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Attention: Robert C. Pierson, Director
Standardization and Non-Power Reactor Project Directorate

Subject: **GE Response to Agenda Items 1,5,9 and 16 Discussed During the GE/NRC Reactor Systems Branch Meeting on November 20-21, 1991**

Enclosed are thirty-four (34) copies of the GE response to the subject item.

It is intended that GE will amend the SSAR, where appropriate, with these responses in a future amendment.

Sincerely,

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Response To Item 1:

RESPONSE 440.187

As discussed in the Response to Question 100.1, the ABWR design assures the stability performance in the normal operating region is more stable than current operating BWRs by incorporating the following design features:

(1) Smaller inlet orifices, which increase the inlet single-phase pressure drop, and, consequently, improve the core and channel stability.

(2) Wider control rod pitch, which increases flow area, and, consequently, reduces the void reactivity coefficient and improves both core and channel stability, and

(3) More steam separators, which reduce the two-phase pressure drop, and improve the stability.

In order to reconfirm this conclusion, a stability analysis based on the procedures developed by the BWROG committee on thermal hydraulic stability (Reference 1) was performed for the ABWR. In this analysis, conservative nuclear conditions, taking into consideration of future core design, were assumed. The results at the most limiting conditions in the normal operating region (i.e.; the intercept of 102 % rod line with all operating RIPs at their minimum speeds, assuming only 9 out of 10 RIPs are in operation) are as follows:

- Core Decay Ratio 0.72,
- Channel Decay ratio 0.36.

These results are also shown in Figure 1 together with the criteria. From Figure 1, it is confirmed that that ABWR is stable in the normal operating region.

It should be noted that the likelihood of operation outside the normal region has also been minimized by the ABWR design. There are ten recirculation pumps served by four power supplies. The Recirculation Flow Control System has a triplicated logic incorporating a minimum speed demand. In addition, each pump has an Adjustable Speed Drive with a fixed minimum speed setpoint.

Furthermore, automatic logics (Figure 2) which prevent plant operation in the region with the least stability margin are also implemented. This design is similar to Option I-A, one of long-term solutions considered by the BWROG. In addition, in order to meet the stability design requirements specified in the ALWR Utility Requirements Document, Option III, LPRM based Oscillation Power Range Monitor (OPRM), which is also one of long-term solutions considered by the BWROG, will be implemented in the ABWR design, when the OPRM design is approved by the NRC.

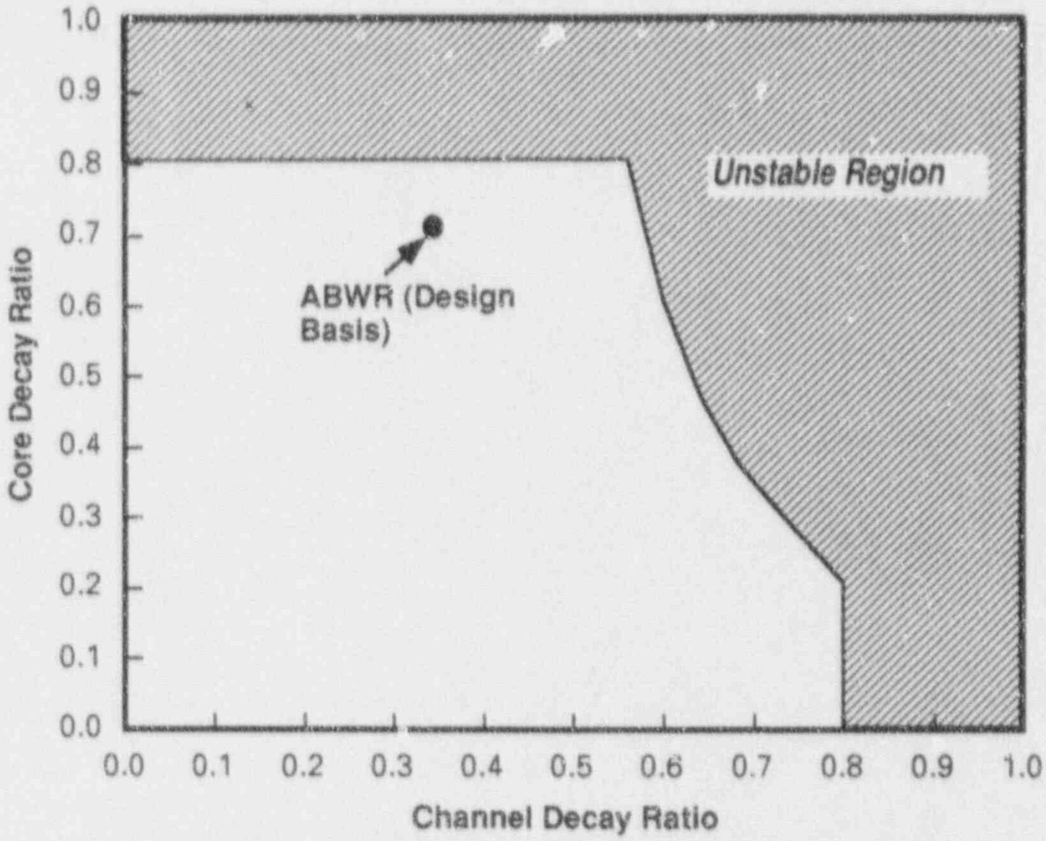
As for issues relates to ATWS stability, they are of no concerns to the ABWR design, since the ABWR design has logic to automatically initiate the SLCS, including automatic initiation of feedwater run back. Furthermore, the ABWR EPG will incorporate any changes recommended by the BWROG.

