Department of Energy
Washington, D.C. 20545

Docket No. 50-537
HQ:S:82:050

JUN 18 1982

Mr. Paul S. Check, Director
CRBR Program Office
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Check:

RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION

Reference: Letter, P. S. Check to J. R. Longenecker, "CRBRP Request for Additional Information," dated March 16 and 25, 1982

This letter formally responds to your request for additional information contained in the reference letter.

Enclosed are responses to Questions CS 430.89 and 490.24 which will also be incorporated into the PSAR Amendment 69; scheduled for submittal later in July.

Sincerely,

John R. Longenecker

John R. Longenecker
Acting Director, Office of the
Clinch River Breeder Reactor
Plant Project
Office of Nuclear Energy

Enclosures

cc: Service List
Standard Distribution
Licensing Distribution

D001

Question CS430.89

(10.3) As explained in Issue No. 1 of NUREG-0133, credit it taken for all valves downstream of the Main Steam Isolation Valve (MSIV) to limit blowdown of a second steam generator in the event of a steam line break upstream of the MSIV. In order to confirm satisfactory performance following such a steam line break provide a tabulation and descriptive text (as appropriate) in the PSAR of all flow paths that branch off the main steam lines between the MSIV's and the turbine stop valves. For each flow path originating at the main steam lines, provide the following information:

a) System identification
b) Maximum steam flow in pounds per hour
c) Type of shut-off valve(s)
d) Size of valve(s)
e) Quality of the valve(s)
f) Design code of the valve(s)
g) Closure time of the valve(s)
h) Actuation mechanism of the valve(s) (i.e., Solenoid operated motor operated, air operated diagram valve, etc.)
i) Motive or power source for the valve actuating mechanism

In the event of the postulated accident, termination of steam flow from all systems identified above, except those that can be used for mitigation of the accident, is required to bring the reactor to a safe cold shutdown. For these systems describe what design features have been incorporated to assure closure of the steam shut-off valve(s). Describe what operator actions (if any) are required.

If the systems that can be used for mitigation of the accident are not available or decision is made to use other means to shut down the reactor describe how these systems are secured to assure positive steam shut-off. Describe what operator actions (if any) are required.

If any of the requested information is presently included in the PSAR text, provide only the references where the information may be found.

Response

Section 15.3.3 of the PSAR addresses steam or feed line pipe break event (updated and attached). Section 5.5 of the PSAR describes the design of the steam generator system. An updated list of steam generator system valves data in provided in the revised Section 5.5.3.4 and figure 5.1-4.

Power operators shall be sized to operate successfully under the maximum differential pressure determined in the design specification.

The main steam isolation valves (superheater outlet isolation valves) are capable of being closed to stop the venting of steam into the steam generator or turbine buildings in case of a steam line pipe break downstream of the isolation valves. The maximum steam flow rate is expected from a steam line break immediately downstream of the isolation valve. The disc and stem will be designed to withstand the forces produced when closing the valve under choke flow conditions.

Figure 5.5-2A shows a main steam isolation valve. It is a conventional gate valve to provide a minimum resistance flow path when the valve is wide open. A closed system hydraulic-pneumatic operator, shown in Figure 5.5-2B, is provided for opening and closing the valve during normal operation or during valve exercising. Upon loss of electrical power, the pneumatic and hydraulic solenoids are opened by springs, which causes pneumatic pressure to shuttle the valve operating cylinder. The oil below the valve operating cylinder returns to the reservoir through the pilot check valves, which are piloted open by pressure acting through the hydraulic solenoid valves. The gate valve is accelerated during the initial period of the blowdown and is decelerated at the end of the closing stroke by a hydraulic damper which enables soft seating of the gate while providing fast closing of the valve. A pressure compensated flow regulator ensures uniform closing times over variations in load. Position switches are provided to indicate gate position remotely. The valve is repositioned by energizing the motor and solenoids.

A superheater bypass valve is installed in parallel with the main superheater outlet isolation valve and check valve for use during plant startup for preheating the BOP steam lines and following plant shutdown to maintain the BOP pressurized. This is an active valve, designed to fail closed.

Each valve used in the SGS will be evaluated as to its performance relative to plant safety and mode of operation in the event of failure (fail open, fail closed, etc.). As part of these evaluations, the need for a pneumatic accumulator adjacent to a valve and solenoid requirements for emergency operation will be determined.

Tests and Inspections

Line valves will be shop tested by the manufacturer for performance according to the design specifications for leakage past seating surfaces and for integrity of the pressure retaining parts. Selected line valves will be manually operated during loop shutdown periods to assure operability.

5.5.2.3.2 Recirculation Pumps

The recirculation pump will be a single stage, centrifugal type, driven by a constant speed, 4.0 KV, 1000 HP motor. It will take suction from the steam drum, and provide 2.22×10^6 pounds of water per hour to the evaporators.

