GE Nuclear Energy

General Electric Company
175 Curtner Avenue, San Jose, CA 95125

August 2, 1994

MFN No. 090-94
Docket No. STN 52-004

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Richard W. Borchardt, Director
Standardization Project Directorate

Subject: NRC Requests for Additional Information (RAIs) on the Simplified Boiling Water Reactor (SBWR) Design

References:
1. Transmittal of Requests for Additional Information (RAIs) Regarding the SBWR Design, Letter from M. Malloy to P. W. Marriott dated April 8, 1994.
2. MFN No. 084-94, RAIs In Process at GE, Letter from P. W. Marriott to R. W. Borchardt, June 15, 1994.

The Reference 1 letter requested additional information regarding the SBWR Design. In partial fulfillment of this request, GE is submitting Attachment 1 to this letter which transmits the responses to the following RAIs in the 440.7 - 440.58 series which focus on thermal-hydraulic testing and analysis activities:

440.13
440.15 - 440.17
440.32 - 440.33
440.36 - 440.58

The GE response to the remaining RAIs in this series will be provided following resumption of NRC's review of the SSAR in design-related areas.

Sincerely,

Rob Aesmit for

P. W. Marriott, Manager
Advanced Plant Technologies
M/C 781, (408) 925-6948

Attachment 1, "Responses to NRC RAIs"

cc: M. Malloy, Project Manager (w/2 copies of Attachment 1)
F. W. Hasselberg, Project Manager (w/1 copy of Attachment 1)

D040
1/1

RAI Number: 440.13

Question:

In SSAR Section 1.2.2.8.1 (Nuclear Fuel), delete the following statement which is not correct: "Fuel design for the SBWR Standard Plant is not within the scope of the certified design."

In SSAR Section 1.2.2.8.2 (Fuel Channel), delete the following statement which is not correct: "Fuel channel design for the SBWR Standard Plant is not within the scope of the certified design."

In SSAR Section 1.2.2.8.3 (Control Rod), delete the following statement which is not correct: "Control rod design for the SBWR Standard Plant is not within the scope of the certified design."

GE Response:

These changes are incorporated in the attached revised SBWR SSAR pages.

The NEMS has no access to the safety-related data base; however safety-related data can be read by the NEMS on the optically-isolated memory portions of the Essential Multiplexing System (EMS) Local Multiplexing Units (LMU). This data can be read by any NEMS multiplexing units that is configured to do so. The NEMS cannot write data to any portion of the EMS.

The NEMS consists of two types of multiplexing units: Local Multiplexing Units (LMU), and Control Room Multiplexing Units (CMU) connected via fiber optic cables. The NEMS also includes network gateways which allow transfer of data between data highway systems.

Throughout the plant, LMUs are located in local plant areas to acquire sensor data and transmit this data to the any equipment that requires it. The LMUs also receive processed signals from the control room for command of control system actuators. CMUs are located in the control room to transmit and receive data for the logic processing units of the plant control systems.

All interconnections are fiber optic data links. Within each NEMS highway, the system uses redundant links for greater reliability.

There are a number of NEMS highway systems that are routed throughout the plant. These systems all have CMUs located in the main control room. Gateways connect the multiple NEMS highway systems to allow for transfer of data between NEMS highway systems.

1.2.2.8 Nuclear Fuel

1.2.2.8.1 Nuclear Fuel

~~Fuel design for the SBWR Standard Plant is not within the scope of the certified design.~~ It is intended that the specific fuel to be used in any facility which has adopted the certified design be in compliance with U.S. NRC approved fuel design criteria. This strategy is intended to permit future use of enhanced/improved fuel designs as they become available. However, this approach is predicated on the assumption that future fuel designs will be extensions of the basic fuel technology that has been developed for boiling water reactors. Key characteristics of this established BWR fuel technology are:

- Uranium oxide based fuel pellets;
- Zirconium-based (or equivalent) fuel cladding;
- All material selected on the basis of BWR operating conditions;
- Multi-rod fuel bundles in an N lattice; and

1.2.2.8.2 Fuel Channel

~~Fuel channel design for the SBWR is not within the scope of the certified design.~~ It is intended that the specific fuel channel to be used in any facility which has adopted the certified design be in compliance with U.S. NRC approved fuel channel design criteria. This strategy is intended to permit future use of enhanced/improved fuel channel designs as they become available. However, this approach is predicated on the assumption that future fuel channel designs will be extensions of the basic technology that has been developed for boiling water reactors. The key characteristic of this established BWR fuel channel technology is the use of zirconium-based (or equivalent) fuel channels which preclude cross-flow in the core region.

The following is a summary of the principal requirements which must be met by the fuel channel supplied to any facility using the certified design:

- The material of the fuel channel shall be shown to be compatible with the reactor environment.
- The channel will be evaluated to ensure that channel deflection does not preclude control rod drive operation.
- The effects of channel bow will be included in the fuel rod critical power evaluations.

1.2.2.8.3 Control Rod

~~Control rod design for the SBWR is not within the scope of the certified design.~~ It is intended that the specific control rod to be used in any facility which has adopted the certified design be in compliance with U.S. NRC approved control rod design criteria. This strategy is intended to permit future use of enhanced/improved control rod designs as they become available. However, this approach is predicated on the assumption that future control rod designs will be extensions of the basic technology that has been developed for boiling water reactors. Key characteristics of this established BWR control rod technology are:

- Control rods perform dual functions of power distribution shaping and reactivity control.
- The control rod has a cruciform cross-sectional envelope shape.
- The control rod has a coupling at the bottom for attachment to the CRD.
- The control rod has an upper bail handle for transporting.

RAI Number: 440.15

Question:

In SSAR Section 3.1.3.7 (page 3.1-30), Criteria 26, Reactivity Control System Redundancy and Capability, and SSAR Section 3.1.3.8 (page 3.1-31), Criteria 27, Combined Reactivity Control Systems Capability, add Reference 9.3.5, Standby Liquid Control System, to the list of SSAR chapters/sections.

GE Response:

GE will make this change to the SBWR SSAR.

RAI Number: 440.16

Question:

GE's response to RAI SRXB.7 was not satisfactory. The staff does not agree with the GE statement that "There are no requirements for a safety-related high pressure injection system..."General Design Criteria (GDC) 33 of 10 CFR 50 Appendix A requires a safety-related system for protection against small breaks in the reactor coolant pressure boundary and the SBWR design does not include a safety-related injection system to satisfy the GDC 33. (Reference SSAR Section 3.1.4.4, Reactor Coolant Make Up.)

GE Response:

The SBWR does not have a safety-related high pressure makeup system. The SBWR, as a passive plant, takes exception to GDC 33. The SBWR relies on the ADS and GDCS system for core flooding if non-safety related makeup from the CRD system is unavailable. The safety advantages of passive technology are expected to provide a lower probability of core damage than non-passive technology in the case of a small break accident.

RAI Number: 440.17

Question:

In SSAR Table 3.2-1 (page 3.2-16), J11, add control rods to the list.

