ATTACHMENT A

Revise the Technical Specifications as follows:

Remove Pages	Insert Pages
2-4	2-4
2-8	2-8
2-10	2-10
3/4 2-14	3/4 2-14
3/4 3-6	3/4 3-6
3/4 3-8	3/4 3-8

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWANCE (TA)	Z	S	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	0	≤ 109% of RTP*	≤ 111.1% of RTP*
b. Low Setpoint	8.3	4.56	0	≤25% of RTP*	≤27.1% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	1.6	0.50	0	≤ 5% of RTP* with a time constant ≥ 2 seconds	≤ 6.3% of RTP* with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	1.6	0.50	0	≤ 5% of RTP* with a time constant ≥ 2 seconds	≤ 6.3% of RTP* with a time constant ≥ 2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤ 25% of RTP*	≤ 30.9% of RTP*
6. Source Range, Neutron Flux	17.0	10.01	0	≤ 10^5 cps	≤ 1.4 x 10^5 cps
7. Overtemperature ΔT	~~7.3~~ 7.0	~~5.18~~ 5.10	See Note 5	See Note 1	See Note 2
8. Overpower ΔT	~~4.8~~ 4.9	~~1.38~~ 1.71	~~1.72~~ 1.49	See Note 3	See Note 4
9. Pressurizer Pressure-Low	3.1	0.71	1.67	≥ 1945 psig***	≥ 1935 psig***
10. Pressurizer Pressure-High	6.2	4.96	0.67	≤ 2375 psig	≤ 2383 psig
11. Pressurizer Water Level-High	8.0	2.18	1.67	≤ 92% of instrument span	≤ 93.8% of instrument span
12. Loss of Flow	2.5	~~2.01~~ 1.39	0.60	≥ 90% of loop design flow**	≥ ~~89.5%~~ 88.8 of loop design flow**

* = RATED THERMAL POWER
** Loop design flow = 88,500 gpm
*** Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 2 seconds for lead and 1 second for lag. Channel calibration shall ensure that these time constants are adjusted to those values.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION (Continued)

T	=	Average temperature, °F;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;
T'	=	$\leq$ 576.2°F (Nominal T_{avg} at RATED THERMAL POWER);
K_3	=	0.00082;
P	=	Pressurizer Pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) For $q_t - q_b$ between -33% and +9%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) For each percent that the magnitude of $q_t - q_b$ exceeds -33%, the ΔT Trip Setpoint shall be automatically reduced by 2.52% of its value at RATED THERMAL POWER; and

(iii) For each percent that the magnitude $q_t - q_b$ exceeds +9%, the ΔT Trip Setpoint shall be automatically reduced by 1.75% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~1.7~~ 1.6% of ΔT span. |

PROPOSED

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION (Continued)

$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s;
K_6	=	0.0012/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$;
T	=	Average Temperature, °F;
T"	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤576.2°F);
S	=	Laplace transform operator, s^{-1}; and
$f_2(\Delta I)$	=	0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~2.6%~~ 2.5% of ΔT span.

NOTE 5: The sensor error for temperature is ~~1.72%~~ 1.49% and 0.73% of span for pressure.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	3 Loops in Operation
Reactor Coolant System T_{avg}	≤ ~~580.2~~ 580.3°F
Pressurizer Pressure	≥ 2220 psia*
Reactor Coolant System Total Flow Rate	≥ 274,800 gpm**

*Limit not applicable during either a THERMAL POWER ramp increase in excess of 5 percent RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of 10% RATED THERMAL POWER.

**Includes a 3.5% flow measurement uncertainty.

PROPOSED

TABLE 3.3-1 (Continued)

b. Above P-6 but below 5% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 5% of RATED THERMAL POWER.

c. Above 5% of RATED THERMAL POWER, POWER OPERATION may continue.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below P-6, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above P-6 setpoint and suspend positive reactivity operations.

b. Above P-6, operation may continue.

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers, suspend all operations involving positive reactivity changes and verify Valve 2CHS-91 is closed and secured in position within the next hour.

ACTION 6 - This Action is not used.

ACTION 7 - With the number of OPERABLE channels* one less than the Total Number of Channels and with the THERMAL POWER level:

a. Less than or equal to 5% of RATED THERMAL POWER, place the inoperable channel in the tripped condition within 1 hour; restore the inoperable channel to operable status within 24 hours after increasing THERMAL POWER above 5% of RATED THERMAL POWER; otherwise reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the following 6 hours.

b. Above 5% of RATED THERMAL POWER, place the inoperable channel in the tripped condition within 1 hour; operation may continue until performance of the next required CHANNEL FUNCTIONAL TEST.

ACTION 8 - With the number of OPERABLE channels one less than the Total Number of Channels and with the THERMAL POWER level above P-9, place the inoperable channel in the tripped condition within 1 hour; operation may continue until performance of the next required CHANNEL FUNCTIONAL TEST.

ACTION 9 - This ACTION is not used

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*An OPERABLE hot leg channel consists of: 1) Three RTD's per hot leg, or 2) two RTD's per hot leg with the failed RTD disconnected and the required bias applied.

PROPOSED

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux	≤ 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux, High Negative Rate	≤ 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux (Below P-10)	NOT APPLICABLE
7. Overtemperature ΔT	≤ ~~5.5~~ 6.0 seconds*
8. Overpower ΔT	≤ ~~5.5~~ 6.0 seconds*
9. Pressurizer Pressure--Low (Above P-7)	≤ 2.0 seconds
10. Pressurizer Pressure--High	≤ 2.0 seconds
11. Pressurizer Water Level--High (Above P-7)	NOT APPLICABLE
12. Loss of Flow - Single Loop (Above P-8)	≤ 1.0 seconds
13. Loss of Flow - Two Loop (Above P-7 and below P-8)	≤ 1.0 seconds
14. Steam Generator Water Level--Low-Low (Loop Stop Valves Open)	≤ 2.0 seconds
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	NOT APPLICABLE
16. Undervoltage-Reactor Coolant Pumps (Above P-7)	≤ 1.5 seconds
17. Underfrequency-Reactor Coolant Pumps (Above P-7)	≤ 0.9 seconds

*Neutron detectors are exempt from response time testing. Response time shall be measured from detector output or input of first electronic component in channel.

ATTACHMENT B

Beaver Valley Power Station, Unit No. 2
Proposed Technical Specification Change No. 38
REACTOR TRIP INSTRUMENTATION REVISION
RTD BYPASS ELIMINATION

A. DESCRIPTION OF AMENDMENT REQUEST

DCP-1469 will incoporate a plant design change during the 2nd refueling outage (September 1990) to remove the existing Reactor Coolant System (RCS) hot leg and cold leg Resistance Temperature Detector (RTD) bypass system and replace it with fast response thermowell mounted RTD'S intalled in the reactor coolant loop piping. The revised design will affect the FSAR and UFSAR response time design basis for the Overtemperature and Overpower delete T and loss of flow reactor trip functions. WCAP-12478 entitled "RTD Bypass Elimination Licensing Report for Beaver Valley Unit 2" (Proprietary) is attached and describes the extensive analyses, evaluation and testing performed to ensure the new design meets all safety, licensing and control requirements necessary for the safe operation of the plant. The proposed technical specification amendment would support this design change by incorporating the revised response times into the applicable reactor trip functions.

B. BACKGROUND

Separate bypass manifolds for each reactor coolant loop hot and cold leg are provided so that individual temperature signals may be developed for use in the reactor control and protection system. The bypass manifold around each steam generator obtains a representative hot leg temperature by mixing the flow from three scoop connections, which extend into the flow stream at locations 120 degrees apart in the cross-sectional plane, on the reactor coolant leg. The hot leg bypass flow exits the manifold to a common return line.

Flow for the cold leg bypass manifold, which bypasses the RCP, is obtained downstream of the pump discharge. Because of the mixing action of the pump, only one connection is required to obtain a representative sample. This connection is located as close as possible to the weld connection at the pump discharge and is in the same relative position in each loop.

The bypass manifold lines join downstream of each of the hot and cold leg manifolds and discharge into a common line. The combined bypass flow passes through a flow indicator before discharging to the suction side of the RCP.

The manifolds are not provided with thermometer wells. Instead, the resistance temperature detectors extend directly into the flow path. Therefore, two isolation valves in series are provided on each side of the bypass manifold to allow for resistance temperature detector maintenance. The valve nearest the connection to the main coolant piping is located above the centerline of the reactor coolant pipe to permit valve repair during cold shutdown. In addition, vents and drains are provided in each manifold to be used, in conjuction with the isolation valve, for maintenance.

The plant design modifications will involve replacing the current RTD bypass system with RTD's installed in each reactor coolant loop to obtain the individual loop temperature signals for input to the reactor control and protection system. The design change will eliminate the bypass line components which have been a source of plant outages as well as occupational radiation exposure

If plant interferences preclude the placement of a thermowell in a scoop, then the scoop will be capped and a new penetration made to accomodate the thermowell as shown in the WCAP Figure 1.2-2.

C. JUSTIFICATION

The NRC has identified (Information Notice 88-13) potential problems resulting from the improper application of packless metal diaphragm valves supplied by Kerotest. The cause of the valve problems was attributed to excessive piping vibration induced by "chugging" during reverse flow through these valves. We have identified twelve (12) Kerotest packless metal diaphragm globe valves located in the RTD bypass manifold piping system which could experience reverse flow. The RTD bypass manifold piping is a source of increased radiation exposure from required maintenance on the bypass piping and from crud traps which increase the radiation exposure throughout the loop compartments. The RTD's in the bypass manifolds are susceptible to physical damage during outage activities and difficulties have been identified with replacement of the direct immersion type RTD's. Problems have also been incountered in torquing the flanged connections in the RTD bypass manifold piping to achieve a leak-tight joint. Therefore, elimination of the RTD bypass piping and associated valves and flanged connections will reduce maintenance and operating problems in addition to reducing personnel radiation exposure.

The new thermowell mounted RTD's have a response time equal to, or better than, the current bypass piping RTD. This will allow the total RCS temperature measurement response time to remain unchanged at 6.0 seconds. As stated in the WCAP, the differences in response time characteristics and instrumentation uncertainties associated with the fast response thermowell RTD system have been analyzed for those UFSAR accident analayses that may be affected by the proposed design change. It was concluceded that a small increase in RCS temperature uncertainty (from 4.0 to 4.1 deg F) can be accomodated by margins in the safety analyses acceptance criteria limits and allocation of generic DNB margin. All other safety analysis assumptions remain valid. Therefore, the applicable UFSAR non-LOCA safety analyses conclusions and acceptance criteria continue to be met. It was determined that for the UFSAR Small and Large Break LOCA events a Peak Clad Temperature (PCT) penalty of 1 degree F was assigned to account for the small increase in RCS temperature uncertainty with the results indicating the analyses continue to meet existing acceptance criteria.

The WCAP provides a discussion of the effects on the plant instrumentation and control functions and concludes that compliance with IEEE 279-1971, applicable general design criteria and industry standards and regulatory guides will not be changed. The mechanical effects are also discussed and it is concluded that the integrity of the reactor coolant piping as a pressure boundary component is maintained by adhering to the applicable ASME Code sections and the pressure retaining capability and fracture prevention characteristics of the piping will not be compromised by these modifications.

D. SAFETY ANALYSIS

Based on the calculations described in WCAP-12478, the following protective function technical specification requiremnts must be revised:

1) Change the Table 2.2-1 values listed for the following functional unit prameters (from/to):

Functional Unit	Allowance (TA)	Z	S	Allowable Value
Overtemperature delta T	(7.3/7.0)	(5.18/5.10)		
Overpower delta T	(4.8/4.9)	(1.38/1.71)	(1.72/1.49)	
Loss of Flow		(2.01/1.39)		(89.5/88.8)

2) Change the Table 2.2-1 maximum allowable value not to be exceeded for:

Note 2 from 1.7 percent to 1.6 percent, and for

Note 4 from 2.6 percent to 2.5 percent

3) Change the Table 2.2-1 note 5 sensor error for temperature from 1.72 percent to 1.49 percent.

4) Change the Table 3.2-1 Reactor Coolant System Tavg from less than or equal to 580.2 F to less than or equal to 580.3 F.

5) Add an * note to Table 3.3-1 Action 7 to identify the use of two hot leg RTD's per loop provided the failed RTD is disconnected and the required bias is applied.

6) Change the Table 3.3-2 response time for the item 7 Overtemperature delta T and the item 8 Overpower delta T from 5.5 seconds to 6 seconds.

Three fast response, narrow range, single element thermowell mounted RTD's will be installed on each hot leg. The thermowells will be located within the three existing RTD bypass manifold scoops and measure the differential temperature and average temperature. One fast response, narrow range, dual-element RTD will be located in each cold leg to replace the cold leg RTD located in the bypass manifold. One element of the RTD will be considered active and the other element will be a spare. The RTD bypass manifold return line will be capped at the nozzle on the crossover leg. The new thermowell mounted RTD's will be used for both control and protection. The average temperature and differential temperature signals used in the control-grade logic will be input into a median signal selector which will select the signal which is in between the highest and lowest values of the three inputs to avoid any adverse plant response as a result of a single signal failure. The 7300 process electronics that require modification will be qualified to the same level as the existing 7300 electronics.

The FSAR accident analyses are affected due to the different response time characteristics and instrumentation uncertainties between the fast response thermowell RTD system and the current RTD bypasss system. However, as shown in the WCAP Table 2.1-1, the total response time remains the same.

	RTD Bypass System (sec)	Fast Response Thermowell RTD System (sec)
RTD Bypass Piping and Thermal Lag	2.0	N/A
RTD Response Time	2.0	4.0
Electronics Delay	2.0	2.0
Total Response Time	6.0	6.0

As described in UFSAR Section 7.2.1.1.4, the hot and cold leg RTD's are inserted into reactor coolant bypass loops. A bypass loop from upstream of the steam generator to downstream of the steam generator is used for the hot leg RTD's and a bypass loop from downstream of the reactor coolant pump to upstream of the pump is used for the cold leg RTD's. The RTD's are located in manifolds within containment and are inserted into the reactor coolant bypass loop flow without thermowells. UFSAR Section

7.2.2.3.2 decribes the monitoring and testing used to demonstrate the accuracy of the RTD temperature measurements. UFSAR Figures 5.1-1 through 5.1-7 illustrate the existing RTD bypass piping configuration. These and other sections of th UFSAR will be revised as shown in Attchment C to address the design modifications as well as describe how the new thermowell mounted RTDs will be used in the control and protection functions of the plant.

The RTD manufacturer, WEED Instruments, Inc. will perform time response testing of each RTD and thermowell prior to installation. Response time testing of the RTD's will also be performed in-situ to demonstrate that the RTD's can satisfy the response time requirement as installed in the plant. Westinghouse has evaluated data taken from other operating plants and determined the appropriate temperature error to account for the effects of temperature streaming and incorporated this error into the safety analysis and calorimetric flow calculations. The spare cold leg RTD element provides sufficient spare capacity to accommodate a single cold leg RTD failure per loop. Provisions in the RTD electronics allow for operation with only two hot leg RTD's in service. Failure of a hot leg RTD would require manual action to defeat the failed signal and rescale the electronics to average the remaining two hot leg signals. The procedure for using the actual plant bias data is provided in Appendix As of the WCAP.

The method for using fast-response RTD's installed in the reactor coolant loop piping as a means for RCS temperature indication has undergone extensive analyses, evaluation and testing as described in WCAP-12478. Incorporating this system into the plant design meets all safety, licensing and control requirements necessary for the safe operation of the plant. The analytical evaluation has been supplemented with in-plant and laboratory testing to further verify system performance. The fast response RTD's to be installed in the reactor coolant loop piping will adequately replace the present hot and cold leg temperature measurement system and enhance ALARA efforts as well as improve plant reliability. Other nuclear plants similar to Beaver Valley Unit 2 have replaced the RTD bypass system with the fast response thermowell RTD system, including Beaver Valley Unit 1, H. B. Robinson and Salem Units 1 and 2. Therefore, based on the above, the proposed changes have been determined to be safe since the design meets all safety, licensing and control requirements necessary for the safe operation of the plant.

E. NO SIGNIFICANT HAZARDS EVALUATION

The no significant hazard considerations involved with the proposed amendment have been evaluated, focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

The commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

The following evaluation is provided for the no significant hazards consideration standards.

1 Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

A plant design change is scheduled for the 2nd refueling outage to remove the existing Reactor Coolant System (RCS) hot leg and cold leg Resistance Temperature Detector (RTD) bypass system and replace it with fast response thermowell mounted RTD's installed in the reactor coolant loop piping. The revised design will affect the FSAR and UFSAR response time design basis for the Overtemperature and Overpower delta T and loss of flow reactor trip functions. The transients which assume protection from these functions are:

Core Thermal Limit Protection
Loss of Electrical Load/Turbine Trip
Uncontrolled RCCA Bank Withdrawal at Power
CVCS Malfunction that results in decrease in the RCS Boron Concentration (Mode 1)
Partial Loss of Forced Reactor Coolant Flow
Reactor Coolant Pump Locked Rotor
Steamline Break for EQ Outside Containment
Reactor Core Response to Excessive Secondary Steam Releases (at Power)

Westinghouse has prepared WCAP-12478 entitled "RTD Bypass Elimination Licensing Report for Beaver Valley Unit 2" (Proprietary) which describes the extensive analyses, evaluation and testing performed to ensure the new design meets all safety, licensing and control requirements necessary for the safe operation of the plant. Attachment C provides revisions to applicable UFSAR Sections to reflect the bypass piping elimination and replacement with the fast response thermowell RTD system. These UFSAR changes are provided as background information for this technical specification change and will be

included in a future UFSAR update. The UFSAR accident analyses are affected due to the different response time characteristics and instrumentation uncertainties between the fast response thermowell RTD system and the current RTD bypass system, however, the total response time remains the same. The new thermowell mounted RTD's will be used for both control and protection. The average temperature and differential temperature signals used in the control-grade logic will be input into a median signal selector which will select the signal which is in between the highest and lowest values of the three inputs to avoid any adverse plant response as a result of a single signal failure. The 7300 process electronics that require modification will be qualified to the same level as the existing 7300 electronics.

