ATTACHMENT 2 TO AEP:NRC:1120

PROPOSED REVISED TECHNICAL SPECIFICATION PAGES

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 27.0*/13.0#
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0
d. Containment Isolation-Phase "A"	$\leq$ 18.0#
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	$\leq$ 48.0*/13.0#
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	$\leq$ 13.0#/23.0##
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0
d. Containment Isolation-Phase "A"	$\leq$ 18.0#/28.0##
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	$\leq$ 13.0#/48.0##
5. Steam Flow in Two Steam Lines - High Coincident with T_{avg} --Low-Low	
a. Safety Injection (ECCS)	$\leq$ 15.0#/25.0##
b. Reactor Trip (from SI)	$\leq$ 5.0
c. Feedwater Isolation	$\leq$ 10.0
d. Containment Isolation-Phase "A"	$\leq$ 20.0#/30.0##
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	$\leq$ 15.0#/50.0##
h. Steam Line Isolation	$\leq$ 13.0

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Flow in Two Steam Lines-High Coincident With Steam Line Pressure-Low	
a. Safety Injection (ECCS)	≤ 13.0#/23.0##
b. Reactor Trip (from SI)	≤ 3.0
c. Feedwater Isolation	≤ 8.0
d. Containment Isolation-Phase "A"	≤ 18.0#/28.0##
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	≤ 14.0#/48.0##
h. Steam Line Isolation	≤ 11.0
7. Containment Pressure--High-High	
a. Containment Spray	≤ 45.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	≤ 10.0
d. Containment Air Recirculation Fan	≤ 660.0
8. Steam Generator Water Level--High-High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	≤ 11.0
9. Steam Generator Water Level--Low-Low	
a. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
b. Turbine Driven Auxiliary Feedwater Pumps	≤ 60.0
10. 4160 volt Emergency Bus Loss of Voltage	
a. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
11. Loss of Main Feedwater Pumps	
a. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
12. Reactor Coolant Pump Bus Undervoltage	
a. Turbine Driven Auxiliary Feedwater Pumps	≤ 60.0

PLANT SYSTEMS

STEAM GENERATOR STOP VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each steam generator stop valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODE 1 - With one steam generator stop valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, reduce power to less than or equal to 5 percent of RATED THERMAL POWER within the next 2 hours.

MODES 2 and 3 - With one steam generator stop valve inoperable, subsequent operation in MODES 2 or 3 may proceed provided:

a. The stop valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5.1 Each steam generator stop valve that is open shall be demonstrated OPERABLE by:

a. Part-stroke exercising the valve at least once per 92 days, and

b. Verifying full closure within 8 seconds on any closure actuation signal while in HOT STANDBY with T_{avg} greater than or equal to 541°F during each reactor shutdown except that verification of full closure within 8 seconds need not be determined more often than once per 92 days.

4.7.1.5.2 The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

4.7.1.5.3 The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 when performing PHYSICS TESTS at the beginning of a cycle provided the steam generator stop valves are maintained closed.

ATTACHMENT 3 TO AEP:NRC:1120

ANALYSIS OF MAIN STEAM LINE BREAK
INSIDE CONTAINMENT (WCAP-11902 SUPPLEMENT 1)

WCAP-11902
Supplement 1

RERATED POWER AND REVISED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNITS 1 & 2 LICENSING REPORT

September 1989

WESTINGHOUSE ELECTRIC CORPORATION
Energy Systems Business Unit
P.O. BOX 355
Pittsburgh, Pennsylvania 15230

TABLE S-2.1-1

COOK NUCLEAR PLANT UNITS 1 AND 2
DESIGN POWER CAPABILITY PARAMETERS FOR RERATING PROGRAM

Parameter	(Unit 1, Original) Case 1	(Unit 2, Current) Case 2
NSSS Power, MWt	3250	3423
Core Power, MWt	3250	3411
RCS Flow, (gpm/loop)*	88,500	-***
Minimum Measured Flow, (total gpm)**	366,400	364,960
RCS Temperatures, °F		
Core Outlet	602.0	-
Vessel Outlet	599.3	-
Core Average	570.5	575.5
Vessel Average	567.8	574.1
Vessel/Core Inlet	536.3	-
Steam Generator Outlet	536.3	-
Zero Load	547.0	547.0
RCS Pressure, psia	2250	2250
Steam Pressure, psia	758	794.4
Steam Flow, (10^6 lb/hr.tot.)	14.12	14.6
Feedwater Temperature, °F	434.8	423.4
% SG Tube Plugging	0	10% avg./ 15% peak

Flow Definitions:

*RCS Flow (Thermal Design Flow) - The conservatively low flow used for thermal/hydraulic design. The design parameters listed above are based on this flow.

**Minimum Measured Flow - The flow specified in the Tech. Specs. which must be confirmed or exceeded by the flow measurements obtained during plant startup and is the flow used in reactor core DNB analyses for plants applying the Improved Thermal Design Procedure.

***Flow values supplied in FSAR for Unit 2 are 141.3 x 10^6 lb/hr for vessel coolant flow, and 134.9 x 10^6 lb/hr for active core flow.

Note: Dashes in Case 2 indicate information which was not contained in the FSAR, and is therefore information which is unavailable to Westinghouse.

TABLE S-2.1-1 (Cont'd)

COOK NUCLEAR PLANT UNITS 1 AND 2
DESIGN POWER CAPABILITY PARAMETERS FOR RERATING PROGRAM

Parameter	(Revised) Case 3	(Revised) Case 4	(Revised) Case 5	(Revised) Case 6
NSSS Power, MWt	3262	3425	3425	3425
Core Power, MWt	3250	3413	3413	3413
RCS Flow, (gpm/loop)*	88,500	88,500	88,500	88,500
Minimum Measured Flow, (total gpm)**	366,400	366,400	366,400	366,400
RCS Temperatures, °F				
Core Outlet	610.1	583.6	614.0	613.9
Vessel Outlet	607.5	580.7	611.2	611.2
Core Average	579.2	549.7	581.8	581.8
Vessel Average	576.3	547.0	578.7	578.7
Vessel/Core Inlet	545.2	513.3	546.2	546.2
Steam Generator Outlet	545.0	513.1	546.0	546.0
Zero Load	547.0	547.0	547.0	547.0
RCS Pressure, psia	2250 or 2100	2250 or 2100	2250 or 2100	2250 or 2100
Steam Pressure, psia	807	603	820	806
Steam Flow, (10^6 lb/hr.tot.)	14.20	14.98	15.07	15.06
Feedwater Temperature, °F	434.8	442.0	442.0	442.0
% SG Tube Plugging (average)	15	10	10	15

Flow Definitions:

*RCS Flow (Thermal Design Flow) - The conservatively low flow used for thermal/hydraulic design. The design parameters listed above are based on this flow.

**Minimum Measured Flow - The flow specified in the Tech. Specs. which must be confirmed or exceeded by the flow measurements obtained during plant startup and is the flow used in reactor core DNB analyses for plants applying the Improved Thermal Design Procedure.

TABLE S-2.1-1 (Cont'd)

COOK NUCLEAR PLANT UNITS 1 AND 2
DESIGN POWER CAPABILITY PARAMETERS FOR RERATING PROGRAM

Parameter	(Revised) Case 7	(Revised) Case 8	(Revised) Case 9	(Revised) Case 10
Power, MWt	3600	3600	3600	3600
Core Power, MWt	3588	3588	3588	3588
RCS Flow, (gpm/loop)*	88,500	88,500	88,500	88,500
Minimum Measured Flow, (total gpm)**	366,400	366,400	366,400	366,400
RCS Temperatures, °F				
Core Outlet	585.4	618.0	585.4	618.1
Vessel Outlet	582.3	615.2	582.3	615.2
Core Average	549.9	584.6	549.9	584.7
Vessel Average	547.0	581.3	547.0	581.3
Vessel/Core Inlet	511.7	547.3	511.7	547.4
Steam Generator Outlet	511.4	547.1	511.4	547.2
Zero Load	547.0	547.0	547.0	547.0
RCS Pressure, psia	2250 or 2100	2250 or 2100	2250 or 2100	2250 or 2100
Steam Pressure, psia	587	820	576	806
Steam Flow, (10^6 lb/hr.tot.)	15.90	16.00	15.89	15.99
Feedwater Temperature, °F	449.0	449.0	449.0	449.0
% SG Tube Plugging (average)	10	10	15	15

Flow Definitions:

*RCS Flow (Thermal Design Flow) - The conservatively low flow used for thermal/hydraulic design. The design parameters listed above are based on this flow.

**Minimum Measured Flow - The flow specified in the Tech. Specs. which must be confirmed or exceeded by the flow measurements obtained during plant startup and is the flow used in reactor core DNB analyses for plants applying the Improved Thermal Design Procedure.

S-3.3 NON-LOCA SAFETY EVALUATION

S-3.3.1 Introduction

This section evaluates the effects of the Cook Nuclear Plant Rerating Program on the non-LOCA transients. The non-LOCA safety evaluation provided within is applicable only for Unit 1, with the exception of the steamline break mass/energy releases (inside and outside containment). The effort performed is to support Unit 1 operation with an uprated core power of 3413 MWt in the range of reactor vessel average temperatures between 547°F and 578.7°F at primary pressure values of 2100 psia <u>or</u> 2250 psia. Table S-2.1-1 (Cases 4 and 5) presents the range of conditions possible for the rerating of Unit 1. The steamline break mass/energy release analyses are performed to support the potential future Unit 1 rerating as well as to bound a potential rerating of Unit 2. Table S-2.1-1 (Cases 7 and 8) presents the range of conditions possible for the future rerating of Unit 2. In addition, the evaluation performed is to support a maximum average steam generator tube plugging level of 10%, with a peak steam generator tube plugging level of 15%.

The following non-LOCA safety evaluation also supports the change and/or relaxation of certain plant parameters to provide Unit 1 with increased operating margin and flexibility. Included in the non-LOCA safety evaluation are:

Increased Most Negative Moderator Temperature Coefficient (MTC)
(Tech Spec 3.1.1.4b)
Degraded ECCS Charging Pump Flow (Tech Spec 4.5.2f)
Increased Main Steamline Isolation Valve (MSIV) Closure Time
(Tech Spec 4.7.1.5b and Tech Spec Table 3.3-5 items 5h, 6h, & 7c)

The evaluation conservatively assumes 0 ppm boron concentration in the Boron Injection Tank (BIT).