The pump and its support will be designed and fabricated per ASME Section III, Class 3 as shown in Table 5.5-6.

Safety/power relief valves are installed on the outlet line of the evaporator units, on the steam drum and on the outlet line from the superheater. These valves all meet the requirements of Section III of the ASME Boiler and Pressure Vessel Code for protection against overpressure. Table 5.5-8 indicated design pressures and valve settings for the steam generator safety/relief valves. Additional valve data is provided in Table 5.5-8a.

5.5.3.5 Steam Generator Module Characteristics

Each evaporator module will produce 1.11 x 10^6 lb/hr of 50% quality steam from subcooled water. Each superheater module will produce 1.11 x 10^6 lb/hr of superheated steam from saturated steam. The thermal hydraulic normal design operating conditions are given in Table 5.5-9.

The steam generator modules will supply the turbine with steam at design conditions over a 40% to 100% thermal power operating range for both clean and fouled conditions. The steam generator modules are also capable of removing reactor decay heat with the natural convection in both the intermediate sodium loop and the recirculaton water loop.

This hockey stick unit is of the same basic design as that of the Atomics International-Modular Steam Generator (AI-MSG) unit which was tested in a test program carried out at the Sodium Component Test Installation. The AI-MSG employed a 158-tube module with an overall length of 66 feet, as compared to the 757-tube CRBRP Steam Generator which has an overall length of 65 feet. The AI-MSG heat exchanger was operated for a total of 4,000 hours including operation both as an evaporator (slightly superheated steam out) and as a once through evaporator-superheater (from sub-cooled liquid to completely superheated steam).

The AI-MSG served as a proof test of the AI prototype hockey-stick steam generator design. The unit was operated for 4,000 hours under steaming conditions; all of these 4,000 hours, the unit was at the same temperature level at which the prototype will operate, with a steam pressure equal to or greater than prototype conditions. Table 5.5-9A compares various design operating conditions for the CRBRP Units to the AI-MSG, and lists the number of hours which the AI-MSG operated under respective conditions. The AI-MSG operated at steam pressures equal to or greater than the CRBRP Units for essentially the whole 4,000 hrs., and at CRBRP superheater inlet temperature for 750 hrs.

Since the AI-MSG unit was operated in the once-through mod, simultaneous simulation of both inlet and outlet CRBRP conditions for the separate CRBRP evaporator and superheater units was not achieved, but operation over the CRBRP temperature and pressure range was achieved on both the sodium and steam conditions for significant portions of the test.

TABLE 5.5-8A

VALVE DATA SUMMARY

(a) VALVE IDENTIFICATION STEAM GENERATOR SYSTEM	(b) MAX FLOW lb/hr	(c) TYPE	(d) SIZE INCHES	(e,f) ASME SECTION III DIVISION	(g) CLOSURE TIME SEC	(h) ACTUATOR MECHANISM	(i) POWER SOURCE
Superheater Outlet (53SGV012)	1.11×10^{6}	Gate	16	Class 3	3 max.	Electro-Hydraulic	1E Electric *
Superheater Bypass (53SGV016)	3.41×10^{4}	Flow Control	4	Class 3	3 max.	Electro-Hydraulic	1E Electric *
Superheater Inlet (53SGV011)	1.11×10^{6}	Gate	12	Class 3	3 max.	Electro-Hydraulic	1E Electric *
Evaporator Inlet (53SGV008)	1.11×10^{6}	Gate	10	Class 3	3 max.	Electro-Hydraulic	1E Electric
Steam Generator Bldg. Feedwater Inlet Isolation (53SGV001)	1.22×10^{6}	Gate	10	Class 3	3	Electro-Hydraulic	1E Electric*
Main Feed Water Inlet (53SGV002)	1.22×10^{6}	Flow Control	10	Class 3	5	Air Diaphram	Instrument Air
Start-up Feedwater Inlet (53SGV003)	2.44×10^{5}	Flow Control	4	Class 3	5	Air Diaphram	Instrument Air
Steam Drum Drain Valves (53SGV014, 15)	1.1×10^{5}	Gate	6	Class 3	3	Electro-Hydraulic	1E Electric*

*Active Function (Safe Position) is 1E Electric

Figure 5.5-2A. Main Steamline Isolation Valve (Superheater Isolation Valve Outlet)

