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December 17, 1985
ST-HL-AE-1546
File No.: G9.17

Mr. Vincent S. Noonan, Project Director
PWR Project Directorate #5
U. S. Nuclear Regulatory Commission
Washington, DC 20555

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
Responses to DSER/FSAR Items Regarding
Auxiliary Feedwater System

Dear Mr. Noonan:

The attachments enclosed provide STP's response to Draft Safety Evaluation Report (DSER) or Final Safety Analysis Report (FSAR) items.

The item numbers listed below correspond to those assigned on STP's internal list of items for completion which includes open and confirmatory DSER items, STP FSAR open items and open NRC questions. This list was given to your Mr. N. Prasad Kadambi on October 8, 1985 by our Mr. M. E. Powell.

The attachments include mark-ups of FSAR pages which will be incorporated in a future FSAR amendment unless otherwise noted below.

The items which are attached to this letter are:

Attachment	Item No.*	Subject
1	Q410.018N-1	Comparison of Auxiliary Feedwater System (AFW) with March 10, 1980 NRC letter.
2	None	Response to TMI item II.E.1.1 regarding AFW.

*Legend
D - DSER Open Item C - DSER Confirmatory Item
F - FSAR Open Item Q - FSAR Question Response Item

B001 1/1

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PROD:

If you should have any questions on this matter, please contact Mr. M. E. Powell at (713) 993-1328.

Very truly yours,

M. R. Wisenburg
Manager, Nuclear Licensing

CAA/yd

Attachments: Responses to DSER/FSAR Items Regarding Auxiliary Feedwater System

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L1/NRC/cj

ATTACHMENT 2
ST-HL-AE- 1546
PAGE 1 OF 3

Question 410.18N

Provide a response to the staff's March 10, 1980 letter to near-term operating license applicants concerning your AFW system design (TMI-2 Task Action Plan, NUREG-0737, Item II.E.1.1). This response should include the following:

(a) A review of the AFW system design against Standard Review Plan Section 10.4.9, and Branch Technical Position ASB 10-1.

(b) A review of the AFW system design, Technical Specifications and operating procedures against the generic short-term and long-term requirements discussed in the March 10, 1980 letter.

(c) The design basis for the AFW flow requirements and verification that the AFW system will meet these requirements (refer to Enclosure 2 of the March 10, 1980 letter).

Response

(a) Tables Q410.18N-1 and Q410.18N-2 summarize the STP conformance to SRP 10.4.9 and BTP ASB 10-1.

(b) The draft STP Technical Specifications were submitted on June 17, 1985 (reference letter ST-HL-AE-1271 to Mr. Hugh L. Thompson from J. H. Goldberg). ~~A review against the~~ Technical Specifications ~~is necessary to complete the response. It is anticipated the response will be provided by the fourth quarter of the year.~~ for the AFWS have been prepared and will be provided by the end of the fourth quarter of 1985.

(c) ~~A response will be provided in the fourth quarter of 1985.~~ The response is provided in FSAR Section 7A, item II.E.1.1

ATTACHMENT 1
ST-HL-AE-1546
PAGE 2 OF 3

Table 410.18N-1

Standard Review Plan, Section 10.4.9

Item	Acceptance Criteria	Related to	STP Position	Reference FSAR Section
1.	II.2	GDC 2	Conforms	10.4.9.2 & 10.4.9.3
2.	II.2	GDC 4	Conforms	3.5, Table 3.5-1, 3.6 & 10.4.9.2
3.	II.3	GDC 5	Conforms	10.4.9.2(1)
4.	II.4	GDC 19	Conforms	7.4.1, 7.4.1.1 & 10.4.9
5.	II.5 (a)	GDC 34 & 44	Conforms	10.4.9.1
	II.5 (b)	GDC 34 & 44	Conforms	10.4.9.1, 10.4.9.2, 10.4.9.3 & Table 10.4-3
	II.5 (c)	GDC 34 & 44	Conforms	10.4.9.2 (paragraph 7 & 11), 6.2.4, 10.4.9.3 and Appendix 10A (later).
6.	II.6	GDC 45	Conforms	6.6
7.	II.7	GDC 46	Conforms	10.4.9.4, 14.2 & STP Tech Specs (later)

Note:

(1) Each unit has an entirely independent Auxiliary Feedwater System.