In summary, the ABWR stability design is consistent with the licensing methodology proposed by the BWROG committee on thermal hydraulic stability. The ABWR will be stable in the normal operating region.

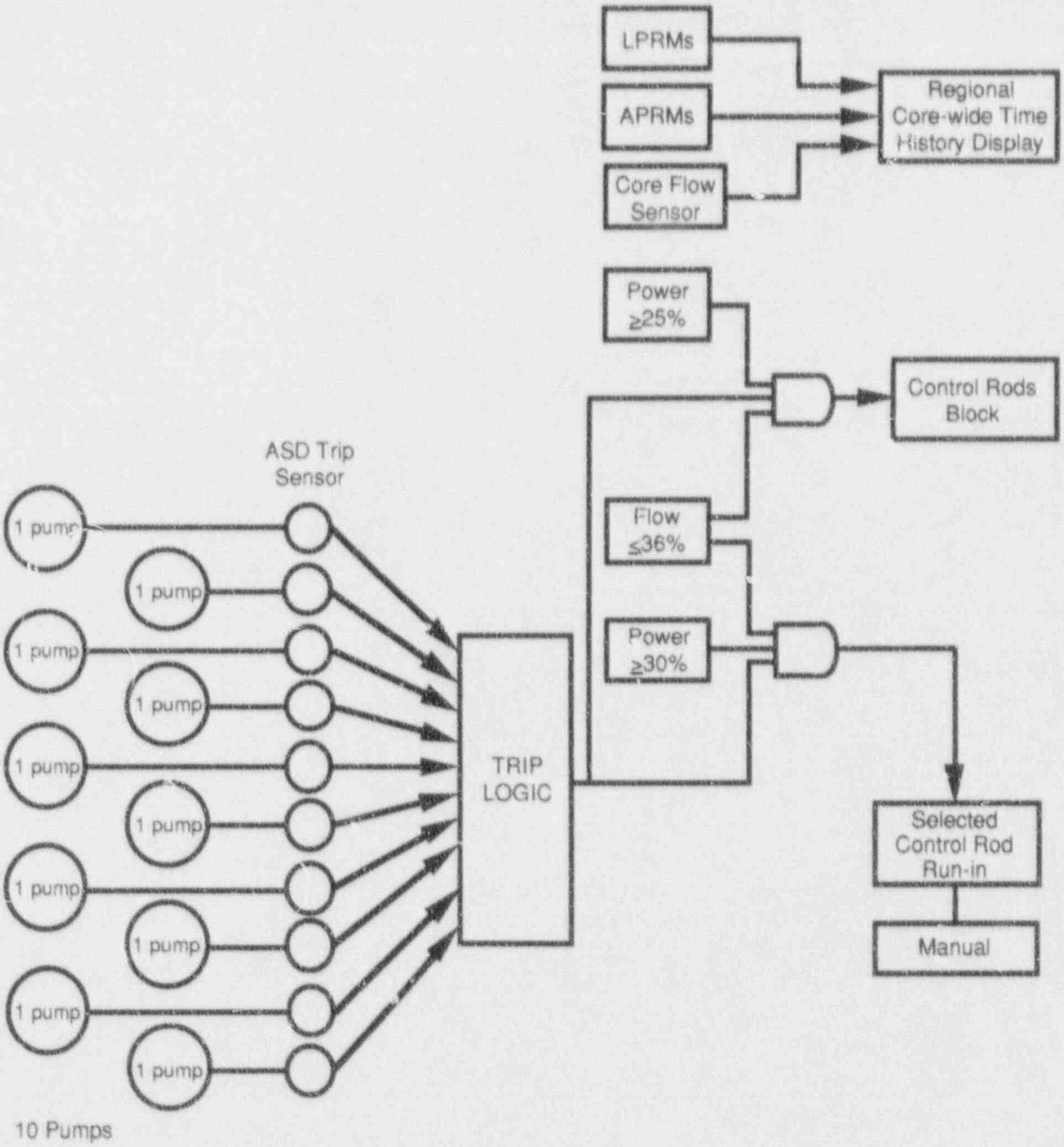
Reference 1:

