Superseded by Amend. #14

The use of the finite element computer program permitted an accurate estimate of the stress pattern at various locations of the structure. The following material properties were used in the program for the various loading conditions:

	Load Conditions	
	D, F, T_O, T_A	P
$E_{concrete}$, Foundation (Psi)	5.0×10^6	5.0×10^6
$E_{concrete}$, Shell (Psi)	2.7×10^6	5.5×10^6
ν concrete (Poisson's Ratio)	0.17	0.17
α concrete (Coeff of Expansion)	0.5×10^{-5}	-
E_{soil} (Psi)	0.1×10^6	0.1×10^6
E_{liner} (Psi)	30×10^6	30×10^6
f_{yliner} (Psi)	34,000	-

(For definition of Load Conditions, see Appendix B.)

The major benefit of the program is the capability to predict shears and moments due to internal restraint and the interaction of the foundation slab relative to the soil. The structure is analyzed assuming an uncracked homogeneous material. This is conservative because the decreased relative stiffness of a cracked section would result in smaller secondary shears and moments.

In arriving at the above-tabulated values of E, the effect of creep is included by using the following equation for long-term loads such as thermal load, dead load and prestress:

$$E_{cs} = E_{ci} \left(\varepsilon_i / (\varepsilon_s + \varepsilon_i) \right)$$

Where:

E_{cs} = sustained modulus of elasticity of concrete.

E_{ci} = instantaneous modulus of elasticity of concrete.

ε_i = instantaneous strain, in/in per psi.

ε_s = creep strain, in/in per psi.

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CONCRETE DESIGN MIXES

Concrete Design Strength	Cement Sks/Yd	Fly Ash Sks/Yd	Aggregates Proportions by Weight Sand	3/4"	1-1/2"	3"	Water
4000 Psi	4.13	1.37	1420	1960	-	-	234
@ 90 Days	3.94	1.32	1283	1024	1091	-	228
	3.74	1.26	1149	662	761	906	217
5000 Psi	5.32	0.94	1333	1918	-	-	259
@ 28 Days	5.10	0.90	1227	996	1095	-	250

Water Reducing Agent - Walter Flood and Co was engaged to perform the necessary strength and shrinkage tests of various water reducing agents to establish the particular additive with the most desirable characteristics for this application. On the basis of these tests, Pozzolith 8, Improved, manufactured by Master Builders, was selected.

5.1.8 CONTAINMENT TESTING

Tests will be made to determine the initial leak tight and structural integrity of the containment.

5.1.8.1 Leak Tight Integrity Tests

The objectives of these tests are:

(a) To determine the initial integrated leak rate for comparison with the 0.2%/day by volume at 55 psig and 283° F specified as the maximum permissible.

(b) To determine the characteristic leak rate variation with pressure so as to allow retesting at pressures less than design pressure.

(c) To institute a performance history summary for both local leak and integrated leak rate tests.

The guidelines established for the tests are:

(a) The methods and equipment used during the initial tests will be such that they can be used for subsequent retests, thus avoiding test result variations due to changes of the methods or equipment.

2. After final assembly and calibration, the transmitter housing will be leak checked using helium and a mass-spectrometer leak detecting device.

(c) Pressurizer Pressure Transmitters

The pressure transmitters whose output signal is used to generate the low-pressure trip signal for reactor trip and safety injection system actuation will be subjected to test conditions at least as severe as those used for the safety injection valves and high-pressure safety injection flow transmitters.

(d) The following additional equipment has been or will be environmental tested for a 24-hour duration at concurrent 55 psig, 283° F and 100% relative humidity:

1. Containment area radiation monitor for containment isolation.

2. Containment air cooler fan motor unit.

3. Containment air cooler high-capacity service water valve.

4. Auxiliary devices such as solenoid valves associated with correct operation of the above items.

5. Samples of control, power and instrument cable were tested in a gamma pool at DBA temperature conditions to a total dose of 4.0×10^7 R which is twice the total 40-year integrated dose plus DBA. Physical tests in accordance with IPCEA were performed and the insulation met all requirements. The control and power cables were then tested to DBA concurrent temperature, pressure and humidity conditions while carrying rated current. All samples passed the electrical and physical tests in accordance with IPCEA.

