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March 25, 1980

Director of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Attn: A. Schwencer, Chief
Operating Reactors Branch No. 1
Division of Operating Reactors
Washington, DC 20555

Reference: Beaver Valley Power Station, Unit No. 1
Docket No. 50-334
NRC Letter Dated January 22, 1980 - Requirements For Auxiliary Feedwater System - Additional Information

Gentlemen:

The attached response is submitted in reply to the NRC request dated January 22, 1980, for additional information regarding requirements for the auxiliary feedwater system at Beaver Valley Power Station.

The response addresses several open items as presented in your letter. However, a complete response to all of the open items will not be available until an engineering evaluation of those items has been completed. The response to those items will be supplied at that time.

Should you have any questions or require additional information, please contact this office.

Very truly yours,

C. N. Dunn
Vice President, Operations

Attachment

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DUQUESNE LIGHT COMPANY
Beaver Valley Power Station
Unit No. 1

Reply to NRC Letter Dated January 22, 1980

1. Response to Recommendation GS-7

The AFW system is initiated automatically by safety injection signal, loss of offsite power and on low low steam generator level. These actuation signals are testable and these signals are the system actuations on which the FSAR Chapter 14 accident analysis are based. The AFW system is also automatically initiated on loss of main feedwater pumps in anticipation of low steam generator level. This anticipatory actuation is not testable during normal operation. The automatic start AFW signals and asociated circuitry are control grade. This instrumentation will be upgraded to safety grade by January 1, 1981, as indicated in our response to NUREG 0578.

Each AFW pump is demonstrated operable every 31 days by:

a. verifying each pump develops a discharge pressure of $\geq$ 1155 psig on recirculation flow.

b. verifying that the pump operates for at least 15 minutes.

c. verifying that the pump operates within the acceptable range of ASME Section XI IWP-3000.

d. verifying the correct position of all flow path valves. Redundant and independent system lineup verification is required for the system flowpath. During performance of the AFW pump surveillance testing, constant communications will be established between the control room and the AFW pump room while any normal discharge valve is closed.

e. cycling each power operated valve in the flowpath through one complete cycle.

f. verifying operability of each river water auxiliary supply valve by cycling each manual river water to AFW system valve through one complete cycle.

g. injecting a simulated signal into the steam generator level channels to test the S/G low and low low setpoints and status lights for automatic AFW pump starts.

On a bimonthly frequency, the Solid State Protection System logic is tested for the automatic AFW pump starts on steam generator low and low low levels.

Following an extended plant outage, AFW flow will be verified from the Primary Demineralized Water Storage Tank to the steam generators.

1. Response to Recommendation GS-7 (continued)

Each AFW pump is demonstrated operable at least once per 18 months by:

a. automatically starting the motor driven AFW pumps by operation of its associated slave relay. In the case of the steam driven AFW pump, operation of the slave relay should cause the steam supply valves to open.

b. in Operational Mode 5, the motor driven AFW pumps are automatically started on a station blackout coupled with a manual Safety Injection signal initiated from the control room.