NEDO-31960, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," June 1991.

Figure 1. ABWR Stability



ABWR is stable in normal operating domain



Notes:
1. Power ≥ 30%: To assure power level below 80% rod line at natural circulation.
2. Flow ≤ 36%: To assure flow rate is higher than that of eight RIPs operations with minimum pump speed

Figure 2. Stability Controls and Protection Logic

A

(5) Loss of AC Power

The RCIC system is designed to perform its function without AC power for at least 8 hours. Supporting systems such as DC power and the water supply will support the RCIC system during this time period. Without AC power, RCIC room cooling will not be available. However, room temperature will not reach the equipment maximum environmental temperature within 8 hours. (also see Subsection 19E.2.1.2.2 for additional information) depressurization systems perform adequate core cooling to prevent excessive fuel clad temperature during LOCA event. Detailed discussion of RCIC meeting this GDC is described in Subsection 3.1.2.

Compliance with GDC 36. The RCIC system is designed such that in-service inspection of the system and its components is carried out in accordance with the intent of ASME Section XI. The RCIC design specification requires layout and arrangement of the containment penetrations, process piping, valves, and other critical equipment outside the reactor vessel, to the maximum practical extent, permit access by personnel and/or appropriate equipment for testing and inspection of system integrity.

Compliance with GDC 37. The RCIC system is designed such that system and its components can be periodically tested to verify operability. Systems operability is demonstrated by preoperational and periodic testings in accordance with RG 1.68 Preoperational test will ensure proper functioning of controls, instrumentation, pumps and valves. Periodic testings confirm systems availability and operability through out the life of the plant. During normal plant operation, a full flow pump test is being performed periodically to assure systems design flow and head requirements are attained. All RCIC systems components are capable of individual functional testings during plant operation. This includes sensors, instrumentation, control logics, pump, valves, and more. Should the need for RCIC operation occur while the system is being tested, the RCIC system and its components will automatically re-aligned to provide cooling water into the reactor. The above test requirements satisfy GDC 37.

5.4.6.1 Design Basis

The reactor core isolation cooling (RCIC) system is a safety system which consists of a turbine, pump, piping, valves, accessories, and instrumentation designed to assure that sufficient reactor water inventory is maintained in the reactor vessel to permit adequate core cooling to take place. This prevents reactor fuel overheating during the following conditions:

(1) a loss-of-coolant (LOCA) event;

(2) vessel isolated and maintained at hot standby;

(3) vessel isolated and accompanied by loss of coolant flow from the reactor feedwater system;

(4) complete plant shutdown with loss of normal feedwater before the reactor is depressurized to a level where the shutdown cooling system can be placed in operation; or

INSERT A

~~(5) loss of AC power for 30 minutes.~~

~~Acceptance criteria II.3 of SRP Section 5.4.6 states that the RCIC system must perform its function without the availability of any AC power. Review Procedure III.7 further requires that there be sufficient battery capability for two hours of operation. While RCIC is designed for 30 minutes of operation during loss-of-ac power, the battery capacity should allow over four hours of operation, which would meet this requirement.~~

During loss of AC power, RCIC when started at water level 2 is capable of preventing water level from dropping below the level which ADS mitigates (Level 1). This accounts for decay heat boil-off and primary system leakages.

Following a reactor scram, steam generation will continue at a reduced rate due to the core fission product decay heat. At this time the turbine bypass system will divert the steam to

RESPONSE TO ITEM 9:

(9) Loss of FW Heating Transient

For ABWR design, the following design requirement is specified for the FW heating system design :

> "No single operator error or equipment failure shall cause loss of more than 55 °C (100 °F) feedwater heating ."