6.1.4 DESIGN ANALYSIS

Ability to meet the core protection criteria is assured by the following design features:

(a) A high-capacity passive system which requires no outside source and will supply large quantities of borated water to rapidly recover the core after a major loss-of-coolant accident up to a break of the largest primary coolant line.

(b) A pumping and water storage system with internal redundancy which will inject borated water to provide core protection

10^{-4}% and above 15% power. High rate-of-change of power alarms are initiated at 1.5 dpm over the operating range of 10^{-8}% to 125% power by the two wide-range channels. In addition, manual or automatic rod withdrawal prohibits between ∿10^{-4}% and 15% power from the two wide-range channels prevent all regulating rods from being withdrawn, but do not prevent insertion.

This is an anticipatory trip which is not required to protect the reactor since the primary trip is high power level trip.

7.2.3.2 High Power Level

A reactor trip at high power level (neutron flux) is provided to shut down the reactor when the indicated neutron power approaches an unsafe value. The high power trip signals are initiated by two-out-of-four coincidence logic from the four power range safety channels. During normal plant operation with all coolant pumps operating, reactor trips are initiated when the reactor power level exceeds a nominal value of ∿107% of indicated full power; this trip level represents a reactor power of no greater than 112% of full power when instrument and calorimetric errors are taken into account. Provisions have been made to select different trip points for various combinations of primary coolant pump operation as described in the low primary coolant flow trip section.

The power range channels are equipped with a range change switch to increase the indicated power by a factor of 10. By use of the range change switch, indicated power is increased to provide full-scale indication at 12.5% power. This action also decreases the overpower trip from 107% to 10.7% to provide overpower trip protection during low power operation.

Pretrip alarms are initiated at 10.4% or 104% of indicated full power depending upon adjustment of the range switch. The pretrip alarm signals are initiated by bistable trip units from the same channels which provide the reactor trip signals. The pretrip alarms provide annunciation in addition to rod withdrawal prohibit signals.

7.2.3.3 Low Reactor Coolant Flow

A reactor trip is provided to protect the core from a power to flow mismatch. Provisions are made in the reactor protective system to permit operation at reduced power if one or more coolant pumps are taken out of service. For this mode of operation, the low flow trip points and the overpower trip points are simultaneously changed by a manual switch equipped with channel separation to the allowable values for the selected pump condition, thus providing a positive means of assuring that the more restrictive settings are used. The switching arrangement is shown on Figure 7-4. The switch settings are readily visible to the operator. The flow measurement signals are provided by summing the output of the differential pressure transmitters to provide an indication

of total coolant flow through the reactor. A reactor trip is initiated by two-out-of-four coincidence logic from either of the four independent measuring channels when the flow function falls below a preselected value.

Pretrip alarms are initiated if the coolant flow function approaches the minimum required for reactor operation at the corresponding power level. A key-operated bypass switch allows this trip to be bypassed for subcritical testing of control rod drive mechanisms. The trip bypass is automatically reset above $10^{-4}\%$ power.

7.2.3.4 High Pressurizer Pressure

A reactor trip for high pressurizer pressure is provided to prevent excessive blowdown of the primary coolant system by relief action through the pressurizer power-operated relief or safety valves.

The trip signals are provided by four narrow range independent pressure transducers measuring the pressurizer pressure. A typical channel diagram is shown on Figure 7-2.

A reactor trip is initiated by two-out-of-four coincidence logic from the four independent measuring channels if the pressurizer pressure exceeds 2400 psia. This signal also opens the power-operated relief valves.

Pretrip alarms are initiated if the pressurizer pressure exceeds 2350 psia.