The differences in response time characteristics and instrumentation uncertainties associated with the fast response thermowell RTD system have been analyzed for those UFSAR accident analyses that may be affected by the proposed design change.

It was concluded that a small increase in RCS temperature uncertainty (from 4.0 to 4.1 deg F) can be accomodated by margins in the safety analyses acceptance criteria limits and allocation of generic DNB margin. All other safety analysis assumptions remains valid. Therefore, the applicable UFSAR non-LOCA safety analyses conclusions and acceptance criteria continue to be met. It was determined that for the UFSAR Small and Large Break LOCA events a Peak Clad Temperature (PCT) penalty of 1 degree F was assigned to account for the small increase in RCS temperature uncertainty with the results indicating the analyses continue to meet existing acceptance criteria.

The effects on the plant instrumentation and control functions were evaluated and found to comply with IEEE 279-1971, applicable general design criteria and industry standards and regulatory guides. The mechanical effects were evaluated and it is concluded that the integrity of the reactor coolant piping as a pressure boundary component is maintained by adhering to the applicable ASME Code sections and the pressure retaining capability and fracture prevention characteristics of the piping will not be compromised. Incorporating this system into the plant design meets all safety, licensing and control requirements necessary for the safe operation of the plant. Therefore, the proposed changes will not introduce any adverse safety considerations or involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated.

2. Does the change create the possiblity of a new or different kind of accident from any accident previously evaluated?

The design change will eliminate the bypass line components which have been a souce of plant outages as well as occupational radiation exposure. The new design will continue to perform the functions satisfied by the current system. The proposed change

will be performed in a manner consistent with the applicable standards, preserve the existing design bases, and will not adversely impact the qualification of any plant systems. This will preclude adverse control/protection systems interactions. The design, installation, and inspection of the new thermowell RTD system will be done in accordance with the applicable ASME Boiler and Pressure Vessel Code criteria. Therefore, the probability for an accident or malfunction of a different type than previously evaluated will not be created.

3. Does the change involve a significant reduction in a margin of safety?

The applicable design bases have been evaluated to ensure the new design will provide the overall reliability, redundancy and diversity assumed available in the plant design for the protection and mitigation of accident and transient conditions. The integrated operation of the new thermowell RTD's is consistent with the assumptions used in the accident analyses. The applicable surveillance requirements will continue to ensure the system functional capability is maintained comparable to the original design standards. The response time measurement provides assurance that the protective functions satisfy the time limits assumed in the accident analyses. The integrity of the reactor coolant piping as a pressure boundary component is maintained by adhering to the applicable ASME Code sections and the pressure retaining capability and fracture prevention characteristics of the piping will not be compromised. Therefore, the proposed changes will not involve a significant reduction in the margin of safety.

F. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The method for using fast response RTD's installed in the reactor coolant loop piping as a means for RCS temperature indication has undergone extensive analyses, evaluation and testing as described in WCAP-12478. Incorporating this system into the plant design meets all safety, licensing and control requirements necessary for the safe operation of the plant. The analytical evaluation has been supplemented with in-plant and laboratory testing to further verify system performance. The fast response RTD's to be installed in the reactor coolant loop piping will adequately replace the present hot and cold leg temperature measurement system and enhance ALARA efforts as well as improve plant reliability. Other nuclear plants similar to Beaver Valley Unit 2 have replaced the RTD bypass system with the fast response thermowell RTD system. Based on the above safety analysis, it is concluded that the activities associated with this license amendment request satisfies the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

G. ENVIRONMENTAL EVALUATION

The proposed changes have been evaluated and it has been determined that the changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c) (9). Therefore, pursuant to 10 CFR 51.22 (b), an environmental assessment of the proposed changes is not requried.

ATTACHMENT C

UFSAR Changes

Beaver Valley Power Station, Unit No. 2

Proposed Technical Specification Change No. 38

5.4.3 Reactor Coolant Piping

5.4.3.1 Design Bases

The RCS piping is designed and fabricated to accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions. Stresses are maintained within the limits as defined in Section III of the ASME Nuclear Power Plant Components Code for Class I components. Code and material requirements are provided in Section 5.2.

Materials of construction are specified to minimize corrosion/erosion and ensure compatibility with the operating environment.

The piping in the RCS is Seismic Category I, Safety Class I and is designed and fabricated in accordance with ASME Section III, Class I requirements.

Stainless steel pipe conforms to ANSI B36.19 for sizes 1/2 inch through 12 inches and wall thickness Schedules 40S through 80S. Stainless steel pipe outside of the scope of ANSI B36.19 conforms to ANSI B36.10.

The minimum wall thickness of the loop pipe and fittings are not less than that calculated using the ASME III Class 1, formula three, of Paragraph NB-3641.1 with an allowable stress value of 17,550 psi. The pipe wall thickness for ~~both bypass and~~ the pressurizer surge line~~s~~ is Schedule 160. The minimum pipe bend radius is five nominal pipe diameters; ovality does not exceed 6 percent.

All butt welds, branch connection nozzle welds, and boss welds shall be of a full penetration design.

Processing and minimization of sensitization are discussed in Sections 5.2.3 and 5.2.5.

Flanges conform to ANSI B16.5.

Socket weld fittings and socket joints conform to ANSI B16.11.

5.4.3.2 Design Description

Principal design data for the reactor coolant piping are given in Table 5.4-7. Beaver Valley Power Station - Unit 2 is provided with loop stop valves on each hot and cold leg. These valves are described in Section 5.4.12.

Reactor coolant loop pipe is seamless forged. Reactor coolant loop fittings are cast seamless without longitudinal or electroslag welds. Pipe and fittings comply with the requirements of the ASME Code, Section II, Parts A and C, Section III, and Section IX.

The RCS piping is specified in the smallest sizes consistent with system requirements. This design philosophy results in the reactor inlet and outlet piping diameters given in Table 5.4-7. The line between the steam generator and the pump suction is larger to reduce pressure drop and improve flow conditions to the pump suction.

The reactor coolant piping and fittings which make up the loops are austenitic stainless steel. All smaller piping which comprise part of the RCS such as the pressurizer surge line, ~~loop bypass lines~~, spray and relief line, loop drains, and connecting lines to other systems are austenitic stainless steel. The thermal sleeve in the charging line connection is also austenitic stainless steel. The nitrogen supply line for the pressurizer relief tank is carbon steel. All joints and connections are welded, except for the pressurizer code safety valves, where flanged joints are used.

All piping connections from auxiliary systems are made above the horizontal centerline of the reactor coolant piping, with the exception of:

1. Residual heat removal (RHR) pump suction lines, which are 45 degrees down from the horizontal centerline. This enables the water level in the RCS to be lowered in the reactor coolant pipe while continuing to operate the residual heat removal system (RHRS), should this be required for maintenance.

2. Loop drain lines and the connection for temporary level measurement of water in the RCS during refueling and maintenance operation.

3. The differential pressure taps for flow measurement, which are downstream of the steam generators on the first 90-degree elbow.

4. The pressurizer surge line, which is attached at the horizontal centerline.

5. The safety injection connections of the hot leg, for which inservice inspection requirements and space limitations dictate location on the horizontal centerline.

6. Two of the three Narrow Range RTD thermowell bosses in each hot leg ~~taps in each resistance temperature detector hot leg connection~~.

7. One hot leg sample connection and loop Wide range RTD thermowells, all located on the horizontal centerline.

Penetrations into the coolant flow path are limited to the following:

1. The spray line inlet connections extend into the cold leg piping in the form of a scoop so that the velocity head of

In the original design these scoops collected a representative temperature sample for the RTD manifold. They now provide a convenient location for narrow range, thermowell mounted RTDs.

the reactor coolant loop flow adds to the spray driving force.

2. The reactor coolant sample system taps protrude into the main stream to obtain a representative sample of the reactor coolant.

3. The resistance temperature detector hot leg ~~bypass connections are~~ scoops ~~which~~ extend into the reactor coolant. ~~to collect a representative temperature sample for the resistance temperature detector manifold~~.

4. The wide range hot and cold leg temperature detectors ~~and the cold leg fast response temperature detectors~~ are located in resistance ~~temperature detector wells~~ thermowells that extend into the reactor coolant pipes.

Insert A

~~Separate bypass manifolds (loops) for each reactor coolant loop hot~~ and cold leg are provided so that individual temperature signals may be developed for use in the reactor control and protection system. The bypass manifold around each steam generator obtains a representative hot leg temperature by mixing the flow from three scoop connections, which extend into the flow steam at locations 120 degrees apart in the cross-sectional plane, on the reactor coolant leg. The hot leg bypass flow exits the manifold to a common return line.

Flow for the cold leg bypass manifold, which bypasses the RCP, is obtained downstream of the pump discharge. Because of the mixing action of the pump, only one connection is required to obtain a representative sample. This connection is located as close as possible to the weld connection at the pump discharge and is in the same relative position in each loop.

The bypass manifold lines join downstream of each of the hot and cold leg manifolds and discharge into a common line. The combined bypass flow passes through a flow indicator before discharging to the suction side of the RCP.

The manifolds are not provided with thermometer wells. Instead, the resistance temperature detectors extend directly into the flow path to reduce the time delay to a minimum. Therefore, two isolation valves in series are provided on each side of the bypass manifold to allow for resistance temperature detector maintenance. The valve nearest the connection to the main coolant piping is located above the centerline of the reactor coolant pipe to permit valve repair during cold shutdown. In addition, vents and drains are provided in each manifold to be used, in conjunction with the isolation valve ~~for maintenance~~.

Signals from these instruments are used to compute the reactor coolant ΔT (temperature of the hot leg, T_{hot}, minus the temperature

INSERT A

Each hot leg has three narrow range, thermowell mounted, fast response RTDs located in approximately the same plane 120 degrees apart. These RTDs extend into the reactor coolant fluid, sensing the temperature at three distinct locations within the hot leg pipe. These three measurements are electronically averaged to provide a representative Thot indication.

The cold leg is provided with a dual element, narrow range, thermowell mounted, fast response RTD. One element provides the cold leg temperature measurement and the other element is an installed spare.

of the cold leg, T_{cold}) and an average reactor coolant temperature (T_{avg}). The T_{avg} for each loop is indicated on the main control board.

The RCS piping includes those sections of piping interconnecting the reactor vessel, steam generator, RCP, and loop stop valves. It also includes the following:

1. Charging line from the system isolation valve up to the branch connections on the reactor coolant loop.

2. Letdown line and excess letdown line from the branch connections on the reactor coolant loop to the system isolation valve.

3. Pressurizer spray lines from the reactor coolant cold legs to the spray nozzle on the pressurizer vessel.

4. Residual heat removal lines to or from the reactor coolant loops up to the designated check valve or isolation valve.

5. Safety injection lines from the designated check valve to the reactor coolant loops.

6. Accumulator lines from the designated check valve to the reactor coolant loops.

~~7. Resistance temperature detector manifold bypass loop piping~~.

7 ~~8~~. Loop fill, loop drain, sample*, and instrument*, lines to or from the designated isolation valve to or from the reactor coolant loops.

8 ~~9~~. Pressurizer surge line from one reactor coolant loop hot leg to the pressurizer vessel surge nozzle.

9 ~~10~~. Loop bypass lines between the loop stop valves of each loop.

10 ~~11~~. ~~Resistance temperature detector scoop element,~~ Pressurizer spray scoop, sample connection* with scoop, reactor coolant RTD thermowell ~~temperature element~~ installation boss, and the ~~temperature element wel~~l itself. RTD thermowell

11 ~~12~~. All branch connection nozzles attached to reactor coolant loops.

12 ~~13~~. Pressure relief lines from nozzles on top of the pressurizer vessel up to and through the power operated pressurizer relief valves (PRVs) and pressurizer safety valves.

13 ~~14~~. Seal injection water to or from the RCP inside the reactor containment.

14 ~~25~~. Auxiliary spray line from the isolation valve to the pressurizer spray line header.

15 ~~26~~. Sample lines*, instrument lines*, and vent lines* from pressurizer to the isolation valve.

16 ~~27~~. Reactor vessel head vent lines from the reactor vessel head to the excess letdown lines.

Note: *Lines with a 3/8-inch flow restricting orifice located below normal pressurizer level qualify as Seismic Category II; in the event of a break in one of these Category II lines, the normal makeup system is capable of providing makeup flow while maintaining pressurizer water level.

Details of the materials of construction and codes used in the fabrication of reactor coolant piping and fittings are discussed in Section 5.2.1.

5.4.3.3 Design Evaluation

Piping load and stress evaluation for normal operating loads, seismic loads, accident loads, and combined normal, accident and seismic loads is discussed in Section 3.9.1.

5.4.3.3.1 Material Corrosion/Erosion Evaluation

The water chemistry is specified and maintained to minimize corrosion. A periodic analysis of the coolant chemical composition is performed to verify that the reactor coolant quality meets the specifications.

The configuration and weld finishes are designed to facilitate inservice inspection as required by ASME Section XI. Pursuant to this, all pressure retaining welds out to the second valve that delineates the RCS boundary are provided with removeable insulation to facilitate examination.

Components constructed with stainless steel will operate satisfactorily under normal plant chemistry conditions in PWR systems, because chlorides, fluorides, and particularly oxygen, are controlled to very low levels (Section 5.2.3).

Periodic analysis of the coolant chemical composition is performed to monitor the adherence of the system to desired reactor coolant water quality listed in Table 5.2-5. Maintenance of the water quality to minimize corrosion is accomplished using the chemical and volume control system (CVCS) and sampling system which are described in Section 9.3.

5.4.3.3.2 Sensitized Stainless Steel

Sensitized stainless steel is discussed in Section 5.2.3.

5.4.3.3.3 Containment Control

Copper, low melting temperature alloys, mercury, and lead are prohibited to preclude contamination of stainless steel and Inconel. Colloidal graphite is the only permissible thread lubricant.

Prior to application of thermal insulation, the austenitic stainless steel surfaces are cleaned and analyzed to a halogen limit of 0.0015 mg Cl/dm^2 and 0.0015 mg F/dm^2.

5.4.3.4 Tests and Inspections

The RCS piping NDE program is given in Table 5.4-8.

Volumetric examination is performed throughout 100 percent of the wall volume of each pipe and fitting in accordance with the applicable requirements of Subsection NB of the ASME Section III code for all pipe 27 1/2 inches and larger. All unacceptable defects are eliminated in accordance with the requirements of the same section of the code.

A liquid penetrant examination is performed on both the entire outside and inside surfaces of each finished fitting in accordance with the criteria of ASME Section III. Acceptance standards are in accordance with the applicable requirements of ASME Section III.

The pressurizer surge [redacted] line~~s~~ conform to SA-376 Grade 304, 304N, or 316 with supplementary requirements S2 (transverse tension tests), and S6 (ultrasonic test). The S2 requirement applies to each length of pipe. The S6 requirement applies to 100 percent of the piping wall volume.

The end of pipe sections, branch ends, and fittings are machined back to provide a smooth weld transition adjacent to the weld path.

In addition, all piping not supplied by the nuclear steam supply system (NSSS) vendor, which forms part of the RCPB, is examined in accordance with the applicable subsection of ASME III.

5.4.4 Main Steam Line Flow Restrictor

5.4.4.1 Design Basis

The outlet nozzle of the steam generator is provided with a flow restrictor designed to limit steam flow in the unlikely event of a break in the main steam line. A large increase in steam flow will create a back pressure which limits further increase in flow. Several protective advantages are thereby provided: rapid rise in

TABLE 5.1-2

REACTOR COOLANT SYSTEM
STEADY STATE FULL POWER OPERATION

Location*	Fluid	Pressure (psig)	Temperature (°F)	Flow gpm**	Flow lb/hr (x 10⁶)
1	Reactor Coolant	2,235.0	609.9	~~109,120~~	~~37.996274~~
2	Reactor Coolant	2,230.7	609.9	~~109,001~~	~~36.9620~~
3	Reactor Coolant	2,190.7	542.3	~~97,342~~	~~36.9620~~
4	Reactor Coolant	2,187.7	542.3	~~97,494~~	~~37.01963~~
5	Reactor Coolant	2,280.1	542.5	~~97,500~~	~~37.0218~~
6	Reactor Coolant	2,275.8	542.5	~~97,433~~	~~36.996274~~
~~7~~	~~Reactor Coolant~~	~~2,230.7~~	~~609.9~~	~~33.8~~	~~0.01146~~
~~8~~	~~Reactor Coolant~~	~~2,230.7~~	~~609.9~~	~~33.8~~	~~0.01146~~
~~9~~	~~Reactor Coolant~~	~~2,230.7~~	~~609.9~~	~~33.8~~	~~0.01146~~
~~10~~	~~Reactor Coolant~~	~~2,204.6~~	~~609.9~~	~~101.4~~	~~0.03438~~
~~11~~	~~Reactor Coolant~~	~~2,199.2~~	~~542.5~~	~~60.9~~	~~0.0231~~
~~12~~	~~Reactor Coolant~~	~~2,188.3~~	~~564.5~~	~~162~~	~~0.0575~~
~~13~~ 7	Reactor Coolant	2,235.0	652.7	2.5	0.00076
~~14~~ 8	Reactor Coolant	2,280.1	542.5	1.0	0.00038
~~15~~ 9	Reactor Coolant	2,235.0	652.7	2.5	0.00076
~~16~~ 10	Reactor Coolant	2,235.0	652.7	-	-
~~17~~ 11	Steam	2,235.0	652.7	-	-
~~18~~ 12	Reactor Coolant	2,235.0	542.5	2.0	0.00076
~~19~~ 13	Steam	2,235.0	652.7	0	0
14,15,16 ~~20,21,22~~	N_2	3.0	120.0	0	0
~~23~~ 17	N_2	3.0	120.0	0	0
~~24~~ 18	Reactor Makeup Water	3.0	120.0	0	0

NOTES:

*Refer to Figure 5.1-1 for these locations.
**At the conditions specified.