See Figure 5.5-2B for valve operator schematic

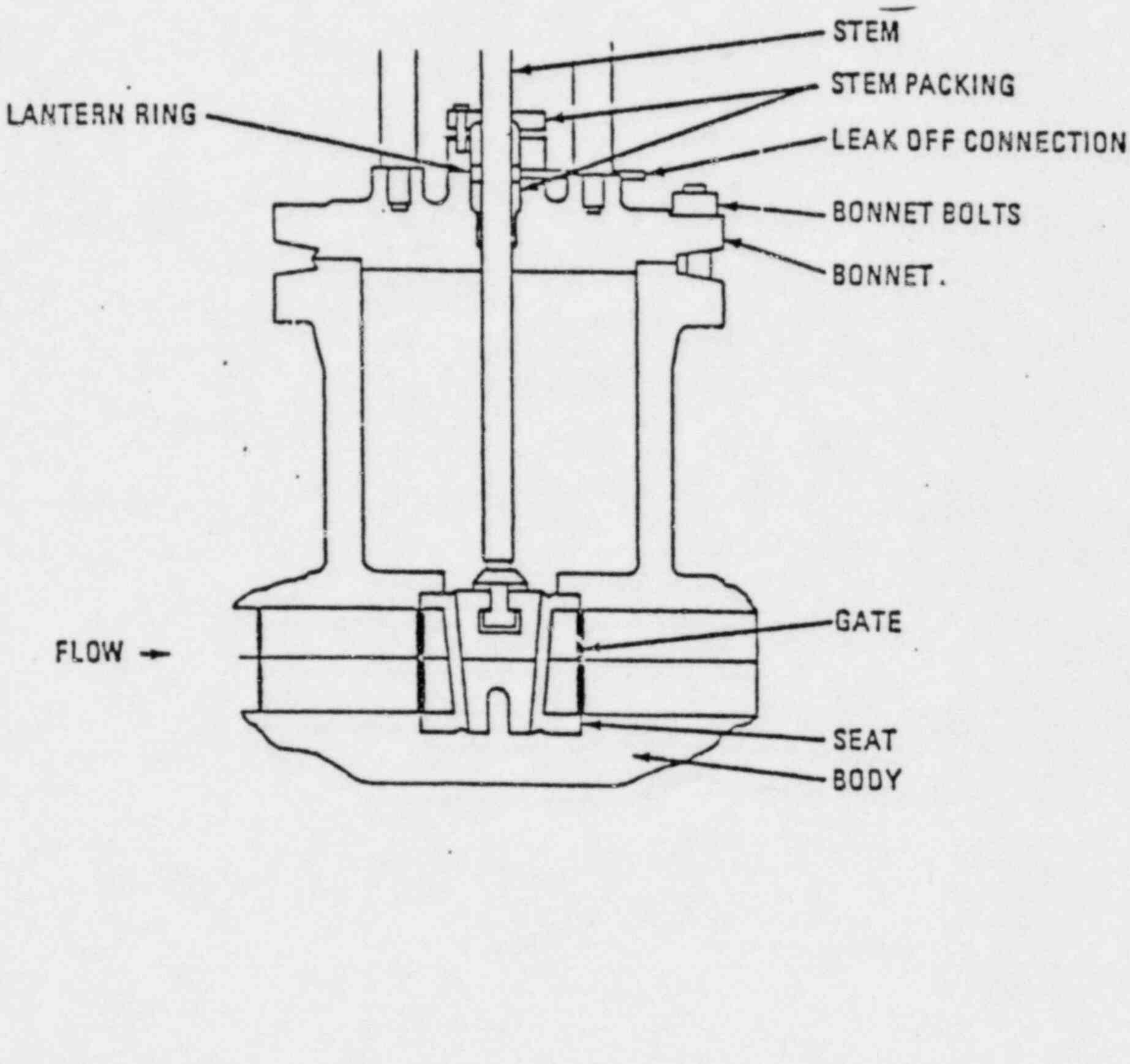
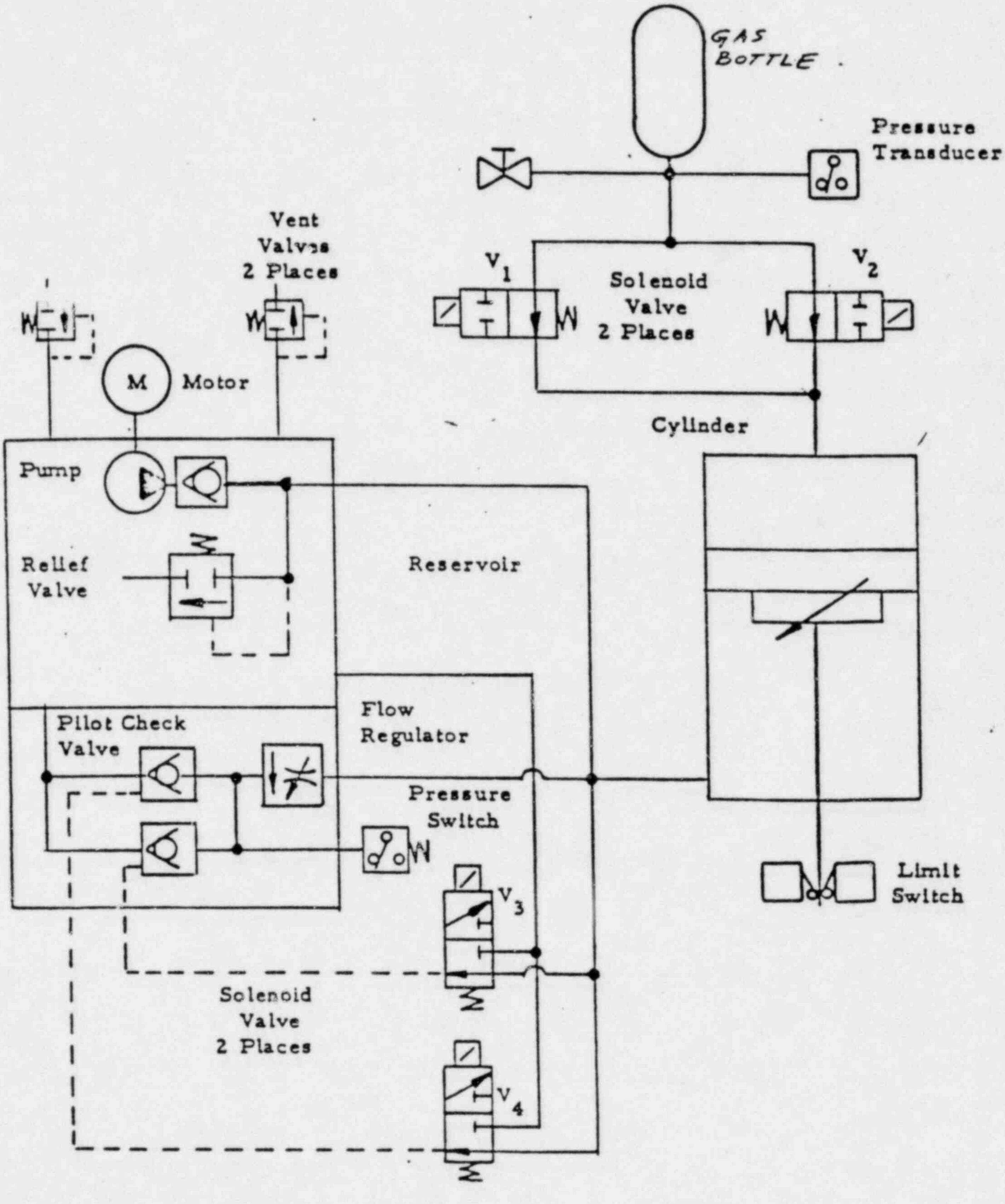