7.2.3.5 Thermal Margin/Low-Pressure Trip

A trip is initiated by a continuously computed function of primary coolant pressure and thermal power to prevent reactor conditions from violating a minimum departure from nuclear boiling ratio (DNBR). At constant coolant flow, the temperature rise in the reactor is a function of power so that the variable trip can be effected by the adjustment of a pressure trip set point with reactor inlet and outlet coolant temperatures. At partial flow conditions, the changes in coolant temperature are such that the low thermal margin protection is continued with no change required in the pressure set point function. The variable pressure trip set point is computed by the function, $P_{Trip} = AT_{Hot} - BT_{Cold} - C$. At design conditions, the values of the constants are A = 57, B = 30, and C = 15,800. These values may be adjusted in the field to accommodate changes in the process conditions. With the function suitably adjusted, reactor operation at less than the 1.3 DNBR is not permitted. A further restriction on operation is that a trip is initiated if the pressure is reduced to 1750 psia. The trip signal is initiated by a two-out-of-four coincidence logic from four independent safety channels, and audible and visual pretrip alarms are actuated to provide for annunciation on approach to reactor trip conditions. A block diagram of the TM/LPT function is shown on Figure 7-5.

An abnormally high main steam flow from either steam generator will cause the secondary pressure to drop rapidly.

Four pressure transmitters on each steam generator actuate trip units which are connected in a two-out-of-four logic to initiate the reactor protective action if the steam generator pressure drops below a preselected value. Signals from two of the four indicating meter relays from either steam generator will trip the reactor and close the main steam isolation valves on both steam generators. Pretrip alarms are also provided.

A key-operated bypass switch allows this trip to be bypassed. This trip bypass is automatically reset above a selected pressure level.

7.2.3.9 Manual Trip

A manual reactor trip is provided to permit the operators to trip the reactor. Manual actuation of either of two reactor trip push-button switches in the main control room causes direct interruption of the a-c power to the d-c power supplies feeding the CRDM electromagnetic clutches.

7.2.4 SIGNAL GENERATION

Four instrument channels are used to generate the signals necessary to initiate the automatic reactor trip action except for the loss of load and high rate-of-change trips where two measuring channels are used. The signal cable routing and readout drawer locations are separated and isolated to provide channel independence.

(1) The high rate-of-change of neutron flux signals are generated by the two wide-range measuring channels (Figure 7-6A) which monitor the flux from source level to 125% of full power. These channels receive signals from flux monitors in the biological shield around the reactor.

(2) The high neutron flux signals are supplied by four linear flux measuring channels (Figure 7-6B) covering the flux range from 0.1% to 125% power. These channels receive signals from ion chambers which monitor the full length of the core and are located in the biological shield around the reactor.

(3) The pressure, flow, thermal margin and water level trips are each actuated from signals generated by separate sets of transmitters. The primary coolant pressure is measured in the pressurizer, flow is measured by monitoring the pressure difference across the steam generators, thermal measurements are taken from the reactor inlet and outlet piping in each loop and combined with primary coolant pressure to determine thermal margin, and steam generator water level and pressure are monitored in each steam generator.

Where the trip is to be allowed only in selected power ranges, a power dependent signal is supplied to the trip modules. Below the selected power levels (15% of rated power or less), these signals provide inhibit action to all of the trips except high neutron flux, low thermal margin, low reactor coolant flow and excess steam flow. The high power rate-of-change trip is inhibited below 10^{-4}% power and also above 15% power. Each neutron flux measurement channel supplies the automatic inhibit signals to trip in the same channel. Therefore, channel separation is maintained.

The CRDM clutches are separated into two groups. The clutches in each group are supplied in parallel with low-voltage, d-c power by an ungrounded feed line. Two a-c to d-c converters supply each feed line so that one converter being cut off does not cause release of the clutches. The converters on each side are each supplied by a line from a preferred a-c bus to assure a continued source of power. Each line passes through two interrupters (each actuated by a separate trip path) in series so that, although both a-c lines must be de-energized to release the clutches, there are two separate means of interrupting each line. This arrangement provides means for the testing of the protective system.

The functions of reactor protective system Relays K1, K2, K3, and K4 shown on Figure 7-1B are listed in Table 7-1.