The evaluation also supports a change to the steam generator water level program. The existing level program is a ramp function from 33% narrow range span (NRS) to 44% NRS from 0% power to 20% power and a constant level at 44% NRS between 20% power and 100% power. The proposed steam generator water level program is a constant level at 44% NRS between 0% power and 100% power.

S-3.3.4.1 Steamline Break Mass/Energy Releases

This section will discuss the analyses of the steamline break event to determine the mass and energy releases inside containment and the superheated mass and energy releases outside containment for the Cook Rerating Program. The analyses were performed to support the range of conditions possible for the rerating of Unit 1 as well as to position Unit 2 for a potential rerating. The analyses also consider the relaxation of certain plant parameters (Section S-3.3-1).

Steamline Break Mass/Energy Releases Inside Containment

The current mass/energy releases for the inside containment analysis is based on work performed for Unit 2, which is applicable for Unit 1. The calculation of the mass/energy release following a steamline break is described in the Cook Unit 2 FSAR Section 14.1.5. The steamline break mass/energy releases were recalculated to address the rerating of both Units and the relaxation of the plant parameters described in Section S-3.3.1.

Steamline ruptures occurring inside a reactor containment structure may result in significant releases of high energy fluid to the containment environment, possibly resulting in high containment temperatures and pressures. The quantitative nature of the releases following a steamline rupture is dependent upon the many possible configurations of the plant steam system and containment designs as well as the plant operating conditions and the size of the rupture. These variations make it difficult to reasonably determine the single "worst case" for both containment pressure and temperature evaluations following a steambreak. The FSAR analysis determined that the limiting scenario of the steambreak cases analyzed for the containment response evaluation were a break size of 0.942 ft^2 occurring at 30% power for the split rupture scenario and a break size of 4.6 ft^2 occurring at full power for the double-ended rupture scenario. (The 30% power split break case was slightly more limiting.) However, it is difficult to conclude if these FSAR cases remain bounding for the range of conditions possible for the reratings of both Units.

Adding to the difficulty in determining the effect of the rerating conditions are the plant parameters changes incorporated into the Cook Rerating Program. The potential changes of certain plant parameters (i.e, relaxed most negative MTC limit, degraded ECCS performance, increased MSIV closure time, and 0 ppm BIT boron concentration requirement) are penalties in the calculation of mass/energy releases. It is not readily apparent as to the total impact of the combination of these changes. As such, a series of steamline breaks, consistent with the cases presented in the FSAR, were analyzed to determine the containment response to a variety of postulated pipe breaks encompassing wide variations in plant operation, safety system performance, and break sizes.

Method of Analysis

The LOFTRAN computer code (Reference 2) was used to calculate the break flows and enthalpies of the release through the steambreak. Blowdown mass/energy releases determined using LOFTRAN include the effects of core power generation, main and auxiliary feedwater additions, engineered safeguards systems, reactor coolant thick metal heat storage, and reverse steam generator heat transfer.

A bounding analysis was performed to address the range of conditions possible for the potential Unit 1 rerating and the potential Unit 2 rerating. The assumptions on the initial conditions are taken to maximize the mass and total energy released. The higher primary temperatures along with the higher uprated power level associated with the Unit 2 rerating parameters are conservative for the mass/energy release calculations. The upper bound temperature of Table S-2.1-1, Case 8 was used. Since the mass blowdown rate is dependent on steam pressure and the steam pressure is less for the lower bound temperature case, the steam pressure of the upper bound temperature case is limiting for the range of operating conditions possible for the reratings of Unit 1 and Unit 2.

The functions which actuate safety injection and steamline isolation during a steamline rupture event are commonly referred to as the Steamline Break Protection System. A plant's steamline break protection system design can have a large effect on steamline break results. The steamline break protection system designs for Unit 1 and Unit 2 are different. Unit 1's design is referred to as an "OLD" steamline break protection system design. Unit 2's design is referred to as a "HYBRID" steamline break protection system design. The two systems have the following characteristics:

Unit 1 - "OLD" Steamline Break Protection

Safety Injection Signals

1. High-high steam flow coincident with low steamline pressure (two out of four lines)

2. High-high steam flow coincident with low-low Tavg (two out of four lines)

3. Two out of three differential pressure signals between a steam line and the remaining steam lines

4. Two out of three low pressurizer pressure signals

5. Two out of three hi containment pressure signals

Steamline Isolation Signals

1. High-high steam flow coincident with low steamline pressure (two out of four lines)

2. High-high steam flow coincident with low-low Tavg (two out of four lines)

3. Two out of four hi-hi containment pressure signals

Unit 2 - "HYBRID" Steamline Break Protection

Safety Injection Signals

1. Low steamline pressure (two out of four lines)

2. Two out of three differential pressure signals between a steam line and the remaining steam lines

3. Two out of three low pressurizer pressure signals

4. Two out of three hi containment pressure signals

Steamline Isolation Signals

1. Low steamline pressure (two out of four lines)

2. High-high steam flow coincident with low-low Tavg (two out of four lines)

3. Two out of four hi-hi containment pressure signals

The only differences between the Unit 1 and Unit 2 designs is the actuations from a high-high steam flow and low-low Tavg signal and the logic associated with the low steamline pressure signal required to actuate safety injection and steamline isolation. For Unit 1, a high-high steam flow coincident with low-low Tavg signal actuates both safety injection and steamline isolation. For Unit 2, a high-high steam flow coincident with low-low Tavg signal actuates only steamline isolation. However, the difference is not significant for the calculation of the mass/energy releases since the analysis does not take credit for any ESF actuations on a high-high steam flow coincident with low-low Tavg signal.

Unit 1's design requires a coincidence between the low steamline pressure and high-high steam flow for protection actuation. Unit 2's design only requires the low steamline pressure signal for protection actuation; no coincidence with steam flow is required.

The coincidence logic required for safety injection initiation and steamline isolation on high-high steam flow and low steam pressure for Unit 1 is more limiting for the calculation of mass/energy releases inside containment than Unit 2's design. Actuation of safety injection and steamline isolation will limit the mass/energy released to the containment. Delaying the safeguards initiation will result in a conservative calculation of the mass/energy

releases for the containment pressure and temperature evaluation. The coincidence requirement for high-high steam flow with low steam pressure of the Unit 1 design increases the likelihood that safeguards initiation might be delayed compared to Unit 2's design where only a low steam pressure signal is required. In the case where the coincidence logic prohibits safety injection and steamline isolation on high-high steam flow with low steam pressure, one of the other signals must be received before the safeguards are initiated. As such, the Unit 1 steamline break protection system design was assumed in this bounding analysis for the calculation of the mass/energy releases inside containment.

Assumptions

A series of steamline breaks were analyzed to determine the most severe break condition for the containment temperature and pressure response. The following assumptions were used in the analysis:

a. Double-ended pipe breaks were assumed to occur at the nozzle of one steam generator and also downstream of the flow restrictor. Split ruptures were assumed to occur at the nozzle of one steam generator.

b. The blowdown is assumed to be dry saturated steam.

c. As discussed above, the Unit 1 steamline break protection system design is assumed. However, credit was not taken for safeguards actuation on high steam line differential pressure or high-high steam flow coincident with low-low Tavg.

d. Steamline isolation is assumed complete 11 seconds after the setpoint is reached for either high-high steam flow coincident with low steam pressure or hi-hi containment pressure. The isolation time allows 8 seconds for valve closure plus 3 seconds for electronic delays and signal processing. The total delay time for steamline isolation of 11 seconds is assumed to support the relaxation of the main steam isolation valve (MSIV) closure time.

e. 4.6 ft^2 and 1.4 ft^2 double-ended pipe breaks were evaluated at 102, 70, 30, and zero percent power levels.

f. Split pipe ruptures were evaluated at 0.86 ft^2, 102% power; 0.908 ft^2, 70% power; 0.942 ft^2, 30% power; and 0.4 ft^2, hot shutdown.

These split break sizes for each power level were modeled because they reflect the largest breaks for which ESF actuations (i.e., steamline isolation, feedwater isolation, and safety injection) must be generated by high containment pressure trips. The high-high steam flow coincident with low steam pressure is not reached for these break sizes or smaller break sizes. (Reference 5)

g. Failure of a main steam isolation valve, failure of a feedwater isolation valve or main feed pump trip, and failure of auxiliary feedwater runout control were considered. Two cases for each break size and power level scenario were evaluated with one case modeling the MSIV failure and the other case modeling the AFW runout control failure. Each case assumed conservative main feedwater addition to bound the feedwater isolation valve or main feed pump trip failure.

h. The auxiliary feedwater system is manually re-aligned by the operator after 10 minutes.

i. A shutdown margin of 1.3% $\Delta k/k$ is assumed. This assumption includes added conservatism with respect to the Unit 1 end-of-life shutdown margin requirement of 1.6% $\Delta k/k$ at no load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position. The Unit 1 end-of-life shutdown margin requirement was used as the basis for this assumption since it is more limiting than the existing Unit 2 shutdown margin requirement.

j. A moderator density coefficient of 0.54 Δk/gm/cc is assumed to support the relaxation of the most negative moderator temperature coefficient limit.

k. Minimum capability for injection of boric acid (2400 ppm) solution corresponding to the most restrictive single failure in the safety injection system. The Emergency Core Cooling System (ECCS) consists of the following systems: 1) the passive accumulators, 2) the low head safety injection (residual heat removal) system, 3) the high head (intermediate head) safety injection system, and 4) the charging safety injection system. Only the charging safety injection system and the passive accumulators are modeled for the steam line break accident analysis.