Figure 5.5-2B Operator Hydraulic-Schematic (Shown In Blowdown Model)



Amend. 69
July 1982

affecting the three steam supply systems and is provided if needed on a per loop basis. By definition, this zone of protection will include the high pressure steam supply system downstream from the individual loop check valves.

7.4.2.1.2 Equipment Design

A high steam flow-to-feedwater flow ratio is indicative of a main steam supply leak down stream from the flow meter or insufficient feedwater flow. The superheater steam outlet valves shall be closed with the appropriate signal supplied by the heat transport instrumentation system (Section 7.5). This action will assure the isolation of any steam system leak common to all three loops and also provide protection against a major steam condenser leak during a steam bypass heat removal operation.

7.4.2.1.3 Initiating Circuits

The OSIS is initiated by the SGAHRS initiation signal described in 7.4.1.1.3. This initiation signal closes the superheater outlet isolation valves in all 3 loops when a High Steam-to-Feedwater Flow Ratio or a Low Steam Drum Water level occurs in any loop. In each Steam Generator System loop, the trip signals for High Steam-to-Feedwater Flow Ratio and the Low Steam Drum Water level are input to a two of three logic network. If two of three trip signals occur in any of the 3 loops, SGAHRS is initiated, and all 3 loops are isolated from the main superheated steam system by closure of the superheater outlet isolation valves.

7.4.2.1.4 Bypasses and Interlocks

Control interlocks and operator overrides associated with the operation of the superheater outlet isolation valves have not been completely defined.

Bypass of OSIS may be required to allow use of the main steam bypass and condenser for reactor heat removal. In case the OSIS is initiated by a leak in the feedwater supply system, the operator may decide to override the closure of certain superheater outlet isolation valves.

7.4.2.1.5 Redundancy and Diversity

Redundancy is provided within the initiating circuits of OSIS. The primary trip function takes place when a high steam-to-feedwater flow ratio is sensed by two of three redundant subsystems on any one level sensed by two of three redundant channels in any one loop provides a backup trip function. Additional redundance is provided by three independnt SGS steam supply loops serving one common turbine header. Any major break in the high pressure steam system external from the individual loop check valves will be sensed as a steam feedwater flow ratio trip signal in all three loops.

7.4.2.1.6 Actuated Device

The superheater outlet isolation and superheater bypass valves utilize a high reliability electro-hydraulic actuator. These valves are designed to fail closed upon loss of electrical supply to the control solenoid.

7.4.2.1.7 Separation

The OSIS Instrumentation and Control System, as part of the Decay Heat Removal System is designed to maintain required isolation and separation between redundant channels (see Section 7.1.2).