108,267	36.796236
108,269	36.796996
96,800	36.796996
96,800	36.796996
96,800	36.796996
96,799	36.796616

INSERT

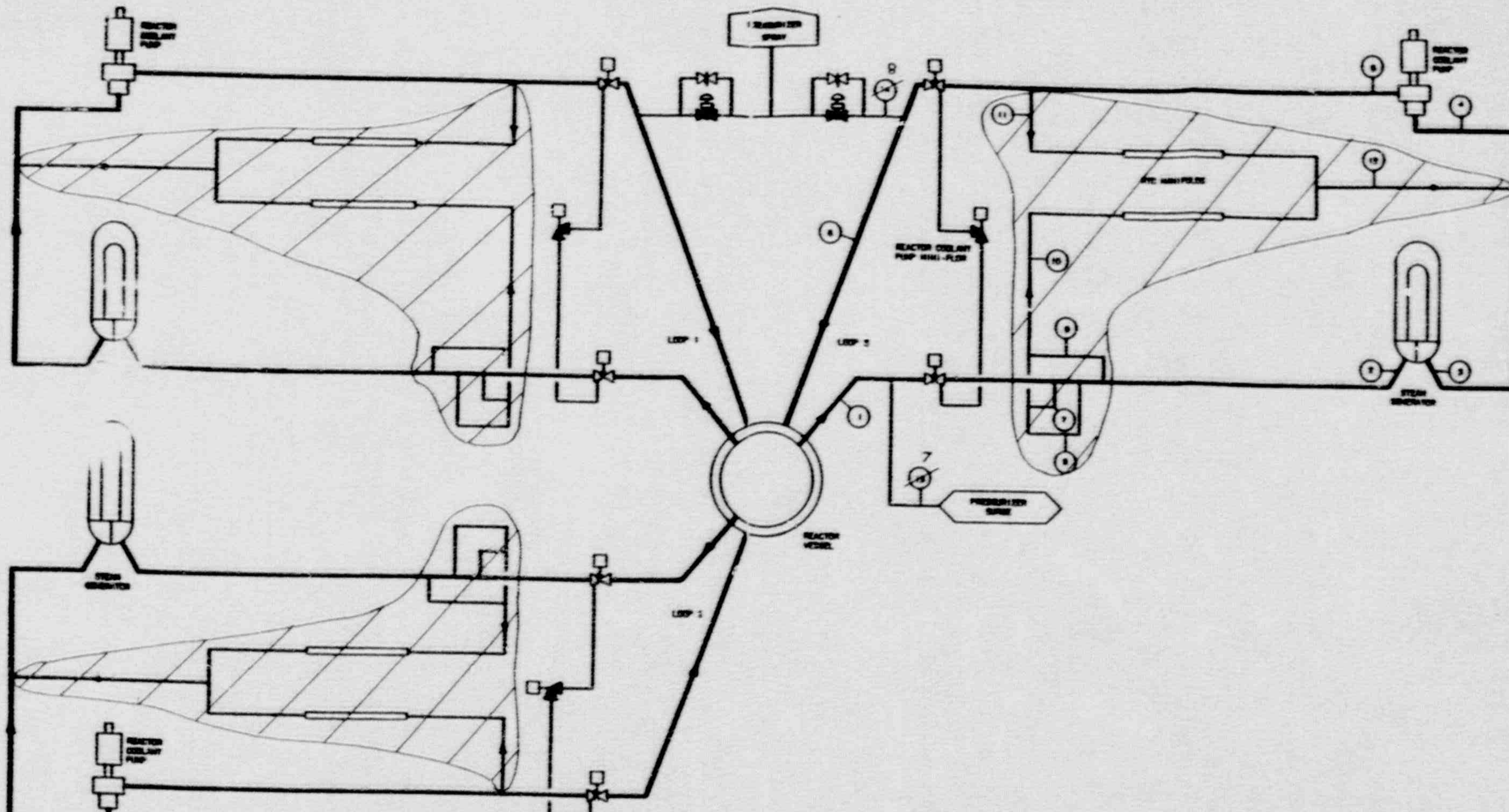

FIGURE 5.1-1 (SH. 1 OF 2)
PROCESS FLOW DIAGRAM
REACTOR COOLANT SYSTEM
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

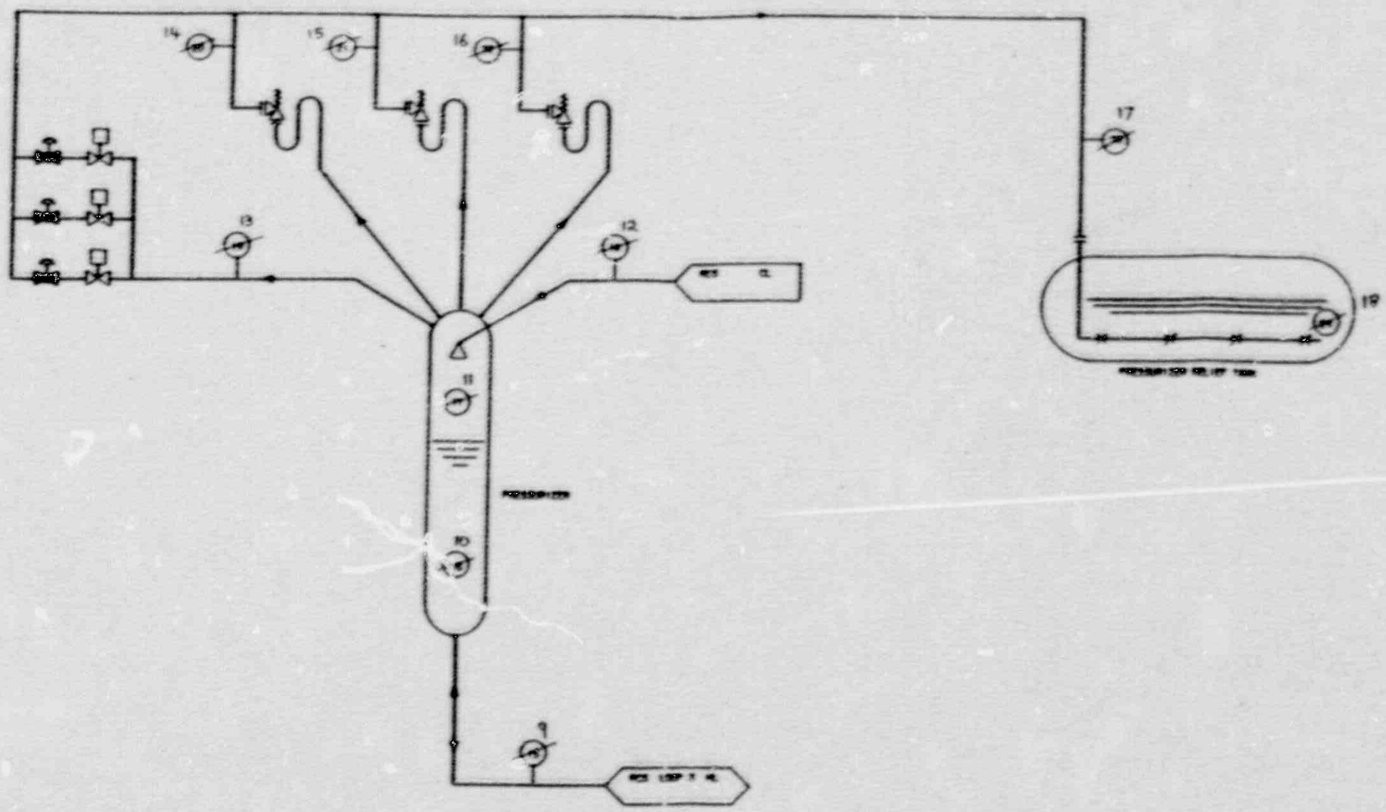

FIGURE 5.1-1 (SH. 2 OF 2)
PROCESS FLOW DIAGRAM
REACTOR COOLANT SYSTEM
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

FERENCE KEY

ID NO.	FSAR FIGURE NO.
-3A	11.2-3
-3B	11.2-4
-4	11.2-5
-5	11.2-6
-1A	10.2-2
-1B	10.2-3
-2	10.2-7
-3	10.4-1
-4	10.2-1
-6	10.2-6
-7	10.2-8
-17	9.4-16
-1	10.4-25
-2	10.4-26
-3	10.4-32
-4	10.4-26A
3-1	10.4-37
3-2	10.4-38
3-3	10.4-39
3-4	10.4-40
-1A	9.2-28
-1B	9.2-29
-1	9.2-17
-2	9.2-18
-3	9.2-19
-4	9.2-20
-1	9.2-1
-2	9.2-2
-3	9.2-3
-4	9.2-4
-5	9.2-5
-1	10.4-3
-2	10.4-4
-3	10.4-5
-4	10.4-6
-1	9.2-22
-3	10.3-5
-4	9.2-27
-1A	9.5-1
-1B	9.5-2
-2	9.5-3
-3	9.5-4
-4A	9.5-5
-4B	9.5-6
-5	9.5-6A
-1	9.3-1
-2	9.3-2
-3	9.3-3
-4	9.3-4

P&ID NO.	FSAR FIGURE NO.
35-2	10.2-9
35-3	10.2-10
36-9	9.5-7
36-10	9.5-12
36-11	9.5-10
36-12A	9.5-8
36-12B	9.5-9
36-13	9.5-11
41A-1	10.4-27
41A-2	10.4-28
41A-3A	10.4-29
41A-3B	10.4-30
41A-4	10.4-31
41C-1	9.2-24
41C-2	9.2-25
41C-3	9.2-26
41D-1	9.3-
41D-2	9.2-
41D-3	9.3
41D-5	9.3
41D-6	9.3
44A-1	9.4-
44A-2	9.4-
44A-4	9.4-3
44B-1	9.4-12
44B-3A	9.4-15
44B-3B	9.4-14
44C-1	9.4-9
44C-2	9.4-10
44D-1A	9.4-4
44D-1B	9.4-5
44D-2	9.4-6
44F-1	9.4-11
44F-2	9.4-7
44F-3	9.4-13
44F-4	9.4-8
44G-1	9.4-17
44G-2	9.4-18
44G-3	9.4-19
44G-4	9.4-20
46-1	6.2-131
59A-1	9.5-2A SH1
59B-1	9.5-2A SH2

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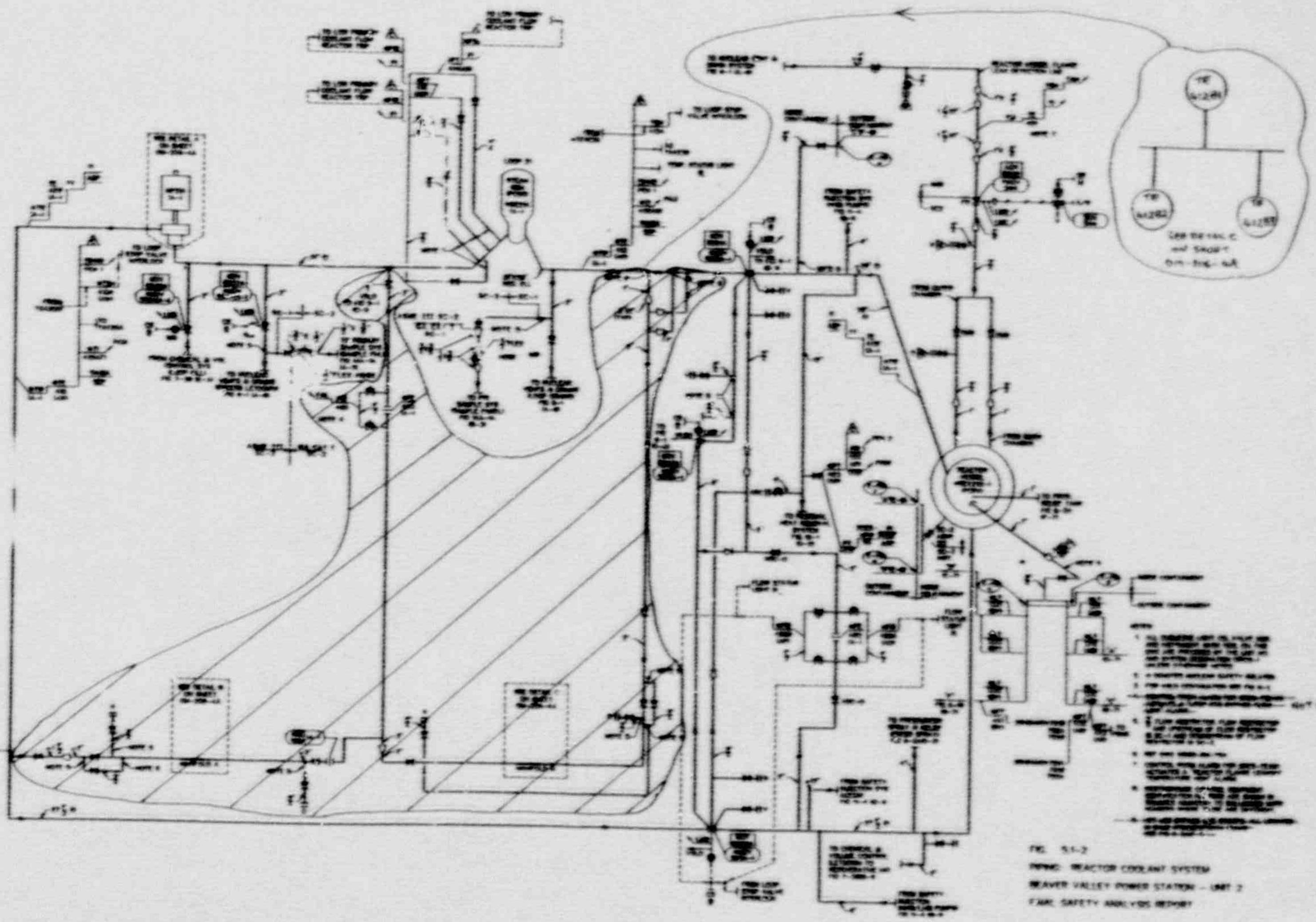


FIG. 5.1-2
PIPING: REACTOR COOLANT SYSTEM
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT

)SS-REFERENCE KEY

P&ID NO.	FSAR FIGURE NO.
25-3A	11.2-3
25-3B	11.2-4
25-4	11.2-5
25-5	11.2-6
26-1A	10.2-2
26-1B	10.2-3
26-2	10.2-7
26-3	10.4-1
26-4	10.2-1
26-6	10.2-6
26-7	10.2-8
26-17	9.4-16
27-1	10.4-25
27-2	10.4-26
27-3	10.4-32
27-4	10.4-26A
27B-1	10.4-37
27B-2	10.4-38
27B-3	10.4-39
27B-4	10.4-40
28-1A	9.2-28
28-1B	9.2-29
29-1	9.2-17
29-2	9.2-18
29-3	9.2-19
29-4	9.2-20
30-1	9.2-1
30-2	9.2-2
30-3	9.2-3
30-4	9.2-4
30-5	9.2-5
31-1	10.4-3
31-2	10.4-4
31-3	10.4-5
31-4	10.4-6
32-1	9.2-22
32-3	10.3-5
32-4	9.2-27
33-1A	9.5-1
33-1B	9.5-2
33-2	9.5-3
33-3	9.5-4
33-4A	9.5-5
33-4B	9.5-6
33-5	9.5-6A
34-1	9.3-1
34-2	9.3-2
34-3	9.3-3
34-4	9.3-4

P&ID NO.	FSAR FIGURE NO.
35-2	10.2-9
35-3	10.2-10
36-9	9.5-7
36-10	9.5-12
36-11	9.5-10
36-12A	9.5-8
36-12B	9.5-9
36-13	9.5-11
41A-1	10.4-27
41A-2	10.4-28
41A-3A	10.4-29
41A-3B	10.4-30
41A-4	10.4-31
41C-1	9.2-24
41C-2	9.2-25
41C-3	9.2-26
41D-1	9.3-17
41D-2	9.2-23
41D-3	9.3-18
41D-5	9.3-20
41D-6	9.3-19
44A-1	9.4-2
44A-2	9.4-1
44A-4	9.4-3
44B-1	9.4-12
44B-3A	9.4-15
44B-3B	9.4-14
44C-1	9.4-9
44C-2	9.4-10
44D-1A	9.4-4
44D-1B	9.4-5
44D-2	9.4-6
44F-1	9.4-11
44F-2	9.4-7
44F-3	9.4-13
44F-4	9.4-8
44G-1	9.4-17
44G-2	9.4-18
44G-3	9.4-19
44G-4	9.4-20
46-1	6.2-131
59A-1	9.5-2A SH1
59B-1	9.5-2A SH2

