VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

July 10, 1980

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attention: Mr. B. Joe Youngblood, Chief
Licensing Branch 1
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Serial No.: 592
NO/LEN/jmj
Docket No. 50-339
License No. NPF-7

Dear Mr. Denton:

NORTH ANNA UNIT 2
AUXILIARY FEEDWATER SYSTEM REQUIREMENTS

We have reviewed your letter of March 10, 1980 that requests information concerning auxiliary feedwater system design basis information and pump flow verification. Our response to these items is provided in Attachment 1.

We believe that the attached response adequately addresses the NRC concerns for North Anna Unit 2 and that a commitment to provide additional information at a later date is not necessary. Our response for North Anna Unit 1 will follow shortly. Should you have any questions or require additional information, please contact us.

Very truly yours,

B. R. Sylvia
Manager - Nuclear
Operations and Maintenance

Attachment

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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 North Anna Units 1 and 2 plants.

- 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 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 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.

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. 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 and 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 350oF. 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	Pressurizer does not become water solid.
Station Blackout	Condition II	(same as LMFW)	Pressurizer does not become water solid.
Steamline Rupture	Condition IV	10 CFR 100 dose limits. Containment design pressure not exceeded	
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.

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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 North Anna Units 1 and 2 plants 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 15.2.8 for the purpose of showing that for a station blackout transient, the peak RCS pressure remains below the criterion for Condition II transients and no fuel failures occur (refer to Table 18-1). 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.

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 15.4.2.2 (see also S15.18). 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 at the low-low level trip setpoint. 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. Auxiliary feedwater flow of 340 gpm was assumed to be delivered to one non-faulted steam generator one minute after reactor trip. Thirty minutes after the accident, the auxiliary feedwater rate is increased to 680 gpm divided between two steam generators, as a result of the realignment of valves by the operator. The criteria listed in Table 18-1 are met.

This analysis 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 one minute 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 main feedwater isolation, 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 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 18-1 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.

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 2910 MWt)	2963 MWt	102% of ESD rating (102% of 2910 MWt)	0% (of rated) - worst case (% of 2775 MWt)
b.	Time delay from event to Rx trip	2 sec (delay after trip)	2 sec	2 sec	0 seconds
c.	AFWS actuation signal/ time delay for AFWS flow	lo-lo SG level 1 minute	NA	low-low SG level 1 minute	Feedwater isolation actuation signal: Safety Injection Signal 0 sec (no delay); Flow to SG: 17 sec after MSLB (which is time of FW isolation)
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	Same as initial level before event
e.	Initial SG inventory	74,600 lbm/SG (at trip)	104,500 lbm/SG @ 525.2°F	94,100 lbm/SG	151,000 lbm/SG
	Rate of change before & after AFWS actuation	See FSAR Figure 15.2.31	N/A	turnaround greater than 2000 sec.	FSAR Table 6.2-11
	decay heat	See FSAR Figure 15.1-6		See FSAR Figure 15.1-6	See FSAR Figure 6.2-1
f.	AFW pump design pressure	1133 psia	1133 psia	1133 psig	1025 psig
g.	Minimum # of SGs which must receive AFW flow	2 of 3	N/A	1 of 3 (1 min after trip) 2 of 3 (operator action after 30 minutes)	N/A
h.	RC pump status	Tripped @ reactor trip	Tripped	All operating	Tripped
i.	Maximum AFW Temperature	120°F	100°F	120°F	120°F
j.	Operator action	none	N/A	10 min.	30 min.
k.	MFW purge volume/temp.	159.6 ft^3/437.6°F	100 ft^3/ 434.3°F	214 ft^3/433.3°F	218 ft^3/441°F
l.	Normal blowdown	none assumed	none assumed	none assumed	none assumed
m.	Sensible heat	see cooldown	Table 2-2	see cooldown	see cooldown
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	680 GPM - constant	variable	340 gpm - 1 min after trip - 680 gpm (after 30 minutes)	900 gpm to broken SG 350 gpm to each intact S/G (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

Based upon review of the flow requirements of Table 2-1, the main feedline break was selected as the most conservative accident in evaluating the AFW pump flow rates based on flow at pressure.

The parameters used are as follows:

1.	Steam Generator Pressure	1133 psia
2.	AFW Temperature	120°F
3.	Flow Rate to any Steam Generator (min)	340 gpm

The motor driven AFW pumps were originally specified to deliver 350 gpm at 1216 psi. The vendor design point included recirculation flow and became 370 gpm at 1214 psi. The turbine driven pump was specified to deliver 700 gpm at 1215 psi. The vendor design point included recirculation flow and became 735 gpm at 1214 psi. Certified pump performance curves from Intersoll-Rand verify that these design points were met.

Recent system pressure drop calculations using 350 gpm as a design flow rate have been compared with the vendor certified pump test curves. The results indicate that a pump head margin exists at the flow required by W (340 gpm), for all three North Anna Unit 2 auxiliary feed pumps.

The margin of pump head or flow assigned for seal leakage and pump wear is essentially zero. Seal leakage for a properly installed and maintained mechanical seal is in the range of 1 to 10 cc/hr. Zero margin for pump wear is justified by the limited pump run time and the quality of the pumped fluid. Further, Technical Specifications 3.7.1.2 and 4.0.5 require frequent operational testing of these pumps in accordance with the ASME XI code. The testing program and data offer a complete pump history which will accurately predict abnormal wear, and identify maintenance requirements due to wear.

The head margin available for recirculation flow and lube-oil cooler flow is listed for each pump on Attachment A. The vendor's suggested flow to each lube-oil cooler is 25 gpm which comes from the first stage of each pump with the resultant loss of flow at the discharge being negligible. Recommended shut off recirculation flow for the turbine pump is 35 gpm, and 20 gpm for the motor driven pumps.

As can be seen on Attachment A and from the information above, there is adequate flow to the steam generators from the auxiliary feedwater pumps for North Anna Unit 2.

ATTACHMENT A

AUXILIARY FEEDWATER SYSTEM DESIGN BASIS
NORTH ANNA UNIT 2

Pump Mark No.	Calculated Head Req'd for Flow of 340 gpm	Pump Head at 340 gpm (certified curve)	Head Margin
2-FW-P2 (Turbine)	2949 ft	3116 ft @ 4100 rpm	167 ft
2-FW-P3A	2774 ft	2900 ft	126 ft
2-FW-P3B	2845 ft	2900 ft	55 ft