The modeling of the safety injection system in LOFTRAN is described in Reference 2. Figure 3.3-52 of WCAP-11902 presents the safety injection flow rates as a function of RCS pressure assumed in the analysis. The flow corresponds to that delivered by one charging pump delivering its full flow to the cold legs. The safety injection flows assumed in this analysis take into account the degradation of the ECCS charging pump performance. No credit has been taken for any borated water that might exists in the injection lines, which must be swept from the lines downstream of the boron injection tank isolation valves prior to the delivery of boric acid to the reactor coolant loops. For this analysis, a boron concentration of 0 ppm for the boron injection tank is assumed.

After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the safety injection charging pump starts. In 27 seconds, the valves are assumed to be in their final position (VCT charging pump suction valve has closed following opening of RWST charging pump suction valve) and the pump is assumed to be at full speed and to draw suction from the RWST. The volume containing the low concentration borated water is swept into the core before the 2400 ppm borated water reaches the core. This delay, described above, is inherently included in the modeling.

l. For the at-power cases, reactor trip is available by safety injection signal, overpower protection signal (high neutron flux reactor trip or OPΔT reactor trip), and low pressurizer pressure reactor trip signal.

m. For reactor coolant pump (RCP) operation, offsite power is assumed available. Continued operation of the reactor coolant pumps maximizes the energy transferred from the reactor coolant system to the steam generators.

n. No steam generator tube plugging is assumed to maximize the heat transfer characteristics.

Single Failure Effects

a. Failure of a main steam isolation valve (MSIV) increases the volume of steam piping which is not isolated from the break. When all valves operate, the piping volume capable of blowing down is located between the steam generator and the first isolation valve. If this valve fails, the volume between the break and the isolation valves in the other steamlines, including safety and relief valve headers and other connecting lines, will feed the break. For the cases which modeled a failure of a MSIV, the steamline volumes associated with Unit 2 were assumed since the volume available for blowdown for this scenario is greater than Unit 1. For the cases which did not model a failure of a MSIV, the steamline volumes associated with Unit 1 were assumed since the volume available for blowdown for this scenario is greater than Unit 2.

b. Failure of a diesel generator would result in the loss of one containment safeguards train resulting in minimum heat removal capability.

c. Failure of a feedwater isolation valve would result in additional inventory in the feedwater line which would not be isolated from the steam generator. The mass in this volume can flash into steam and exit through the break. For consistency with the FSAR steamline break mass/energy release analysis, all cases conservatively assumed failure of the feedwater isolation valve, which resulted in the additional inventory available for release through the steambreak and in higher than normal main feedwater flows.

d. Failure of the auxiliary feedwater runout control equipment would result in higher auxiliary feedwater flows entering the steam generator prior to re-alignment of the AFW system. For cases where the runout control operates properly, a bounding constant AFW flow of 670 gpm to the faulted steam generator was assumed. This value was increased to 1325 gpm to simulate a failure of the runout control.

Results

The steamline break mass/energy releases inside containment were calculated to account for the range of conditions possible for the potential reratings of Unit 1 and Unit 2 and for the relaxation of certain plant parameters. One set of mass/energy releases were calculated to bound the reratings for both Units incorporating the limiting steamline break protection design of Unit 1. The analysis assumptions support relaxation of the most negative moderator temperature coefficient limit, degradation of the charging pump performance of the Emergency Core Cooling System, extension of the main steam isolation valve closure time, and relaxation of the minimum BIT boron concentration requirement.

Section S-3.4.2.1 presents the containment integrity evaluation for a main steamline break using the mass/energy releases calculated here. As discussed in Section S-3.4.2.1, the limiting scenarios of the steambreak cases analyzed for the containment response evaluation were a break size of 4.6 ft^2 occurring at 102% power with a main steamline isolation failure for the double-ended rupture scenario and a break size of 0.86 ft^2 occurring at 102% power with an auxiliary feedwater runout protection failure for the split rupture scenario. Table S-3.3-4 presents the mass/energy releases for these limiting steambreak cases of the containment response evaluation.

S-3.3.6 REFERENCES

1. Augustine, D. B., and Cecchett, D. L., "Reduced Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Unit 1 Licensing Report," WCAP-11902, October 1988.

2. Burnett, T, W.. T., et al., "LOFTRAN Code Description," WCAP-7907-A, April 1, 1984.

3. Butler, J. C., and Love, D. S, "Steamline Break Mass/Energy Releases for Equipment Qualification Outside Containment," WCAP-10961, Rev. 1 (proprietary) and WCAP-11184 (nonproprietary), October, 1985.

4. Hollingsworth, S. D., and Wood, D. C., "Reactor Core Response To Excessive Secondary Steam Releases," WCAP-9227, January 1978.

5. Land, R. E., "Mass and Energy Releases Following a Steam Line Rupture;" WCAP-8860, September 1976.

6. "American Electric Power Service Corporation Donald C. Cook Nuclear Plant Unit 1: Safety Evaluation for Including Uncertainty Due to Operator Readability of Pressurizer Pressure Instrumentation," AEP-89-216, Letter from J. C. Hoebel (W) to R. B. Bennett (AEPSC), September 1989.

TABLE S-3.3-4

STEAMLINE BREAK
MASS/ENERGY RELEASES INSIDE CONTAINMENT
102% POWER DER (4.6 FT^2) BREAK
FAILURE - MSIV

TIME (SEC)	MASS (LBM/SEC)	ENERGY (BTU x 10^6/SEC)
0.00	0.00	0.0
0.20	10430.00	1.250
3.60	6552.00	7.883
6.60	5612.00	6.748
12.80	4978.00	5.974
13.00	4913.00	5.895
13.20	4847.00	5.816
13.40	4781.00	5.737
13.60	4716.00	5.660
14.00	4587.00	5.504
14.40	4458.00	5.350
14.80	4332.00	5.198
15.00	4269.00	5.123
15.20	4206.00	5.047
15.60	4083.00	4.899
15.80	4022.00	4.826
16.00	3961.00	4.753
16.60	3782.00	4.538
17.20	3606.00	4.328
17.60	3492.00	4.190
17.80	3435.00	4.122
18.40	3268.00	3.921
18.60	3213.00	3.856
18.80	3158.00	3.790
19.20	3050.00	3.660
23.80	1876.00	2.251
28.80	1623.00	1.421
30.40	1575.00	1.883
36.40	1461.00	1.746
39.20	1431.00	1.708
50.70	1369.00	1.634
57.20	1356.00	1.618
106.20	1331.00	1.588
109.20	1331.00	1.587
111.20	1184.00	1.409
118.20	308.70	0.358
125.20	188.10	0.217
136.20	98.97	0.114
602.70	93.24	0.107

TABLE S-3.3-4 (Cont'd)

STEAMLINE BREAK
MASS/ENERGY RELEASES INSIDE CONTAINMENT
102% POWER SPLIT (0.86 FT^2) BREAK
FAILURE - AUXILIARY FEEDWATER RUNOUT PROTECTION

TIME (SEC)	MASS (LBM/SEC)	ENERGY (BTU x 10^6/SEC)
0.00	0.00	0.0000
0.20	1394.00	1.6690
1.60	1366.00	1.6370
2.00	1358.00	1.6270
2.40	1350.00	1.6170
2.80	1342.00	1.6080
4.20	1316.00	1.5770
4.40	1312.00	1.5730
8.60	1550.00	1.8540
9.40	1575.00	1.8840
12.00	1632.00	1.9500
12.60	1638.00	1.9570
15.80	1635.00	1.9530
18.00	1618.00	1.9340
21.40	1458.00	1.7460
22.60	1400.00	1.6790
23.60	1357.00	1.6280
23.80	1349.00	1.6180
25.00	1302.00	1.5630
32.00	1103.00	1.3260
32.20	1098.00	1.3210
33.80	1064.00	1.2810
42.00	928.70	1.1180
42.60	920.80	1.1090
43.20	913.10	1.1000
43.80	905.70	1.0910
44.40	898.40	1.0820
55.20	799.10	0.9625
67.20	732.60	0.8823
80.20	691.30	0.8325
82.20	686.60	0.8269
96.20	662.50	0.7977
98.70	659.50	0.7941
118.20	645.70	0.7775
124.20	643.60	0.7749
282.70	633.20	0.7623
285.20	633.10	0.7622
290.20	615.00	0.7402
292.70	579.70	0.6977
297.70	556.60	0.6695
302.70	490.40	0.5896
320.20	304.70	0.3643

TABLE S-3.3-4 (Cont'd)

STEAMLINE BREAK
MASS/ENERGY RELEASES INSIDE CONTAINMENT
102% POWER SPLIT (0.86 FT^2) BREAK
FAILURE - AUXILIARY FEEDWATER RUNOUT PROTECTION

TIME (SEC)	MASS (LBM/SEC)	ENERGY (BTU x 10^6/SEC)
330.20	238.70	0.2845
340.20	206.50	0.2456
352.70	190.20	0.2259
525.20	181.90	0.2160
535.20	182.00	0.2160
600.20	182.10	0.2162
605.20	190.70	0.2258

S-3.4.2.1 Main Steamline Break (MSLB) Containment Integrity

Introduction and Background

An evaluation was performed to determine the impact of reduced temperature and pressure operation on the Donald C. Cook Nuclear Plant Unit 1 Long-Term Main Steamline Break Containment Integrity analysis. This evaluation is documented

in Section 3.4.2 of WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT," and it was concluded that reduced temperature and pressure operation did not have an adverse impact on the analysis results and conclusions. This Section documents the analysis performed for both Donald C. Cook Nuclear Plant Units 1 & 2 to determine the impact of the rerated conditions described in Section S-2.1 on Containment Integrity following a Main Steamline Break.

A series of main steamline split and double-ended breaks were analyzed as a part of the original licensing basis for Donald C. Cook Nuclear Plant Unit 2 to determine the most severe break condition for containment temperature and pressure response for this design basis event. The analysis and evaluation are discussed in Reference 1. These results documented in the FSAR show that the most limiting double-ended break was the 4.6 square foot break, occurring at 102% power with main steam isolation valve failure. The most limiting split break was the 0.942 square foot break, occurring at 30% power with the failure of auxiliary feedwater runout protection. The calculated peak temperatures for these cases were 319.1°F and 328.1°F respectively. Additional generic sensitivities discussed in Reference 2, illustrate that other smaller breaks were not limiting.