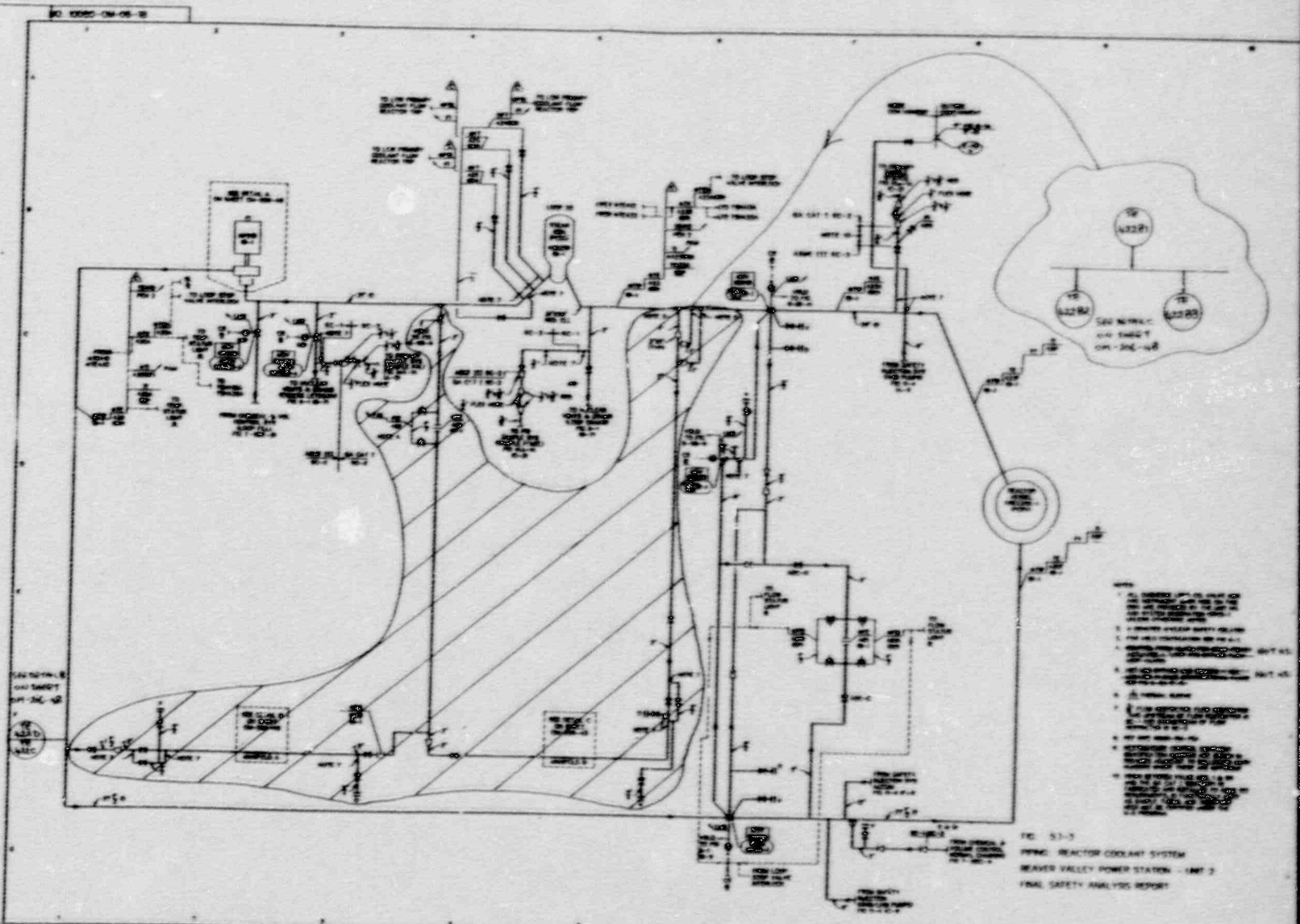
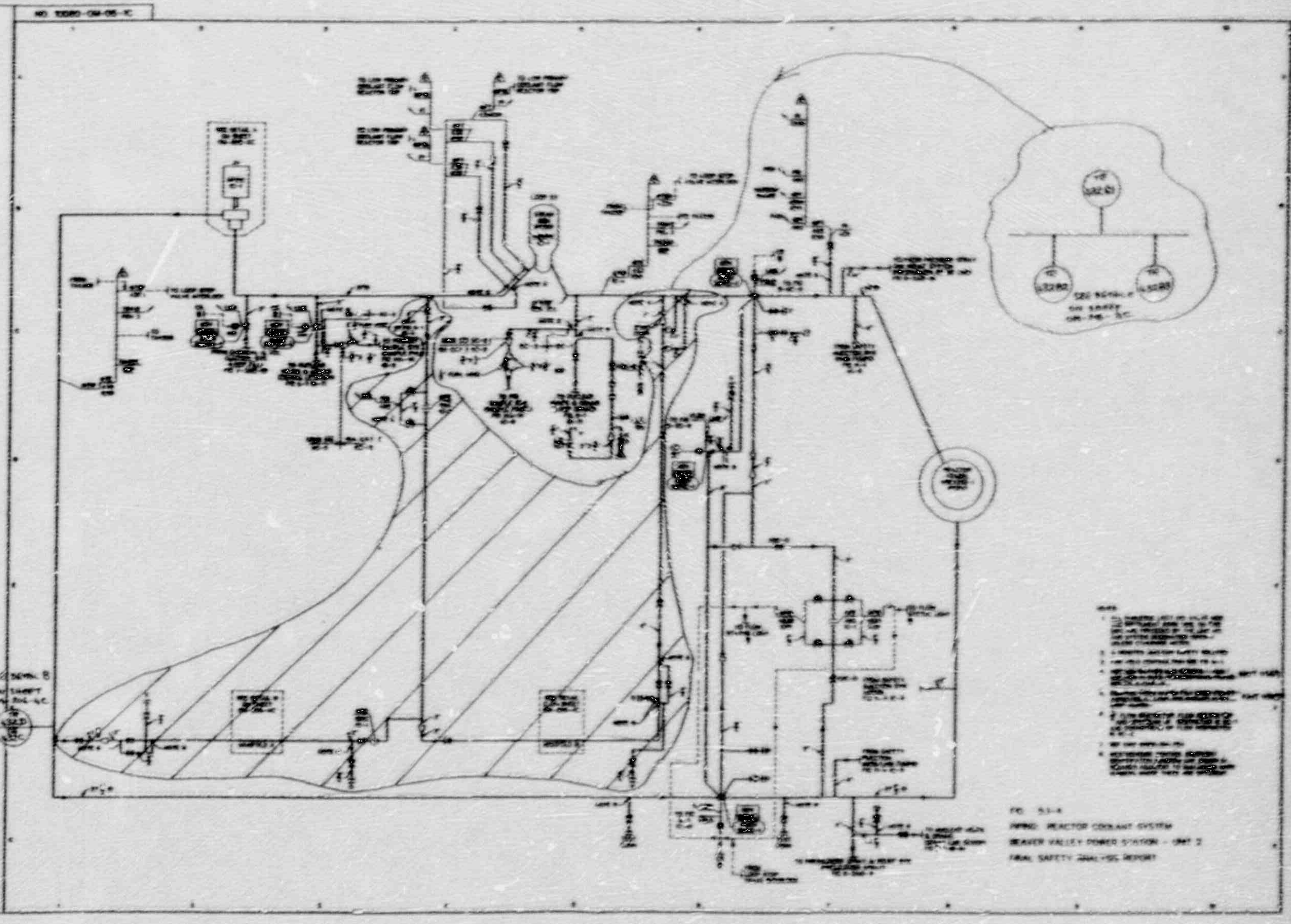


FIG. 5.1-3
PIPING: REACTOR COOLANT SYSTEM
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT

REFERENCE KEY

P&ID NO.	FSAR FIGURE NO.	P&ID NO.	FSAR FIGURE NO.
25-3A	11.2-3	35-2	10.2-9
25-3B	11.2-4	35-3	10.2-10
25-4	11.2-5		
25-5	11.2-6	36-9	9.5-7
		36-10	9.5-12
26-1A	10.2-2	36-11	9.5-10
26-1B	10.2-3	36-12A	9.5-8
26-2	10.2-7	36-12B	9.5-9
26-3	10.4-1	36-13	9.5-11
26-4	10.2-1		
26-6	10.2-6	41A-1	10.4-27
26-7	10.2-8	41A-2	10.4-28
26-17	9.4-16	41A-3A	10.4-29
		41A-3B	10.4-30
27-1	10.4-25	41A-4	10.4-31
27-2	10.4-26		
27-3	10.4-32	41C-1	9.2-24
27-4	10.4-26A	41C-2	9.2-25
27B-1	10.4-37	41C-3	9.2-26
27B-2	10.4-38		
27B-3	10.4-39	41D-1	9.3-17
27B-4	10.4-40	41D-2	9.2-23
		41D-3	9.3-18
28-1A	9.2-28	41D-5	9.3-20
28-1B	9.2-29	41D-6	9.3-19
29-1	9.2-17		
29-2	9.2-18	44A-1	9.4-2
29-3	9.2-19	44A-2	9.4-1
29-4	9.2-20	44A-4	9.4-3
30-1	9.2-1		
30-2	9.2-2	44B-1	9.4-12
30-3	9.2-3	44B-3A	9.4-15
30-4	9.2-4	44B-3B	9.4-14
30-5	9.2-5		
31-1	10.4-3	44C-1	9.4-9
31-2	10.4-4	44C-2	9.4-10
31-3	10.4-5		
31-4	10.4-6	44D-1A	9.4-4
32-1	9.2-22	44D-1B	9.4-5
32-3	10.3-5	44D-2	9.4-6
32-4	9.2-27		
33-1A	9.5-1	44F-1	9.4-11
33-1B	9.5-2	44F-2	9.4-7
33-2	9.5-3	44F-3	9.4-13
33-3	9.5-4	44F-4	9.4-8
33-4A	9.5-5	44G-1	9.4-17
33-4B	9.5-6	44G-2	9.4-18
33-5	9.5-6A	44G-3	9.4-19
34-1	9.3-1	44G-4	9.4-20
34-2	9.3-2	46-1	6.2-131
34-3	9.3-3		
34-4	9.3-4	59A-1	9.5-2A SH1
		59B-1	9.5-2A SH2



REFERENCE KEY

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25-5	11.2-6
26-1A	10.2-2
26-1B	10.2-3
26-2	10.2-7
26-3	10.4-1
26-4	10.2-1
26-6	10.2-6
26-7	10.2-8
26-17	9.4-16
27-1	10.4-25
27-2	10.4-26
27-3	10.4-32
27-4	10.4-26A
27B-1	10.4-37
27B-2	10.4-38
27B-3	10.4-39
27B-4	10.4-40
28-1A	9.2-28
28-1B	9.2-29
29-1	9.2-17
29-2	9.2-18
29-3	9.2-19
29-4	9.2-20
30-1	9.2-1
30-2	9.2-2
30-3	9.2-3
30-4	9.2-4
30-5	9.2-5
31-1	10.4-3
31-2	10.4-4
31-3	10.4-5
31-4	10.4-6
32-1	9.2-22
32-3	10.3-5
32-4	9.2-27
33-1A	9.5-1
33-1B	9.5-2
33-2	9.5-3
33-3	9.5-4
33-4A	9.5-5
33-4B	9.5-6
33-5	9.5-6A
34-1	9.3-1
34-2	9.3-2
34-3	9.3-3
34-4	9.3-4

P&ID NO.	FSAR FIGURE NO.
35-2	10.2-9
35-3	10.2-10
36-9	9.5-7
36-10	9.5-12
36-11	9.5-10
36-12A	9.5-8
36-12B	9.5-9
36-13	9.5-11
41A-1	10.4-27
41A-2	10.4-28
41A-3A	10.4-29
41A-3B	10.4-30
41A-4	10.4-31
41C-1	9.2-24
41C-2	9.2-25
41C-3	9.2-26
41D-1	[illegible]
41D-2	[illegible]
41D-3	[illegible]-5
41D-5	[illegible]-20
41D-6	
44A-1	
44A-2	
44A-4	9.4-3
44B-1	9.4-12
44B-3A	9.4-15
44B-3B	9.4-14
44C-1	9.4-9
44C-2	9.4-10
44D-1A	9.4-4
44D-1B	9.4-5
44D-2	9.4-6
44F-1	9.4-11
44F-2	9.4-7
44F-3	9.4-13
44F-4	9.4-8
44G-1	9.4-17
44G-2	9.4-18
44G-3	9.4-19
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59A-1	9.5-2A SH1
59B-1	9.5-2A SH2

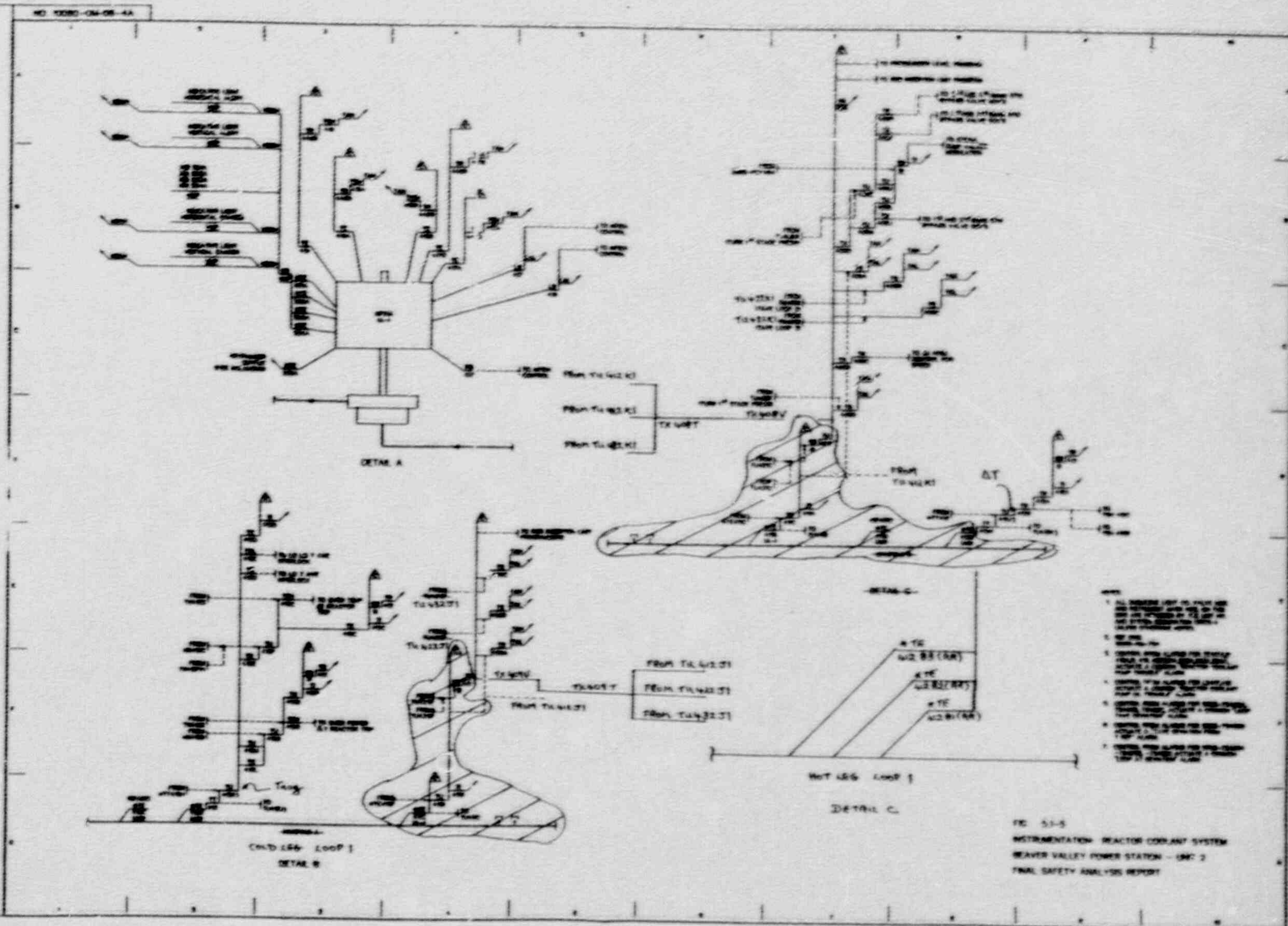

FIG. 5.1-5
INSTRUMENTATION REACTOR COOLANT SYSTEM
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT

REFERENCE KEY

P&ID NO.	FSAR FIGURE NO.
25-3A	11.2-3
25-3B	11.2-4
25-4	11.2-5
25-5	11.2-6
26-1A	10.2-[illegible]
26-1B	10.2-3
26-2	10.2-7
26-3	10.4-1
26-4	10.2-1
26-6	10.2-6
26-7	10.2-8
26-17	9.4-16
27-1	10.4-25
27-2	10.4-26
27-3	10.4-32
27-4	10.4-[illegible]8A
27B-1	10.4-37
27B-2	10.4-38
27B-3	10.[illegible]-39
27B-4	[illegible]
28-1A	9.[illegible]
28-1B	9.[illegible]-29
29-1	9.2-17
29-2	9.2-18
29-3	9.2-19
29-4	9.2-20
30-1	9.2-1
30-2	9.2-2
30-3	9.2-3
30-4	9.2-4
30-5	9.2-5
31-1	10.4-3
31-2	10.4-4
31-3	10.4-5
31-4	10.4-6
32-1	9.2-22
32-3	10.3-5
32-4	9.2-27
33-1A	9.5-1
33-1B	9.5-2
33-2	9.5-3
33-3	9.5-4
33-4A	9.5-5
33-4B	9.5-6
33-5	9.5-6A
34-1	9.3-1
34-2	9.3-2
34-3	9.3-3
34-4	9.3-4

P&ID NO.	FSAR FIGURE NO.
35-2	10.2-9
35-3	10.2-10
36-9	9.5-7
36-10	9.5-12
36-11	9.5-10
36-12A	9.5-8
36-12B	9.5-9
36-13	9.5-11
41A-1	10.4-27
41A-2	10.4-28
41A-3A	10.4-29
41A-3B	10.4-30
41A-4	10.4-31
41C-1	9.2-24
41C-2	9.2-25
41C-3	9.2-26
41D-[illegible]	[illegible]
[illegible]-2	[illegible]
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44A-1	9.4-2
44A-2	9.4-1
44A-4	9.4-3
44B-1	9.4-12
44B-3A	9.4-15
44B-3B	9.4-14
44C-1	9.4-9
44C-2	9.4-10
44D-1A	9.4-4
44D-1B	9.4-5
44D-2	9.4-6
44F-1	9.4-11
44F-2	9.4-7
44F-3	9.4-13
44F-4	9.4-8
44G-1	9.4-17
44G-2	9.4-18
44G-3	9.4-19
44G-4	9.4-20
46-1	8.2-131
59A-1	9.5-2A SH1
59B-1	9.5-2A SH2