7.4.2.1.8 Operator Information

Indication of the superheater outlet isolation valve position is supplied to the control room. Indicator lamps are used for open-close position indication to the plant operator.

7.4.2.2 Design Analysis

To provide a high degree of assurance that the OSIS will operate when necessary, and in time to provide adequate isolation, the power for the system is taken from energy sources of high reliability which are readily available. As a safety related system, the instrumentation and controls critical to OSIS operation are subject to the safety criteria identified in Section 7.1.2.

Redundant monitoring and control equipment will be provided to ensure that a single failure will not impair the capability of the OSIS instrumentation and Control System to perform its intended safety function. The system will be designed for fail safe operation and control equipment, where practical, will assure a failed position consistent with its intended safety function.

7.4.3 Remote Shutdown System

A Remote Shutdown System is provided. It consists of the following provisions:

10.3 MAIN STEAM SUPPLY SYSTEM

The Main Steam Supply System is shown in Figure 10.3.1.

10.3.1 Design Bases

The Main Steam Supply System includes steam piping and components downstream of the steam piping anchor at the steam generator building penetration and conveys superheated steam from each of the three steam generator loops to the high pressure turbine. Each steam generator loop is designed to furnish approximately 1,110,000 pounds per hour of 1535 psig, 906°F steam at the superheater outlet nozzle.

The portion of the Main Steam Supply System downstream of the piping anchor at the steam generator building penetration up to the turbine stop valves and including the turbine by-pass piping is designed to ANSI B31.1. The piping and component upstream of that point are safety related and designed in accordance with ASME Code Section III as discussed in Section 5.5. Piping downstream of the turbine by-pass valves and the isolation valve for the steam seal regulator are designed in accordance with ANSI B31.1, or manufacturer's standard. This portion of the system has no safety function, accordingly, no special precautions have been taken for protection from environmental effects.

A turbine by-pass system bypasses up to 80 percent of the rated steam flow (975 MWt) directly from the main steam header to the condenser and the deaerator.

No safety-related equipment is located in the turbine building. Therefore, a main steam line break cannot jeopardize any safety-related equipment. The ventilation system for the turbine generator building is not safety-related and effluent resulting from a main steam line break will not affect the HVAC system for any vital area.

10.3.2 Description

Three separate lines convey the superheated steam from the three steam generator loops to the main steam header. Following temperature and pressure equalization in the main steam header, the steam is carried to the turbine by four parallel pipes. Each of these pipes contains a stop valve and a turbine governor control valve.

The turbine bypass is connected to the main steam header located before the turbine stop valves. Figure 10.3-1 shows a diagrammatic arrangement of the Main Steam Piping System.