GE Response:

GE will make this change to the SBWR SSAR.

RAI Number: 440.32

Question:

What is meant by "overshoot the relief valve setpoint" in SSAR Section 5.2.2.3.3, Safety/Relief Valve Capacity? Figure 5.2-6 shows that peak vessel pressure is independent of valve capacity. How can the peak pressure be completely independent of the valve capacity? How many (minimum number of) safety-relief valves are required to meet the ASME Boiler and Pressure Vessel Code limit of 1375 psig? Explain in detail the significance of the operation of low and high set point valves shown in the figure.

GE Response:

Please see attached revised SSAR Section 5.2.2.3.3.

- Reclosure pressure setpoint (% of opening setpoint) both modes:
  - maximum safety limit (used in analysis): 98
  - minimum operational limit: 88

The opening and reclosure setpoints are assumed at a conservatively high level above the nominal setpoints. This is to account for initial setpoint errors and any instrument setpoint drift that might occur during operation. Typically, the setpoints in the analysis (on average) are assumed to be at least 1% above the actual nominal setpoints. Conservative SRV response characteristics are also assumed; therefore, the analysis conservatively bounds all SRV operating conditions.

Safety/Relief Valve Capacity

Sizing of the SRV capacity is based on establishing an adequate margin from the peak vessel bottom pressure to the vessel code limit (9.481 MPa gauge) in response to the reference transients.

The method used to determine total valve capacity is as follows.

Whenever the system pressure increases to the valve spring set pressure of a group of valves, these valves are assumed to begin opening and to reach full open at 103% of the valve spring set pressure. The lift characteristics assumed are shown in Figure 5.2-3.

5.2.2.3.3 Evaluation of Results

Safety/Relief Valve Capacity

The required SRV capacity is determined by analyzing the pressure rise from a turbine/generator trip with bypass failure transient with pressure scram. Results of this analysis are given in Figure 5.2-5. The peak vessel bottom pressure calculated is 8.73 MPa absolute (1266 psia), which is well below the acceptance limit of 9.481 MPa gauge (1375 psig). Figure 5.2-5 shows the MSIV isolation transient with flux scram, which is slightly milder than the turbine trip. The pressurization is not dynamic and ~~does not significantly overshoot the relief valve setpoint~~ ceases to increase once a single relief valve opens. Figure 5.2-6 shows that peak vessel pressure is only a function of the valve setpoint.

This is because the higher steam volume-to-power ratio of the SBWR causes the pressure rate prior to scram to be much lower than operating BWRs [0.34 MPa/sec (38 psi/sec) versus 0.57 MPa/sec (82 psi/sec)]. After a scram, the pressure rates due to stored energy release are correspondingly lower. The peak pressure in these events is the relief valve setpoint because, at these low pressurization rates, with large margin to the SRV setpoint, the pressure increase is effectively terminated by any single relief valve opening. As shown on Figure 5.2-6, the peak pressure is the same when 1, 2, 3, or 4 low setpoint valves are available. If all low setpoint valves are assumed to fail, the peak

pressure steps up to a higher value, which is essentially the high pressure setpoint, provided any high setpoint valve is available.

Pressure Drop in Inlet and Discharge

Pressure drop in the piping from the reactor vessel to the valves is taken into account in calculating the maximum vessel pressures. Pressure drop in the discharge piping to the suppression pool is limited by proper discharge line sizing to prevent backpressure on each SRV from exceeding 40% of the valve inlet pressure, thus assuring choked flow in the valve orifice and no reduction of valve capacity due to the discharge piping. Each SRV has its own separate discharge line.

5.2.2.3.4 System Reliability

The system is designed to satisfy the requirements of Section III of the ASME Code. The consequences of failure are discussed in Subsection 15.1.4 of this report.

5.2.2.4 Testing and Inspection Requirements

The inspection and testing of applicable SRVs utilizes a quality assurance program which complies with Appendix B of 10CFR50.

The SRVs are tested at a suitable test facility in accordance with quality control procedures to detect defects and to prove operability prior to installation. The following tests are conducted:

(1) hydrostatic test at specified test conditions (ASME Code requirement based on design pressure and temperature);

(2) thermally stabilize the SRV to perform quantitative steam leakage testing at 1.03 MPa (150 psi) below the SRV nameplate value with an acceptance criterion not to exceed 0.45 kg/hr (1 lb/hr) leakage;

(3) full flow SRV test for set pressures and blowdown where the valve is pressurized with saturated steam, with the pressure rising to the valve set pressure (during production testing the SRV is adjusted to open at the nameplate setpressure ±1%); and

(4) response time test where each SRV is tested to demonstrate acceptable response time based on system requirements.

The valves are installed as received from the factory. The valve manufacturer certifies that design and performance requirements have been met. This includes capacity and blowdown requirements. The setpoints are adjusted, verified, and indicated on the valves by the vendor. Specified manual and automatic initiated signal for power actuation (relief mode) of each SRV is verified during the preoperational test program.

RAI Number: 440.33

Question:

In SSAR Figure 5.2-1, add units to all the numerical values shown in the figure.

GE Response:

All units of numeric values in Figure 5.2-1 are mm and there is a note to that effect on the attached revised figure.