Table 410.18N-2

Branch Technical Position ASB 10-1

Item	ASB 10-1 Position	Related to	STP Position	Reference FSAR Sections
1.	B.1	Independency & Diversity	Conforms	10.4.9.2
2.	B.2	Diverse & Separate Motive Power	Conforms	10.4.9.2, 7.4.1.1, Table 10.1-1
3.	B.3	Train Separation & Crossconnect	Meets the intent.(1)	10.4.9.2, Fig. 10.4.9.-1, 10.4.9.1.4, 10.4.9.3
4.	B.4	Redundancy	Conforms	10.4.9.1.4, 10.4.9.3
5.	B.5	AFW Flow Following HELB	Conforms	10.4.9.1.4

Note

1. The STP AFW system with its four independent trains is designed to function (provide the required AFW flow) following a postulated piping failure with or without off site power available considering, at the same time, any single failure.

   Additionally the AFW trains are provided with a cross-connect for use during nonsafety-actuated AFW system operation. This allows one, two, or three operating pumps to feed all four SGs. In addition, the cross-connect valves are provided with manual actuators which would allow any operable AFW pump to be aligned with any effective SG during an extreme accident and failure combination.

II.E.1.1 AUXILIARY FEEDWATER SYSTEM EVALUATION

Position

The Office of Nuclear Reactor Regulation is requiring reevaluation of the auxiliary feedwater (AFW) systems for all PWR operating plant licensees and operating license applications. This action includes:

(1) Perform a simplified AFW system reliability analysis that uses event-tree and fault-tree logic techniques to determine the potential for AFW system failure under various loss-of-main-feedwater-transient conditions. Particular emphasis is given to determining potential failures that could result from human errors, common causes, single-point vulnerabilities, and test and maintenance outages;

(2) Perform a deterministic review of the AFW system using the acceptance criteria of Standard Review Plan Section 10.4.9 and associated Branch Technical Position ASB 10-1 as principal guidance; and

(3) Reevaluate the AFW system flowrate design bases and criteria.

Clarification

Operating Plant Licenses--Items 1 and 2 above have been completed for Westinghouse (W), Combustion Engineering (C-E), and two Babcock and Wilcox (B&W) operating plants (Rancho Seco, short-term only, and TMI-1). As a result of staff review of items 1 and 2, letters were issued to these plants that required the implementation of certain short- and long-term AFW system upgrade requirements. Included in these letters was a request for additional information regarding item 3 above. The staff is now in the process of evaluating licensees' responses and commitments to these letters.

The remaining B&W operating plants (Oconee 1-3, Crystal River 3, ANO-1, and Davis-Besse 1) have submitted the analysis described in item 1 above. The analysis is presently undergoing staff review. When the results of the staff reviews are complete, each of the remaining B&W plants will receive a letter specifying the short- and long-term AFW system upgrade requirements based on item 1 above. Included in these letters will be a request for additional information regarding items 2 and 3 above.

Operating License Applicants--Operating license applicants have been requested to respond to staff letters of March 10, 1980 (W and C-E) and April 24, 1980 (B&W). These responses will be reviewed during the normal review process for these applications.

STP Response

The following information responds to the NRC letter of March 10, 1980, enclosure 2, relating to the Auxiliary Feedwater System Design Bases.

Question 1

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

1) Loss of Main Feedwater (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 feedline break
8) Main steamline 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 proceeed with an orderly cooldown of the plant to the reactor coolant conditions 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 South Texas Units 1 & 2.

- Loss of Main Feedwater Transient
  - Loss of main feedwater with offsite power available
  - Loss of Offsite Power - LOOP (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 LOOP with the consequential shutdown of the system pumps, auxiliaries, and controls

These transients are discussed in Sections 15.2.6 and 15.2.7.

Loss of main feedwater transients are characterized by a 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 a high initial power level, 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/or 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 overpressurization 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 LOOP 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 LOOP, except that reactor coolant pump heat input is not a consideration in the LOOP transient following loss of power to the reactor coolant pump bus.