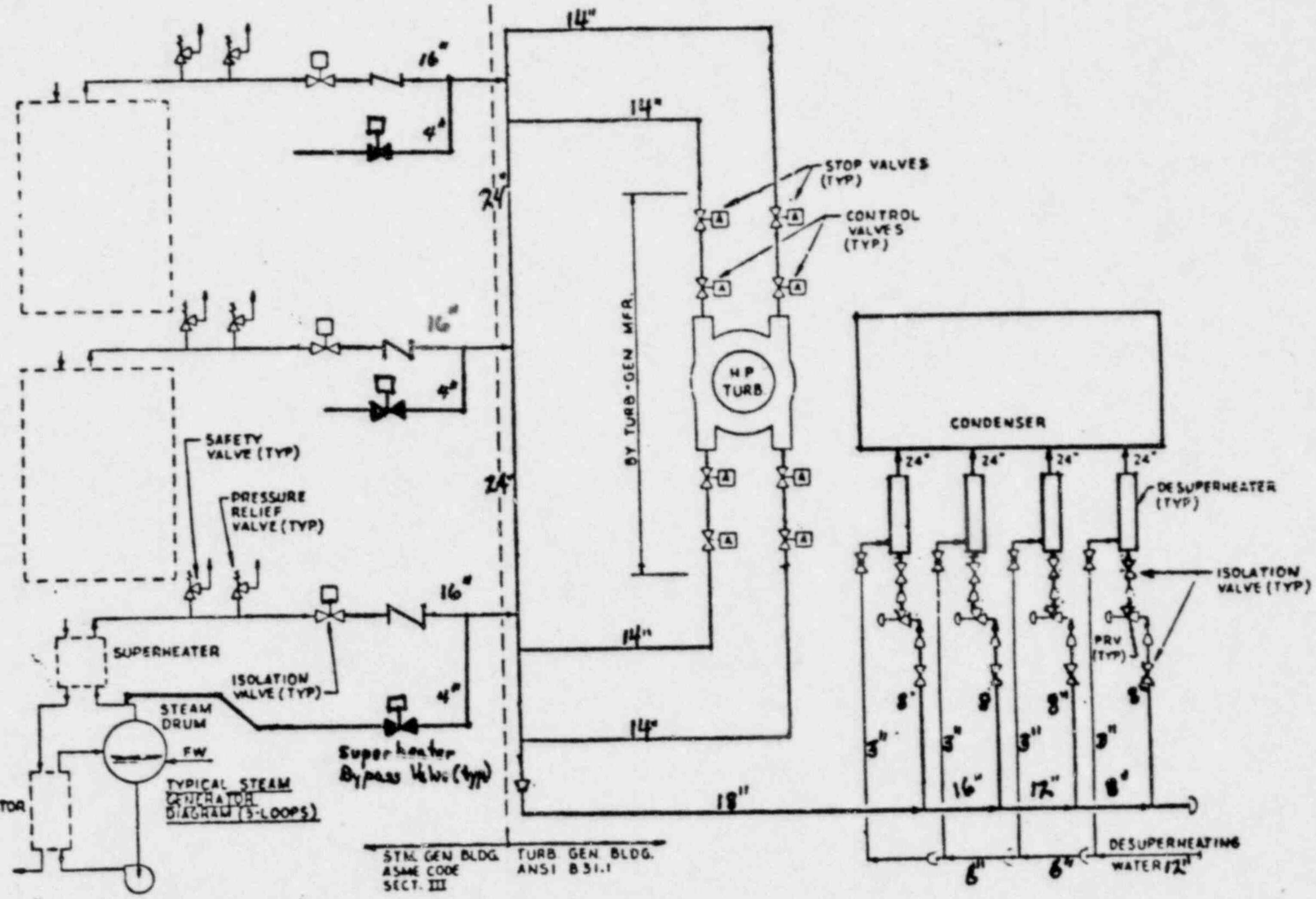


Figure 10.3-1. Basic Flow Diagram - Main Steam and Steam Dump System

Amend. 58
Nov. 1980

15.3.3 Extremely Unlikely Events

15.3.3.1 Steam or Feed Line Pipe Break

15.3.3.1.1 Identification of Causes and Accident Description

The breakage of a steam or feed pipe in the steam generator system is considered an extremely unlikely event. If such a break should occur, the resulting accident might have one of several forms, depending on where the break is located in the system, its size and whether or not it is insolatable. It should be noted that a reactor trip by the Plant Protection System will shut down the reactor before any of the steam system temperature changes have been transported back to the reactor core (at pony motor speed approximately 150 seconds) hence no problem results with immediate reactor safety. The event instead is considered in the plant design for its effect on plant component service life through thermal transient-induced stress.

The plant has incorporated design features to protect against the steam line break. For instance the Superheater Outlet Isolation Valve and Superheater Bypass Valve in each loop are active valves and will close within 3 seconds following a steam line break. Closing of these valves in the failed loop will prevent blowdown of more than one loop through the postulated pipe break. The valves in the failed loop will close by either a Low Superheater Outlet Pressure (< 1100 psig) or a High Steam/Feedwater Flow Mismatch. When a high steam/feedwater flow ratio occurs, the Superheater Outlet Isolation Valves and Superheater Bypass Valves in the other two loops will close. A detailed description of the Outlet Steam Isolation Subsystem (OSIS) is presented in Section 7.4.2. The superheater Outlet Check Valve provides additional back-up to prevent blowdown but is not relied upon in any analysis. The Superheater Bypass Valve is normally closed during operation.

In the event of failure of an active valve to close, the Superheater Outlet and Bypass Valves in the other two loops preclude their blowdown.

Breaks at the following locations have been investigated:

a. Main steam line rupture.

b. Steam line from a superheater to the main steam header

c. Saturated steam line between the steam drum and the superheater.

d. Feedline break

e. Recirculation line break

The saturated steam line break has been selected as the most severe thermal transients of the events presented above. Analysis results for this event are presented in Figure 15.3.3.1-1. All of the above cases are summarized as follows:

Main steam line rupture:

A steam break at the main steam header would, if not isolated, produce a severe cold leg temperature transient in all three loops consisting of a down transient due to initial excess cooling followed by an up-transient after dryout. It is not plausible, however, to assume that isolation would fail to occur in all three loops, hence for case (a) automatic isolation was assumed at three seconds with isolation initiated by the Plant Protection System (PPS).

Once the superheater outlet isolation valves close, the plant achieves a new operating point based on steam load through the safety valves and hence no other excessive plant temperatures are produced. As noted below, a reactor shutdown is initiated by the PPS based on either the primary shutdown system (steam/feed flow mismatch) or secondary system (Low Drum Level), terminating high power operation before excessive loss of water inventory. Either the high steam-to-feedwater flow ratio or the Low Steam Drum Water Level Trip also activates the steam generator auxiliary heat removal system (SGAHRS) as noted below and discussed in Section 5.6. All three loops would provide heat removal from the core. With the superheat steam line isolated, pressure in the steam system will build up to the relief setpoint. The drum water level will drop due to steam venting and the low steam drum water level trip will then activate SGAHRS if it has not been activated earlier in the transient by the High Steam to Feedwater Flow Ratio.