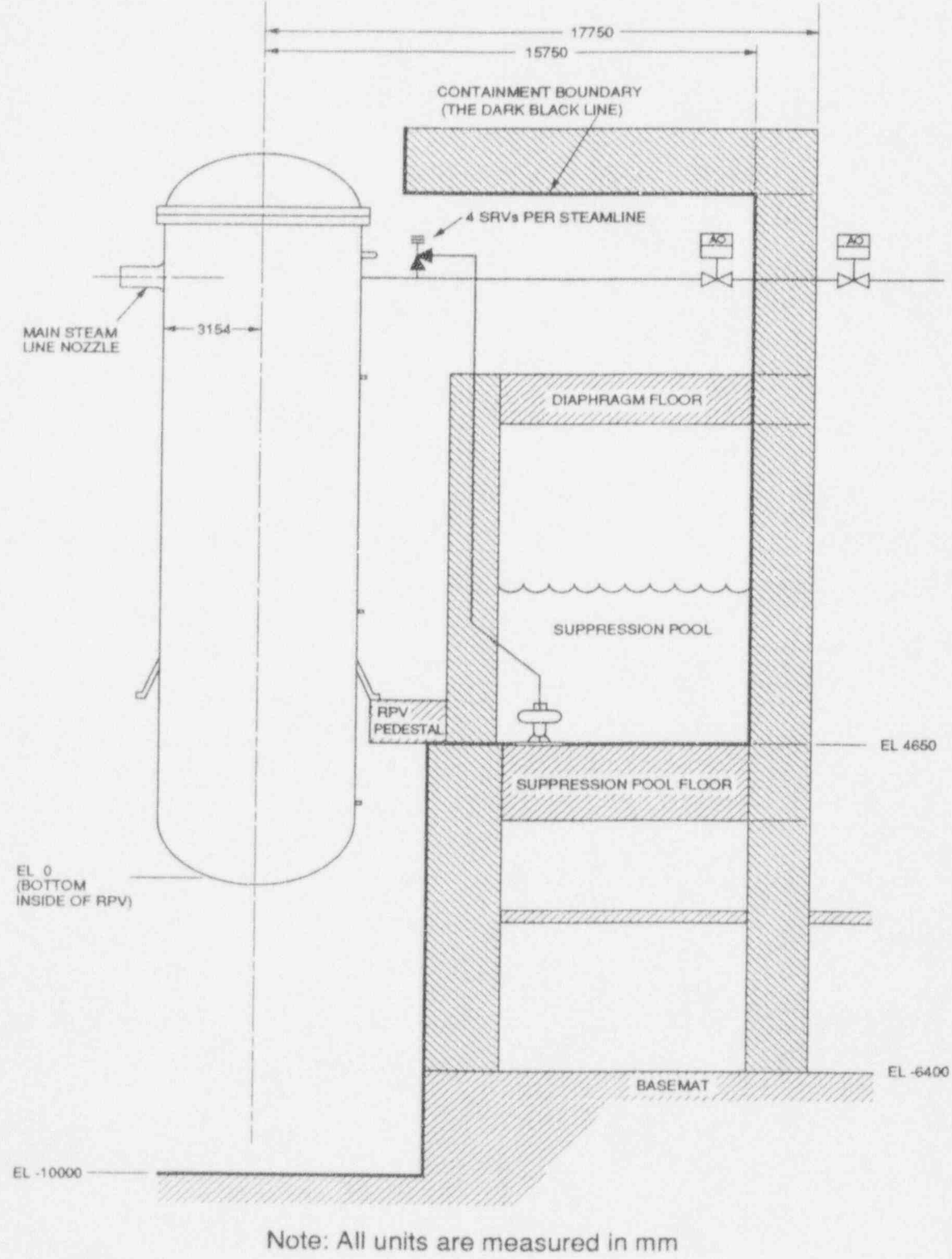


Figure 5.2-1 Safety/Relief Valve Schematic Elevation

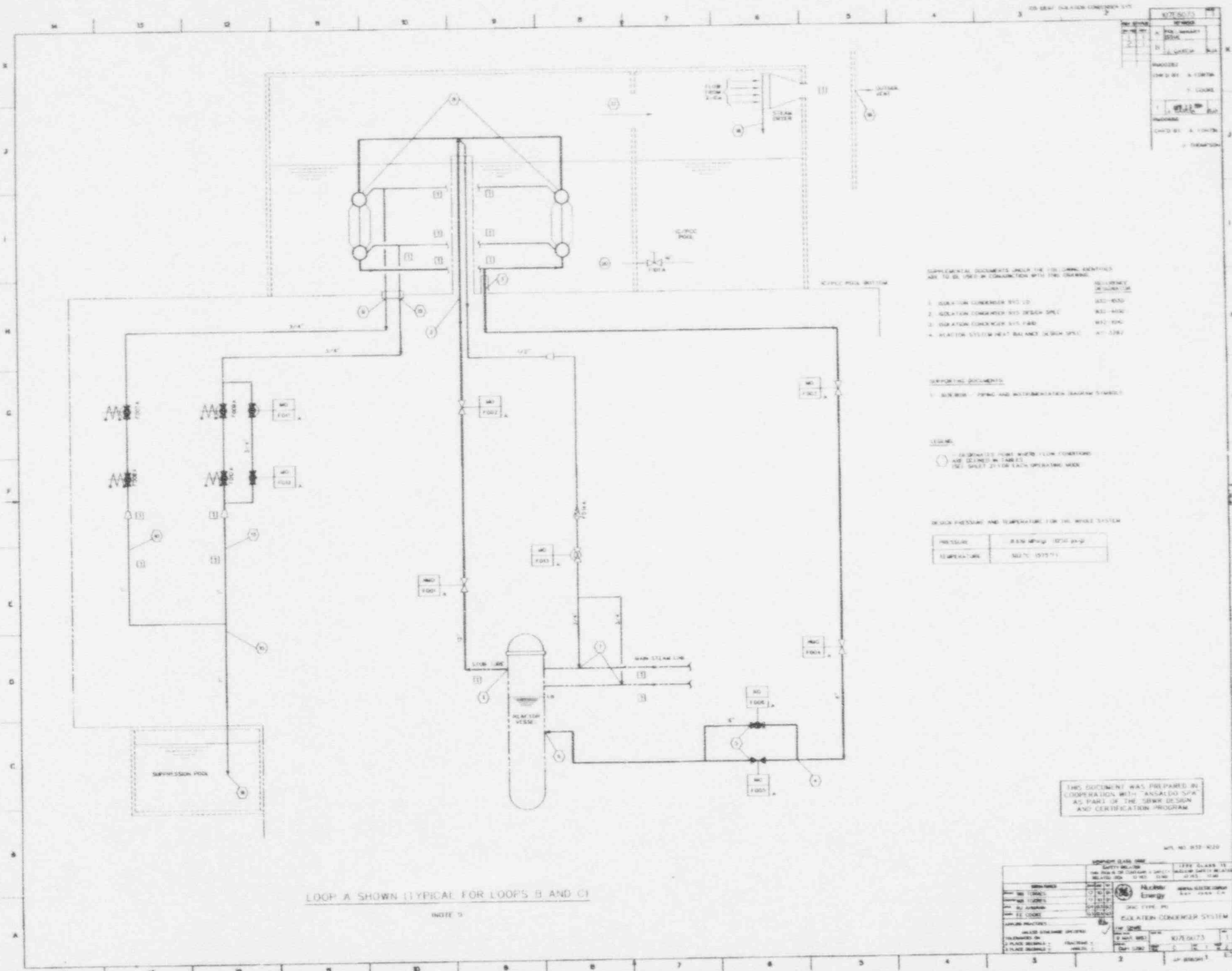
RAI Number: 440.36

Question:

Provide a diagram showing the isolation condenser system (ICS) design parameters: Pressures, temperature, and flow rates. Submit the process flow diagram for the ICS.

GE Response:

The process flow diagram 107E6073 Rev. 1 is attached.



SUPPLEMENTAL DOCUMENTS UNDER THE FOLLOWING IDENTITIES ARE TO BE USED IN CONJUNCTION WITH THIS DRAWING
1. ISOLATION CONDENSER SYS
2. ISOLATION CONDENSER SYS DESIGN SPEC
3. ISOLATION CONDENSER SYS P&ID
4. REACTOR SYSTEM HEAT BALANCE DESIGN SPEC
SUPPORTING DOCUMENTS
PIPING AND INSTRUMENTATION DIAGRAM SYMBOLS
LEGEND
DESIGN PRESSURE AND TEMPERATURE FOR THE WHOLE SYSTEM
PRESSURE
TEMPERATURE
THIS DOCUMENT WAS PREPARED IN COOPERATION WITH "ANSALDO SPA" AS PART OF THE SBWR DESIGN AND CERTIFICATION PROGRAM
SUPPRESSION POOL
REACTOR VESSEL
MAIN STEAM LINE
IC/PCC POOL
OUTSIDE VENT
Nuclear Energy
ISOLATION CONDENSER SYSTEM
107E6073
LOOP A SHOWN (TYPICAL FOR LOOPS B AND C)

107E6073

OPERATING CONDITIONS — NOTES 2 AND 3

[1]

MODE A: PLANT STARTUP
IN THE STEAM REGION, STARTING FROM COLD AMBIENT CONDITIONS, THE TEMPERATURE AND PRESSURE ARE THE SAME AS THE REACTOR PRESSURE VESSEL. DURING PLANT HEATUP THE CONDENSATE GRADUALLY FILLS THE REGION FROM THE RETURN VALVES UP TO THE IC TOP.