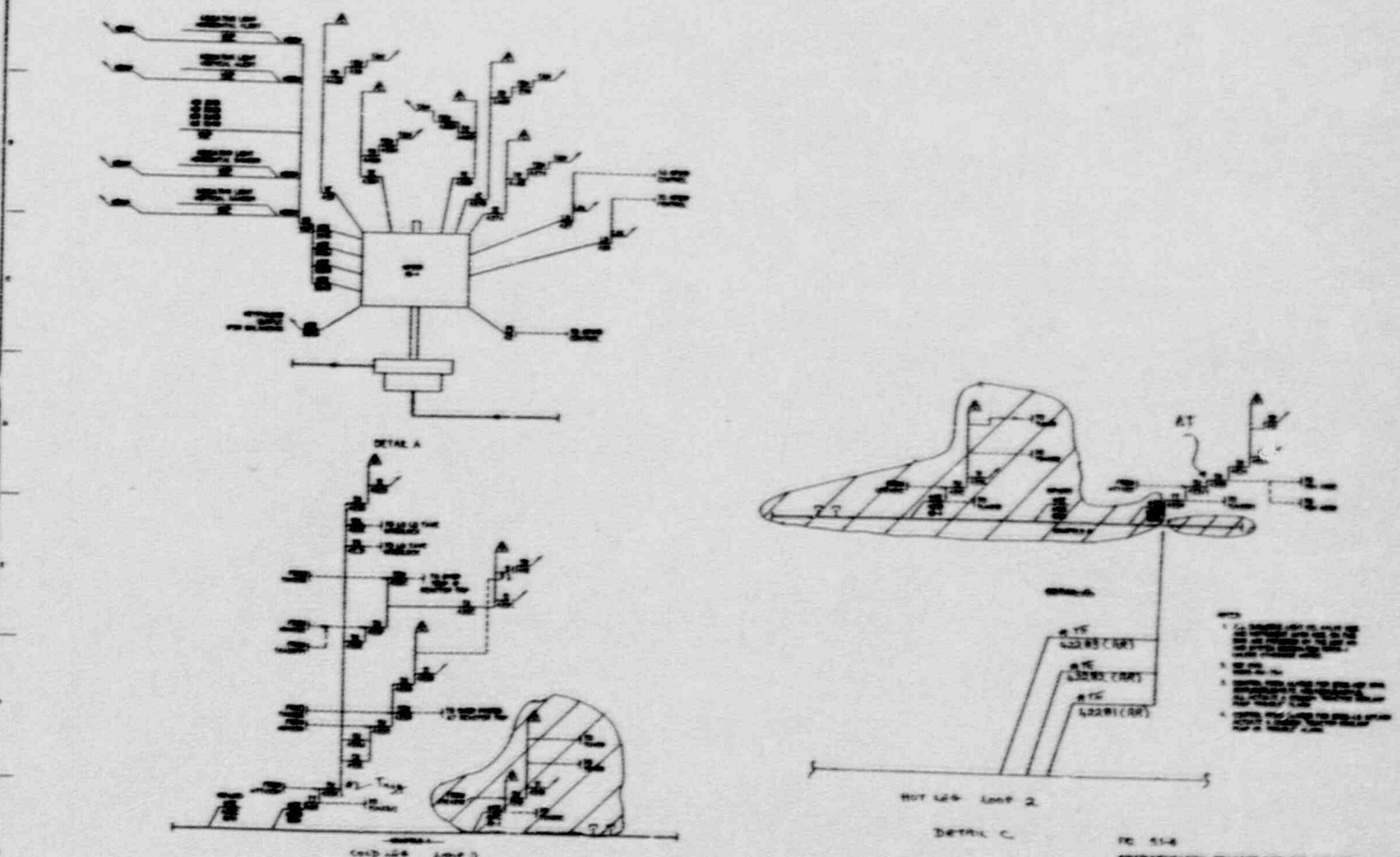

FIG. 5.1-6
INSTRUMENTATION REACTOR COOLANT SYSTEM
BEAVER VALLEY POWER STATION – UNIT 2
FINAL SAFETY ANALYSIS REPORT

S-REFERENCE KEY

P&ID NO.	FSAR FIGURE NO.	P&ID NO.	FSAR FIGURE NO.
25-3A	11.2-3	35-2	10.2-9
25-3B	11.2-4	35-3	10.2-10
25-4	11.2-5	36-9	9.5-7
25-5	11.2-6	36-10	9.5-12
26-1A	10.2-2	36-11	9.5-10
26-1B	10.2-3	36-12A	9.5-8
26-2	10.2-7	36-12B	9.5-9
26-3	10.4-1	36-13	9.5-11
26-4	10.2-1	41A-1	10.4-27
26-6	10.2-6	41A-2	10.4-28
26-[illegible]	10.2-8	41A-3A	10.4-29
26-17	9.4-16	41A-3B	10.4-30
27-1	10.4-25	41A-4	10.4-31
27-2	10.4-26	41C-1	9.2-24
27-3	10.4-32	41C-2	9.2-25
28-[illegible]	[illegible]	41C-3	9.2-26
2[illegible]	[illegible]	41D-1	9.3-17
29-[illegible]	[illegible]	[illegible]-2	[illegible]-23
29-2	[illegible]	41D-3	9.3-[illegible]
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30-4	9.2-4	44B-1	9.4-12
30-5	9.2-5	44B-3A	9.4-15
31-1	10.4-3	44B-3B	9.4-14
31-2	10.4-4	44C-1	9.4-9
31-3	10.4-5	44C-2	9.4-10
31-4	10.4-6	44D-1A	9.4-4
32-1	9.2-22	44D-1B	9.4-5
32-3	10.3-5	44D-2	9.4-6
32-4	9.2-27	44F-1	9.4-11
33-1A	9.5-1	44F-2	9.4-7
33-1B	9.5-2	44F-3	9.4-13
33-2	9.5-3	44F-4	9.4-8
33-3	9.5-4	44G-1	9.4-17
33-4A	9.5-5	44G-2	9.4-18
33-4B	9.5-6	44G-3	9.4-19
34-1	9.3-1	46-1	6.2-131
34-2	9.3-2		
34-3	9.3-3		
34-4	9.3-4		

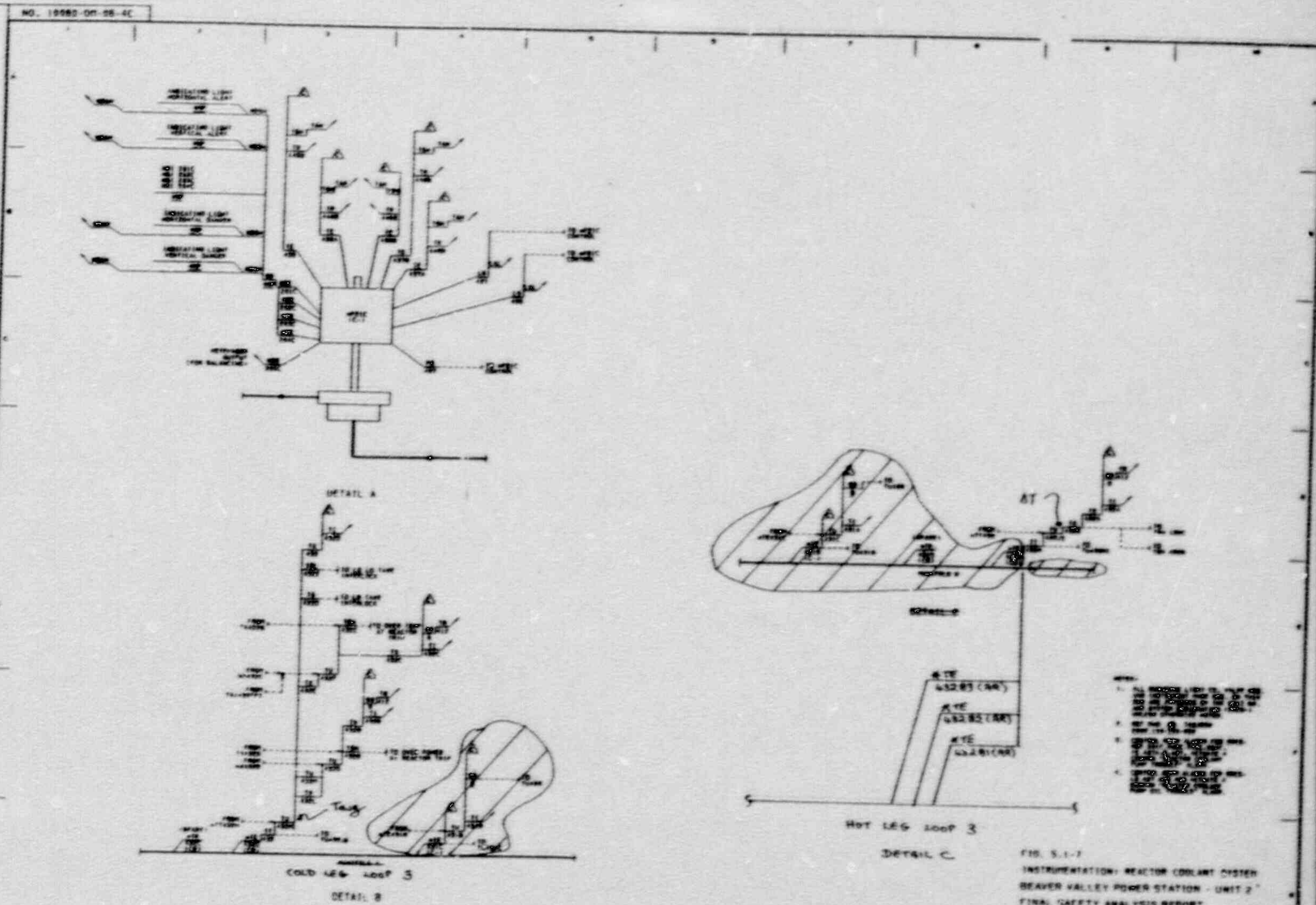

FIG. 5.1-7
INSTRUMENTATION: REACTOR COOLANT SYSTEM
BEAVER VALLEY POWER STATION - UNIT 2
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7.2.1.1.4 Coolant Temperature Sensor Arrangement

Insert B

~~Resistance Temperature Detector Bypass Manifold~~

~~Separate bypass manifolds (loops) for each reactor coolant loop hot and cold leg are provided so that individual temperature signals may be developed for use in the reactor control and protection system. The bypass manifold around each steam generator obtains a representative hot leg temperature by mixing the flow from three scoop connections, which extend into the flow stream at locations 120° apart in the cross-sectional plane, on the reactor coolant hot leg. The hot leg bypass flow exits from the manifold to a common return line.~~

~~Flow for the cold leg bypass manifold is obtained downstream of the pump discharge. Because of the mixing action of the pump, only one connection is required to obtain a representative sample. This connection is located as close as possible to the weld connection at the pump discharge and is in the same relative position in each loop.~~

~~The bypass manifold lines join downstream of each of the hot and cold leg manifolds and discharge into a common line. The combined bypass flows pass through a flow indicator before discharging to the suction side of the RCP.~~

~~The manifolds are not provided with thermometer wells. Instead, the resistance temperature detectors extend directly into the flow path to reduce the time delay to a minimum. Therefore, two isolation valves in series are provided on each side of the bypass manifold to allow for RTD maintenance. The valve nearest the connection to the main coolant piping is located above the elevation of the reactor vessel nozzles to permit valve repair during cold shutdown without draining the RCS. In addition, vents and drains are provided in each manifold to be used, in conjunction with the isolation valve, for maintenance.~~

Signals from these instruments are used to compute the reactor coolant ΔT (temperature of the hot leg, T_{hot}, minus the temperature at the cold leg, T_{cold},) and an average reactor coolant temperature (T_{avg}). The T_{avg} for each loop is indicated on the main control board.

Wide Range Cold Leg and Hot Leg Temperatures

Wide range ~~T~~tTemperature detectors, located in the thermometer wells in the cold and hot leg piping of each loop, supply signals to wide range temperature recorders. This information is used by the operator to control coolant temperature during start-up and shutdown.

INSERT B

The hot and cold leg temperature signals required for input to the protection and control functions are obtained using thermowell mounted RTDs installed in each reactor coolant loop.

The hot leg temperature measurement in each loop is accomplished using three fast response narrow range RTDs mounted in thermowells. Two of the three thermowells in each loop are located within the scoops previously used to supply temperature samples to the RTD bypass manifold. The third RTD could not be located within the scoop due to structual interferences and is located upstream from the scoop plane. The two scoops used to accommodate the thermowells were modified by machining a flow hole in the end of the scoop to facilitate the flow of water through the existing holes in the leading edge of the scoop and passed the temperature sensitive tip of the RTD.

Due to temperature streaming the temperatures measured by the three hot leg RTDs are different and therefore these signals are electronically averaged to generate a hot leg average temperature. Provisions were made in the RTD electronics to allow for operation with only two RTDs in service. The two RTD measurement can be biased to correct for the difference compared with the three RTD average.

The cold leg temperature measurement in each loop is accomplished by one fast response, narrow range, dual element RTD. The original cold leg RTD bypass penetration nozzle was modified to accept the thermowell. Temperature streaming in the cold leg is not a concern due to the mixing action of the the reactor coolant pump. Therefore, only a single temperature measurement is required in each cold leg.

Any reactor trip will actuate an alarm and an indicator in the main control room. Such protective actions are indicated and identified down to the channel level.

Alarms and indicators are also used to alert the operator of deviations from normal operating conditions so that he may take appropriate corrective action to avoid a reactor trip. Actuation of any rod stop or trip of any reactor trip channel will actuate an alarm.

<u>System Repair</u>

The system is designed to facilitate the recognition, location, replacement, and repair of malfunctioning components or modules. The capability for testing was previously discussed in Section 7.2.2.2.3.

7.2.2.3 Specific Control and Protection Interactions

7.2.2.3.1 Neutron Flux

Four power range neutron flux channels are provided for overpower protection. An isolation signal is also provided for automatic rod control. If any channel fails in such a way as to produce a low output, that channel is incapable of proper overpower protection but a two out of four overpower trip logic ensures an overpower trip, if needed, even with an independent failure in another channel.

In addition, channel deviation signals in the control system will give an alarm if any neutron flux channel deviates significantly from the average of the flux signals. Also, the control system will respond only to rapid changes in indicated neutron flux. Slow changes or drifts are compensated by the temperature control signals. Finally, an overpower signal from any nuclear power range channel will block manual and automatic rod withdrawal. The set point for this rod is below the reactor trip set point.

7.2.2.3.2 Coolant Temperature

The accuracy of the RTD ~~bypass~~ loop temperature measurements is demonstrated during BVPS-2 start-up tests by comparing the temperature measurements from all ~~bypass loop~~ RTDs with one another, as well as with the temperature measurements obtained from the wide range RTD located in the hot leg and cold leg piping of each loop. The comparisons are done with the RCS in an isothermal condition. The linearity of the ΔT versus plant power is not important as far as reactor protection is concerned. The RTS set points are based on percentages of the indicated ΔT at nominal full power rather than on absolute values of ΔT. This is done to account for loop differences which are inherent. Therefore, the percent ΔT scheme is relative, not absolute, and provides better protective action without the expense of accuracy. For this reason, the linearity of the ΔT signals, as a function of power, is of importance rather than the

absolute values of the ΔT. As part of the BVPS-2 start-up tests, the ~~bypass~~ loop RTD signals will be compared with the core exit thermocouple signals during isothermal RCS conditions.

Plant control is based upon signals derived from protection system channels after isolation, by isolation amplifiers such that no feedback effect can perturb the protection channels.

Insert C

~~Since control is based on the average temperature of the loop with the highest temperature, the control rods are always moved based upon the most pessimistic temperature measurement with respect to margins to DNB. A spurious low average temperature measurement from any loop temperature control channel will cause no control action. A spurious high average temperature measurement will cause rod insertion (safe direction).~~

~~Individual low flow alarms, with individual status lights for each reactor coolant loop bypass flow, are provided on the main control board. The alarm and status lights provide the operator with immediate indication of a low flow condition in the bypass loops associated with any reactor coolant loop.~~

~~Flow will be monitored:~~

1. ~~Prior to restoring temperature channels to normal service following reopening of bypass loop stop valves whenever a bypass loop has been out of service,~~

2. ~~On a periodic basis, and~~

3. ~~Following any bypass loop low flow alarm.~~

In addition, channel deviation signals in the control system will give an alarm if any temperature channel deviates significantly from the median ~~auctioneered (highest)~~ value. Automatic rod withdrawal blocks and turbine runback (power demand reduction) will also occur if any two out of the three overtemperature or overpower ΔT channels indicate an adverse condition.

7.2.2.3.3 Pressurizer Pressure

The pressurizer pressure protection channel signals are used for high and low pressure protection and as inputs to the overtemperature ΔT trip protection function. Separate control channels are used to control pressurizer spray and heaters and pressurizer power-operated relief valves (PORVs). Pressurizer pressure is sensed by fast response pressure transmitters.

A spurious high pressure signal from one channel can cause decreasing pressure by actuation of either spray or relief valves. Additional redundancy is provided in the low pressurizer pressure reactor trip

INSERT C

The input signals (one per loop) to the Reactor Control System are obtained from electronically isolated protection Tavg and Delta-T signals. A Median Signal Selector (MSS) is implemented in the Reactor Control System, one for Tavg and one for Delta-T. The MSS receives three signals as input and selects the median signal for input to the appropriate control systems. Any single failure, high or low, in a calculated temperature will not result in an adverse control system response since the failed high or low temperature signal will be rejected by the MSS.

Hence, the implementation of a MSS in the Reactor Control System in conjuction with two out of three protection logic satisfies the requirements of IEEE 279-1971, Section 4.7, "Control and Protection System Interaction".

The response time allocated for measuring RCS hot and cold leg temperatures using thermowell mounted fast response RTDs is four seconds. This response time does not include the process electronics.