Rupture in a Steam Line Between a Superheater and the Main Steam Header:

This event results from a break occurring in the superheater exit steam line upstream of the isolation valve. A similar event follows from a break downstream of the isolation valve (including a break in the main steam line) if the isolation valve fails to close. For these cases. isolation can still be effectively accomplished by the superheater inlet isolation valve, either by manual initiation or automatically when steam drum pressure falls below 500 psig. Consequently, a break in the superheater-to-header line has an effect similar to the preceding main steam line break case, but its effects are limited to a single loop.

Saturated Steam Line Break:

In the saturated steam line break, case (c) above, the break may be located such that loss of water in the affected steam drum cannot be prevented. Isolation valves on the modules could still be closed, but safety valve outflow will still lead to module dryout. Consequently, no credit is taken for isolation in these cases.

As steam is removed from the system by the break, increased flashing of water into steam within the steam generator occurs, removing additional heat and causing the sodium temperature initially to decrease at the evaporator exit. A plant shutdown, when initiated by low steam feed flow, will cause coastdown of the intermediate sodium pump, and hence will amplify the initial decrease in evaporator exit temperature. Subsequently, when most of the moisture has been discharged from the steam generator, both evaporators and superheater will dry out, and the evaporator exit sodium temperature will increase to approach the intermediate hot leg temperature. The cold leg temperature increase will eventually be transported back to the reactor inlet, after being conducted through the IHX of the affected loop. Due to extended transport delays at pony motor flowrates, the temperature increase

An alternate location for this break is at the exit of one evaporator module. Closure of the other isolation valves, including the inlet valve on the affected module, would lead to a dryout of the generator similar to previous cases. If the inlet isolation valve on the module does not close, the contents of the drum would be dumped through the affected module, producing a severe temperature down-transient on that module. The remaining module will dry out and its resulting increase in sodium exit temperature will mix with that from the faulted module to attenuate the net intermediate cold leg temperature transient.

For the steam and feed break cases, the following conditions have been applied to assure a conservative analysis:

a. The largest possible break size is assumed, corresponding to the full guillotine severance of the pipe involved.

b. The earliest PPS trip is used to predict the largest span for the sodium temperature transient for cases in which the intermediate cold leg temperature is considered.

c. The transients were run from a starting point at the 1121 MWt reactor power design condition (stretch power).

d. Credit has not been taken for heat storage in shell and structural metal in active or unheated parts of the modules in mitigating the thermal transients. Credit was taken only for 75% of the tube metal in the heated part of the modules.

e. No isolation was performed on the affected unit during the drum to superheater break, feed break and recirculation line break cases and the steam generator was allowed to go to full dryout.

The action of the Plant Protection System (PPS) in the above cases is the following:

Primary Shutdown System

a. Reactor and plant trip - steam-feedwater flow ratio

Secondary Shutdown System

a. Reactor and plant trip - high evaporator outlet temperature

Question CS490.24

The W-2 test is a slow, overpowered test (about 5 cents/s ramp rate) conducted on full length FFTF geometry fuel pins in the Sodium Loop Safety Facility (SLSF) by the Hanford Engineering Development Laboratory (HEDL). It has not been fully examined or analyzed; nevertheless, the test has several important implications for the CRBR.

First, there is the puzzle of very early cladding breaches, possibly as early as ten seconds into the transient, and with a breach definitely confirmed at about 15 seconds into the transient. These early failures were unexpected because of the low fluence that had been accumulated by the cladding.

Second, gross fuel expulsion occured about as predicted by all of the prediction methods (as to time) at about 22 seconds into the transient. However, the site of the expulsion was apparently at axial midplane, which was unexpected.

Third, it is speculated that the site of expulsion may have been influenced by the early failure, which is presumed to have occured at midplane.

The applicant is requested to comment on: 1) the implications of the early cladding breaches with respect to the adequacy of performance evaluation models in cladding failure criteria being used for the CRBR, and 2) the implications of the midplane site of the fuel expulsion and of the influence the early failure may have had on the location of the site, for beyond-design energetics.

Response

1) Evaluation of the W-2 Sodium Loop Safety Facility (SLSF) test was not intended to be used by the CRBRP as a primary requisite to test the validity of the CRBRP methodology in predicting incipient failure threshold (time). Since the completion of the test, considerable effort has been expended by the safety community reviewing the test results, however, complete test examination and interpretation of test instrumentation has not been reported in the open literature, although a preliminary data report is available. Once the W-2 test has been fully examined and it can be determined that the test will provide a useful benchmark relative to predicting cladding breach initiation, the incipient failure threshold time can be evaluated using the CRBRP methodology. A schedule for the release of the available testing information will be provided to NRC by July 31, 1982.