TABLE 7-1

REACTOR PROTECTIVE SYSTEM RELAYS

Contact	Function	Contact	Function
K1-1	Oscillograph	K2-1	***Rod Rundown
K1-2	Spare	K2-2	Rod Rundown
K1-3	*Trip Lockout	K2-3	Trip Lockout
K1-4	Trip Lockout	K2-4	Trip Lockout
K1-5	Diesel Generator Start(1-1)	K2-5	Diesel Generator Start(1-2)
K1-6	Trips Turbine	K2-6	Trips Turbine
K1-7	**Trip Reset Lockout	K2-7	Trip Reset Lockout
K1-8	Annunciator	K2-8	Spare
K3-1	Rod Rundown	K4-1	Spare
K3-2	Rod Rundown	K4-2	Spare
K3-3	Trip Lockout	K4-3	Trip Lockout
K3-4	Trip Lockout	K4-4	Trip Lockout
K3-5	Diesel Generator Start(1-1)	K4-5	Diesel Generator Start(1-2)
K3-6	Trips Turbine	K4-6	Trips Turbine
K3-7	Spare	K4-7	Spare
K3-8	Spare	K4-8	Spare

*Trip Lockout - The reactor trip reset push button must be depressed to permit reactor start-up following a trip.

**Trip Reset Lockout - The reactor trip cannot be reset within 30 seconds following a reactor trip. This function prevents the reactor trip being reset while the control rods are still descending following a reactor trip.

***Rod Rundown - The control rods receive a "rods in" signal following a reactor trip which causes any "stuck" rods to be driven to the bottom of the core.

(13) The d-c clutch power supply circuits operate ungrounded so that single grounds have no effect. The clutches are supplied in two groups by separate pairs of power supplies to further reduce the possibility of clutches being improperly held. The clutch impedances and load requirements are such that the application of any other local available voltage will not prevent clutch release, eg, connection of the clutch supply circuit to the battery distribution circuit would cause the distribution circuit fuse to blow due to excessive current drain. Connection to a 115 volt a-c circuit would have similar effects. Connection to available low-voltage d-c, such as the nuclear instrumentation power supplies, would have no effect since these power supplies have insufficient capacity to supply the load.

7.2.8 POWER SOURCES

The power for the protective system is supplied from four separate independent preferred a-c buses. Each preferred bus is supplied from the battery system through an inverter to assure an uninterrupted, transient-free source of power.

Each preferred bus also has provision for connection to an instrument a-c bus to permit servicing of the inverters.

The distribution circuits to the preferred buses are provided with circuit protective devices to assure that individual circuit faults are isolated.

7.2.9 PHYSICAL SEPARATION

The location of sensors and the connection of sensing lines to the process loop are effected to assure channel separation and preclude the possibility of single events negating a system action. The process transmitters located inside the containment, which are required for short-term operation following a DBA, are housed in cabinets for mechanical and environmental protection. Protection from excessive temperature and humidity conditions is provided during a DBA for a period of time sufficient to complete their assigned function. The routing of cables from these cabinets is arranged so that the cables are separated from each other and from power cabling to reduce the danger of common event failures. This includes separation at the containment penetration areas. In the control room, the four nuclear instrumentation and protective system trip channels are located in individual compartments. Mechanical and thermal barriers between these compartments reduce the possibility of common event failure. Outputs from the components in this area to the control boards are buffered so that shorting, grounding, or the application of the highest available local voltages does not cause channel malfunction.

with some reactor parameters at values which would normally cause a trip. For these special operations, zero power mode bypass switches may be used to bypass the low flow, low steam generator level, low steam generator pressure, and the low thermal margin/low-pressure trip functions.** These bypasses are automatically removed above 10^{-4}% power.

Interlocks to prohibit regulating group withdrawal are provided to prevent the reactor from reaching undesirable conditions. These interlocks are summarized in Table 7-3.

TABLE 7-3

Withdrawal Prohibit Conditions	Manual Individual Control Mode	Manual Sequential Group Control Mode	Automatic Sequential Group Control Mode
T_{avg} Deviation			X
T_{avg} - T_{ref} Deviation			X
Pretrip Overpower	X	X	X
Rod Drop			X
High Start-Up Rate (Between 10^{-4}% and 15% Power)	X	X	X

The part-length rods may be moved manually either individually or as a group. A selector switch prevents simultaneous manual movement of the part-length and any other rods. The part-length rods have upper and lower limits of travel.