The reference steam and power conversion system shown in Figures 10.1-1 to 10.1-3 meets thie requirement. In fact, the FW temperature drop based on the reference heat balance shown in Figure 10.1-2 is as follows:

- isolation of one low pessure heater	< 15 °F
- isolation of one high pressure heater	< 28 °F
- isolation of one low pressure heater string	< 53 °F
- isolation of one high pressure heater string	< 53 °F

Therefore, the use of 100 °F temperature drop in the transient analysis is conservative.

A drop of 150 °F occurred at a domestic BWR was a unique condition for that particular plant design. That unique condition will not occur in the ABWR design.

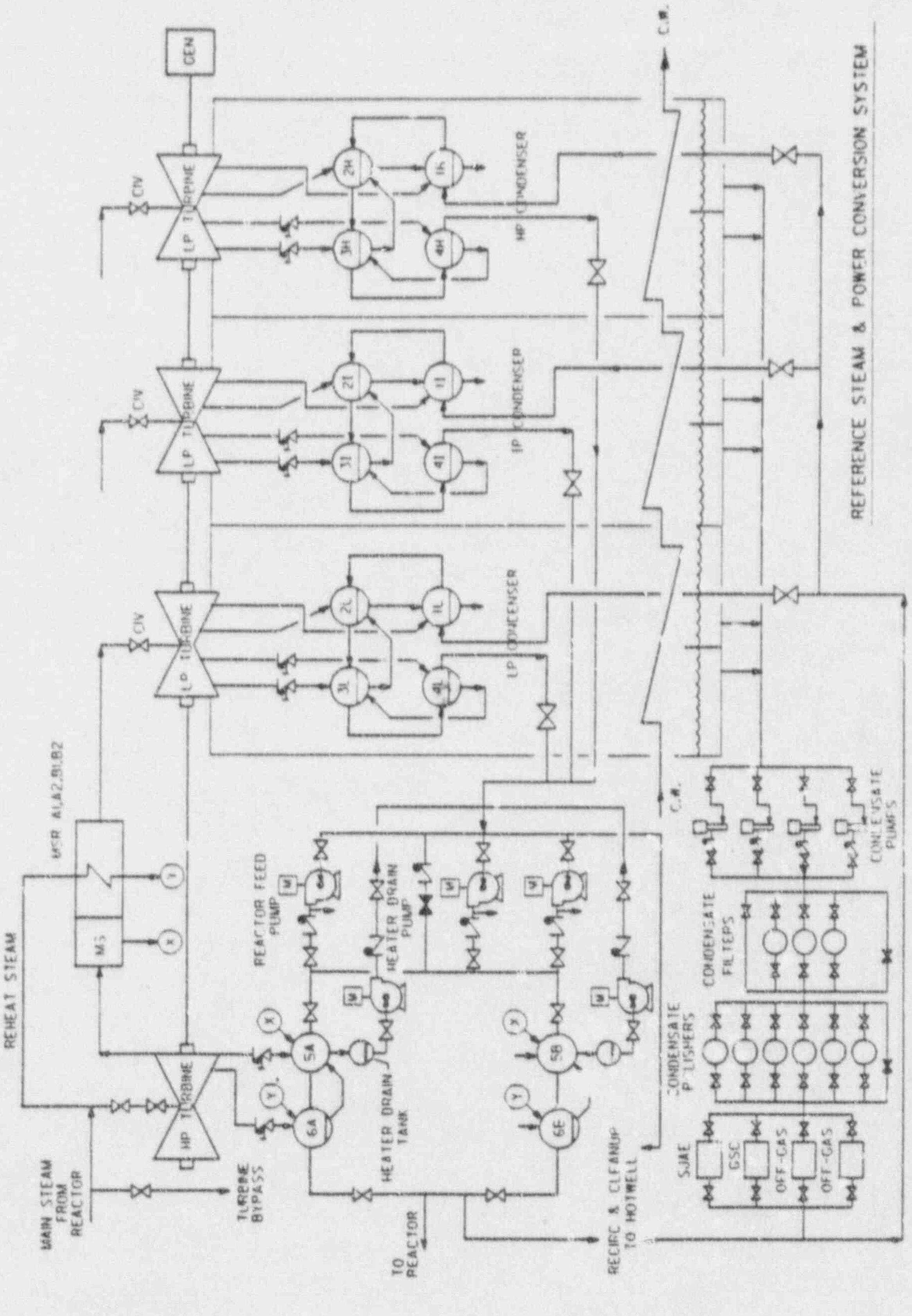


Figure 10.1-1 REFERENCE STEAM & POWER CONVERSION SYSTEM

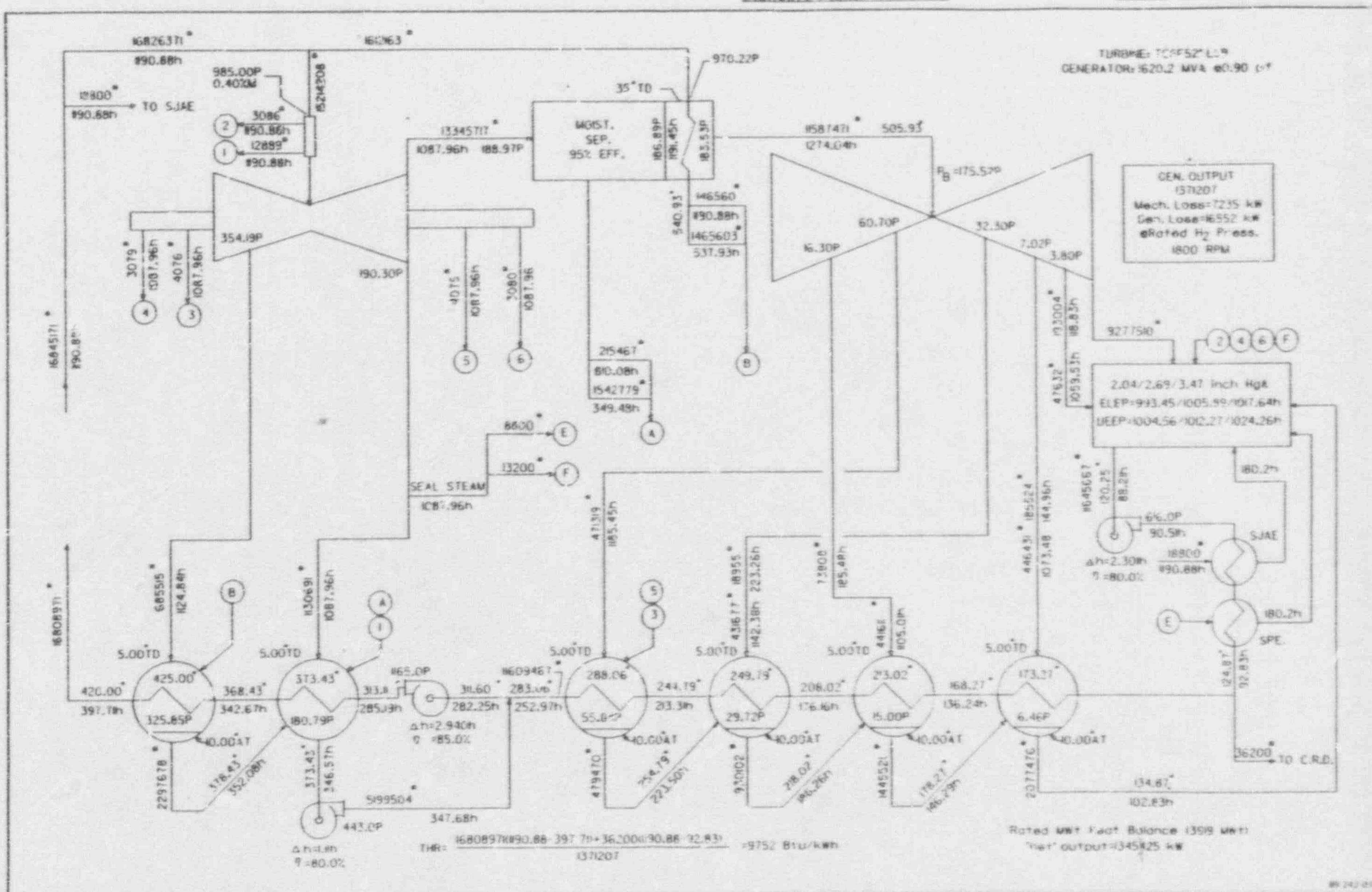


Figure 10.1-2 REFERENCE HEAT BALANCE FOR GUARANTEED REACTOR RATING

		ΔT/stage	ΔT/heater	ΔT/string
	124.87° F			
		43.40° F	14.467 °F	
1	168.27° F			
		39.75° F	13.250° F	
2	208.02° F			
		36.77° F	12.257° F	
3	244.79° F			
		38.27° F	12.757° F	
4	283.06° F			52.731° F
	313.11° F			
5	368.43° F	55.32° F	27.66 °F	
6	420.00° F	51.57° F	25.785° F	
				53.445 °F