<u>Secondary System Pipe Ruptures</u>

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 to the faulted steam generator. With a "typical" headered AFS arrangement, such situations can result in the injection of a disproportionately large fraction of the total auxiliary feedwater flow (the system preferentially pumps water to the lowest pressure region) to the faulted loop rather than to the effective steam generators which are at relatively high pressure. However, the South Texas units have four auxiliary feedwater pumps, with associated independent piping trains. Each auxiliary feedwater train delivers flow to a different steam generator. This arrangement allows the flow from only one auxiliary feedwater pump to spill through a break and ensures that sufficient flow will be delivered to the remaining effective steam generators. 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 an excessive release of mass and energy to containment. Thus, steamline rupture conditions establish the upper limt 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 nonfaulted loops, 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 limit, control, or terminate 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

Although the AFS must be designed to cope with a complete loss of ac power, i.e., the loss of both offsite and onsite ac power sources, this event is not considered to be a design basis event for overall plant design by current industry standards and government regulations.

The South Texas AFS provided three motor-driven pumps and one turbine-driven pump. Each pump is capable of delivering a minimum of 550 gpm at a pressure equivalent to the accumulation pressure of the lowest setpoint of the steam generator safety valves. The AFS is designed with diversity in pump motive power sources and essential instrumentation and control power sources. The AFS is capable of delivering the required flow of 550 gpm to at least one steam generator, assuming the loss of both onsite and offsite ac power.

Loss-of-Coolant Accident (LOCA)

The loss of coolant accidents discussed in Section 15.6.5 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 LOCAs 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 safe 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 RCS 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. The Auxiliary Feedwater System is provided with a seismic Category I Auxiliary Feedwater Storage tank which is sized with sufficient capacity for 4 hours of standby, followed by a 10 hour natural circulation cooldown, with an additonal 8 hour soak period.

Response to 1.b

Table II.E.1.1-1 summarizes the criteria which are the general design bases for each event, discussed in the response to Question 1.a. Specific assumptions used in the analyses to verify that the design bases are met are discussed in response to Question 2. (See also the response to NRC Question 410.18N.)

The primary function of the Auxiliary Feedwater System is to provide sufficient heat removal capability following reactor trip and 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 evaluated 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.

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 errors 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 actuation of AFWS flow.

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 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 removal 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 in the FSAR. Specifically, they include:

- Loss of Main Feedwater/Loss of Offsite Power (LOOP)
- 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 South Texas Units 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 LOOP , assuming an available auxiliary pump having a diverse (non-ac) power supply. The LOCA analysis, as discussed in response to Question 1.b, incorporates the system flow 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/Loss o, Offsite Power (LOOP)

The Loss of Main Feedwater/ LOOP events were analyzed for the South Texas Units and are presented in FSAR Section 15.2.7. The difference between the two events is that, for the LOOP case, power to the Reactor Coolant Pumps is assumed tc be lost following reactor trip. The acceptance criteria for these ANS Condition II events, as listed in Table II.E.1.1-1, are all met. The following assumptions, concerning the AFWS, have been made. Sixty seconds following generation of the low-low steam generator water level signal, auxiliary feedwater is initiated. One auxiliary feedwater pump is assumed to provide auxiliary feedwater to one steam generator at a rate of 515 gpm. It takes approximately 78 seconds to eliminate the 90 cubic foot purge volume before the relatively cold auxiliary feedwater (160°F) reaches the steam generator. Table II.E.1.1-2 summarizes the assumptions used in these analyses. In addition, FSAR Section 15.2.7 provides more detail concerning the Loss of Main Feedwater/ LOOP analysis.

Rupture of a Main Feedwater Pipe

The Main Feedwater Pipe Rupture event was analyzed for the South Texas Units and is presented in FSAR Section 15.2.8. Cases were analyzed both with and without offsite power available. The acceptance criteria for this ANS Condition IV event, as listed in Table II.E.1.1-1, are all met. The following assumptions, concerning the AFWS, have been made. Sixty seconds following generation of the low-low steam generator water level signal, auxiliary feedwater is initiated. One auxiliary feedwater pump is assumed to provide auxiliary feedwater to one nonfaulted steam generator at a rate of 540 gpm. It takes approximately 83 seconds to eliminate the 100 cubic foot purge volume before the relatively cold auxiliary feedwater (160°F) reaches the steam generator. Table II.E.1.1-2 summarizes the assumptions used in this analysis. In addition, FSAR Section 15.2.8 provides more detail concerning the Main Feedwater Pipe Rupture analysis.

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 four 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. Table II.E.1.1-2 summarizes the assumptions used in this analysis. At 30 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 II.E.1.1-1 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 prevously discussed transients meet the flow requirements of plant cooldown. This operation, however, defines the basis for tank size, based on the required cooldown duration, maximum decay heat input and maximum stored heat in the system. As previously discussed in the response to Question 1.a, the Auxiliary Feedwater System (AFWS) partially cools the system to the point where the RHRS may complete the cooldown, i.e., 350°F in the RCS. Table II.E.1.1-2 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 II.E.1.1-3 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.