Purpose

The purpose of the analysis documented in the following paragraphs is to demonstrate that the peak containment temperature resulting from a design basis main steamline break will not exceed the equipment qualification temperature criterion for Donald C. Cook Nuclear Plants Units 1 and 2, at the rerated conditions. The containment pressure response generated for the LOCA Containment Integrity analysis for the double-ended pump suction RCS break case (Reference 3) bounds the Main Steamline Break containment pressure response, and therefore is not a concern here. This analysis assumes reduced safety injection flow, due to degradation of ECCS performance, closure of the RHR crosstie valves and the current containment heat sink information.

Analytical Assumptions

The analysis performed for the Rerating Program is consistent with the Reference 1 analysis except for assumptions directly related to the rerating parameters. The analytical effort provides bounding system calculations for both Units 1 & 2 at the rerated plant conditions described in Section S-2.1.

A spectrum of split breaks is analyzed at 0.86 ft^2, 102% power; 0.908 ft^2, 70% power; 0.942 ft^2, 30% power and 0.4 ft^2, hot shutdown. Double-ended breaks of 1.4 ft^2 and 4.6 ft^2 are analyzed at power levels of 102%, 70%, 30% and zero power levels.

The break sizes analyzed in the present analysis are based on the current FSAR analysis. As in the FSAR analysis, loss of one containment safeguards train was also assumed for all the cases in addition to the single failure assumed in the mass and energy release calculations.

The following cases were analyzed for containment response:

A. <u>Split break cases</u>

1) 0.86 ft^2, 102% power, MSIV failure
2) 0.86 ft^2, 102% power, AFRP failure
3) 0.908 ft^2, 70% power, MSIV failure
4) 0.908 ft^2, 70% power, AFRP failure
5) 0.942 ft^2, 30% power, MSIV failure
6) 0.942 ft^2, 30% power, AFRP failure
7) 0.40 ft^2, hot shutdown, MSIV failure
8) 0.40 ft^2, hot shutdown, AFRP failure

Note: MSIV - Main Steam Isolation Valve
AFRP - Auxiliary Feedwater Runout Protection

B. Double-ended rupture cases*

1) 4.6 ft^2, 102% power, MSIV failure
2) 4.6 ft^2, 102% power, AFRP failure
3) 4.6 ft^2, 70% power, MSIV failure
4) 4.6 ft^2, 70% power, AFRP failure
5) 4.6 ft^2, 30% power, MSIV failure
6) 4.6 ft^2, hot shutdown, MSIV failure
7) 1.4 ft^2, 102% power, MSIV failure
8) 1.4 ft^2, 102% power, AFRP failure
9) 1.4 ft^2, 70% power, MSIV failure
10) 1.4 ft^2, 30% power, MSIV failure
11) 1.4 ft^2, hot shutdown, MSIV failure

Note: *The limiting 4.6 ft^2 double-ended failure cases (102% and 70% power), with MSIV failure were analyzed with AFRP failure and found to be less limiting than the corresponding MSIV failure cases. Therefore only the most limiting 1.4 ft^2 (102% power) was analyzed with AFRP failure.

The mass and energy releases to the containment as a result of the postulated accident are calculated using the LOFTRAN computer code (Reference 4). The mass and energy releases are calculated using two different failures for each case namely, 1) failure of the auxiliary feedwater runout protection and 2) failure of the main steam isolation valve. As in Reference 1, no credit is taken for entrainment. Section S-3.3.4.1 presents additional details regarding the calculation of the inside containment steamline break mass and energy releases.

The LOTIC-III computer code (Reference 5) is used to calculate the consequence of these releases, in particular the peak containment temperature.

The main steam line break containment integrity calculations are performed with an additional failure of one of the containment safeguards trains, which results in minimum spray flow (this includes a 10% degradation in the spray pump flow). Where applicable, input data consistent with that of the LOCA containment integrity analysis (Reference 3) is used.

The total initial ice mass assumed is 2.11×10^6 lbs.

The initial conditions in the containment are a temperature of 120°F in the lower and dead ended compartments, a temperature of 27°F in the ice condenser, and a temperature of 57°F in the upper compartment. All volumes are at a pressure of 0.3 psig and a relative humidity of 15%.

The refueling water storage tank (RWST) temperature is assumed to be 100°F.

A spray pump flow of 1900 gpm to the upper compartment and 900 gpm to the lower compartment is assumed, at a temperature of 100°F.

The spray flow is initiated 45.0 seconds after the containment reaches the hi-hi pressure signal of 3.5 psig. This setpoint includes instrument uncertainties.

Results

The results of the analysis show that the maximum calculated containment temperature is 324.9°F for the 4.6 ft^2 double ended rupture at 102% of the full power. The mass and energy calculations for this case are based on the main steam isolation valve failure.

The maximum containment temperature calculated for the limiting small split break (0.86 ft^2 at 102% of full power) is 324.4°F. The auxiliary feedwater runout protection failure is assumed for this case. Table S-3.4-1 and Figures S-3.4-1 through S-3.4-4 show the results for the two limiting cases.

Comparison of these results to the current FSAR results with respect to the peak containment temperature indicates that the FSAR result was more limiting. This is due to the lower mass and energy releases inside containment, calculated for the present analysis. The peak temperature shown in the FSAR for the limiting split break case (0.86 ft^2 at 102% of full power, with auxiliary feedwater runout protection failure) is higher than the

present case. However, the FSAR results for the limiting double-ended rupture case (4.6 ft^2 at 102% power, with main steam isolation valve failure) is lower than the present double-ended results. A detailed study of the results shows that even though the mass and energy releases within containment are lower in both the present cases, the double-ended break results in a higher temperature due to reduced flows from the lower compartment into the ice-condenser.

The peak occurs very early in the transient (within the first ten seconds). At this early time the only heat removal systems that exist are the containment wall heat sinks and the heat flow between the compartments. In the present case, heat removal by the walls is better (due to more detailed modeling of the walls), but the heat flow from the lower compartment into the ice-condenser is lower (due to the lower initial temperature assumed in the ice-condenser and the upper compartment, which affects the driving force through the ice-condenser).

Conclusions

The main steamline break containment integrity analysis has been performed consistent with the current licensing basis analysis and Donald C. Cook Nuclear Plant Units 1 & 2 rerating program, considering the present plant operating conditions. The results of this analysis are bounded by the current FSAR results. This analysis therefore demonstrates that the containment heat removal systems function to rapidly reduce the containment pressure and temperature in the event of a main steamline break accident.

S-3.4.3 References

1. Westinghouse letter # NS-TMA-1946, 9/20/78, " American Electric Power Projects Donald C. Cook Unit 2 (Docket 50-316) Response to Question 022.9".

2. Westinghouse letter #AEP-80-525, 3/10/80, "Response to NRC Question 022.17 - AMP's steamline break analysis".

3. WCAP-11908," Containment Integrity Analysis for Donald C. Cook Nuclear Plant Units 1 and 2", July 1988.

4. WCAP-7907-P-A (Proprietary), "LOFTRAN Code Description", April 1984.

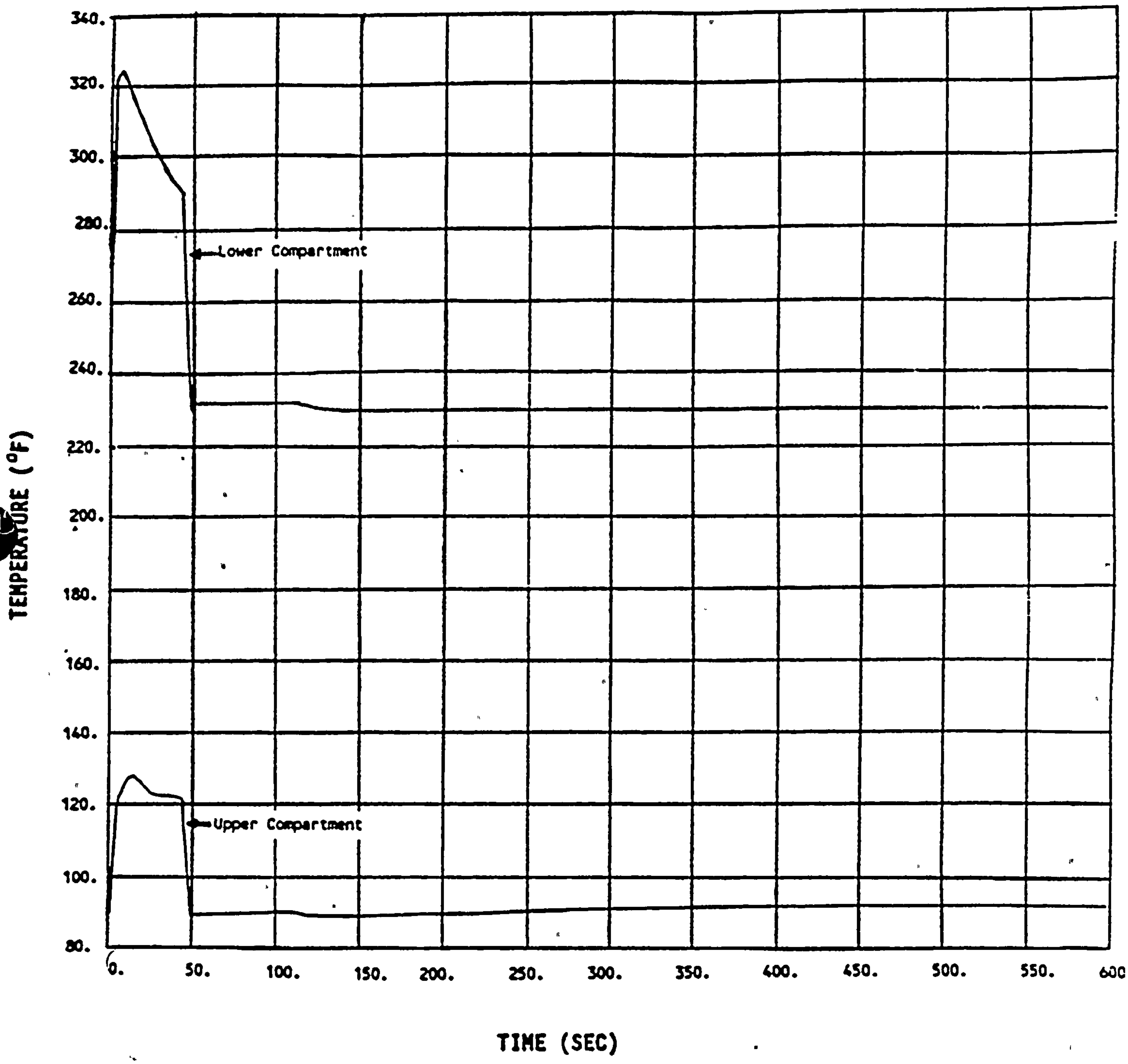
5. WCAP-8354-P-A (Proprietary), Supplement 2, "Long Term Ice Condenser Containment Code - LOTIC-3 Code", February 1979.