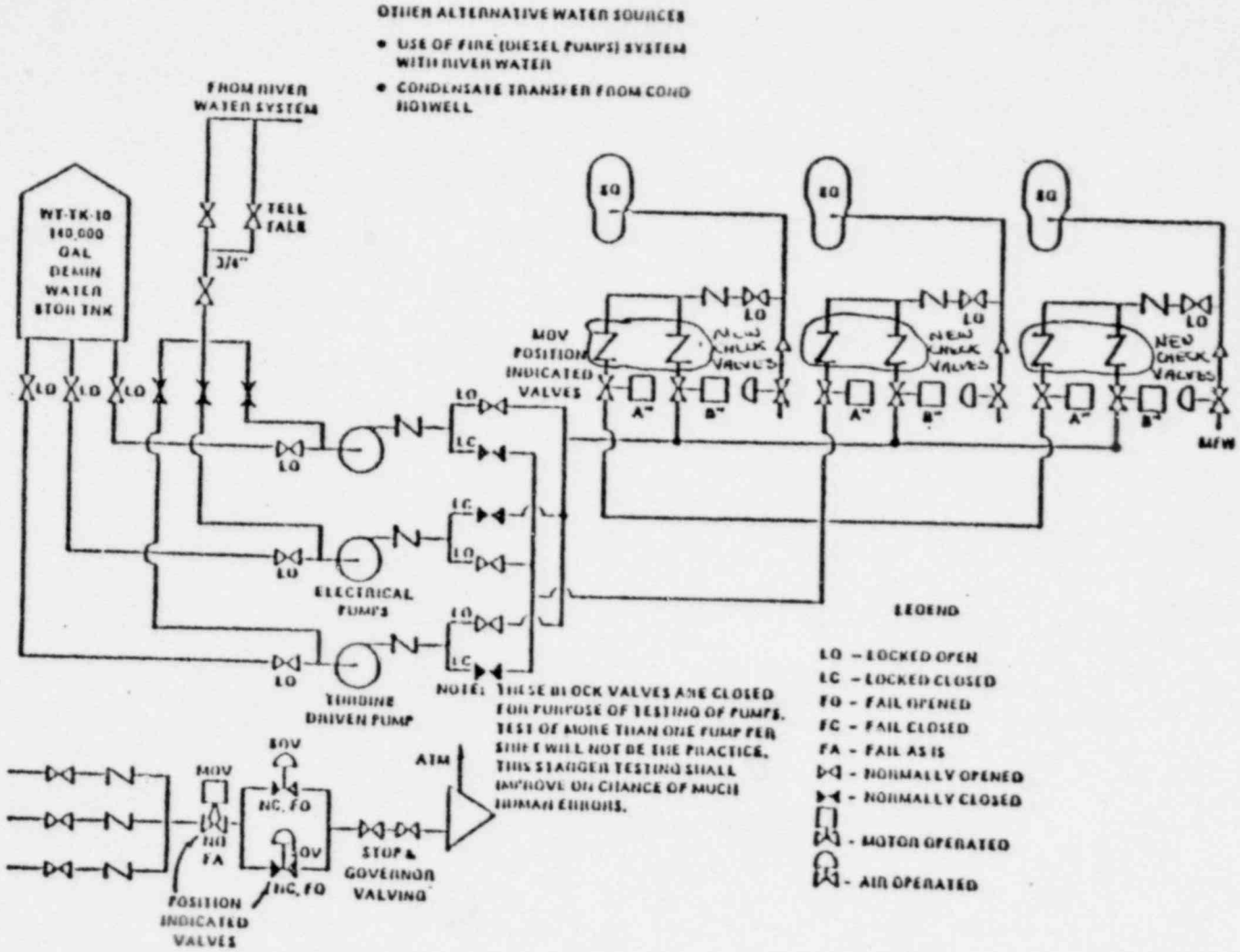
2. Response to Plant Specific Recommendation No. 7

The station staff has completed its review of the AFW system discharge block valves for the normal, transient and accident conditions, including the effects of high energy line breaks, and the valve lineup is shown in the attached figure. It should be noted that during the present outage six additional check valves were installed and were taken into consideration during this review. These check valves are noted on the attached figure. The block valve alignment which is submitted follows the NRC Standard Review Plan 10.4.9 for the AFW system.

Attached is Appendix 1 which provides the basis for AFW system flow requirements. This information is in response to Enclosure 2 of your October 11, 1979, letter.

3. Response to Long Term Recommendation GL-5

As stated earlier, we will upgrade the automatic initiation of AFW system to safety grade requirements by January 1, 1981.



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	Auxiliary Feedwater System	Beaver Valley Unit 1

APPENDIX 1

Design Basis for Auxiliary Feedwater System

Question 1

a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:

1) Loss of Main Feed (LMFW)
2) LMFW w/loss of offsite AC power
3) LMFW w/loss of onsite and offsite AC power
4) Plant cooldown
5) Turbine trip with and without bypass
6) Main steam isolation valve closure
7) Main feed line break
8) Main steam line break
9) Small break LOCA
10) Other transient or accident conditions not listed above.

b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

1) Maximum RCS pressure (PORV or safety valve actuation)
2) Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
3) RCS cooling rate limit to avoid excessive coolant shrinkage
4) Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

Response to 1.a

The Auxiliary Feedwater System serves as a backup system for supplying feedwater to the secondary side of the steam generators at times when the feedwater system is not available, thereby maintaining the heat sink capabilities of the steam generator. As an Engineered Safeguards System, the Auxiliary Feedwater System is directly relied upon to prevent core damage and system overpressurization in the event of transients such as a loss of normal feedwater or a secondary system pipe rupture, and to provide a means for plant cooldown following any plant transient.

Following a reactor trip, decay heat is dissipated by evaporating water in the steam generators and venting the generated steam either to the condensers through the steam dump or to the atmosphere through the steam generator safety valves or the power-operated relief valves. Steam generator water inventory must be maintained at a level sufficient to ensure adequate heat transfer and continuation of the decay heat removal process. The water level is maintained under these circumstances by the Auxiliary Feedwater System which delivers an emergency water supply to the steam generators. The Auxiliary Feedwater System must be capable of functioning for extended periods, allowing time either to restore normal feedwater flow or to proceed with an orderly cooldown of the plant to

the reactor coolant temperature where the Residual Heat Removal System can assume the burden of decay heat removal. The Auxiliary Feedwater System flow and the emergency water supply capacity must be sufficient to remove core decay heat, reactor coolant pump heat, and sensible heat during the plant cooldown. The Auxiliary Feedwater System can also be used to maintain the steam generator water levels above the tubes following a LOCA. In the latter function, the water head in the steam generators serves as a barrier to prevent leakage of fission products from the Reactor Coolant System into the secondary plant.