7. Steam dump control

   a. Permits BVPS-2 to accept a sudden loss of load without incurring reactor trip. Steam is dumped to the condenser as necessary to accommodate excess power generation in the reactor during turbine load reduction transients.

   b. Ensures that stored energy and residual heat are removed following a reactor trip to bring BVPS-2 to equilibrium no load conditions without actuation of the steam generator safety valves.

   c. Maintains BVPS-2 at no load conditions and permits a manually controlled cooldown of the nuclear plant.

8. Incore instrumentation

   Provides information on the neutron flux distribution and on the core outlet temperatures at selected core locations.

7.7.1.1 Reactor Control System

The reactor control system enables BVPS-2 to follow load changes automatically including the acceptance of step load increases or decreases of 10-percent, and ramp increases or decreases of 5-percent/min within the load range of 15 to 100-percent without reactor trip, steam dump, or pressure relief (subject to possible xenon limitations). The system is also capable of restoring coolant average temperature to within the programmed temperature deadband following a change in load. Manual control rod operation may be performed at any time.

The reactor control system controls the reactor coolant average temperature by regulation of control rod bank position. The reactor coolant loop average temperatures are determined from hot leg and cold leg measurements in each reactor coolant loop. There is an average coolant temperature (T_{avg}) computed for each loop, where:

$$T_{avg} = \frac{T_{hot} + T_{cold}}{2} \qquad (7.7\text{-}1)$$

The error between the programmed reference temperature (based on turbine impulse chamber pressure) and the ~~highest~~ median of the T_{avg} measured temperatures (which is processed through a lead-lag compensation unit) from each of the reactor coolant loops constitutes the primary control signal, as shown in general on Figure 7.7-1 and in more detail on the functional diagram, Figure 7.2-1, Sheet 9. The system is capable of restoring coolant average temperature to the programmed value following a change in load. The programmed coolant temperature increases linearly with turbine load from zero power to

the full power condition. The median T_{avg} signal is also supplied ~~a signal~~ to the pressurizer level control, steam dump control, and rod insertion limit monitoring control systems.

~~The temperature channels needed to derive the temperature input signals for the reactor control system are physically separated from the temperature channels used to derive appropriate protection signals.~~ The temperature inputs to the control systems are derived using the median signal selector.

An additional control input signal is derived from the reactor power versus turbine load mismatch signal. This additional control input signal improves system performance by enhancing response and reducing transient peaks.

7.7.1.2 Rod Control System

7.7.1.2.1 Rod Control System

The rod control system receives rod speed and direction signals from the T_{avg} control system. The rod speed demand signal varies over the corresponding range of 8 to 72 steps/min depending on the magnitude of the input signal. Manual control is provided to move a control bank in or out at a prescribed fixed speed.

When the turbine load reaches approximately 15-percent of rated load, the operator may select the automatic mode, and rod motion is then controlled by the reactor control system. A permissive interlock, C-5 (Table 7.7-1), derived from measurements of turbine impulse chamber and pressure, prevents automatic control when the turbine load is below 15-percent. In the automatic mode, the rods are again withdrawn (or inserted) in a predetermined programmed sequence by the automatic programming equipment. The manual and automatic controls are further interlocked with the control interlocks (Table 7.7-1).

The shutdown banks are always in the fully withdrawn position during normal operation, and are moved to this position at a constant speed by manual control prior to criticality. A reactor trip signal causes them to fall by gravity into the core. There are two shutdown banks.

The control banks are the only rods that can be manipulated under automatic control. Each control bank is divided into two groups to obtain smaller incremental reactivity changes per step. All RCCAs in a group are electrically paralleled to move simultaneously. There is individual position indication for each RCCA.

Power to rod drive mechanisms is supplied by two motor-generator sets operating from two separate 480 V three-phase buses. Each generator is the synchronous type and is driven by a 200 hp induction motor. The ac power is distributed to the rod control power cabinets through the two series-connected reactor trip breakers.

1. The low alarm alerts the operator of an approach to the rod insertion limits requiring boron addition by following normal procedures with the chemical and volume control system (CVCS), and

2. The low-low alarm alerts the operator to take immediate action to add boron to the RCS by any one of several alternate methods.

The purpose of the control bank rod insertion monitor is to give warning to the operator of excessive rod insertion. The insertion limit maintains sufficient core reactivity shutdown margin following reactor trip, provides a limit on the maximum inserted rod worth in the unlikely event of a hypothetical rod ejection, and limits rod insertion such that acceptable nuclear peaking factors are maintained. Since the amount of shutdown reactivity required for the design shutdown margin following a reactor trip increases with increasing power, the allowable rod insertion limits must be decreased (the rods must be withdrawn further) with increasing power. Two parameters which are proportional to power are used as inputs to the insertion monitor. These are the ΔT between the hot leg and the cold leg, which is a direct function of reactor power, and T_{avg}, which is programmed as a function of power. The rod insertion monitor uses parameters for each control rod bank as follows:

$$Z_{LL} = A(\Delta T)_{\underset{median}{\cancel{auct}}} + B(T_{avg})_{\underset{median}{\cancel{auct}}} + C \qquad (7.7\text{-}2)$$

where:

Z_{LL} = Maximum permissible insertion limit for affected control bank

$(\Delta T)_{\underset{median}{\cancel{auct}}}$ = ~~Highest~~ median ΔT of all loops

$(T_{avg})_{\underset{median}{\cancel{auct}}}$ = ~~Highest~~ median T_{avg} of all loops

A,B,C = Constants chosen to maintain $Z_{LL} \geq$ actual limit based on physics calculations

The control rod bank demand position (Z) is compared to Z as follows:

If $Z - Z_{LL} \leq D$, a low alarm is actuated

If $Z - Z_{LL} \leq E$, a low-low alarm is actuated

~~Since the highest values of T_{avg} and ΔT are chosen by auctioneering, a conservatively high representation of power is used in the insertion limit calculation~~.

Actuation of the low alarm alerts the operator of an approach to a reduced shutdown reactivity situation. Administrative procedures require the operator to add boron through the CVCS. Actuation of the low-low alarm requires the operator to initiate emergency boration procedures. The value for E is chosen such that the low-low alarm would normally be actuated before the insertion limit is reached. The value for D is chosen to allow the operator to follow normal boration procedures. Figure 7.7-2 shows a block diagram representation of the control rod bank insertion monitor. The monitor is shown in more detail on the functional diagram, Figure 7.2-1, Sheet 9. In addition to the rod insertion monitor for the control banks, the BVPS-2 computer, which monitors individual rod positions, provides an alarm that is associated with the rod deviation alarm discussed in Section 7.7.1.3.4. This alarm is provided to warn the operator if any shutdown RCCA leaves the fully withdrawn position.

Rod insertion limits are established by:

1. Establishing the allowed rod reactivity insertion at full power consistent with the purposes given previously,

2. Establishing the differential reactivity worth of the control rods when moved in normal sequence,

3. Establishing the change in reactivity with power level by relating power level to rod position, or

4. Linearizing the resultant limit curve. All key nuclear parameters in this procedure are measured as part of the initial and periodic physics testing program.

Any unexpected change in the position of the control bank under automatic control, or a change in coolant temperature under manual control, provides a direct and immediate indication of a change in the reactivity status of the reactor. In addition, samples are taken periodically of coolant boron concentration. Variations in concentration during core life provide an additional check on the reactivity status of the reactor, including core depletion.

7.7.1.3.4 Rod Deviation Alarms

The demanded and measured rod position signals are displayed on the main control board. They are also monitored by the BVPS-2 computer, which provides a visual printout and an audible alarm whenever an individual rod position signal deviates from the other rods in the bank by a preset limit. The alarm can be set with appropriate

7.7.1.8 Steam Dump Control

The steam dump system, as described in Section 10.4.4, is designed to accept an 85 to 100-percent loss of net load without tripping the reactor.

The automatic steam dump system is able to accommodate this abnormal load rejection and to reduce the effects of the transient imposed upon the RCS. By passing 90-percent of full load main steam directly to the condenser and atmosphere, an artificial load is thereby maintained on the primary system. The rod control system can then reduce the reactor temperature to a new equilibrium value without causing overtemperature and/or overpressure conditions. The steam dump steam flow capacity is 90-percent of full load steam flow at full load steam pressure.

If the difference between the reference T_{avg} (T_{ref}) based on turbine impulse chamber pressure and the lead/lag compensated median [redacted] T_{avg} [redacted] exceeds a predetermined amount, and the interlock mentioned as follows is satisfied, a demand signal will actuate the steam dump to maintain the RCS temperature within control range until a new equilibrium condition is reached.

To prevent actuation of steam dump on small load perturbations, an independent load rejection sensing circuit is provided. This circuit senses the rate of decrease in the turbine load as detected by the turbine impulse chamber pressure. It is provided to unblock the dump valves when the rate of load rejection exceeds a preset value corresponding to a 10-percent step load decrease or a sustained ramp load decrease of 5-percent/min.

A block diagram of the steam dump control system is shown on Figure 7.7-7.

7.7.1.8.1 Load Rejection Steam Dump Controller

This circuit prevents large increase in reactor coolant temperature following a large, sudden load decrease. The error signal is a difference between the lead/lag compensated [redacted] median T_{avg} and the reference T_{avg} is based on turbine impulse chamber pressure.

The T_{avg} signal is the same as that used in the [redacted] Rod Control system. The lead/lag compensation for the T_{avg} signal is to compensate for lags in the BVPS-2 thermal response and in valve positioning. Following a sudden load decrease, T_{ref} is immediately decreased and T_{avg} tends to increase, thus generating an immediate demand signal for steam dump. Since control rods are available, in this situation steam dump terminates as the error comes within the maneuvering capability of the control rods.

7.7.1.8.2 Plant Trip Steam Dump Controller

Following a reactor trip, the load rejection steam dump controller is defeated and the reactor trip steam dump controller becomes active. Since control rods are not available in this situation, the demand signal is the error signal between the lead/lag compensated ~~auctioneered~~ median T_{avg} and the no load reference T_{avg}. When the error signal exceeds a predetermined set point, the dump valves are tripped open in a prescribed sequence. As the error signal reduces in magnitude indicating that the RCS T_{avg} is being reduced toward the reference no-load value, the dump valves are modulated by the BVPS-2 trip controller to regulate the rate of removal decay heat and thus gradually establish the equilibrium hot standby condition.

Following a reactor trip only, sufficient steam dump capacity is necessary to maintain steam pressure below the steam generator relief valve set point (approximately 40-percent capacity to the condenser), the two groups of valves are opened. The error signal determines whether a group is to be tripped open or modulated open. The valves are modulated when the error is below the trip-open set points.

7.7.1.8.3 Steam Header Pressure Controller

Residual heat removal is maintained by the steam generator pressure controller (manually selected), which controls the amount of steam flow to the condensers. This controller operates a portion of the same steam dump valves to the condensers, which are used during the initial transient following turbine reactor trip or load rejection.

7.7.1.9 Incore Instrumentation

The incore instrumentation system consists of chromel-alumel thermocouples, at fixed core outlet positions, and moveable miniature neutron detectors, which can be positioned at the center of selected fuel assemblies anywhere along the length of the fuel assembly vertical axis. The basic system for insertion of these detectors is shown on Figure 7.7-8.

7.7.1.9.1 Thermocouples

The chromel-alumel thermocouples are inserted into guide tubes that penetrate the reactor vessel head through seal assemblies and terminate at the exit flow end of the fuel assemblies. The thermocouples are provided with two primary seals, a conoseal and swage type seal from conduit to head. The thermocouples are supported in guide tubes in the upper core support assembly. Thermocouple readings are monitored by the computer, with backup readout provided by a precision indicator with manual point selection located in the main control room. Information from the incore instrumentation is available even if the BVPS-2 computer is not in service.

The BVPS-2 control systems will prevent an undesirable condition in the operation of the nuclear plant that, if reached, will be protected by reactor trip. The description and analysis of this protection is covered in Section 7.2. Worst-case failure modes of the BVPS-2 control systems are postulated in the analysis of off-design operational transients and accidents covered in Chapter 15, such as the following:

1. Uncontrolled RCCA withdrawal from a subcritical condition,

2. Uncontrolled RCCA withdrawal at power,

3. Misalignment of RCCA,

4. Loss of external electrical load and/or turbine trip,

5. Loss of all ac power to the station auxiliaries (station blackout),

6. Excessive heat removal due to feedwater system malfunctions,

7. Excessive load increase incident, and

8. Accidental depressurization of the RCS.

These analyses will show that a reactor trip set point is reached in time to protect the health and safety of the public under these postulated incidents and that the resulting coolant temperatures will produce a DNBR well above the limiting value of 1.30. Thus, there will be no clad damage and no release of fission products to the RCS under the assumption of these postulated worst-case failure modes of the BVPS-2 control system.

7.7.2.1 Separation of Protection and Control Systems

In some cases, it is advantageous to employ control signals derived from individual protection channels through isolation amplifiers contained in the protection channel. As such, a failure in the control circuitry does not adversely affect the protection channel. Test results have shown that postulated faults on the isolated output portion of the circuit (nonprotection side of the circuit) will not affect the input (protection) side of the circuit.

Where a single random failure can cause a control system action that results in a condition requiring protective action and can also prevent proper action of a protection system channel designed to protect against the condition, the remaining redundant protection channels are capable of providing the protective action even when degraded by a second random failure. This meets the applicable requirements in Paragraph 4.7 of IEEE Standard 279-1971.

Insert D

INSERT D

The loop Tavg and Delta-T channel required inputs to the steam dump system, reactor control system, the control rod insertion monitor and the pressurizer level control system are electrically isolated prior to being routed to the control cabinets. A median signal is then calculated for Tavg and Delta-T in the control cabinets utilizing a Median Signal Selector (MSS) for input to the appropriate control systems.

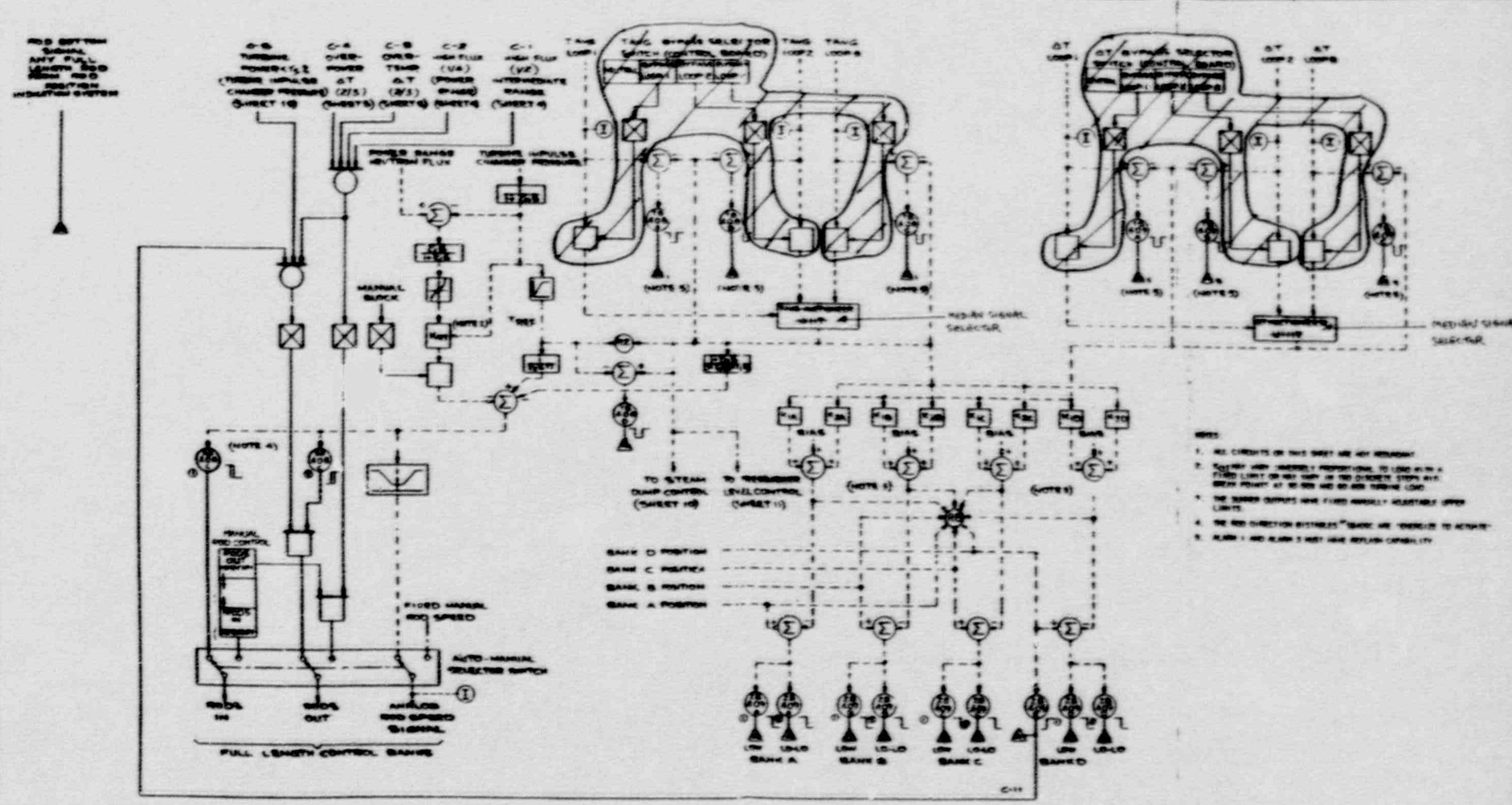


FIGURE 7.2-1(SH. 9 OF 18)
FUNCTIONAL DIAGRAM ROD CONTROLS & ROD BLOCKS
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