* SAME AS REACTOR PRESSURE VESSEL [1]

MODE B1: NORMAL PLANT OPERATION WITH ICS IN READY STANDBY (REACTOR AT 100% POWER)

POSITION		( )	(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	( )	(12)	(13)	( )	(15)	(16)	(17)	(18)	(19)
FLOW kg/s x 10^3		0.333	0.325	-	-	-	-	0.162	-	-	-	-	-	-	-	-	-	-	-	-	
TEMPERATURE °C		*	*	10	10	10	*	*	10	10	10	-	10	10	-	10	10	AMB	AMB	AMB	
PRESSURE	kPa (g) (psig)	*	7089 (1025)	7118 (1032)	7251 (1052)	7127 (1034)	7117 (1032)	6950 (1008)	7075 (1027)	7157 (1038)	13.8 (2)	-	7137 (1035)	13.8 (2)	-	13.8 (2)	33.8 (4.9)	0 (0)	0 (0)	0 (0)	

MODE B2: NORMAL PLANT OPERATION WITH ICS IN READY STANDBY (REACTOR AT 105% OF RATED STEAM FLOW)

POSITION		( )	(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	( )	(12)	(13)	( )	(15)	(16)	(17)	(18)	(19)
FLOW kg/s x 10^3		0.338	0.334	-	-	-	-	0.170	-	-	-	-	-	-	-	-	-	-	-	-	
TEMPERATURE °C		*	*	10	10	10	*	*	10	10	10	-	10	10	-	10	10	AMB	AMB	AMB	
PRESSURE	kPa (g) (psig)	*	7172 (1040)	7201 (1047)	7358 (1067)	7232 (1049)	7221 (1047)	7043 (1022)	7183 (1043)	7241 (1050)	13.8 (2)	-	7241 (1050)	13.8 (2)	-	13.8 (2)	33.8 (4.9)	0 (0)	0 (0)	0 (0)	

MODE C1: REACTOR ISOLATION WITH ICS INITIAL OPERATION (TRANSIENT CONDITION, WITHOUT FEEDWATER)

POSITION		( )	(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	( )	(12)	(13)	( )	(15)	(16)	(17)	(18)	(19)
FLOW kg/s x 10^3		28.2	28.2	347	347	347	347	0	347	-	-	-	-	-	-	-	-	-	-	-	
TEMPERATURE °C		*	285	10	10	10	*	*	286	10	10	-	10	10	-	10	10	AMB	AMB	AMB	
PRESSURE	kPa (g) (psig)	*	7934 (1151)	7950 (1153)	7981 (1158)	7946 (1153)	7934 (1151)	*	7934 (1151)	7962 (1155)	13.8 (2)	-	7278 (1057)	13.8 (2)	-	13.8 (2)	33.8 (4.9)	0 (0)	0 (0)	0 (0)	

MODE C2: REACTOR ISOLATION AT QUASI-STEADY STATE CONDITION AFTER MODE C1 TRANSIENT

POSITION		( )	(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	( )	(12)	(13)	( )	(15)	(16)	(17)	(18)	(19)
FLOW kg/s x 10^3		101.5	—			—	101.5	0	50.75	-	-	-	-	-	-	-	-	80.4	2.44	158.12	
TEMPERATURE °C		*	—					—	*	10	10	-	10	10	-	10	10	100	100	100	
PRESSURE	kPa (g) (psig)	*	7058.7 (1023.7)	7038.7 (1020.8)	7125.7 (1033.4)	7124.8 (1033.3)	7116.9 (1032.3)	*	7043 (1021)	7101 (1030)	13.8 (2)	-	7037 (1020)	13.8 (2)	-	13.8 (2)	33.8 (4.9)	+3.5 (+0.5)	+2.8 (+0.4)	0 (0)	

MODE D: STATION BLACKOUT WITH ICS OPERATION — NOTE 4

POSITION		( )	(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	( )	(12)	(13)	( )	(15)	(16)	(17)	(18)	(19)
FLOW kg/s x 10^3		101.5	—			—	101.5	-	101.5	-	-	-	-	-	-	-	-	80.4	2.58	156.12	
TEMPERATURE °C		*	—					—	*	10	10	-	10	10	-	10	10	100	100	100	
PRESSURE	kPa (g) (psig)	*	7058.2 (1023.8)	7048.7 (1023)	7108.2 (1035.3)	7108 (1033.5)	7116.8 (1032.3)	7088.4 (1025.3)	7044 (1023)	7102 (1030)	13.8 (2)	-	7078 (1027)	13.8 (2)	-	13.8 (2)	33.8 (4.9)	+3.5 (+0.5)	+2.8 (+0.4)	0 (0)	

MODE E: ICS OPERATION WITH AUTOMATIC VENTING

POSITION		( )	(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	( )	(12)	(13)	( )	(15)	(16)	(17)	(18)	(19)
FLOW kg/s x 10^3		12.43	12.43	11.2	—	—	11.2	0	0.21	0	6	-	1.22	1.23	-	1.22	1.22	8.87	0.296	17.44	
TEMPERATURE °C		*	—					—	*	10	10	-	282	282	-	282	282	100	100	100	
PRESSURE	kPa (g) (psig)	*	7513.9 (1089.8)	7513.6 (1089.7)	7588.5 (1097.7)	7588.5 (1097.7)	7583.8 (1097)	7518 (1090)	7543.6 (1089.7)	7572 (1098)	<3448 (<500)	-	7483 (1082)	6536 (948)	-	<3448 (<500)	<3448 (<500)	-	-	0 (0)	

MODE F: IC PIPE RUPTURE
FOLLOWING AN IC PIPE RUPTURE THE FOUR ISOLATION VALVES CLOSE AND AUTOMATICALLY ISOLATE THE LEAKING IC, STARTING FROM ACTUAL CONDITIONS AND DEPENDING ON THE KIND OF RUPTURE. THE IC IS DRAINED. THE FLOW RATE, TEMPERATURE AND PRESSURE VALUES ARE NOT MEANINGFUL IN THIS MODE OF OPERATION EXCEPT THE PRESSURE RISES TO [illegible] kPa(g) ([illegible] psig) AT (17) AND (18) DURING IC PIPE OR TUBE RUPTURE IN THE IC POOL.