DESIGN CONDITIONS

The reactor plant conditions which impose safety-related performance requirements on the design of the Auxiliary Feedwater System are as follows for the Beaver Valley Unit No. 1 plant.

- Loss of Main Feedwater Transient
  - Loss of main feedwater with offsite power available
  - Station blackout (i.e., loss of main feedwater without offsite power available)
- Secondary System Pipe Ruptures
  - Feedline rupture
  - Steamline rupture
- Loss of all AC Power
- Loss of Coolant Accident (LOCA)
- Cooldown

Loss of Main Feedwater Transients

The design loss of main feedwater transients are those caused by:

- Interruptions of the Main Feedwater System flow due to a malfunction in the feedwater or condensate system
- Loss of offsite power or blackout with the consequential shutdown of the system pumps, auxiliaries, and controls

Loss of main feedwater transients are characterized by a rapid reduction in steam generator water levels which results in a reactor trip, a turbine trip, and auxiliary feedwater actuation by the protection system logic. Following reactor trip from high power, the power quickly falls to decay heat levels. The water levels continue to decrease, progressively uncovering the steam generator tubes as decay heat is transferred and discharged in the form of steam either through the steam dump valves to the condenser or through the steam generator safety or power-operated relief valves to the atmosphere. The reactor coolant temperature

increases as the residual heat in excess of that dissipated through the steam generators is absorbed. With increased temperature, the volume of reactor coolant expands and begins filling the pressurizer. Without the addition of sufficient auxiliary feedwater, further expansion will result in water being discharged through the pressurizer safety and relief valves. If the temperature rise and the resulting volumetric expansion of the primary coolant are permitted to continue, then (1) pressurizer safety valve capacities may be exceeded causing over-pressurization of the Reactor Coolant System and/or (2) the continuing loss of fluid from the primary coolant system may result in bulk boiling in the Reactor Coolant System and eventually in core uncovering, loss of natural circulation, and core damage. If such a situation were ever to occur, the Emergency Core Cooling System would be ineffectual because the primary coolant system pressure exceeds the shutoff head of the safety injection pumps, the nitrogen over-pressure in the accumulator tanks, and the design pressure of the Residual Heat Removal Loop. Hence, the timely introduction of sufficient auxiliary feedwater is necessary to arrest the decrease in the steam generator water levels, to reverse the rise in reactor coolant temperature, to prevent the pressurizer from filling to a water solid condition, and eventually to establish stable hot standby conditions. Subsequently, a decision may be made to proceed with plant cooldown if the problem cannot be satisfactorily corrected.

The blackout transient differs from a simple loss of main feedwater in that emergency power sources must be relied upon to operate vital equipment. The loss of power to the electric driven condenser circulating water pumps results in a loss of condenser vacuum and condenser dump valves. Hence, steam formed by decay heat is relieved through the steam generator safety valves or the power-operated relief valves. The calculated transient is similar for both the loss of main feedwater and the blackout, except that reactor coolant pump heat input is not a consideration in the blackout transient following loss of power to the reactor coolant pump bus.

The station blackout transient serves as the basis for the minimum flow required for the smallest capacity single auxiliary feedwater pump for the Beaver Valley Plant. The pump is sized so that any single pump will provide sufficient flow against the steam generator safety valve set pressure (with 3% accumulation) to prevent water relief from the pressurizer. The same criterion is met for the loss of feedwater transient by the operation of any two pumps, where A/C power is available.

Secondary System Pipe Ruptures

The feedwater line rupture accident not only results in the loss of feedwater flow to the steam generators but also results in the complete blowdown of one steam generator within a short time if the rupture should occur downstream of the last nonreturn valve in the main or auxiliary feedwater piping to an individual steam generator. Another significant result of a feedline rupture may be the spilling of auxiliary feedwater out the break as a consequence of the fact that the

auxiliary feedwater branch line may be connected to the main feedwater line in the region of the postulated break. Such situations can result in the spilling of a disproportionately large fraction of the total auxiliary feedwater flow because the system preferentially pumps water to the lowest pressure region in the faulted loop rather than to the effective steam generators which are at relatively high pressure. The system design must allow for terminating, limiting, or minimizing that fraction of auxiliary feedwater flow which is delivered to a faulted loop or spilled through a break in order to ensure that sufficient flow will be delivered to the remaining effective steam generator(s). The concerns are similar for the main feedwater line rupture as those explained for the loss of main feedwater transients.