TABLE S-3.4-1

MAIN STEAMLINE BREAKS

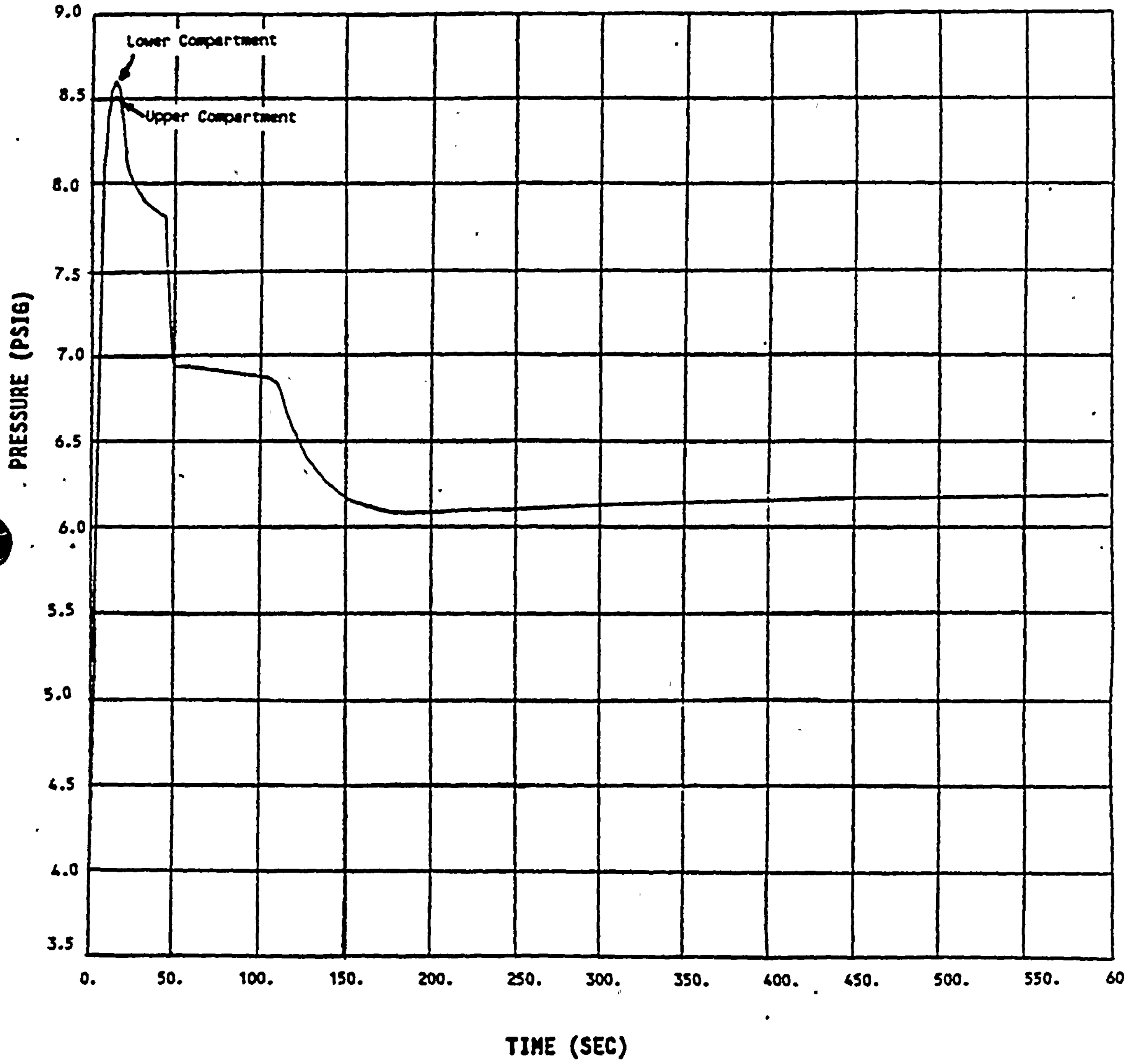
Type of Break	Double-Ended Rupture	Split Break
Break Size (FT^2)	4.6	0.86
Type of Failure	MSIV	AFRP
T_{max} (°F)	324.9	324.4
Time of T_{max} (sec)	6.39	50.72
P_{max} (psig)	8.62	7.24
Time of P_{max} (sec)	14.01	50.72

Note: MSIV - Main Steam Isolation Valve
AFRP - Auxiliary Feedwater Runout Protection