7.3.2.2 Primary Pressure Regulating

Two independent pressure channels provide suppressed range (1500 to 2500 psia) signals for control of the pressurizer heaters and spray valves. The output of either controller may be manually selected to perform the control function. During normal operation, a small group of heaters is proportionally controlled to maintain operating pressure. If the pressure falls below the proportional band, all of the heaters are energized. Above the normal operating range, the spray valves are proportionally opened to increase the spray flow rate as pressure rises. A small continuous flow is maintained through the spray lines at all times to keep the pipes warm and reduce thermal shock as the control valves open.

**Four bypass switches will be provided. Each bypass switch will remove all four trip functions from one of the four protective system channels.

7.4.2 SYSTEM DESCRIPTION

7.4.2.1 Process Instrumentation

The following process instruments are associated with the reactor protective, reactor control or primary plant controls:

Temperature - Temperature measurements are made with precision resistance temperature detectors (RTDs) which provide a signal to the remote temperature indicating control and safety devices.

The following is a brief description of each of the temperature measurement channels:

(a) Hot Leg Temperature

Each primary hot leg contains five temperature measurement channels. Four of these channels provide a hot leg temperature signal to the thermal margin/low-pressure trip set point computers. The other hot leg temperature measurement channel provides a signal to the loop T_{avg} and ΔT summing computers in the reactor regulating system. The five hot leg temperatures are indicated on the control panel. A high-temperature alarm from the one control channel is provided to alert the operator to a high-temperature condition.

(b) Cold Leg Temperature

Each primary cold leg contains three temperature measurement channels. Two of the channels provide a cold leg temperature signal to the thermal margin/low-pressure trip set point computers. These two channels also provide cold leg temperature indication on the control panel. The other cold leg temperature measurement channel provides either a signal to the loop T_{avg} and ΔT summing computers or a wide-range temperature recorder on the main control panel. A switch is provided to select the cold leg temperature measurement channel for loop T_{avg} and ΔT. The cold leg channel not selected for loop T_{avg} and ΔT is automatically switched to the wide-range temperature recorder.

(c) Loop Average Temperature

Each loop is provided with an average temperature summer. The T_{avg} summer receives inputs from the control channel hot leg temperature detector and the selected cold leg temperature detector and provides an average temperature output to the reactor regulating system and to a recorder.

The temperature recorders are equipped with two pens. One pen records the average temperature and the other pen records the programmed reference temperature signal (T_{ref}) corresponding to turbine load. (First-Stage Pressure)

The contact output of each trip unit is fed to a single channel of the reactor protective system. Thus, with two wide-range logarithmic channels, a separate rate trip signal is fed to Channels A, B, C and D of the reactor protective system. The $<10^{-4}\%$ of full power rate-of-change bypass is initiated by the wide-range channel level signal. The level signal is fed to two trip units set to trip above $10^{-4}\%$. Contacts from each trip unit open above $10^{-4}\%$ to remove the rate trip bypass and to remove the zero power manually actuated bypass associated with a single channel. The zero power manual actuated bypass allows control rod drop testing, or rod withdrawal for other tests during shutdown. The trips bypassed are low flow, low steam generator level, low steam generator pressure, thermal margin/low pressure. These trips are automatically reactivated above $10^{-4}\%$ full power.

The >15% full power rate-of-change trip bypass for a particular channel is initiated by a bistable trip unit in the power range safety channel. Above 15% full power, the bistable trip unit resets closing a contact in parallel with the rate trip contact associated with that channel (A, B, C or D). This method of rate trip bypass permits maximum independence of rate trip channels.