2) The TOP event with fuel expulsion at the core midplane has been analyzed extensively, and the results are documented in Reference QCS490.24-1. The analyses have shown that the midplane fuel expulsion would not result in a sustained superprompt critical excursion, whether fission gas or fuel

vapor pressure causes the fuel expulsion. Reference QCS490.24-1 also contains the results of an alternative SAS/FCI analysis to provide insite into the margin available. This less rigorous analysis assumed the superprompt critical excursion based on SAS/FCI calculations at near prompt critical, despite the fact the SAS/FCI calculations at such conditions were considered unrealistically conservative (Reference QCS490.24-2). The resulting work energy was calculated to be 33 MJ at sodium impact with the reactor head, which is well below the SMBDB value at 101 MJ.

The NRC question speculated that the site of fuel expulsion in the W-2 test may have been influenced by the early cladding breach which was not predicted by current analytical models. This implies that the fuel expulsion site may not be determined accurately within the current models. To address the implication, PLUTO2 calculations have been performed to confirm that the midplane fuel expulsion, which has been analyzed as mentioned above, is the most energetic case. The results of these PLUTO2 calculations are plotted in Figure QCS490.24-1. Examination of Figure QCS490.24-1 shows that the midplane fuel expulsion yields essentially the highest peak positive reactivity feedback from fuel motion. Therefore, it can be said that an early cladding breach may cause at worst fuel expulsion at the midplane, which has been analyzed from the standpoint of the whole core response (Reference QCS490.24-1).

References:

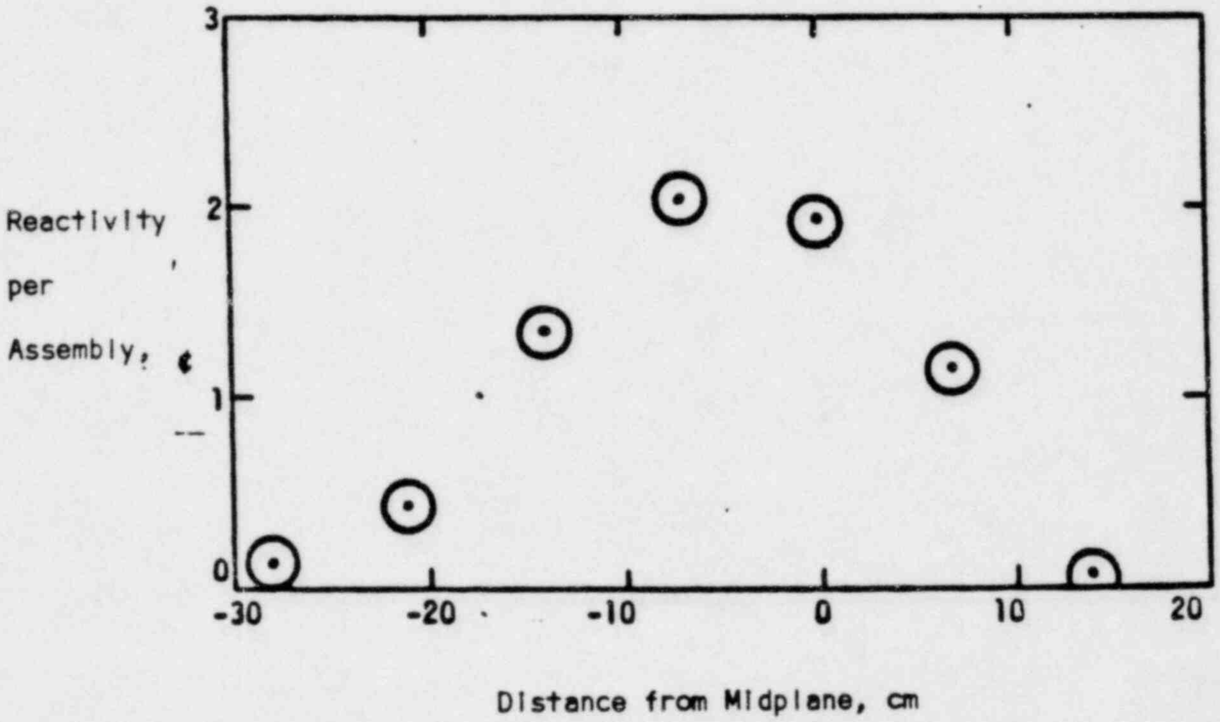
QCS490.24-1 S. K. Rhow, et al., "An Assessment of HCDA Energetics in the CRBRP Heterogeneous Reactor Core, "CRBRP-GEFR-00523, December 1981.

QCS490.24-2 J. L. McElroy, et al, "An Analysis of Hypothetical Core Disruptive Events in the Clinch River Breeder Reactor Plant," CRBRP-GEFR-00103, General Electric Co., April 1978.

Figure QCS490.24-1

PLUTO2 Prediction of Peak Fuel

Motion Reactivity vs. Failure Location



Amend. 69
July 1982