCTION WITH ECP 10-0466 AND ECP 10-0470.
4. DELETED
5. ALL PIPE BEND RADII ARE 5D UNLESS NOTED OTHERWISE.
6. INSULATION IS REFLECTIVE METALLIC (TRANSCO) EXCEPT AT THE CORE HOLE, WHICH IS NUKON AND FROM THE FLANGES AT THE D-RING WALL TO THE REACTOR HEAD, WHICH IS DARCHEM.
7. DELETED
8. DELETED
9. DELETED
10. ALL FLANGES ARE TO MATCH INSIDE DIAMETER OF PIPE.
11. ALL PIPING SHOWN ON THIS DRAWING IS SEISMIC CATEGORY I AND Q-LISTED.
12. WRAP STAINLESS STEEL SHIM AROUND PIPE TO PREVENT CARBON STEEL CLAMP FROM CONTACTING PIPE. CUT SHIM TO FIT, OVERLAP AT TOP.

DAVIS-BESSE NUCLEAR POWER STATION
HANGER LOCATION ISOMETRIC R.V.HEAD
TO HOT LEG VENT LINE
CONTAINMENT BUILDING
HL-1170

Q

FIGURE 3.6-56A
REVISION 30
OCTOBER 2014

DB 07-16-14 DFN=H:/PLDSGN/HL1170.DGN