The rate-of-change of power pretrip alarm utilizes a single bistable trip unit (containing two sets of relay contacts) in each wide-range logarithmic channel. Each set of contacts feeds an auxiliary bistable trip unit in one of the channels of reactor protective system. The auxiliary trip unit in turn initiates the rod withdrawal prohibit signal and pretrip alarm. The signal to the auxiliary trip unit is bypassed below $10^{-4}\%$ and above 15% of full power to avoid spurious alarms and rod withdrawal prohibits.

Reset of the trip units operates as described for the start-up channels.

<u>Power Range Safety Channels</u> - The four power range channels measure flux linearly over the range of 1% to 125% of full power. The detector assembly consists of two uncompensated ion chambers for each channel. One detector extends axially along the lower half of the core while the other, which is located directly above it, monitors flux from the upper half of the core. The upper and lower sections have a total active length of 12 feet. The d-c current signal from each of the ion chambers is fed directly to the control room drawer assembly without preamplification. Integral shielded cable is used within the region of high neutron and gamma flux.

The signal from each chamber (top and bottom) is fed to independent amplifiers. The output of the amplifiers is indicated, compared and summed. The individual amplifier output is indicated on the amplifier drawer. The outputs are compared with each other to indicate axial flux tilts. The summed output of the two amplifiers is indicated, recorded and compared with averaged summed outputs

TABLE 9-8A (Contd)

Valve No	Valve Description	Normal Position	Shutdown Position	Position After Loss of Air
Service Water System (Figure 9-1 and Figure 9-2)				
0823	Component Cool HX Dischg	O	O	O
0824	Return From Containment Coolers	O	O	O
0825	Eng Safe Room Cooler Supply	C	C	O
0826	Component Cool HX Dischg	O	O	O
0835	Turbine LO Cooler Stop Bypass	C	C	O
0836	Turbine LO Cooler Stop Bypass	C	C	O
0838	Normal Cont Cooler Control	O	O	C
0839	Generator H_2 Cooler Stop Bypass	C	C	O
0843	Normal Cont Cooler Control	O	O	C
0844	Critical Service Wtr Header Iso	O	O	O
0845	Critical Service Wtr Header Iso	O	O	O
0846	Critical Service Wtr Header Cross-connect	O	O	O
0847	Supply to Containment Coolers	O	O	O
0852	Generator Exciter Cooler Supply Bypass	C	C	O
0857	Critical Service Wtr Header Cross-connect	O	O	O
0861	8" Return From Cont Coolers	C	C	O
0862	Containment Cooler Supply	O	O	O
0863	Normal Cont Cooler Control	O	O	C
0864	8" Return From Cont Coolers	C	C	O
0865	Containment Cooler Supply	O	O	O
0867	8" Return From Cont Coolers	C	C	O
0869	Containment Cooler Supply	O	O	O
0870	Containment Cooler Supply	O	O	O
0872	Normal Cont Cooler Control	O	O	C
0873	8" Return From Cont Coolers	C	C	O
0876	Diesel Generator Cool Supply	O	O	O
0877	Diesel Generator Cool Supply	O	O	O
0879	Backup Cool Safeguards Pumps	C	C	C
0880	Backup Cool Safeguards Pumps	C	C	C
0884	Diesel Generator Cool Supply	C	C	O
0885	Diesel Generator Cool Supply	C	C	O
1318	Service Wtr Pump Header Isolation	O	O	O
1319	Service Wtr Pump Header Isolation	O	O	O
1359	Noncritical Service Wtr Header Isolation	O	O	C
Component Cooling System (Figure 9-6)				
0909	Letdown HX Return	O	O	O
0910	Component Cool to Cont Isolation	O	O	O
0911	Component Cool From Cont Isolation	O	O	O
0913	Supply Safeguards Pumps	C	O	O

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TABLE 9-8A (Contd)

Valve No	Valve Description	Normal Position	Shutdown Position	Position After Loss of Air
0915	Comp Cool Surge Tk Vent	O	O	C
0918	Comp Cool Surge Tk Makeup	C	C	C
0937	Supply to Shutdown HX	C	O	O
0938	Supply to Shutdown HX	C	O	O
0940	Component Cool From Cont Isolation	O	O	O
0944	Supply to Spent Fuel HX	O	O	C
0945	Supply to Comp Cool HX	O	O	O
0946	Supply to Comp Cool HX	O	O	O
0947	Supply to Safeguards Pumps	O	O	O
0948	Supply to Safeguards Pumps	O	O	O
0949	Supply to Safeguards Pumps	O	O	O
0950	Return From Safeguards Pumps	C	O	O
0951	Return From Safeguards Pumps	C	O	O