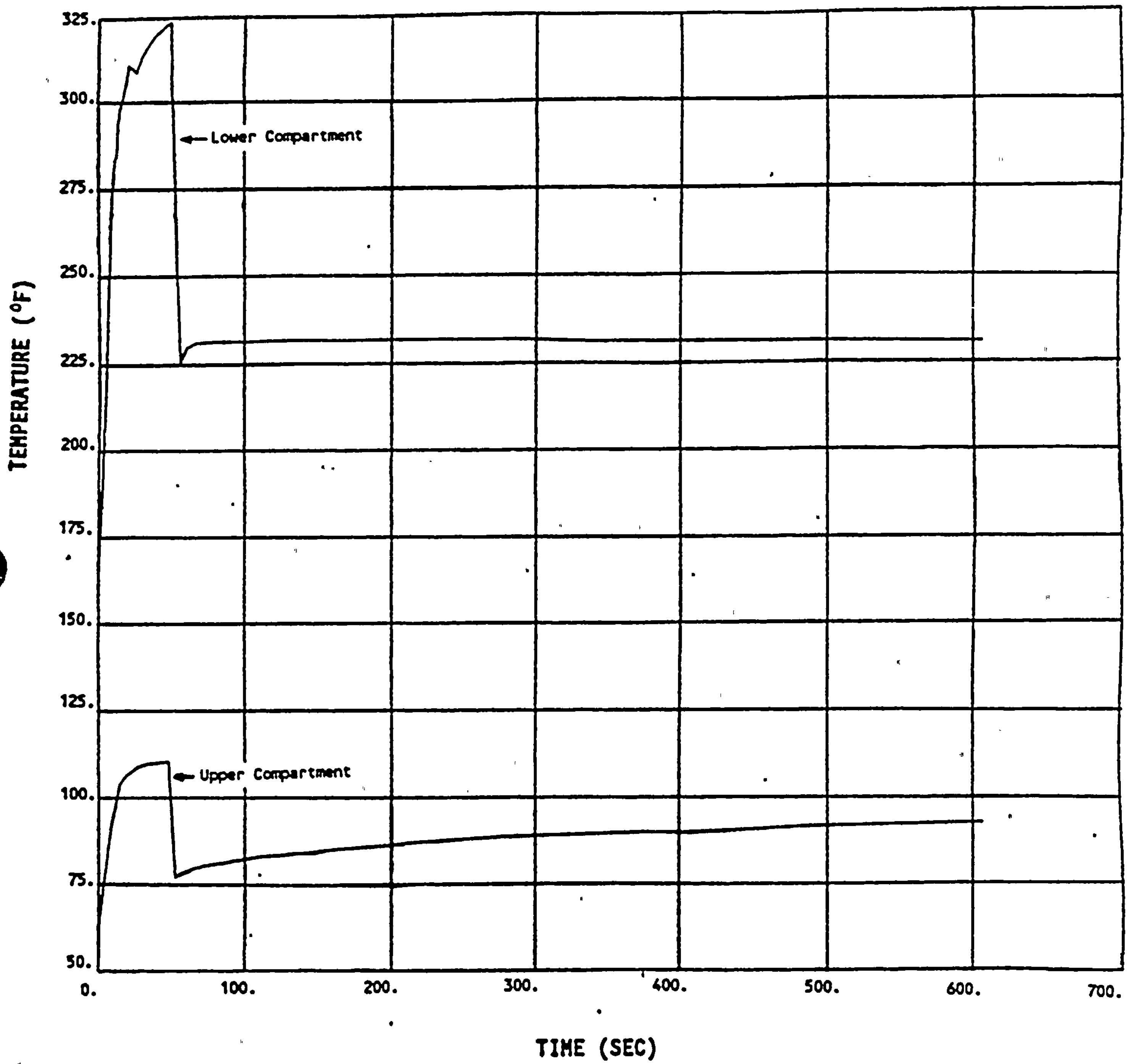
COMPARTMENT TEMPERATURE

Figure S-3.4-1 - 4.6 ft^2 Double-Ended Rupture, 102% Power, MSIV Failure



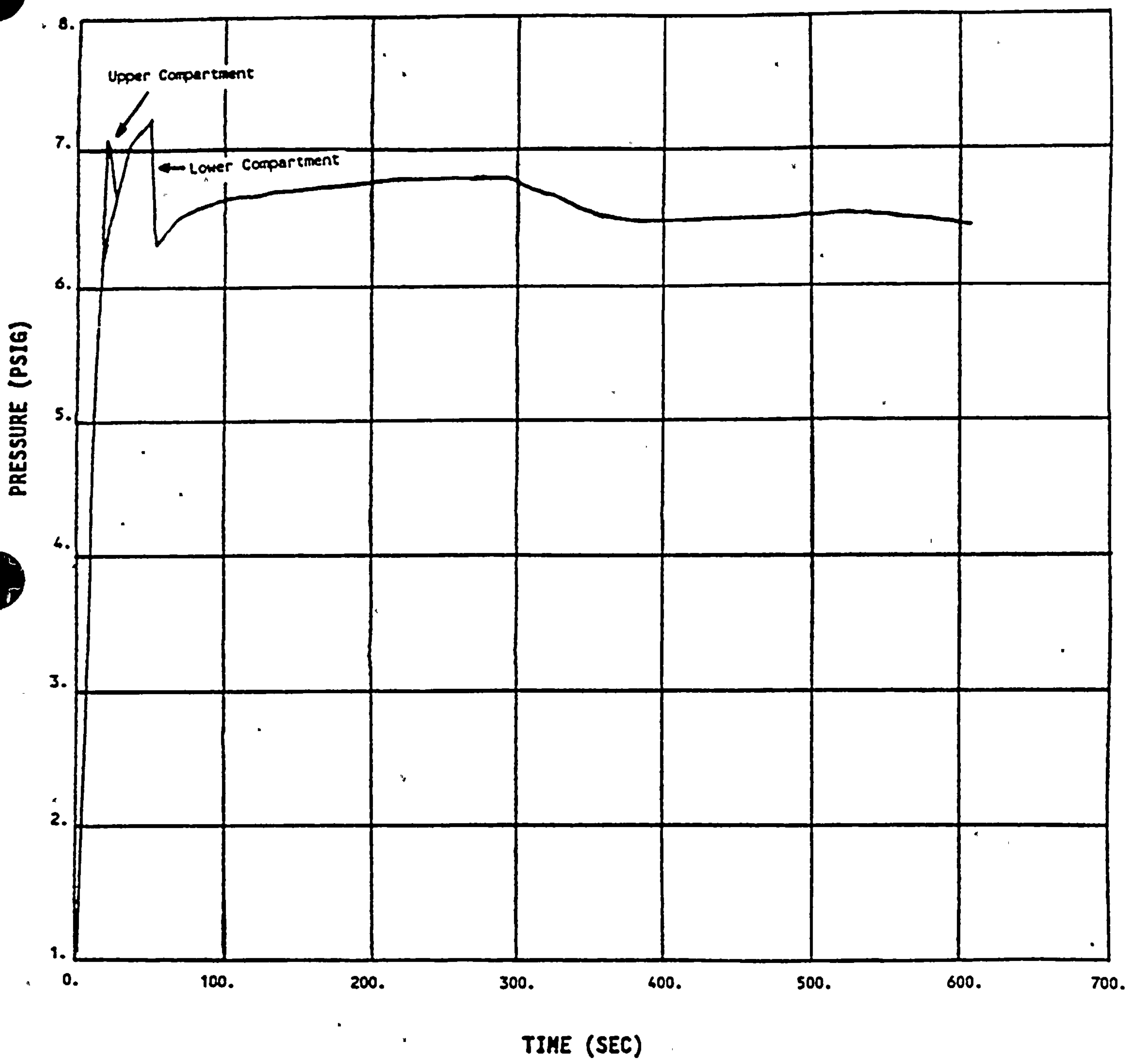
COMPARTMENT PRESSURE

Figure S-3.4-2 - 4.6 ft^2 Double-Ended Rupture, 102% Power, MSIV Failure



COMPARTMENT TEMPERATURE

Figure S-3.4-3 - 0.86 ft^2 Split Break, 102% Power, AFRP Failure



COMPARTMENT PRESSURE

Figure S-3.4-4 - 0.86 ft^2 Split Break, 102% Power, AFRP Failure

ATTACHMENT 6 TO AEP:NRC:1071E

FIGURE 3.3-52 AND JUSTIFICATION FOR PRESSURIZER LEVEL FROM WCAP 11902

WESTINGHOUSE CLASS III

WCAP-11902

REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT

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October 1988

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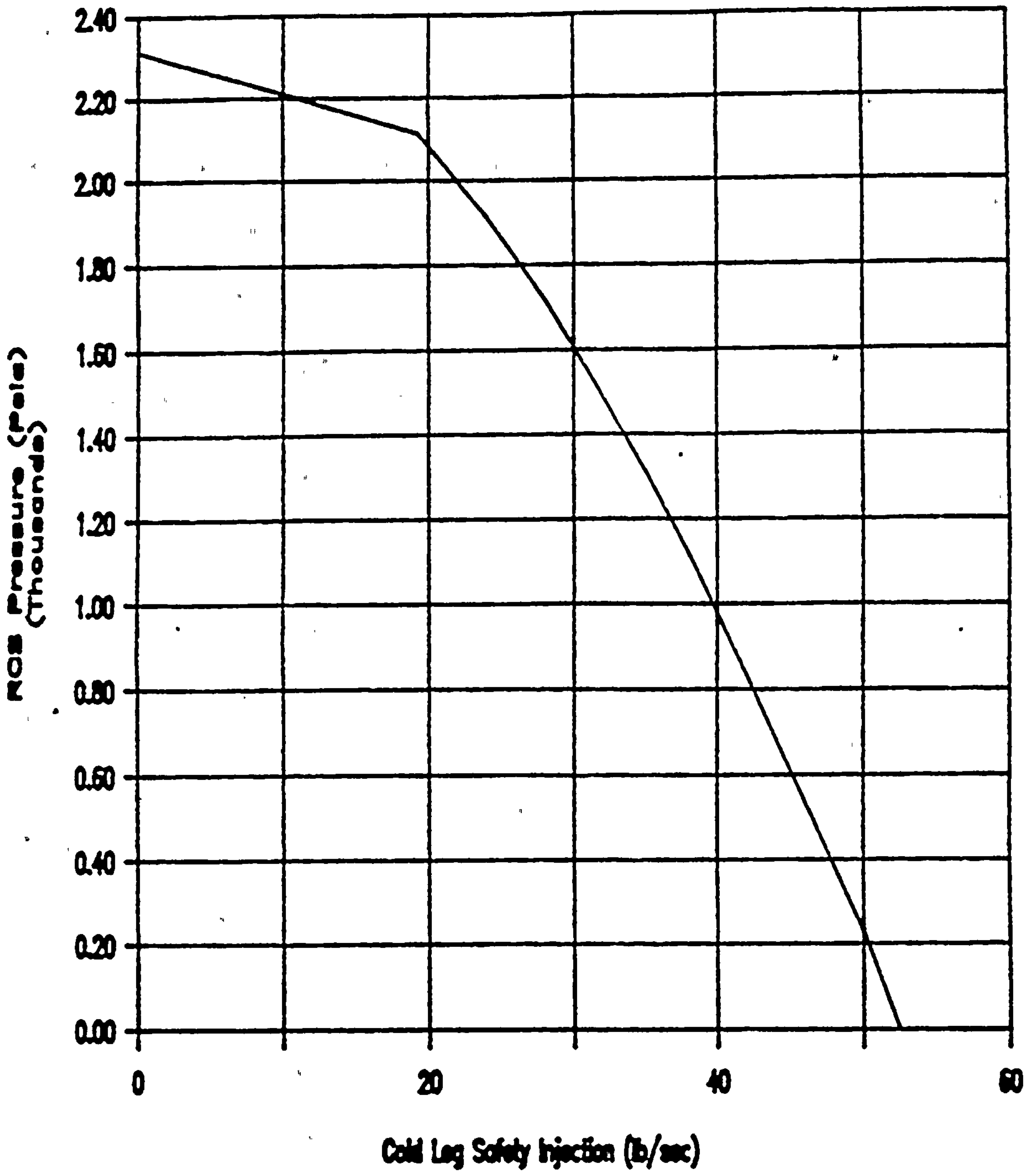

Figure 3.3-52 Safety Injection Flow Supplied by One Charging Pump

ATTACHMENT 4 TO AEP:NRC:1120

ANALYSIS OF STEAM LINE BREAK CORE RESPONSE

S-3.3.4.13 Rupture of a Steam Pipe

The rupture of a steam pipe event was analyzed in Section 3.3.4.13 of WCAP-11902 to support the reduced temperature and pressure operation as well as to bound the range of conditions possible for the rerating of Unit 1. Table S-3.3-3 presents the initial conditions assumed in the WCAP-11902 analysis.

The relaxation of the Technical Specification most negative moderator temperature coefficient refers to the core MTC limit in the unrodded configuration. This MTC limit relaxation is incorporated into the steamline

break core response analysis. The WCAP-11902 analysis assumed a negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position. The reactivity feedback assumption is adjusted to conservatively predict the return to power transient. Verification is performed to show that the reactivity feedback employed in the analysis is conservative.

The analysis conservatively assumed the minimum capability for injection of boric acid solution corresponding to the most limiting single failure in the Emergency Core Cooling System (ECCS). The analysis assumed that the safety injection flow was provided by one charging pump. The analysis assumed degraded performance of the charging pump. Figure 3.3-52 of WCAP-11902 presents the safety injection flow rates as a function of RCS pressure, which takes into account the degraded performance of this ECCS (charging pump) system. The analysis also conservatively assumed a boron concentration of 0 ppm for the boron injection tank (BIT). As such, the analysis supports degradation of the charging pump performance and positions Unit 1 for relaxation of the minimum BIT boron concentration requirement.

The steamline break core response analysis assumed steamline isolation to occur within 11 seconds from receipt of the signal generated by high steam flow coincident with low steam pressure. The 11 second delay is assumed to account for signal processing and electronic delay plus the closure time of the main steamline isolation valves (MSIV). The analysis models only the total delay from the time the setpoint is reached until the time the MSIV is fully closed. Although the WCAP-11902 analysis specified that a MSIV closure time of 7 seconds was assumed, margin is available in the total delay time assumed to support an 8 second MSIV closure time. The 8 second MSIV closure time represents an increase of 3 seconds from the existing Technical Specification limit (5 seconds). As such, the WCAP-11902 steamline break core response analysis supports a relaxation of the MSIV closure time requirement.