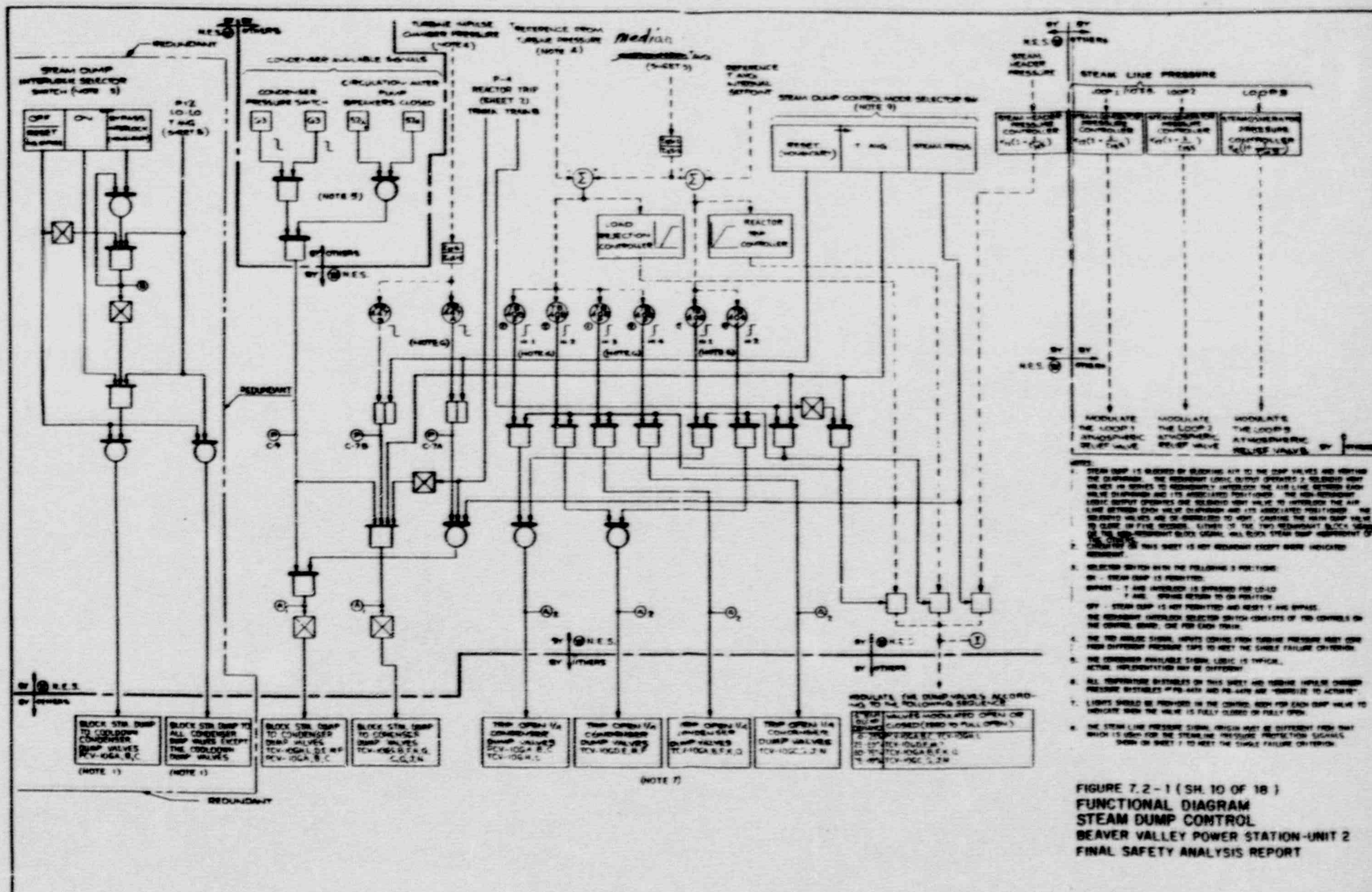
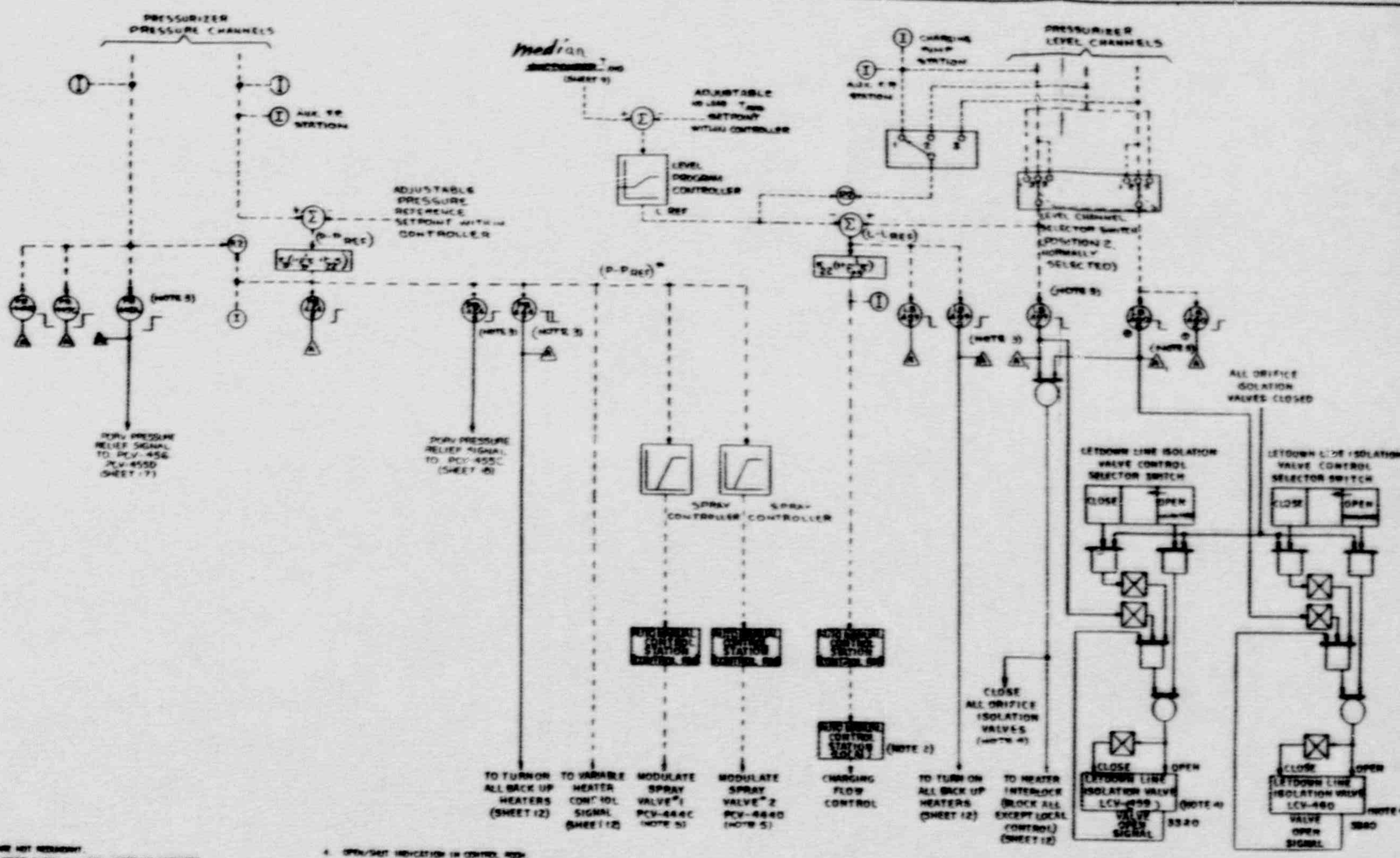


FIGURE 7.2-1 (SH. 10 OF 18)
FUNCTIONAL DIAGRAM
STEAM DUMP CONTROL
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT



NOTES:

1. ALL CIRCUITS ON THIS SHEET ARE NOT REDUNDANT.
2. LOCAL CONTROL OVERRIDES ALL OTHER SIGNALS. LOCAL OVERRIDE ACTUATES ALARM IN CONTROL ROOM.
3. PRESSURE BISTABLES NO. PB-444B, PB-444F, AND PB-445A AND LEVEL BISTABLES LB-459C, LB-460B AND LB-460C ARE "ENERGIZE TO ACTUATE".
4. OPEN/SHUT INDICATION IN CONTROL ROOM.
5. A LIGHT SHOULD BE PROVIDED IN THE CONTROL ROOM FOR EACH SPRAY VALVE TO INDICATE WHEN THE VALVE IS NOT FULLY CLOSED.

FIGURE 7.2-1 (SH. 11 OF 18)
FUNCTIONAL DIAGRAM
PRESSURIZER PRESSURE & LEVEL CONTROL
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

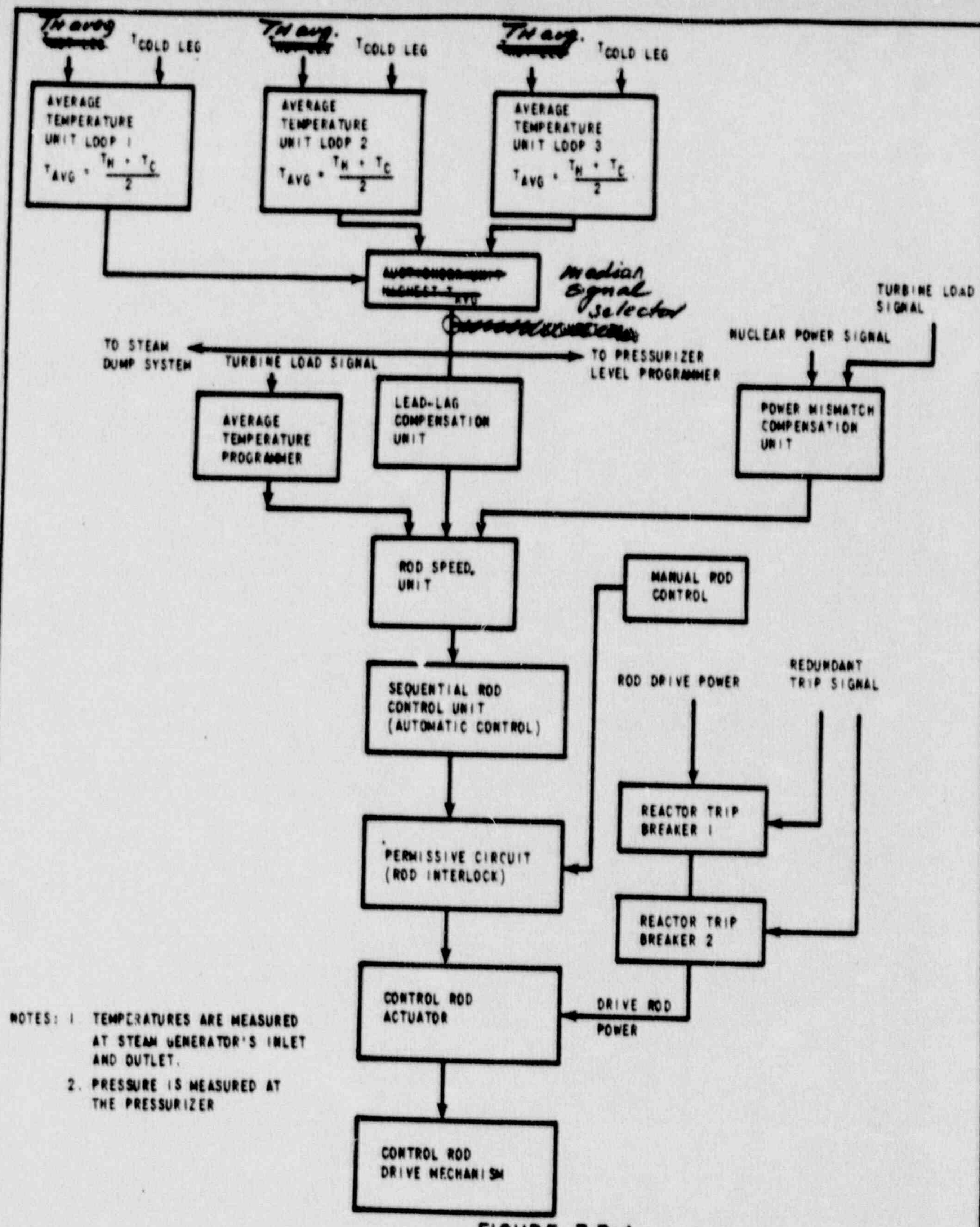
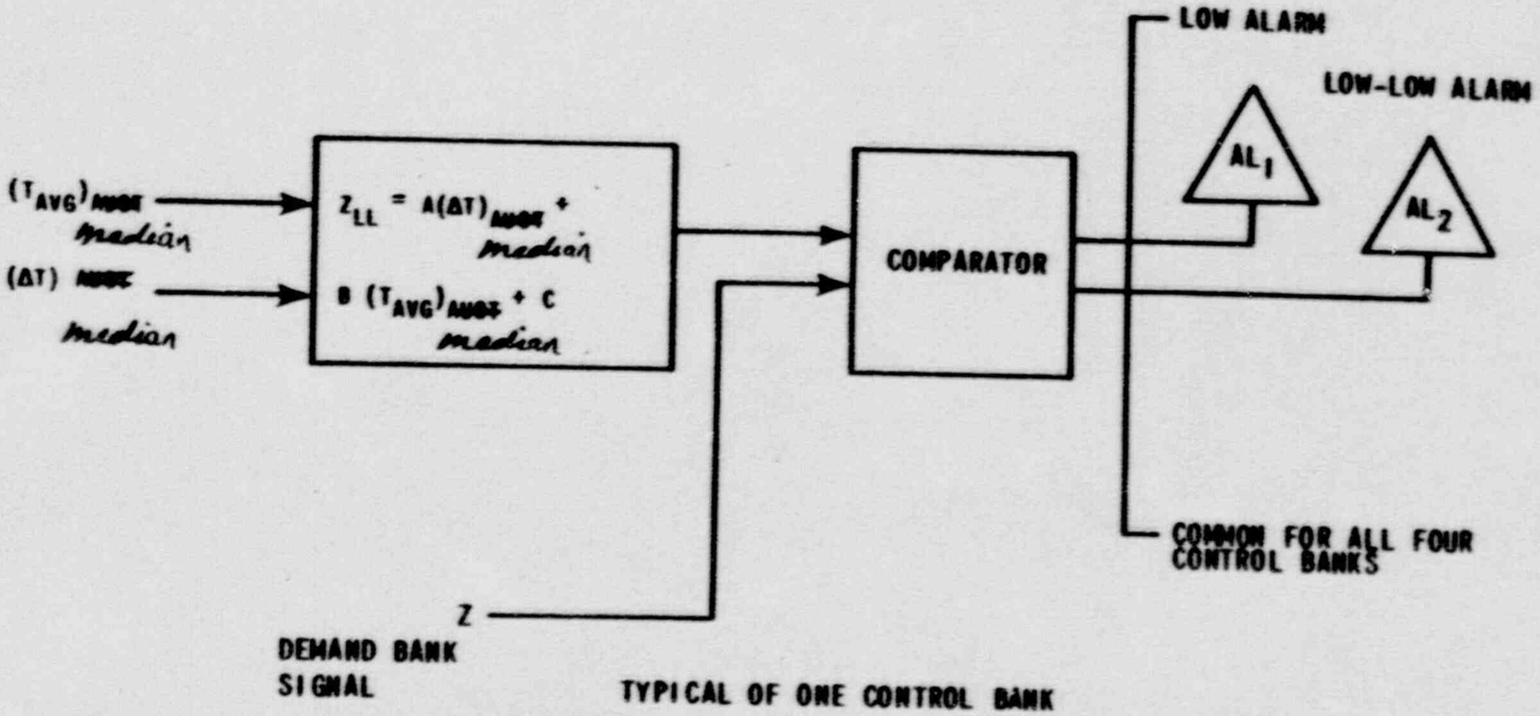