Main steamline rupture accident conditions are characterized initially by plant cooldown and, for breaks inside containment, by increasing containment pressure and temperature. Auxiliary feedwater is not needed during the early phase of the transient but flow to the faulted loop will contribute to the release of mass and energy to containment. Thus, steamline rupture conditions establish the upper limit on auxiliary feedwater flow delivered to a faulted loop. Eventually, however, the Reactor Coolant System will heat up again and auxiliary feedwater flow will be required to be delivered to the unfaulted loop, but at somewhat lower rates than for the loss of feedwater transients described previously. Provisions must be made in the design of the Auxiliary Feedwater System to allow limitation, control, or termination of the auxiliary feedwater flow to the faulted loop as necessary in order to prevent containment overpressurization following a steamline break inside containment, and to ensure the minimum flow to the remaining unfaulted loops.

Loss of All AC Power

The loss of all AC power is postulated as resulting from accident conditions wherein not only onsite and offsite AC power is lost but also AC emergency power is lost as an assumed common mode failure. Battery power for operation of protection circuits is assumed available. The impact on the Auxiliary Feedwater System is the necessity for providing both an auxiliary feedwater pump power and control source which are not dependent on AC power and which are capable of maintaining the plant at hot shutdown until AC power is restored.

Loss-of-Coolant Accident (LOCA)

The loss of coolant accidents do not impose on the auxiliary feedwater system any flow requirements in addition to those required by the other accidents addressed in this response. The following description of the small LOCA is provided here for the sake of completeness to explain the role of the auxiliary feedwater system in this transient.

Small LOCA's are characterized by relatively slow rates of decrease in reactor coolant system pressure and liquid volume. The principal contribution from the Auxiliary Feedwater System following such small LOCAs is basically the same as the system's function during hot shutdown or

following spurious safety injection signal which trips the reactor. Maintaining a water level inventory in the secondary side of the steam generators provides a heat sink for removing decay heat and establishes the capability for providing a buoyancy head for natural circulation. The auxiliary feedwater system may be utilized to assist in a system cooldown and depressurization following a small LOCA while bringing the reactor to a cold shutdown condition.

Cooldown

The cooldown function performed by the Auxiliary Feedwater System is a partial one since the reactor coolant system is reduced from normal zero load temperatures to a hot leg temperature of approximately 350°F. The latter is the maximum temperature recommended for placing the Residual Heat Removal System (RHRS) into service. The RHR system completes the cooldown to cold shutdown conditions.

Cooldown may be required following expected transients, following an accident such as a main feedline break, or during a normal cooldown prior to refueling or performing reactor plant maintenance. If the reactor is tripped following extended operation at rated power level, the AFWS is capable of delivering sufficient AFW to remove decay heat and reactor coolant pump (RCP) heat following reactor trip while maintaining the steam generator (SG) water level. Following transients or accidents, the recommended cooldown rate is consistent with expected needs and at the same time does not impose additional requirements on the capacities of the auxiliary feedwater pumps, considering a single failure. In any event, the process consists of being able to dissipate plant sensible heat in addition to the decay heat produced by the reactor core.

Response to 1.b

Table 1B-1 summarizes the criteria which are the general design bases for each event, discussed in the response to Question 1.a, above. Specific assumptions used in the analyses to verify that the design bases are met are discussed in response to Question 2.

The primary function of the Auxiliary Feedwater System is to provide sufficient heat removal capability for heatup accidents following reactor trip to remove the decay heat generated by the core and prevent system overpressurization. Other plant protection systems are designed to meet short term or pre-trip fuel failure criteria. The effects of excessive coolant shrinkage are bounded by the analysis of the rupture of a main steam pipe transient. The maximum flow requirements determined by other bases are incorporated into this analysis, resulting in no additional flow requirements.

TABLE 1B-1

Criteria for Auxiliary Feedwater System Design Basis Conditions

Condition or Transient	Classification*	Criteria*	Additional Design Criteria
Loss of Main Feedwater	Condition II	Peak RCS pressure not to exceed design pressure. No consequential fuel failures	
Station Blackout	Condition II	(same as LMFW)	Pressurizer does not fill with 1 single motor driven aux. feed pump feeding 2 SGs.
Feedline Rupture	Condition IV	10 CFR 100 dose limits. Containment design pressure not exceeded	Core does not uncover
Loss of all A/C Power	N/A	Note 1	Same as blackout assuming turbine driven pump
Loss of Coolant	Condition III	10 CFR 100 dose limits 10 CFR 50 PCT limits	
	Condition IV	10 CFR 100 dose limits 10 CFR 50 PCT limits	
Cooldown	N/A		100°F/hr 547°F to 350°F

*Ref: ANSI N18.2 (This information provided for those transients performed in the FSAR).

Note 1 Although this transient establishes the basis for AFW pump powered by a diverse power source, this is not evaluated relative to typical criteria since multiple failures must be assumed to postulate this transient.