MODE G: MAINTENANCE
IN THIS MODE OF OPERATION THE FLOW RATE, TEMPERATURE AND PRESSURE VALUES ARE NOT MEANINGFUL.

MODE H & J: VALVE PREOPERATIONAL AND OPERATIONAL READINESS TESTS [1]
PRESSURE, TEMPERATURE AND FLOW CONDITIONS ARE THE SAME AS MODE B1 OR B2. THE ICS DOES NOT COME INTO OPERATION.

VALVE POSITIONS

VALVE NUMBER	MODE A PLANT STARTUP	MODE B NORMAL PLANT OPERATION	MODE C REACTOR ISOLATION WITH IC OPERATION	MODE D STATION BLACKOUT WITH IC OPERATION	MODE E IC OPERATION WITH VENTING	MODE F IC PIPE RUPTURE	MODE G MAINTENANCE: DRAINED FOR MAINTENANCE	MODE G MAINTENANCE: DURING MAINTENANCE	MODE H VALVE PREOPERATIONAL TEST NOTE 6 [1]	MODE J VALVE OPERATIONAL READINESS TEST NOTE 5 [1]
F001A,B,C F002A,B,C F003A,B,C F004A,B,C	O	O	O	O	O	C *	O	C	O/C/O EACH VALVE INDIVIDUALLY WITH VALVES F005 AND F006 CLOSED	O/C/O EACH VALVE INDIVIDUALLY WITH VALVES F005 AND F006 CLOSED
F005A,B,C	C	C	O	C	O	O	O	C	C/O/C WITH VALVE F003 OR F004 CLOSED	C/O/C WITH VALVE F003 OR F004 CLOSED
[1] F006A,B,C	C	C	C/O **	O	C	C/O **	O	C		
F007A,B,C F008A,B,C	C	C	C	C	C	C	O	C	C/O/C EACH VALVE INDIVIDUALLY WITH VALVES F005 AND F006 CLOSED	C
F009A,B,C F010A,B,C *****	C	C	C	C	C/O/C***	C	O	C		C
F011A,B,C F012A,B,C *****	C	C	C	C	C/O/C***	C	O	C		
F013A,B,C	O	O	O	O	O	O ****	O	C	O/C/O	O
F014A,B,C	O	O	O	C	O	O	O	C	*RESET OPEN	O

O: OPEN C: CLOSED A: VALVE IN LOOP A B: VALVE IN LOOP B C: VALVE IN LOOP C

* AFTER IC PIPE RUPTURE THE ISOLATION SIGNAL CLOSES THE ISOLATION VALVES

[1] ** OPENS IF REACTOR WATER LEVEL IS BELOW L1 IN "STARTUP" AND "RUN" REACTOR OPERATING MODES

[1] *** TO DEPRESSURIZE THE REACTOR [illegible] THE ICS, OPEN AND RECLOSE THE VENT VALVES F009 AND F010 OR F011 AND F012 AFTER [illegible] HOURS OF IC OPERATION. THE TIME OF VENT OPENING IS [illegible] MINUTES TO PURGE THE IC NONCONDENSABLE GASES

**** THE REVERSE FLOW [illegible] THE PURGE LINE IS PREVENTED BY THE EXCESS FLOW VALVE

***** BOTH OF THESE VALVES CAN BE OPENED TOGETHER ONLY WHEN THE CONDENSATE DRAIN VALVES F005 AND F006 ARE OPEN

NOTES

1. SEE SHEET 2 FOR CONDITIONS THAT APPLY AT THE POSITIONS DESIGNATED WITHIN THE SYMBOL ( )
2. SEE SHEET 1 FOR DIAGRAM LOCATIONS FOR THE POSITIONS ( ) SHOWN IN THIS TABLE
3. SEE "VALVE POSITION" TABLE ON THIS SHEET (SHEET 3) WHICH DEFINES THE OPEN/CLOSED POSITION OF THE VALVES SHOWN ON THE DIAGRAM (SHEET 1) FOR THE EACH OPERATING MODE
4. GATE VALVE F001 CLOSED AND GLOBE VALVE F006 OPEN (STATION BLACKOUT WITH LOSS OF SAFETY GRADE BATTERY POWER). VESSEL PRESSURE AND TEMPERATURE AT NOMINAL CONDITIONS (SAME AS MODE C2) AND FLOW RATE CORRESPONDING TO WATER LEVEL IN THE LOWER DRUM BELOW THE TUBES
5. TESTS DONE WITH THE REACTOR SYSTEM PRESSURIZED AND WITH THE REACTOR MODE SWITCH IN THE "RUN" POSITION
6. TESTS DONE WITH THE REACTOR SYSTEM DEPRESSURIZED AND THE MODE SWITCH IN THE "REFUEL" POSITION