The WCAP-11902 steamline break core response analysis is performed at Hot Zero Power, with a corresponding initial steam generator level at 33% NRS. Increasing the initial level to 44% NRS insignificantly impacts the results of

the analysis. Increasing the water level will not have an unacceptable effect on the minimum DNBR for the double-ended rupture (4.6 ft^2, 1.4 ft^2) steamline break core response analysis. This evaluation is based on sensitivity studies presented in WCAP-9227, "Reactor Core Response to Excessive Secondary Steam Releases" (Reference 4). Although this report was not used in support of the WCAP-11902 analysis, the conclusions presented are generic in nature and as such can be applied to Cook Unit 1.

Thus, the safety analysis and conclusions presented in Section 3.3.4.13 remain applicable for the parameters of the Unit 1 rerating.

ATTACHMENT 1 TO AEP:NRC:1107

REASONS AND 10 CFR 50.92 ANALYSES FOR CHANGES TO THE

DONALD C. COOK NUCLEAR PLANT

UNITS 1 AND 2 TECHNICAL SPECIFICATIONS

Introduction

Many of the T/Ss applicable during Mode 6 refueling are vague and subject to interpretation. Consequently, on the advice of our former Project Manager, and in an attempt to avoid any future confusion over the intent of the T/Ss, we are requesting the T/S changes presented in this submittal to clarify any ambiguities that may exist. These changes may be categorized into four types:

Clarification of Mode 6
Clarification of Applicability of T/Ss
Clarification of the Intent of T/Ss
Changes to Reflect Industry Norms

Clarification of Mode 6 - T/S 1.4

T/S 1.4 defines operational mode as follows, "An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1." In Table 1.1, Mode 6 refueling has a footnote stating, "Reactor vessel head unbolted or removed and fuel in the vessel." As such, when no fuel is in the vessel the plant is not in Mode 6, or any other mode. Hence, conditions such as movement of irradiated fuel with no fuel assemblies in the reactor vessel are categorized as "other conditions specified for each specification," as discussed in T/S 3/4.0.1 and T/S 3/4.0.4.

The fact that both the statements, "At all times" and "All Modes" can be found in the applicability statements of the Cook Nuclear Plant's T/Ss supports this position. Clearly there must be times when the units are not in a mode to necessitate the use of the phrase "At all times."

Consequently, to clarify the definition of operational mode we are proposing to modify T/S 1.4 as follows, "An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1 with fuel in the reactor vessel." The phrase "with fuel in the reactor vessel" was taken from the most recent version of MERITS, and was verbally agreed to by the NRC in T/S improvement meetings.

Clarification of Applicability of T/Ss

As a result of the proposed clarification of the definition of Mode 6 and change to the definition of operational mode, we are proposing to add statements to the applicability section of numerous T/Ss that state conditions when the T/Ss are applicable, but the unit is not in a mode.

T/S 3.1.1.3, Bases 3/4.1.1.3

T/S 3.1.1.3 states the LCO for boron dilution. Currently this LCO is only applicable when the units are in a mode. We are proposing to add the following condition under APPLICABILITY, "During movement of irradiated fuel assemblies within containment." When no fuel is in the reactor vessel, reactivity changes in the Reactor Coolant System are not a concern. However, to ensure that the Reactor Coolant System is capable of responding to reactivity changes that may occur if/when irradiated fuel is inserted into the core as a result of an unanticipated event during fuel handling, this condition is being added.

The following statement is being added to Bases 3/4.1.1.3 for clarification, "When no fuel is in the reactor vessel, reactivity changes in the Reactor Coolant System are not a concern. However, prior to any irradiated fuel being moved into containment the minimum flow rate shall be established."

We are also making a correction to the Unit 2 Bases of this section. In AEP:NRC:0916W we requested that the minimum required Reactor Coolant System flow rate for dilution and Mode 6 operation be decreased from 3000 GPM to 2000 GPM. This T/S change was issued on February 9, 1989. While the change in flow rate was made in both Unit 1 and Unit 2 T/Ss 3/4.1.1.3 and 3/4.9.8.1, and the Unit 1 Bases, it was inadvertently not made in the Unit 2 Bases. Consequently, we would like to correct the 3000 GPM stated in Bases 3/4.1.1.3 to 2000 GPM.

T/Ss 3.1.2.1, 3.1.2.3, 3.1.2.5, 3.1.2.7, Bases 3/4.1.2

The T/Ss in 3/4.1.2 address the Boration Systems. Currently, T/Ss 3.1.2.1, 3.1.2.3, 3.1.2.5 and 3.2.1.7 are applicable during Modes 5 and 6. We are proposing to add the following condition to the applicability of each of these T/Ss, "During movement of fuel assemblies within containment."

As explained for the proposed change to T/S 3.1.1.3, this condition is being added to ensure that the Reactor Coolant System is capable of responding to reactivity changes that may result from an unanticipated fuel handling event requiring insertion of fuel into the vessel. However, when no fuel is in the reactor vessel, reactivity changes in the Reactor Coolant System are not a concern.

The following statements are being added to Bases 3/4.1.2 for clarification:

"Negative reactivity control in the Reactor Coolant System is not a concern when no fuel is in the reactor vessel. However, prior to any irradiated fuel being moved into containment the minimum flow paths shall be operable."

"The OPERABILITY of the boron injection system during Mode 6 ensures that this system is available for reactivity control. When no fuel is in the vessel RCS reactivity is not a concern. However, prior to any irradiated fuel being moved into containment the requirements of Specifications 3.1.1.3, 3.1.2.1, 3.1.2.3, 3.1.2.5, 3.1.2.7 shall be met."

T/S 3/4.3.3.1, Table 3.3-6, Item 2

The following condition is being proposed to be added to the APPLICABILITY of Item 2 in T/S 3/4.3.3.1, Table 3.3-6, "Fuel in Containment."

The primary purpose of the Radiation Monitoring Instrumentation listed as Item 2 of Specification 3.3.3.1 is to detect abnormal levels of radiation in the containment building and to actuate equipment to minimize the release of radioactive material to the outside environment in the event of a fuel handling accident within the containment building. Even with no fuel in the vessel, they are needed if fuel is being transported, or is in the containment building for any reason. While in a condition with no fuel in the containment building, the radiation monitoring instrumentation channels are not necessary.

This, as well as changes to Action 22 of this table, are elaborated on in Bases 3/4.3.3.1. The changes to Action 22 are discussed later in this submittal.

Since the page orientation of Table 3.3-6 has been changed all three pages of the table are being submitted.

T/S 3.8.1.2

To ensure that the A.C. electrical power sources that may be needed to mitigate the consequences of a fuel handling accident are available, the following statements are being proposed to be added to the APPLICABILITY of T/S 3.8.1.2:

"During movement of irradiated fuel with no fuel assemblies in the reactor vessel.

During loaded crane operation over irradiated fuel assemblies with no fuel assemblies in the reactor vessel."

In addition, we are requesting that the ACTION statement be modified to read as follows:

"With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes,* movement of irradiated fuel, or crane operation with loads over the fuel storage pool, until the minimum required A.C. electrical power sources are restored to OPERABLE status."

This change to the ACTION statement makes it more restrictive than the current ACTION statement. The current ACTION statement does not require the suspension of movement of irradiated fuel, or crane operation with loads over the fuel storage pool.

T/S 3.8.2.2

To ensure that the A.C. electrical busses that may be needed to mitigate the consequences of a fuel handling accident are available, the following statements are being added to the APPLICABILITY of T/S 3.8.2.2:

"During movement of irradiated fuel with no fuel assemblies in the reactor vessel.

During loaded crane operation over irradiated fuel assemblies with no fuel assemblies in the reactor vessel."

The ACTION statement of this specification is also being changed to reflect these statements. However, this is being addressed later in the submittal since the proposed change goes beyond reflecting the revised APPLICABILITY statement.

T/S 3.8.2.4

As in T/S 3.8.1.2 and T/S 3.8.2.2 the APPLICABILITY statement is being changed to address movement of irradiated fuel and loaded crane operation as follows:

"During movement of irradiated fuel with no fuel assemblies in the reactor vessel.

During loaded crane operation over irradiated fuel assemblies with no fuel assemblies in the reactor vessel."

We are also proposing changes in the ACTION statement to reflect the revised APPLICABILITY statement. However, as in T/S 3.8.2.2, since we are proposing other changes as well to this ACTION statement these changes will be addressed later in this submittal.

T/S 3.9.8.1, Bases 3/4.9.8

T/S 3.9.8.1 states that, "At least one residual heat removal loop shall be in operation" and is applicable in Mode 6. We are proposing to also make it applicable, "During movement of irradiated fuel within containment."

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during Mode 6, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

Likewise, we are proposing to make T/S 3.9.8.2 also applicable "During movement of irradiated fuel within containment." T/S 3.9.8.2 currently states that, "Two independent Residual Heat Removal (RHR) loops shall be OPERABLE" and is applicable in, "MODE 6 when the water level above the top of the reactor pressure vessel flange is less than 23 feet."

We present the justification for both proposed changes in Bases 3/4.9.8 as follows:

"When no fuel is in the vessel the residual heat removal system is not needed to remove decay heat. However, prior to any irradiated fuel being moved into containment the provisions of Specification 3.9.8.1 and 3.9.8.2 shall be met."

Bases 3/4.9.1

We are proposing to add a statement to the Bases to emphasize that T/S 3.9.1 is not applicable when no fuel is in the vessel. It reads as follows:

"When no fuel is in the vessel, maintaining the reactor subcritical and reactivity control in the water volume is not a concern."

Bases 3/4.9.2

When no fuel is in the vessel there is no need for neutron flux monitors. For clarification and emphasis we are proposing to add the following statement to Bases 3/4.9.2, "When no fuel is in the vessel this is not a concern."

Bases 3/4.9.9

T/S 3/4.9.9 states:

"The Containment Purge and Exhaust Isolation System shall be OPERABLE.

APPLICABILITY: During core alterations or movement of irradiated fuel within containment."

We are proposing to add to Bases 3/4.9.9, which addresses this T/S, the following statement for clarification and emphasis:

"When no irradiated fuel is in containment, a radiation hazard that potentially could have environmental consequences is impossible."

Clarification of the Intent of T/Ss

The changes that are being proposed to these T/Ss are not intended to change the meaning of the existing T/Ss, but to clarify them. The way these T/Ss are currently worded, they are subject to interpretations that may put an unreasonable burden on the plant without providing any additional safety. Consequently, we are requesting these changes so that the T/Ss clearly state their intent.