Question 2

Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a above including:

a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.

b. Time delay from initiating event to reactor trip.

c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).

d. Minimum steam generator water level when initiating event occurs.

e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences -- identify reactor decay heat rate used.

f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

g. Minimum number of steam generators that must receive AFW flow; e.g., 1 out of 2? 2 out of 4?

h. RC flow condition -- continued operation of RC pumps or natural circulation.

i. Maximum AFW inlet temperature.

j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removel due to blowdown.

k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

l. Operating condition of steam generator normal blowdown following initiating event.

m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

Response to 2

Analyses have been performed for the limiting transients which define the AFWS performance requirements. These analyses have been provided for review and have been approved in the Applicant's FSAR. Specifically, they include:

- Loss of Main Feedwater (Station Blackout)
- Rupture of a Main Feedwater Pipe
- Rupture of a Main Steam Pipe Inside Containment

In addition to the above analyses, calculations have been performed specifically for the Beaver Valley Unit No. 1 plant to determine the plant cooldown flow (storage capacity) requirements. The Loss of All AC Power is evaluated via a comparison to the transient results of a Blackout, assuming an available auxiliary pump having a diverse (non-AC) power supply. The LOCA analysis, as discussed in response 1.b, incorporates the system flows requirements as defined by other transients, and therefore is not performed for the purpose of specifying AFWS flow requirements. Each of the analyses listed above are explained in further detail in the following sections of this response.

Loss of Main Feedwater (Blackout)

A loss of feedwater, assuming a loss of power to the reactor coolant pumps, was performed in FSAR Section 14.1.8 for the purpose of showing that for a station blackout transient, a single motor driven auxiliary feedwater pump delivering flow to two steam generators does not result in filling the pressurizer. Furthermore, the peak RCS pressure remains below the criterion for Condition II transients and no fuel failures occur (refer to Table 1B-1). Table 2-1 summarizes the assumptions used in this analysis. The transient analysis begins at the time of reactor trip. This can be done because the trip occurs on a steam generator level signal, hence the core power, temperatures and steam generator level at time of reactor trip do not depend on the event sequence prior to trip. Although the time from the loss of feedwater until the reactor trip occurs cannot be determined from this analysis, this delay is expected to be 20-30 seconds. The analysis assumes that the plant is initially operating at 102% (calorimetric error) of the Engineered Safeguards design (ESD) rating shown on the table, a very conservative assumption in defining decay heat and stored energy in the RCS. The reactor is assumed to be tripped on low-low steam generator level, allowing for level uncertainty. The FSAR shows that there is a considerable margin with respect to filling the pressurizer. A loss of normal feedwater transient with the assumption that the two smallest auxiliary feedwater pumps and reactor coolant pumps are running even results in more margin.

This analysis establishes the capacity of the smallest single pump and also establishes train association of equipment so that this analysis remains valid assuming the most limiting single failure.

Rupture of Main Feedwater Pipe

The double ended rupture of a main feedwater pipe downstream of the main feedwater line check valve is analyzed in FSAR Section 14.2.5.2. Table 2-1 summarizes the assumptions used in this analysis. Reactor trip is assumed to occur when the unaffected steam generators are at the low level setpoint (adjusted for errors) and the faulted loop is assumed to be empty. This conservative assumption maximizes the stored heat prior to reactor trip and minimizes the ability of the steam generator to remove heat from the RCS following reactor trip due to a conservatively small total steam generator inventory. As in the loss of normal feedwater analysis, the initial power rating was assumed to be 102% of the ESD rating. The FSAR analysis shows that no auxiliary feedwater flow is assumed until 10 minutes after the break. At this time it is assumed that the operator has isolated the AFWS from the break and the minimum flow requirement of 350 gpm (total) commences. The criteria listed in Table 1B-1 are met.

This analysis may establish the capacity of single pumps, establishes requirements for layout to preclude indefinite loss of auxiliary feedwater to the postulated break, and establishes train association requirements for equipment so that the AFWS can deliver the minimum flow required in 10 minutes following operator actions assuming the worst single failure.

Rupture of a Main Steam Pipe Inside Containment

Because the steamline break transient is a cooldown, the AFWS is not needed to remove heat in the short term. Furthermore, addition of excessive auxiliary feedwater to the faulted steam generator will affect the peak containment pressure following a steamline break inside containment. This transient is performed at three power levels for several break sizes. Auxiliary feedwater is assumed to be initiated at the time of the break, independent of system actuation signals. The maximum flow is used for this analysis, considering a pump runout. Table 2-1 summarizes the assumptions used in this analysis. At 10 minutes after the break, it is assumed that the operator has isolated the AFWS from the faulted steam generator which subsequently blows down to ambient pressure. The criteria stated in Table 1B-1 are met.