Main Steam, Main and Auxiliary Turbine Systems (Figure 9-9)

Valve No	Valve Description	Normal Position	Shutdown Position	Position After Loss of Air
0501	Main Steam Isolation Valve	O	O	As Is (Accumulator)
0510	Main Steam Isolation Valve	O	O	As Is (Accumulator)
0511	Steam Bypass Valve	C	Open for Bleed	C (Note I)
0521	Steam to Aux Turbine Feed Pump	C	O	O

Service and Instrument Air Systems (Figure 9-8)

Valve 1211, Instrument Air Supply to Containment, is open during reactor operation or reactor shutdown and fails open on loss of air.

Process Sampling System (Figure 9-16)

Air-operated process sampling valves are normally closed unless sampling a specific point. All air-operated valves fail close.

Radioactive Waste Treatment System (Figures 11-2 and 11-3)

All air-operated valves in the radioactive waste treatment system, including liquid and gas discharge stop valves, fail close upon loss of instrument air.

Heating, Ventilation and Air Conditioning

Reference Section 9.8.4.

Shield Cooling System

During normal reactor operation and reactor shutdown, one of two air-operated shield cooling supply valves is open. Upon loss of instrument air, both supply valves fail open.

Note 1: A hand operator is provided on the steam bypass valve.

TABLE 9-13

DESIGN PARAMETERS
Chemical and Volume Control System

1.1 General

Normal Letdown Flow	40 Gpm
Normal Purification Flow Rate	40 Gpm
Normal Charging Flow	44 Gpm
Primary Coolant Pump Controlled Bleed-Off (4 Pumps)	4 Gpm
Normal Letdown Temperature at Loop	547.8° F
Normal Charging Temperature at Loop	425° F
Ion Exchanger Operating Temperature	120° F

1.2 Regenerative Heat Exchanger - E56

Quantity	1
Type	Shell and Tube, Vertical
Normal Heat Transfer	6.6×10^6 Btu/Hr
Code	ASME III, Class A
Shell Side (Letdown)	
Fluid	Primary Coolant, 1 Wt % Boric Acid, Maximum
Design Pressure	2485 Psig
Design Temperature	650° F
Materials	Stainless Steel
Tube Side (Charging)	
Fluid	Primary Coolant, 6-1/4 Wt % Boric Acid, Maximum
Design Pressure	2735 Psig
Design Temperature	650° F
Materials	Stainless Steel

TABLE 9-13 (Contd)

Operating Parameters - Regenerative Heat Exchanger

Shell Side (Letdown)	Normal	Maximum Unbalanced Charging With Heat Transfer	Maximum Purification	Maximum Unbalanced Letdown
Flow - Gpm	40	40	120	120
Inlet Temp - °F	547.8	547.8	547.8	547.8
Outlet Temp - °F	238	176	361	450
Tube Side (Charging)				
Flow - Gpm	44	133	124	33
Inlet Temp - °F	120	120	120	120
Outlet Temp - °F	425	240	325	516
Heat Transfer - Btu/Hr	6.6×10^6	7.81×10^6	12.41×10^6	6.73×10^6

1.3 Letdown Orifice - R02003, R02004 and R02005

Quantity	3
Capacity, Each	40 Gpm
Design Pressure	2485 Psig
Design Temperature	550° F
Normal Temperature of Fluid	238° F
Maximum Temperature of Fluid	470° F
Normal Downstream Pressure	470 Psig
Normal Upstream Pressure	1525 Psig
Material	Stainless Steel
Fluid	Primary Coolant, 1 Wt % Boric Acid, Maximum

1.4 Letdown Heat Exchanger - E58

Quantity	1
Type	Shell and Tube, Horizontal