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

The South Texas Auxiliary Feedwater System flow design capabilities, considering various single failures, are documented in the Failure Mode Analysis for the AFW System provided in Table 10.4-3 of the South Texas FSAR.

The South Texas AFW pump sizing is based on delivering the required flow at the lowest steam generator safety relief valve set pressure plus accumulation (1339 psia). The required flow does not include a continuous recirculation flow because of the system use of Automatic Recirculation Control (ARC) valves which provide 100% forward flow when the flowrate is above the pump minimum flow requirements. Likewise, the seal leakage is not considered in the pump design flow since the pumps are provided with mechanical seals. The AFW pump design wear margin is based on head rather than flow, and when converted to flow this wear margin is approximately 4%.

TABLE II.E.1.1-1

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 +10% No consequential fuel failures	Pressurizer does not fill
Loss of Offsite Power	Condition II	(same as LMFW)	Pressurizer does not fill
Feedline Rupture	Condition IV	10CFR100 dose limits Containment design pressure not exceeded	Core does not uncover
Loss of all A/C Power	N/A	Note 1 Containment design pressure not exceeded	Same as blackout assuming turbine driven pump
Loss of Coolant	Condition III	10 CFR 100 dose limits 10 CFR 100 PCT limits	
	Condition IV	10 CFR 100 dose limits 10 CFR 100 PCT limits	
Cooldown	N/A		100°F/hr 567°F to 350°F

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

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

ATTACHMENT 2
ST-HL-AE-1546
PAGE 16 OF 20

9318Q:10/121085

TABLE II.E.1.1-2

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 NSSS power	102% of nominal rating (102% of 3817 MWt)	4100 MWt	102% of nominal rating (102% of 3817 MWt)	0, 30, 70, 100% (percent of 3817 MWt)
b.	Time delay from event to Rx trip	23 sec	2 sec	17.3 sec	variable
c.	AFWS actuation signal/time delay for AFWS flow	low-low SG level/ 1 minute	N/A	Low-low SG level/ 1 minute	Assumed immediately @ 0 sec (no delay)
d.	SG water level at time of reactor trip	(low-low SG level) 27.9% NR span 91,344 lbm	N/A	(low-low SG level) 27.9% NR span 86,625 lbm	N/A
e.	Initial SG inventory	118,829 lbm/SG	98,100 lbm/SG at 556.3°F	Broken Loop - 128,874 lbm Intact Loop - 118,829 lbm	consistent with power
	Rate of change before & after AFWS actuation	See FSAR Figure 15.2-10	N/A	See Figure II.E.1.1-1	N/A
	Decay heat	ANS-5.1-1979 + 2σ	N/A	ANS-5.1-1979 + 2σ	ANS + 20%
f.	AFW pump design pressure	1339 psia	1339 psia	1339 psia	N/A
g.	Minimum # of SGs which must receive AFW flow	1 of 4	N/A	1 of 4	N/A

ATTACHMENT 2
ST.HL.AE. 1546
PAGE 17 OF 20

TABLE II.E.1.1-2 (Continued)

SUMMARY OF ASSUMPTIONS USED IN AFWS DESIGN VERIFICATION ANALYSES

Transient	Loss of Feedwater/ Station Blackout	Cooldown	Main Feedline Break	Main Steamline Break (Containment)
h. RC pump status	*Tripped at reactor trip (2 second delay)	Tripped	*Tripped @ reactor trip (2 second delay)	All operating
i. Maximum AFW temperature	160°F**	120°F	160°F**	Same temperature as main feedwater at initial operating power
j. Operator action	None	N/A	None	Aux. feed flow terminated after 30 minutes
k. MFW purge volume/ S/G and temperature	90 ft^3/440°F	0 ft^3/440°F	100 ft^3/440°F	450 ft^3/loop (for dryout time)
l. Normal blowdown	none assumed	none assumed	none assumed	none assumed
m. Sensible heat	see cooldown	Table II.E.1.1-3	see cooldown	N/A
n. Time at standby/time to cooldown to RHR	2 hr/10 hrs*	2 hr/5 hrs	2 hr/5 hrs	N/A
o. AFW flowrate	515 gpm*** constant	variable	540 gpm constant	1210 gpm (constant) to broken SG