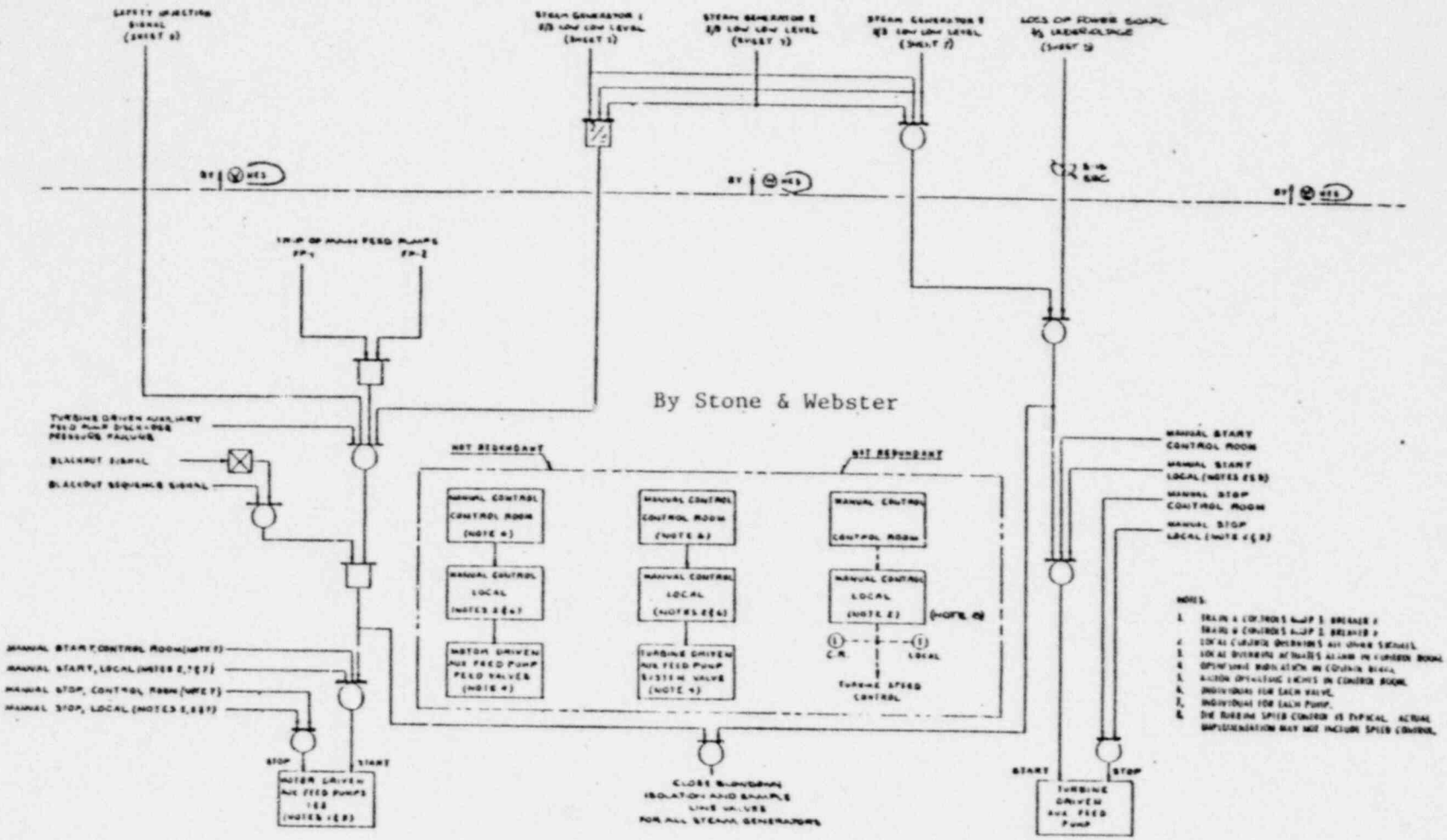
This transient establishes the maximum allowable auxiliary feedwater flow rate to a single faulted steam generator assuming all pumps operating, establishes the basis for runout protection, if needed, and establishes layout requirements so that the flow requirements may be met considering the worst single failure.

Plant Cooldown

Maximum and minimum flow requirements from the previously discussed transients meet the flow requirements of plant cooldown. This operation, however, defines the basis for tankage size, based on the required cooldown duration, maximum decay heat input and maximum stored heat in the system. As previously discussed in response 1A, the auxiliary feedwater system partially cools the system to the point where the RHRS may complete the cooldown, i.e., 350°F in the RCS. Table 2-1 shows the assumptions used to determine the cooldown heat capacity of the auxiliary feedwater system.

The cooldown is assumed to commence at the maximum rated power, and maximum trip delays and decay heat source terms are assumed when the reactor is tripped. Primary metal, primary water, secondary system metal and secondary system water are all included in the stored heat to be removed by the AFWS. See Table 2-2 for the items constituting the sensible heat stored in the NSSS.