A	DAR-3292
	107E6073

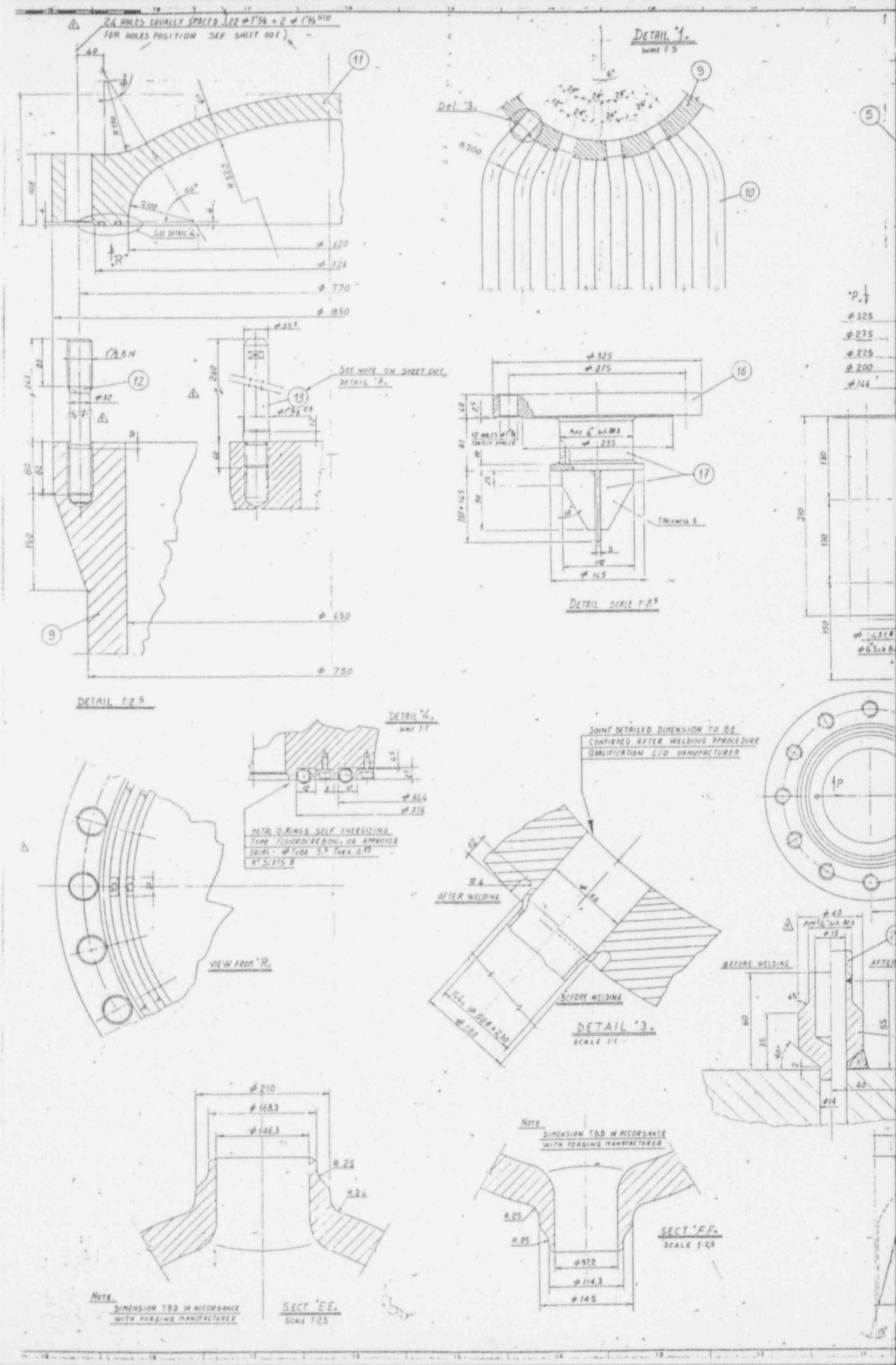
RAI Number: 440.37

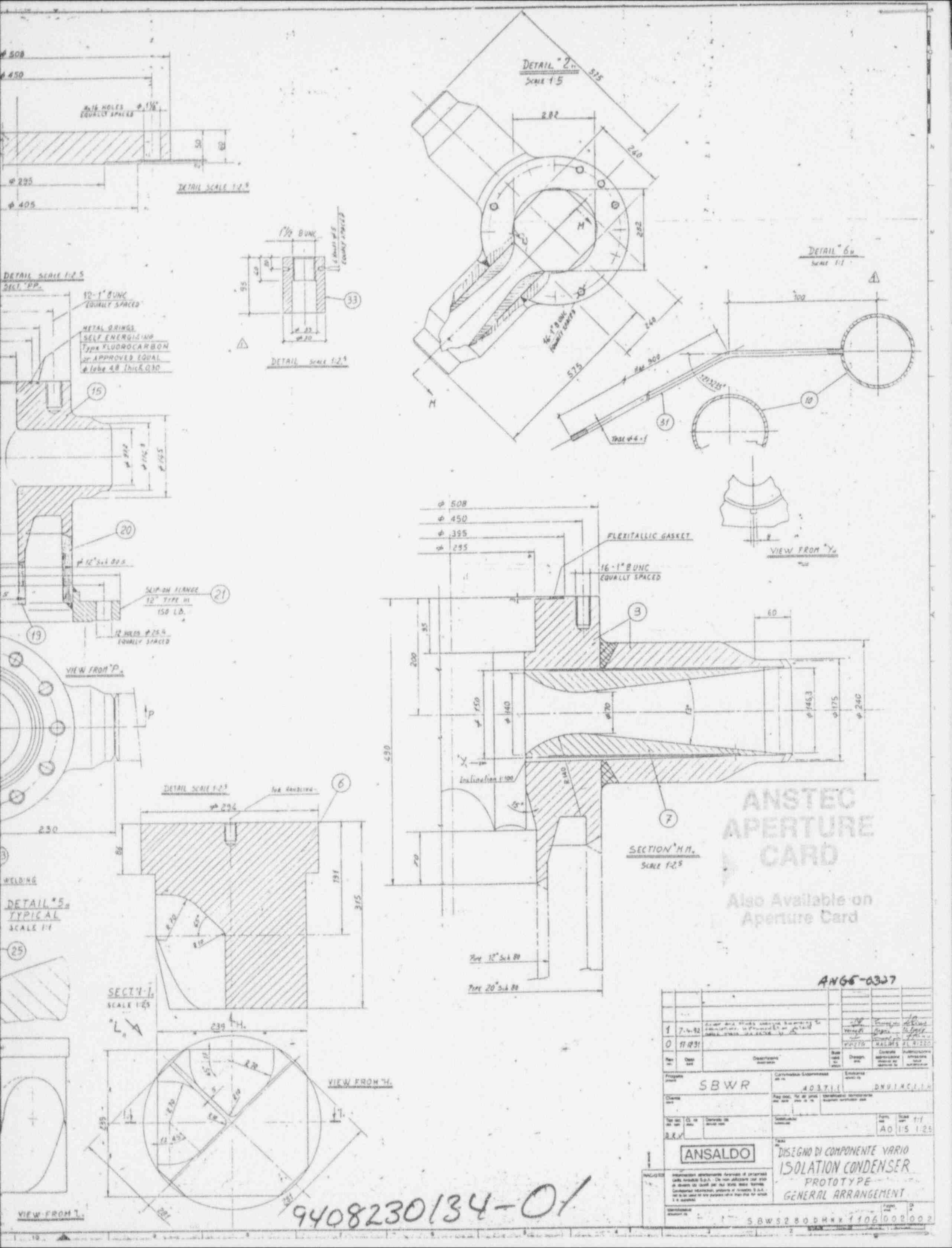
Question:

The isolation condenser for the SBWR is a vertical heat exchanger which is significantly different from the IC in operating plants. Provide a detailed description and drawing of the SBWR IC.