* with offsite power not available
** 120°F is maximum temperature
*** system design is 550 gpm per train

ATTACHMENT 2
ST-HL-AE-1546
PAGE 18 OF 20

TABLE II.E.1.1-3

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

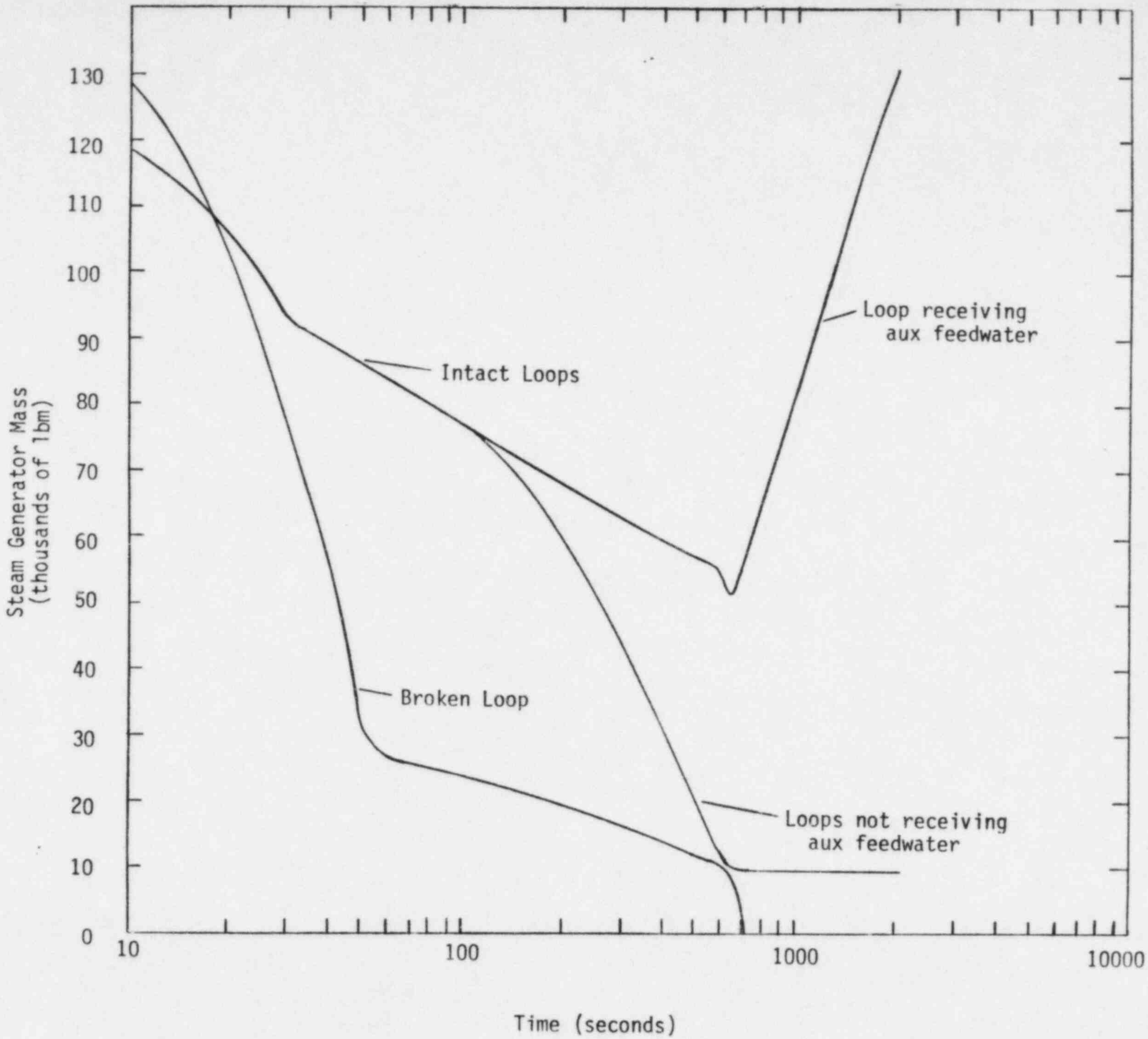

Figure II.E.1.1-1 Feedline Break

ATTACHMENT 2
ST-HL-AE-1546
PAGE 20 OF 20