This operation is analyzed to establish minimum tank size requirements for auxiliary feedwater fluid source which are normally aligned.



NOTE: Control of pump start/stop and auxiliary feedwater flow from the control room can be disabled and transferred to the emergency shutdown panel.

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Figure 2-1	Auxiliary Feedwater Actuation Logic	Beaver Valley Unit 1

TABLE 2-1

Summary of Assumptions Used in AFWS Design Verification Analyses

	Transient	Loss of Feedwater (station blackout)	Cooldown	Main Feedline Break	Main Steamline Break (containment)
a.	Max reactor power	102% of ESD rating (102% of 2774 MWt)	2660 MWt	102% of ESD rating (102% of 2774 MWt)	0, 30, 102% of rated (% of 2660 MWt)
b.	Time delay from event to Rx trip	2 sec	2 sec	2 sec	variable
c.	AFWS actuation signal/time delay for AFWS flow	lo-lo SG level 1 minute	NA	low-low SG level	Assumed immediately 0 sec (no delay)
d.	SG water level at time of reactor trip	(lo-lo SG level) 0% NR span	NA	(lo SG level + steam-feed mismatch) 2 @ 20% NR span 1 @ tube sheet	N/A
e.	Initial SG inventory	60,700 lbm/SG (at trip)	100,832 lbm/SG @ 516.8°F	93100 lbm/SG	consistent with power
	Rate of change before & after AFWS actuation	See FSAR Figure 14.1.31	N/A	turnaround ~ 2055 sec.	N/A
	decay heat	See FSAR Figure 14D-6		See FSAR Figure 14D-6	See FSAR Figure 14D-6
f.	AFW pump design pressure	1133 psia	1133 psia	1133 psia	N/A
g.	Minimum # of SGs which must receive AFW flow	2 of 3	N/A	2 of 3	N/A
h.	RC pump status	Tripped @ reactor trip	All operating	Tripped @ reactor trip	All operating
i.	Maximum AFW temperature	120°F	100°F	120°F	equal to main feed temperature
j.	Operator action	none	N/A	10 min.	10 min.
k.	MFW purge volume/temp.	100 ft^3/435°F	150 ft^3/ 436.2°F	385 ft^3/436.2°F	800 ft^3/loop (for dryout time)
l.	Normal blowdown	none assumed	none assumed	none assumed	none assumed
m.	Sensible heat	see cooldown	Table 2-2	see cooldown	N/A
n.	Time at standby/time to cooldown to RHR	2 hr/4 hr	2 hr/4 hr	2 hr/4 hr	N/A
o.	AFW flow rate	350 GPM - constant (min. requirement)	variable	350 gpm - constant (after 10 min.) (min. requirement)	1595 GPM (constant) to broken SG. (max. requirement)

TABLE 2-2

Summary of Sensible Heat Sources

Primary Water Sources (initially at rated power temperature and inventory)
- RCS fluid
- Pressurizer fluid (liquid and vapor)

Primary Metal Sources (initially at rated power temperature)
- Reactor coolant piping, pumps and reactor vessel
- Pressurizer
- Steam generator tube metal and tube sheet
- Steam generator metal below tube sheet
- Reactor vessel internals

Secondary Water Sources (initially at rated power temperature and inventory)
- Steam generator fluid (liquid and vapor)
- Main feedwater purge fluid between steam generator and AFWS piping.

Secondary Metal Sources (initially at rated power temperature)
- All steam generator metal above tube sheet, excluding tubes.

Question 3

Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

Response to 3

Figure 3-1 schematically shows the major features and components of the Auxiliary Feedwater System for the Beaver Valley Unit No. 1 Plant. Flow rates for all of the design transients described in Response 2 have been met by the system for the worst single failure. The flows for those single failures considered are tabulated for the various transients in Table 3-1, including the following:

A. A/C Train Failure
B. Turbine Driven Pump Failue
C. Motor Driven Pump Failure
D. AFWS check valve failure (failure to close on reverse flow).

Operator intervention within 10 minutes is required in order to meet the minimum flow requirements on the Feedline Rupture and the maximum flow requirements for the Main Steamline Break Inside Containment.

This evaluation demonstrates that Beaver Valley is in compliance with Long Term Recommendation No. 2 in that the AFWS maintains the capability to supply the required AFW flow to the steam generator(s) assuming a pipe break anywhere in the AFW pump discharge lines concurrent with a single active failure assuming that the operator takes action to isolate the break within 10 minutes.