T/S 3.3.3.1, Table 3.3-6, TABLE NOTATION, ACTION 22

We are proposing to add the words, "and Specification 3.9.9 is applicable" to ACTION 22. Specification 3.9.9 states:

"The Containment Purge and Isolation System shall be OPERABLE.

APPLICABILITY: During core alterations or movement of irradiated fuel within the containment."

The addition of this statement to Action 22 emphasizes that when core alterations or movement of irradiated fuel are not taking place the Containment Purge and Exhaust System is not required.

As previously stated, the primary purpose of the Radiation Monitoring Instrumentation listed as Item 2 is to detect abnormal levels of radiation in the containment building and to automatically isolate the Containment Purge and Exhaust valves. However, when no fuel is present within the containment building, protection against a fuel handling accident is unnecessary. Therefore the automatic isolation of the Containment Purge and Exhaust System is not required and the containment RMS Channels listed in Item 2 of Specification 3.3.3.1 are not required to be OPERABLE.

This T/S is further clarified by the addition of the following sentences to Bases 3/4.3.3.1:

"When the Containment Area, Particulate and Noble Gas Radiation Monitoring Instrumentation Channels detect an abnormally high radiation level they initiate containment isolation. However, when no fuel is in containment it is highly unlikely for a radiation hazard that could potentially have environmental consequences to develop."

<u>T/S 3/4.4.7, Table 4.4-3</u>

T/S 3.4.7 states the LCO as:

"The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-1.

<u>APPLICABILITY</u>: At all times."

The SURVEILLANCE REQUIREMENTS of 4.4.7 state:-

"The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-3."

We are proposing to modify TABLE 4.4-3 by making the ** footnote read as follows:

"**Not required when the Reactor Coolant System is drained to half loop and RHR is removed from service."

The reason we are adding the words, "and RHR is removed from service" to this footnote is that when the Reactor Coolant System is at half loop and there is no forced circulation it is impossible to take a representative coolant sample. Any sample that would be obtained would be impacted by the effects of settlement and stagnation.

However, we recognize the fact that T/S 3.4.7 requires that the Reactor Coolant System chemistry be maintained at all times. Consequently, we address this in the paragraph being added to Bases 3/4.4.7. It states:

"Since corrosion inhibitors are added to the reactor coolant, the threat of corrosion is minimized during this condition. Since it is impossible to obtain a representative coolant sample when the Reactor Coolant System is drained to half loop and RHR removed from service, sampling is not required under these conditions. However, prior to fuel being removed from or returned to the reactor vessel the Reactor Coolant chemistry shall be determined to be within the limits by analysis of those parameters specified in Table 3.4-1."

T/S 3.7.2.1

This specification presently reads as follows:

"3.7.2.1 The temperature of both the primary and secondary coolants in the steam generators shall be greater than 70°F when the pressure of either coolant in the steam generator is greater than 200 psig.

APPLICABILITY: At all times."

The SURVEILLANCE REQUIREMENTS go on to state: "The pressure in each side of the steam generator shall be determined to be less than 200 psig at least once per hour when the temperature of either the primary or secondary coolant is less than 70°F." When coolant pressure is low (less than 200 psig) it is impossible to get an accurate pressure measurement, although the operators intuitively know and engineering analysis can determine the approximate pressure. However, accurate measurements can be taken when the pressure is greater than 200 psig.

Based upon this, we are also proposing to change the SURVEILLANCE REQUIREMENTS to read as follows:

"When temperature of either the primary or secondary coolants is less than 70°F, perform one of the following:

a. Verify primary and secondary pressure is less than 200 psig hourly,

b. Verify controls are in place which prevent pressure from exceeding 200 psig on the primary and secondary side once per 12 hours.

This proposed change makes the T/S more conservative since we are adding the SURVEILLANCE REQUIREMENT to verify that controls are in place.

In addition, we are writing out the mathematical symbols in this specification ($>$, $\leq$, $<$).

T/S 3/4.9.8.1

We are proposing to add the following statement to ACTION a.:

"Immediately initiate a corrective action to return at least one residual heat removal loop to OPERABLE status as soon as possible."

It is intuitively obvious that if at least one residual heat removal loop is not available during the applicable operating conditions (Mode 6 and during movement of irradiated fuel within containment) that immediate corrective action should be taken. This has been the position of Cook Nuclear Plant and is being added for clarification and emphasis.

Changes to Reflect Industry Norms

The purpose of the following T/S changes are to make Cook Nuclear Plant's T/Ss reflect the NRC's interpretations and the industry's standard T/Ss.

T/S 3.1.2.3, Bases 3/4.1.2 and T/S 3.4.9.3, Bases 3/4.4.9

We are proposing that T/S 3.1.2.3 ACTION b. be changed from:

"With more than one charging pump OPERABLE or with a safety injection pump(s) OPERABLE when the temperature of any RCS cold leg is less than or equal to 170°F, unless the reactor vessel head is removed, remove the additional charging pump(s) and the safety injection pump(s) motor circuit breakers from the electrical power circuit within one hour."

to:

"With more than one charging pump OPERABLE or with a safety injection pump(s) OPERABLE when the temperature of any RCS cold leg is less than or equal to 170°F, unless the reactor vessel head is not secured to the vessel, remove the additional charging pump(s) motor circuit breakers from the electrical power circuit within one hour."

Likewise we are proposing to change the APPLICABILITY of T/S 3.4.9.3 from :

"When the temperature of one or more of the RCS cold legs is less than or equal to 170°F, except when the reactor vessel head is removed."

to:

"When the temperature of one or more of the RCS cold legs is less than or equal to 170°F, except when the reactor vessel head is not secured to the vessel."

The justification for changing these statements from saying, "the reactor vessel head is removed" to, "the reactor vessel head is not secured to the vessel" is stated in the following paragraph which we propose to add to Bases 3/4.1.2 and Bases 3/4.4.9:

When the vessel head is not secured to the vessel, it is not a pressure barrier. This includes times when the reactor vessel head without the studs is resting on the reactor vessel flange or when the studs are inserted resting on the "mailboxes." (The mailboxes" are devices which prevent the studs from engaging the threads in the reactor vessel flange.)

The pressure required to lift the head off the reactor with the studs removed is 14.29 psig. The pressure required to lift the head off the reactor with the added weight of the studs inserted resting on the mailboxes (approximately 38,000 lbs.) is 16.5 psig. Both of these pressure values are significantly less than the lift setting specified by the T/S for the PORVs and the RHR safety valve. As a result, cold overpressurization is not a concern if the reactor vessel head without studs is just lying on the vessel flange or if the reactor vessel head is lying on the vessel flange with the studs inserted resting on the mailboxes. T/S 3.1.2.3 and 3.4.9.3 are met in either case.

For the intent of meeting Action (b) of T/S 3.1.2.3, having either of the above conditions, i.e., the reactor vessle head with studs removed lying on the vessel flange or the reactor vessel head lying on the vessel flange with the studs inserted resting on the mailboxes, is equivalent to having the reactor vessel head removed.

This T/S clarification as it relates to T/S 3.4.9.3 was discussed with the NRC resident inspector, Mr. B. Jorgensen. Mr. Jorgensen was in agreement with our interpretation. For T/S 3.1.2.3, our interpretation was discussed with Mr. David Passehl - NRC resident inspector and he agreed with our interpretation.

<u>T/S 3.8.2.2 and T/S 3.8.2.4</u>

We are proposing to remove the existing ACTION statement from 3.8.2.2, which reads as follows:

"With less than the above complement of A.C. busses OPERABLE and energized, establish CONTAINMENT INTEGRITY within 8 hours."

In its place, we are proposing to use the ACTION statement from the Westinghouse Standard Technical Specifications Revision 5, which reads as follows:

"With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over the fuel storage pool, and within 8 hours, depressurize and vent the

Reactor Coolant System through a greater than or equal to two square inch vent. In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible."

Similarly, we are proposing to replace the ACTION of T/S 3.8.2.4, which reads as follows:

"With less than the above complement of D.C. equipment and bus OPERABLE, establish CONTAINMENT INTEGRITY within 8 hours."

with:

"With the required battery banks and/or full-capacity charger inoperable, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over the fuel storage pool, initiate corrective action to restore the required battery bank and full-capacity charger to OPERABLE status as soon as possible, and within 8 hours, depressurize and vent the Reactor Coolant System through a greater than two square inch vent."

This statement is identical to that in the Westinghouse Standard Technical Specifications Revision 5, except that we added, "or crane operation with loads over the fuel storage pool." The addition of this statement makes the proposed ACTION more conservative than that in the Standard T/Ss.

It is our position that the proposed ACTION statements enhance plant safety. The current statements call for CONTAINMENT INTEGRITY, but do not require the activities that could possibly lead to a radiation accident to be halted, or that corrective actions be taken. Establishing CONTAINMENT INTEGRITY causes major scheduling problems during an outage, and does not provide any significant safety benefit over the proposed ACTION.

Analysis of Significant Hazards

Per 10 CFR 50.92, a proposed amendment will not involve significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of a previously evaluated accident,

(2) create the possibility of a new or different kind of accident from any previously evaluated, or

(3) involve a significant reduction in a margin of safety.

Criterion 1

The proposed changes do not increase the probability or consequences of a previously evaluated accident. Their intent is to provide clarification and remove ambiguities, and reflect NRC and industry interpretations and norms. They do not affect the accident analysis. Consequently, we believe that these changes do not increase the probability or consequences of a previously analyzed accident.

Criterion 2

The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated. They do not require physical alteration of the plant or changes in parameters governing normal plant operation. We therefore believe these changes do not create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated.

Criterion 3

The proposed changes are consistent with NRC and industry interpretations and norms. We therefore believe the proposed changes do not significantly reduce a margin of safety.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve a significant hazards consideration. The first of these examples refers to changes which are purely administrative. Since the proposed changes are consistent with NRC and industry interpretations and norms, and are intended to provide clarification, we believe these changes fall within the scope of this example.

ATTACHMENT 2 TO AEP:NRC:1107

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