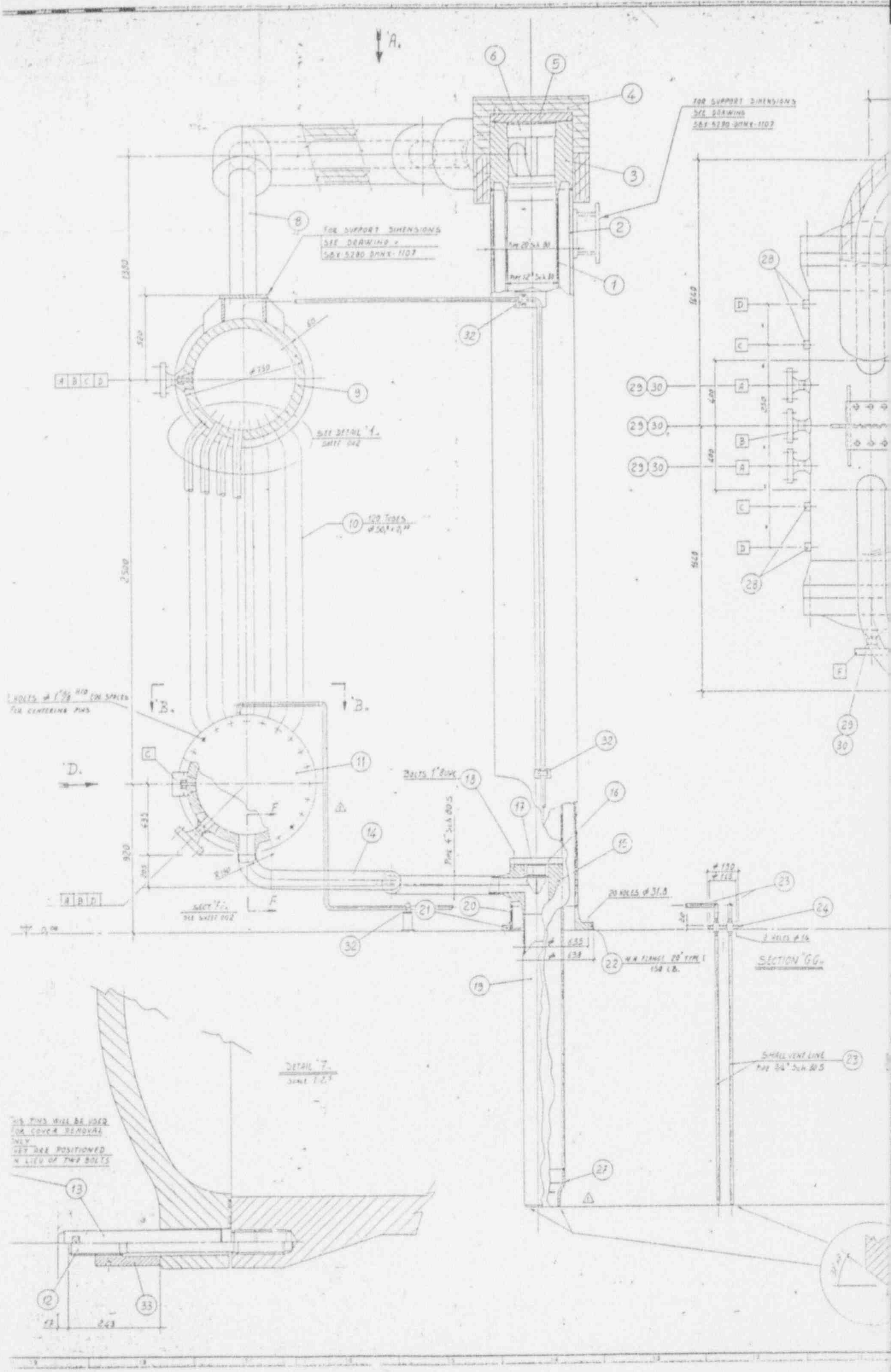
GE Response:

The Isolation Condenser System (ICS) specification 25A5013 Rev. 1 (attached) contains a detailed description of the isolation condenser system. A copy of the Isolation Condenser (IC) drawing is attached.