TABLE 3-1

Auxiliary Feedwater Flow(1) to Steam Cenerators
Following an Accident/Transient with Selected Single Failure - GPM

	Accident/Transient	Single Failure: Elec. Train Failure	TD Pump Failure	MD Pump FAilure	CV(2) Failure
		A	B	C	D
1.	Loss of Main FW	1050	700	1050	1400
2.	Feedline Rupture	(3)	(3)	(3)	(3)
3.	Blackout	1050	700	1050	1400
4.	Cooldown	1050	700	1050	1400
5.	Main Steamline Rupture	(3)	(3)	(3)	(3)
6.	Main steamline	<1440	<1440	<1440	<1440

Notes:

(1) Items 1 thru 5 are minimum expected flows to intact loops; item 6 is maximum possible flow to the faulted loop.

(2) Including only those CVs in the AFWS. "Failure" is interpreted as failure to close on reverse flow; failure of the CV to open to permit flow in the normal direction is not considered.

(3) Ten minute operator action is required to isolate AFW flow to faulted loop. Prior to operator action, flow is 0 gpm to unfaulted loops; after operator action, flow is >350 gpm to unfaulted loops.

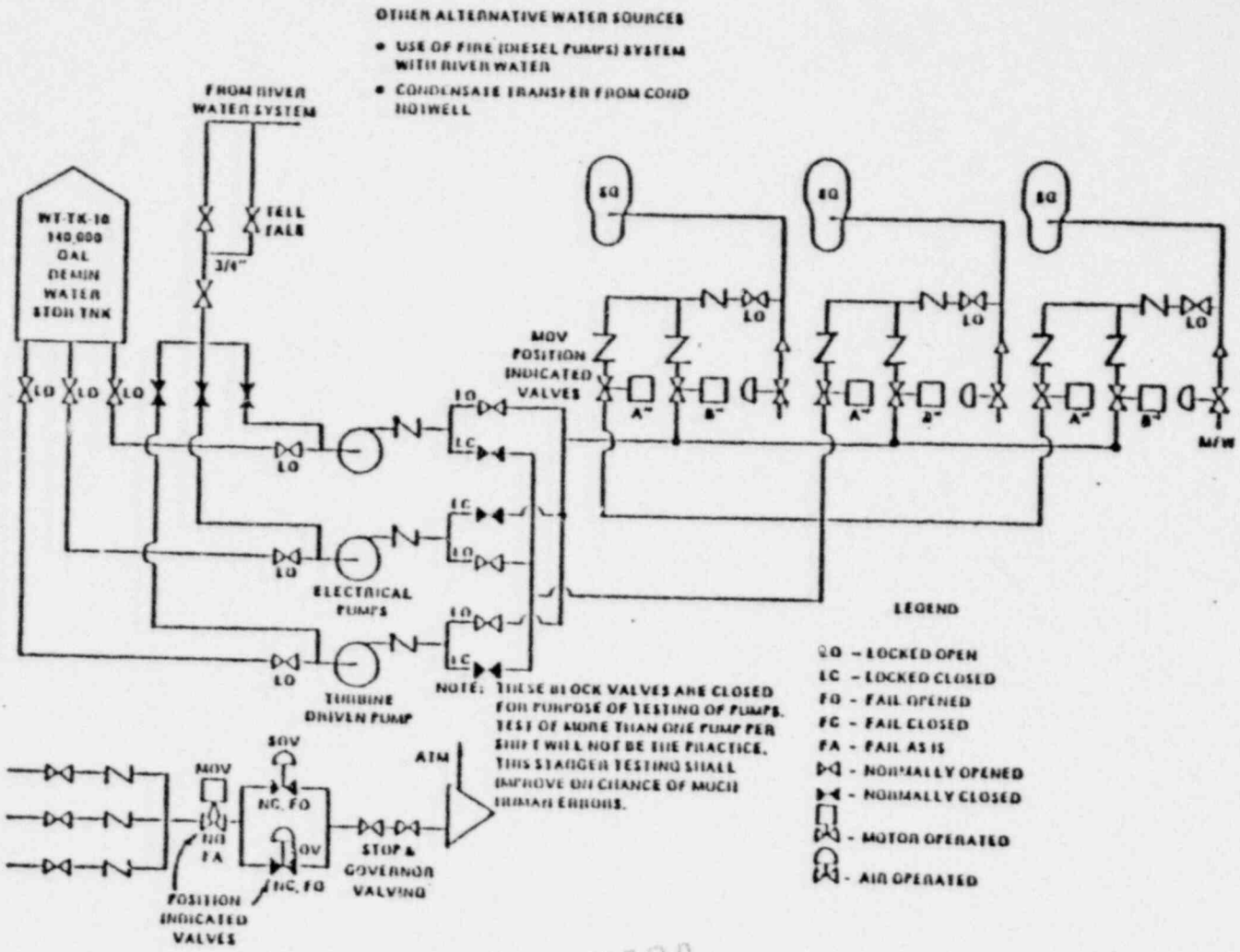

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Figure 3-1	Auxiliary Feedwater System	Beaver Valley Unit 1