4.13.9 The automatic flow control range shall be from 70% to 100% rated power (100% rod line).

4.13.10 The minimum RIP speed shall be greater than or equal to 450 RPM.

4.14 Core Flow Measurement Requirements

4.14.1 Core flow measurement shall be provided to deliver inputs for scram trip as shown in Figures 1 and 2.

4.14.2 The required measurement accuracy shall be within the requirements specified in Section 2.1.2.c.

4.14.3 The design basis maximum sensor response time shall be less than or equal to 0.25 second. (Analysis condition for E/PA = 1.0 second)

4.15 Feedwater Requirements

4.15.1 Trip of main feedwater pumps shall be initiated upon the condition of high vessel water level (Level 8). This function may be designed as a non-safety related trip. However, the design of this trip function shall be highly reliable.

4.15.2 The trip signal shall be the same signal to be supplied for the high vessel water level turbine trip (see Section 4.10).

4.15.3 The maximum feedwater runout capacity with a dome pressure of 74.9 Kg/cm^2g (1065 psig) shall be less than or equal to 130 percent of rated. The change of flow below the pressure specified above shall be less than 2.8% flow/Kg/cm^2 (0.2% flow/psi). E/P analysis may take credit of the maximum flow limit (110%) imposed by the feedwater control system.

4.15.4 Following a trip of one main feedwater pump, the minimum feedwater available to the vessel shall be greater than or equal to 75% of rated.

4.15.5 A six-heater feedwater heating system shall be designed to provide at least 215.5°C (420°F) feedwater at the rated condition.

* 4.15.6 No single operator error or equipment failure shall cause loss of more than 55°C (100°F) feedwater heating.

4.15.7 The 1σ (standard deviation) uncertainty for the feedwater flow measurement system shall be less than or equal to 1.76% of rated feedwater flow.

4.16 Auxiliary Water Makeup Requirements

4.16.1 The Reactor Core Isolation Cooling (RCIC) system shall be initiated upon the condition of low vessel water level (Level 2).

RESPONSE TO ITEM 16:

Issue 16 Capability of RCIC/RHR Systems to mitigate ATWS

(Containment Response for ATWS with Failure of Reactivity Control)

About October 17, the NRC forwarded the following question:

> "During the GE presentation to the staff on the ABWR PRA on August 6, 1991, GE referred to an INEL analysis which showed that RCIC was capable of preventing core damage. INEL performed the analysis of a high pressure ATWS with very low makeup flow to support GE's PRA assessment of the ABWR during degraded conditions. (Ref. DOE/ID 10211, October 1988.) The conclusion of the analysis was that based upon a constant vessel superheat of 175K, the equivalent of 3.45 heat exchangers are necessary to keep the peak containment pressure below the design pressure. Confirm that the three heat exchangers as presently designed having (sic.) sufficient heat removal capacity to mitigate ATWS."

'... ference report does indeed note that 3.45 heat exchangers would be required to maintain the peak containment pressure below the design pressure. However, this is not the correct limit to use in determining the success criteria for the PRA. An accident in which both the rod insertion and boron injection fails is well beyond the design basis. A more appropriate limit on the containment performance is the factored loads or service level C criterion. For the ABWR service level C corresponds to a pressure above 80 psig (Ref. 2, section 19E.2.3.2). Table 3 of reference 1 indicates that the peak containment pressure using 3 heat exchangers is 72 psig.

Furthermore, even if the more restrictive criteria were to apply, the time until the containment design pressure is reached is approximately 6 hours. This allows ample time for the operators to provide boron injection using alternate means, which would reduce the power generation rate well below that which could be removed using three trains of RHR. Therefore, no change is required in the ABWR success criteria.

References: 1. K. C. Wagner, "Analysis of a High Pressure ATWS with Very Low Make-up Flow", DOE/ID-10211, October 1988.

2. ABWR Standard Safety Analysis Report.