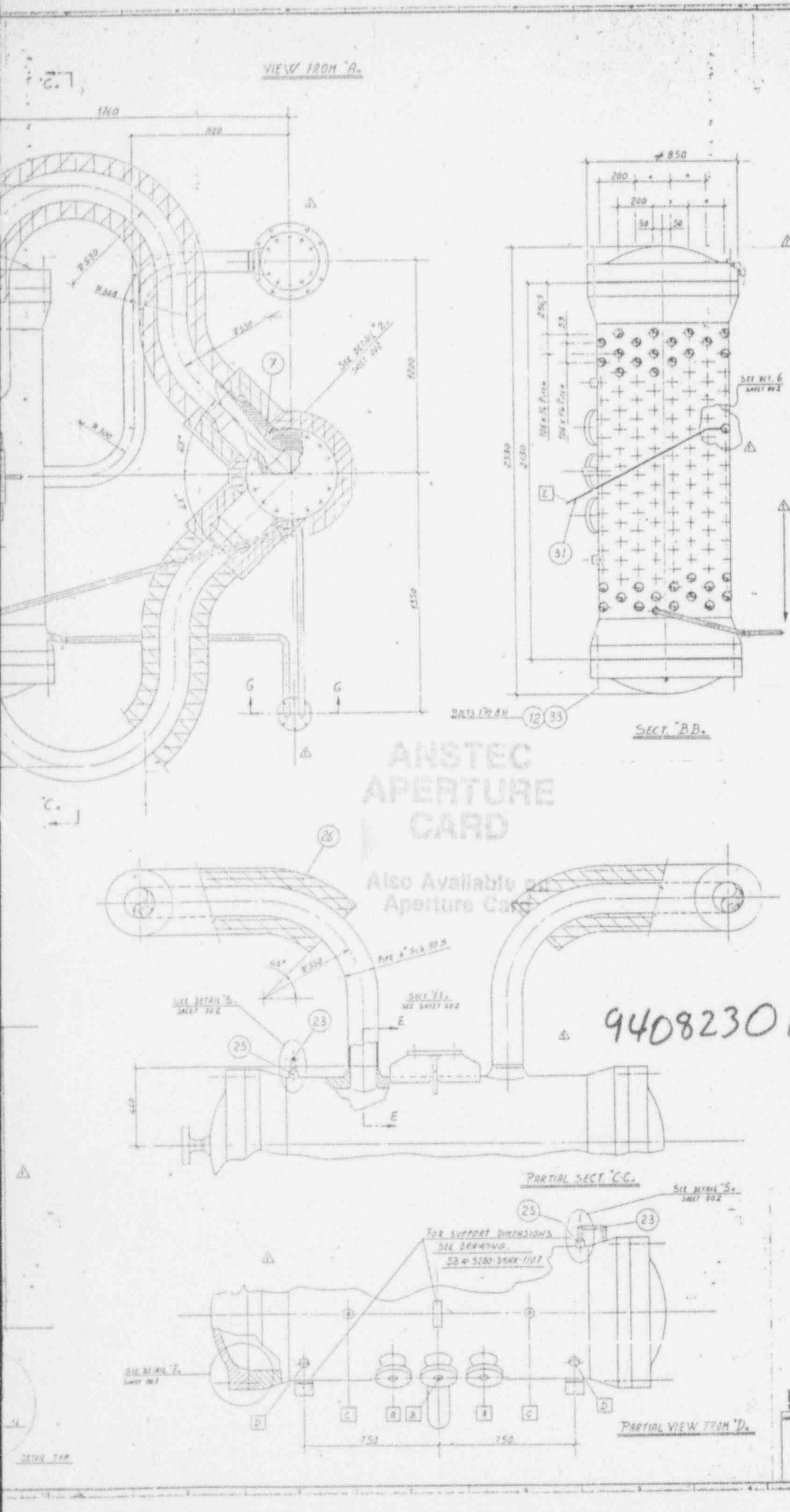

24 HOLES EQUALLY SPACED (22 ⌀ 1"3/4 + 2 ⌀ 1"3/4 H10
FOR HOLES POSITION SEE SHEET 001)
SEE DETAIL 4.
⌀ 620
⌀ 726
⌀ 770
⌀ 850
DETAIL "1.
SCALE 1:5
Det. "3.
R 200
1½ 8 N
⌀ 32
SEE NOTE ON SHEET 001
DETAIL "7.
⌀ 325
⌀ 275
⌀ 225
⌀ 200
⌀ 145
THICKNESS 5
DETAIL SCALE 1:2.5
⌀ 630
⌀ 750
DETAIL 1:2.5
DETAIL "4.
SCALE 1:1
⌀ 664
⌀ 716
METAL 'O' RINGS SELF ENERGIZING
TYPE FLUOROCARBON, OR APPROVED
EQUAL. ⌀ TUBE 3.5 THK. 0.13
N° SLOTS 8
VIEW FROM "R.
JOINT DETAILED DIMENSION TO BE
CONFIRMED AFTER WELDING PROCEDURE
QUALIFICATION C/O MANUFACTURER
AFTER WELDING
BEFORE WELDING
DETAIL "3.
SCALE 1:1
BEFORE WELDING
⌀ 210
⌀ 168.3
⌀ 146.3
R 25
NOTE
DIMENSION TBD IN ACCORDANCE
WITH FORGING MANUFACTURER
SECT "EE.
SCALE 1:2.5
NOTE
DIMENSION TBD IN ACCORDANCE
WITH FORGING MANUFACTURER
⌀ 97.2
⌀ 114.3
⌀ 145
SECT "FF.
SCALE 1:2.5



DETAIL "2" SCALE 1:5
DETAIL "6" SCALE 1:1
DETAIL SCALE 1:2.5
DETAIL SCALE 1:2.5 SECT. "PP."
12-1" 8UNC EQUALLY SPACED
METAL O-RINGS SELF ENERGIZING Type FLUOROCARBON or APPROVED EQUAL
1½ 8UNC
DETAIL SCALE 1:2.5
16-1" 8UNC EQUALLY SPACED
VIEW FROM "Y"
FLEXITALLIC GASKET
16-1" 8UNC EQUALLY SPACED
SLIP-ON FLANGE 12" TYPE III 150 LB.
12 HOLES ϕ25.4 EQUALLY SPACED
VIEW FROM "P"
DETAIL SCALE 1:2.5
SECTION "H.H." SCALE 1:2.5
Pipe 12" Sch 80
Pipe 20" Sch 80
DETAIL "5" TYPICAL SCALE 1:1
WELDING
SECT "I-I" SCALE 1:2.5
VIEW FROM "H"
VIEW FROM "L"
ANSTEC APERTURE CARD
Also Available on Aperture Card
ANG5-0327
SBWR
403.7.1.1
ANSALDO
DISEGNO DI COMPONENTE VARIO
ISOLATION CONDENSER
PROTOTYPE
GENERAL ARRANGEMENT
A0 1:5 1:2.5
SBWS280DHxx1106 002 002
9408230134-01



A.
FOR SUPPORT DIMENSIONS
SEE DRAWING
SBX-5280-DMNX-1107
FOR SUPPORT DIMENSIONS
SEE DRAWING
SBX-5280 DMNX-1107
SEE DETAIL '1.
SHEET 002
120 TUBES
B.
B.
D.
BOLTS 1" 8UNC
SECT. 'F'.
SEE SHEET 002
20 HOLES ⌀31.8
N.H. FLANGE 20" TYPE C
150 LB.
SECTION 'GG'.
DETAIL '1.
SMALL VENT LINE
THIS PINS WILL BE USED
FOR COVER REMOVAL
ONLY
THEY ARE POSITIONED
IN LIEU OF TWO BOLTS



9408230134-02

ITEM	QTY	DESCRIPTION	MATERIAL	DETAIL
1	1	MAIN STEAM LINE	SA-333 Gr. 6	
2	1	GUARD PIPE	SA-333 Gr. 6	
3	1	STEAM DISTRIBUTOR	SA 508 Cl. 3	
4	16	BOLT FOR DISTRIBUTOR COVER	SA 540	
5	1	DISTRIBUTOR COVER	SA 508 Cl. 3	
6	1	FLOW DISTRIBUTOR DEVICE	SA 182 - F 306	
7	2	VENTURI INSERT	SB 564 (INCONEL 600)	
8	2	FEED LINE	SB 167 (INCONEL 600)	
9	2	HEADER	SB 564 (INCONEL 600)	
10	120	TUBE	SB 163 (INCONEL 600)	
11	4	HEADER COVER	SB 564 (INCONEL 600)	
12	96	BOLT FOR HEADER COVER	SB 637 (INCONEL 718)	
13	8	CENTERING PIN	SB 637 (INCONEL 718)	
14	1	DRAIN LINE	SB 167 (INCONEL 600)	
15	1	DRAIN "T" FORGING	SA 182 F 304 L	
16	1	DRAIN "T" FORGING COVER	SA 182 - F 304 L	
17	1	FLOW PLUG	SA 312/240 TP 304	
18	12	BOLT FOR "T" FORGING COVER	SA 193 B8	
19	1	MAIN DRAIN LINE	SA 312 TP 304 L	
20	1	DRAIN PENETRATION PIPE	SA 312 TP 304	
21	1	DRAIN FLANGE	SA 182 TP 304	
22	1	GUARD PIPE FLANGE	SA 105	
23	—	VENT LINE	SA 312 TP 304 L	
24	1	VENT FLANGE	SA 240 TP 304	
25	2	VENT NOZZLE	SB 564 (INCONEL 600)	
26	—	REMOVABLE INSULATION	—	NOT IN THE SCOPE OF SUPPLY
27	1	CONCENTRIC REDUCER 16"×12"	SA 333 Gr. 6	
28	8	THREADED OUTLET 1/4" NPT	SA 182 F 304 L	
29	7	WELDOLETS 2"	SA 182 F 304 L	
30	7	WELDING NECK FLANGE 2" 1500 LB	SA 182 F 304 L	
31	11	TUBE	SA 182 F 304 L	NOT IN THE SCOPE OF SUPPLY
32	—	VENT LINE SUPPORT	—	
33	96	THREADED BUSH	SB 637 (INCONEL 718)	

KEY	
A	TUBE BUNDLE INSTRUMENTATION CABLE NOZZLES
B	HEADER INSTRUMENTATION CABLE NOZZLE
C	PRESSURE GAUGE CONNECTION NOZZLE
D	THERMOCOUPLE CONNECTION NOZZLE
E	PRESSURE GAUGE OUTLET (TYP.)
F	COVER INSTRUMENTATION CABLE NOZZLE

ANGE-0327

Rev.	Date	Description				
1	7-4-92	[illegible]				
0	13-12-91					

Progetto: SBWR

Commessa: 4037.11 — DNV1NC11H

Tipo doc.: DRV — Form.: A0 — Scala: 1:10

ANSALDO

Titolo: DISEGNO DI COMPONENTE VARIO

ISOLATION CONDENSER PROTOTYPE GENERAL ARRANGEMENT

Identificativo: SBW5280DMNX1104001 002