FIGURE 7.7-1
SIMPLIFIED BLOCK DIAGRAM
ROD CONTROL SYSTEM
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT



NOTE: 1. ANALOG CIRCUITRY IS USED FOR THE COMPARATOR NETWORK.
2. COMPARISON IS DONE FOR ALL CONTROL BANKS

FIGURE 7.7-2
CONTROL BANK ROD
INSERTION MONITOR
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

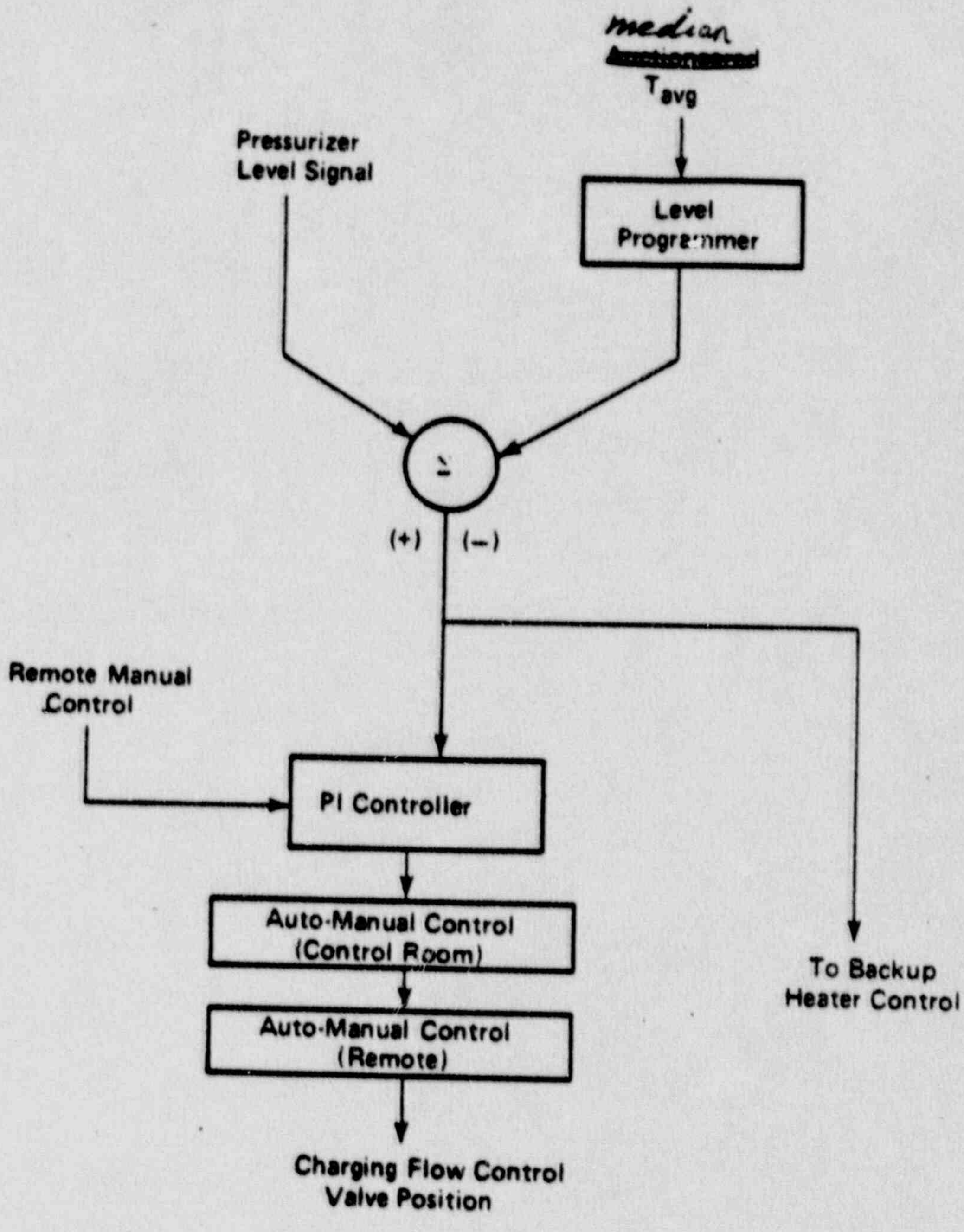


FIGURE 7.7-5
BLOCK DIAGRAM OF PRESSURIZER
LEVEL CONTROL SYSTEM
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

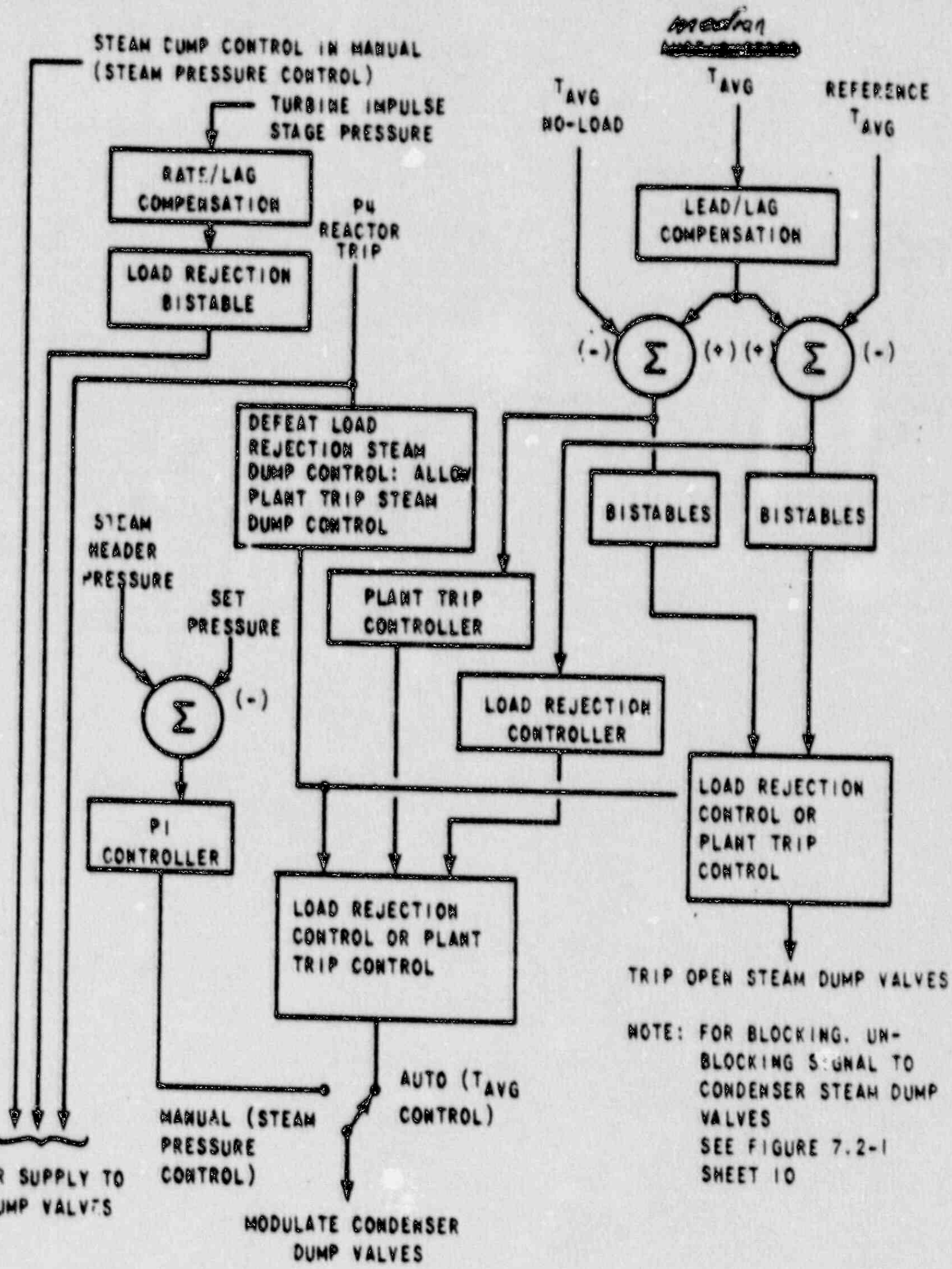


FIGURE 7.7-7
BLOCK DIAGRAM OF STEAM DUMP CONTROL SYSTEM
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

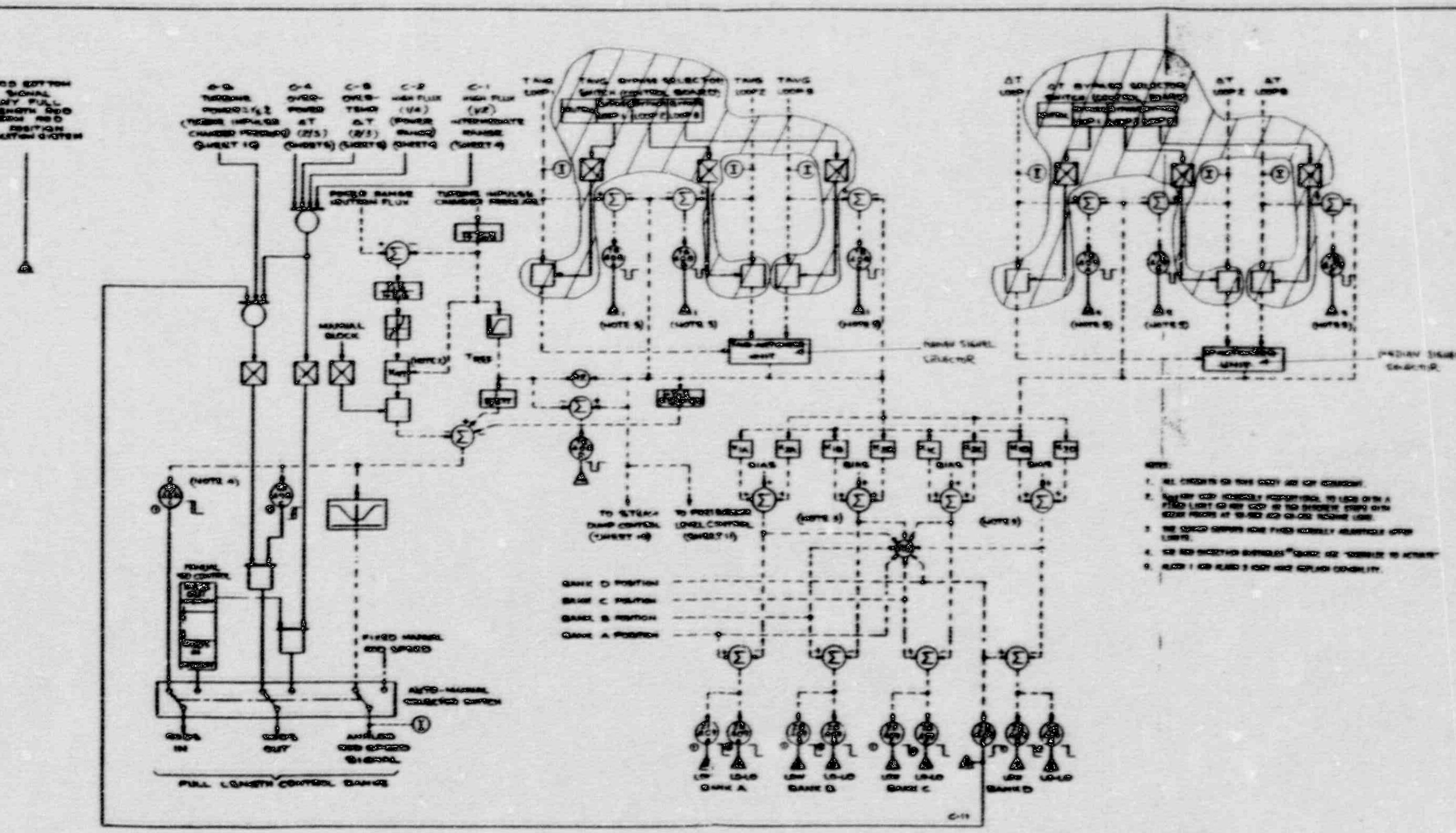


FIGURE 7.3-14
FUNCTIONAL DIAGRAM
ROD CONTROLS & ROD BLOCKS
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

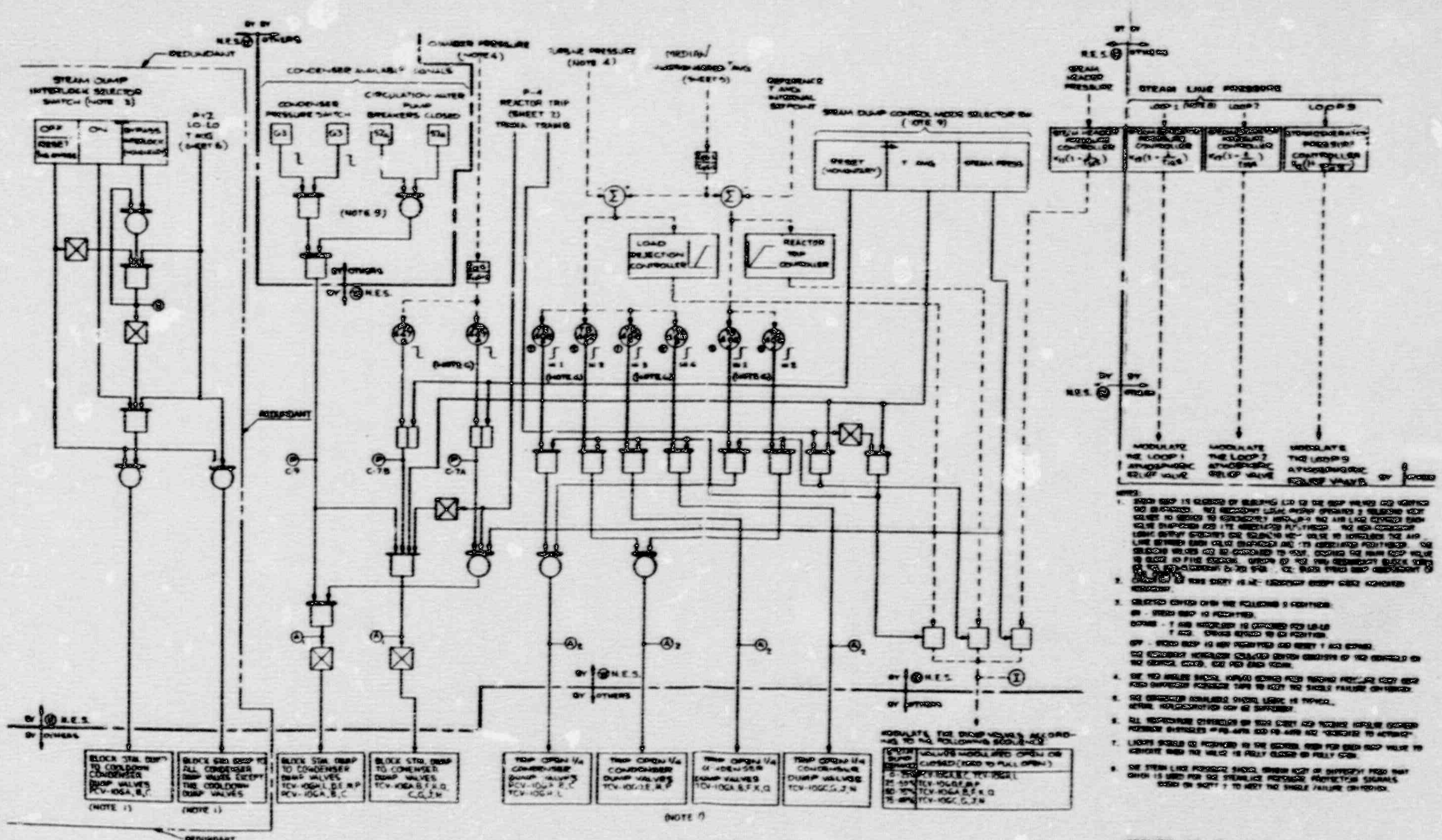


FIGURE 7.3-15
FUNCTIONAL DIAGRAM
STEAM DUMP CONTROL
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

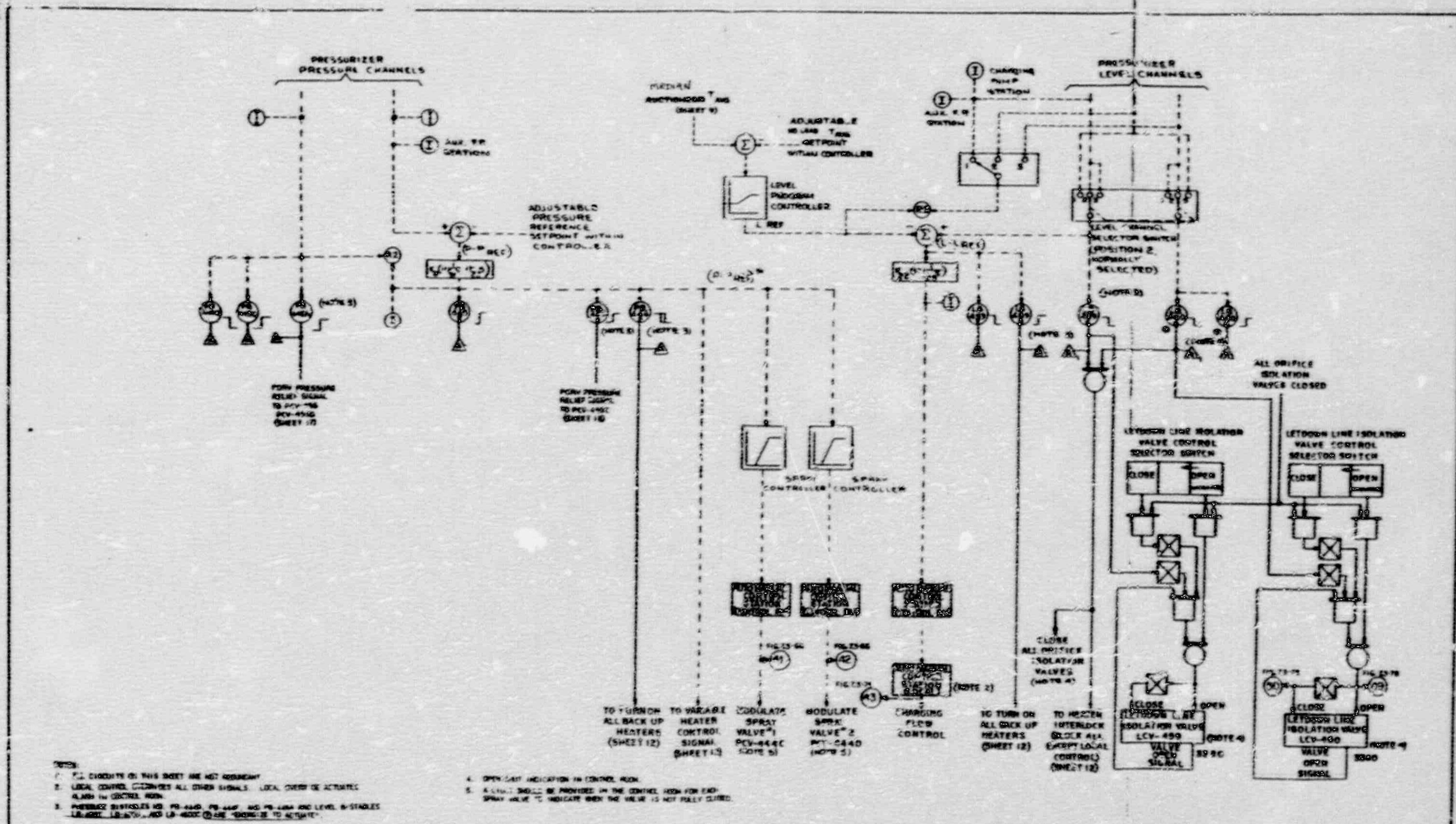


FIGURE 7.3-16
FUNCTIONAL DIAGRAM
PRESSURIZER PRESSURE
& LEVEL CONTROL
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT