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DENTON,H.R. Office of Nuclear Reactor Regulation, Director

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SUBJECT: Forwards responses to questions contained in Encl 2 to DG. Eisenhut 791030 ltr re flow design basis of auxiliary feedwater sys.

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Donald C. Cook Nuclear Plant Unit Nos. 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
Auxiliary Feedwater System Flow Design Basis

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Denton:

The attachment to this letter contains our responses to the questions contained in Enclosure 2 to Mr. Eisenhut's October 30, 1979 letter concerning the flow design basis of the Auxiliary Feedwater System.

Very truly yours,

R. S. Hunter
Vice President

cc: R. C. Callen
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ATTACHMENT TO AEP:NRC:0300C

Responses are for D. C. Cook Units 1 & 2, except as noted, using the same format as enclosure 2 to Mr. Eisenhut's October 30, 1979 letter.

Response to Question 1:

The Auxiliary Feedwater System (AFS) serves as an emergency backup system for supplying feedwater to the secondary side of the steam generators at times when the main feedwater system is not available, thereby maintaining an adequate heat sink. As an Engineered Safeguards System, the AFS is one of the mitigating systems which would prevent possible 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 and decay heat removal 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 system or to the atmosphere through the steam generator safety valves or the power-operated relief valves. Steam generator water inventory is 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 AFS which delivers an alternate water supply to the steam generators after the main feedwater pumps are tripped. The AFS is capable of functioning for extended periods, thereby allowing time either to restore normal feedwater flow or to proceed with an orderly primary system cooldown to conditions where the Residual Heat Removal System (RHR) can be employed for decay heat removal. The AFS flow and water supply capacity are sufficient to remove core decay heat, reactor coolant pump heat, and sensible heat during the plant cooldown.

1.a The plant transient and accident conditions which impose safety-related performance requirements on the design of the AFS including AFS flow requirements are as follows:

A. Loss of Main Feedwater Transient

i) Loss of main feedwater with offsite power available
ii) Station blackout (i.e., loss of main feedwater without offsite power available)

B. Secondary System Ruptures

i) Main Steamline rupture
ii) Main Feedline rupture (Unit 2 FSAR analysis only)

C. Loss of all AC Power

D. Loss of Coolant Accident (LOCA)

E. Cooldown

A. 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 main feedwater or condensate system.

- Loss of offsite power to the station 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, feedwater isolation, and auxiliary feedwater actuation by the Reactor Protection System logic. Following reactor trip from full power, the power quickly falls to decay heat levels. The steam generator 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, continued unarrested temperature rise could potentially cause challenges to the RCS overpressurization protection system (relief/safety valves). Hence, the timely introduction of sufficient auxiliary feedwater mitigates the decrease in the steam generator water levels, reverses the rise in reactor coolant temperature, prevents the pressurizer from filling to a water solid condition, and establishes stable hot standby conditions.

The loss of offsite AC power transient differs from a simple loss of main feedwater in that emergency power sources are 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, for this case steam formed by the 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.

B. Secondary System Ruptures

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 and flow to the faulted loop will contribute to an excessive 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 loops, but at somewhat lower rates than for the loss of feedwater transients described previously. Provisions are made in the design of the AFS to limit, control, and/or terminate the auxiliary feedwater flow to the faulted loop as necessary in order to limit containment pressure rise following a steamline break inside containment and to ensure the flow to the remaining unfaulted loops is sufficient to terminate the RCS heatup.

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 is connected to the main feedwater line. Such situations can result in the loss of a fraction of the total auxiliary feedwater flow because the system preferentially pumps water to the lowest pressure region in the faulted loop. The AFS design allows for terminating, limiting, and 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 design flow requirements are similar for the main feedwater line rupture as those explained for the loss of main feedwater transients.

C. Loss of All AC Power

The loss of all AC power is postulated as resulting from accident conditions wherein not only offsite AC power is lost but also onsite AC emergency power is lost. DC power supplied by the redundant onsite safety related station batteries for operation of protection circuits is assumed available. For the Cook Plant, the Turbine Driven Auxiliary Feedwater Pump Train operates independent of onsite and offsite AC power. This AFS train is DC powered by its own dedicated safety related battery. The impact of a total loss of all AC power on the AFS is accounted for by

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.

D. 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 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.

E. Cooldown

The cooldown function performed by the AFS is a partial one since the reactor coolant system is reduced from normal no load temperatures to a hot leg temperature of approximately 350°F. The latter is the maximum temperature recommended for placing the RHR system into service. The RHR system completes the cooldown to cold shutdown conditions.

Cooldown may be required following expected transients, following an accident, or to accomplish a normal cooldown prior to refueling or performing reactor plant maintenance. If the reactor is tripped following extended operation at rated power level, the AFS is capable of delivering sufficient AFW to remove decay heat and reactor coolant pump (RCP) heat following reactor trip while maintaining adequate steam generator (SG) water inventory. 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 AFS, 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.

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 AFS 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 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 for the AFS.

TABLE 1B-1

Criteria for Auxiliary Feedwater System Design Basis Conditions

Condition or Transient	Classification*	Criteria*	Additional Design Criteria
Loss of Main Feedwater (LMFW)	Condition II	Peak RCS pressure not to exceed design pressure. No consequential fuel failures.	
LMFW with loss of offsite	Condition II	(same as LMFW).	Pressurizer does not fill water solid with a single motor driven auxiliary feed pump feeding 2 SGs.
Steamline Rupture	Condition IV	10 CFR 100 dose limits. Containment design pressure not exceeded.	
Feedline Rupture (Unit-2 FSAR Analysis)	Condition IV	RCS design pressure not exceeded. 10 CFR 100 dose limits.	
Loss of all AC Power	NA	Note 1	Same as blackout assuming turbine driven pump operates
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	NA		$100^{o}F/hr$ $547^{o}F$ to RHR

*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. With the turbine driven pump train operable from its' own DC power source, this transient is bounded by the LMFW.

Response to Question 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 Donald C. Cook Nuclear Plant FSAR. Specifically, they include:

- Loss of Main Feedwater/Station Blackout (similar for flow design basis)

- Rupture of a Main Feedwater Line (Unit 2 FSAR analysis)

- Rupture of a Main Steam Line Inside Containment

The Loss of All AC Power is discussed via a comparison to the transient results of a Station Blackout, assuming an available auxiliary feedwater pump having an independent DC power supply. The LOCA analysis, as discussed in the Response 1.b above, 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. In addition to the above analyses, calculations have been performed specifically to determine the requirements for plant cooldown flow for the purpose of determining the storage capacity of the condensate storage tank.

Loss of Main Feedwater/Station Blackout (similar for flow design basis)

A loss of feedwater, assuming a loss of power to the reactor coolant pumps, was performed in FSAR Section 14.1.9 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 and Table 2.2 summarize the assumptions used in this analysis for Cook Unit 1 and Unit 2, respectively. 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 at most 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. In Unit 1, the reactor is assumed to be tripped on steam flow/ feed flow mismatch coincident with low steam generator water level, allowing for level uncertainty. In Unit 2, the reactor is assumed to be tripped on low-low steam generator level, allowing for level uncertainty. The FSAR

analysis 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 results in even more margin with respect to filling the pressurizer.

Rupture of Main Feedwater Pipe (Unit 2 FSAR Analysis)

The double ended rupture of a main feedwater pipe downstream of the main feedwater line check valve is analyzed for Cook Unit 2. Table 2.2 includes the assumptions used in this analysis. The assumption is made that reactor trip occurs when the steam generators are at the low level setpoint, adjusted for errors, coincident with steam flow/feed flow mismatch and the faulted loop is assumed to be empty. This conservative assumption maximizes the primary system heat input 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. The initial power rating was assumed to be 102% of rated power. The auxiliary feedwater is injected into the unfaulted steam generators one (1) minute after the reactor trip setpoint is reached. The criteria listed in Table 1B-1 are met.

This analysis establishes the capacity of single pumps, requirements for layout to preclude indefinite loss of auxiliary feedwater to the postulated break, and train association requirements for equipment so that the AFWS can deliver the minimum flow required assuming the worst single failure. This applies to both Units 1 and 2 which have identical AFS.

Rupture of a Main Steam Pipe Inside Containment

Because the steam line break transient is initially a cooldown, the AFS is not needed to remove heat in the short term. Furthermore, addition of excessive auxiliary feedwater to the faulted steam generator could affect the peak containment pressure following a steam line break inside containment. This transient in Unit 2 is performed at four power levels for several break sizes. Auxiliary feedwater is assumed to be initiated at the time of the break. Maximum flow is used for this analysis, considering failure of the runout protection for the largest pump. Tables 2-1 and 2-2 include the assumptions used in this analysis. At 10 minutes after the break, it is assumed that the operator has isolated the AFS 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, the basis for runout protection, and the 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 AFS partially cools the system to the point where the RHR system may complete the cooldown. Tables 2-1 and 2-2 show the assumptions used to determine the cooldown heat capacity of the auxiliary feedwater system. These assumptions were used in the cooldown calculations for D. C. Cook Unit 1. They are also applicable to Unit 2 since the two units have an identical AFS and condensate storage tanks.

The cooldown is assumed to commence at rated power, and maximum trip delays and decay heat source terms are assumed when the reactor is tripped during the cooldown process. Primary metal, primary water, secondary system metal and secondary system water are all included in the stored heat to be removed by the AFS. See Table 2-3 for the items constituting the sensible heat stored in the NSSS.

TABLE 2-1

Summary of Assumptions Used in AFWS Design Verification Analyses
For D. C. Cook Unit 1

Transient	Loss of Feedwater (Station Blackout)	Cooldown	Main Steamline Break (Containment)
a. Max Reactor Power	102% of ESD Rating (102% of 3391 MWt)	3250 MWt	102% of ESD Rating (102% of 3391 MWt)
b. Time Delay From Reactor Trip Signal To Rod Motion	2 Seconds	2 Seconds	NA
c. AFWS Actuation Signal/ Time Delay For AFWS Flow	Low-Low SG Level/ 1 Minute	NA	AFWS Initiated At 10 Seconds.
d. SG Water Level At Time of Reactor Trip	0% NR Span (Low SG Level + Steam-feed Mismatch)	NA	NA
e. Initial SG Inventory	63,000 lbm/SG (At Trip)	87,935 lbm/SG @ 512.1°F	117,500 lbm/SG
Rate of Change Before & After Actuation	See FSAR Figure 14.1.9-1	NA	NA
Decay Heat	ANS + 20%	NA	ANS + 20%
f. AFW Pump Design	1187 psia	1187 psia	NA
g. Minimum No. of SGs Which Must Receive AFW Flow	2 of 4 (4 of 4 Receive Water)	NA	NA
h. RC Pump Status	Tripped @ Reactor Trip	Tripped	All Operating
i. Maximum AFW Temperature	100°F	80°F	100°F
j. Operator Action	None	NA	10 Min.
k. MFW Purge Volume/ Temperature	200 ft^3 per loop/ 431°F (bounding volume)	None Assumed	1250 ft^3 per loop/ 435°F (bounding volume)

TABLE 2-1
(CONTINUED)

Transient	Loss of Feedwater (Station Blackout)	Cooldown	Main Steamline Break (Containment)
l. Normal Blowdown	None Assumed	None Assumed	None Assumed
m. Sensible Heat	See Cooldown	Table 2-3	NA
n. Time At Standby/ Time To Cooldown To RHR	2 hr/4 hr	2 hr/4 hr	NA

TABLE 2-2

Summary of Assumptions Used in AFWS Design Verification Analyses for D. C. Cook Unit 2

Transient	Loss of Feedwater (Station Blackout)	Cooldown*	Main Feedline Break	Main Steamline Break (Containment)
a. Maximum Reactor Power	102% of ESD Rating (102% of 3556.1 MWt)	3250 MWt	102% of Rated Power (102% of 3403 MWt)	0, 30, 70, 102% of Rated (% of 3403 MWt).
b. Time Delay from Rx Trip Signal to Rod Motion	2 Seconds	2 Seconds	2 Seconds (20.1 Sec. From Event to Rx Trip)	2 Seconds (Time from Event to Rx Trip is Variable)
c. AFWS Actuation Signal/Time Delay for AFWS Flow	Low-Low SG Level/ 1 Minute	NA	Low-Low SG Level/ 1 Minute	Assumed Immediately 0 Sec. (No Delay)
d. SG Water Level at Time of Reactor Trip	0% NR Span (Low-Low SG Level)	NA	3 @ 20% NR Span 1 @ Tube Sheet (Low SG Level + Steamfeed Mismatch)	NA
e. Initial SG Inventory	56,442 lbm/SG (At Trip)	87,925 lbm/SG @ 512.1°F	97,914 lbm/SG	Depending on Power Level
[illegible] Rate of Change Before & After AFWS Actuation	See FSAR Figure 14.1.9-1	NA	Turnaround 1700 Sec.	NA
Decay Heat	See FSAR Figure 14.1-6 ANS + 20%	NA	See FSAR Figure 14.1-6	See FSAR Figure 14.1-6
f. AFW Pump Design	1187 psia	1187 psia	1187 psia	NA
g. Minimum No. of SGs Which Must Receive AFW Flow	2 of 4	NA	2 of 4 (3 of 4 Receive Water)	NA
h. RC Pump Status	Tripped @ Reactor Trip	Tripped	Tripped @ Reactor Trip	All Operating
i. Maximum AFW Temperature	$120^{\circ}F$	$80^{\circ}F$	$120^{\circ}F$	Equal to Main Feedwater Temperature
j. Operator Action	None	NA	None	10 Min.

TABLE 2-2
(CONTINUED)

Transient	Loss of Feedwater (Station Blackout)	Cooldown*	Main Feedline Break	Main Steamline Break (Containment)
k. MFW Purge Volume/Temperature	79.4 ft^3 per Loop/ 431°F	None Assumed	78.6 ft^3 per Loop/ 431°F	800 ft^3/Loop (For Dryout Time) (Bounding Volume)
l. Normal Blowdown	None Assumed	None Assumed	None Assumed	None Assumed
m. Sensible Heat	See Cooldown	Table 2-3	See Cooldown	NA
n. Time At Standby/Time To Cooldown to RHR	2 hr/4 hr	2 hr/4 hr	2 hr/4 hr	NA

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* Cooldown analyzed for Unit 1 and repeated here for information.