A 4/10/78

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REC: KNIGHTON G W
NRC

ORG: JURGENSEN R W
AMER ELEC PWR SVC

DOCDATE: 03/28/78
DATE RCVD: 04/05/78

DOCTYPE: LETTER NOTARIZED: NO

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SUBJECT:
REQUEST REMOVAL OF ROBERT S. HUNTER FROM MAILLING LIST AND ADDING ROBERT W. JURGENSEN IN HIS PLACE.

PLANT NAME: COOK - UNIT 2
COOK - UNIT 1

REVIEWER INITIAL: XRS
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REGULATORY DOCKET FILE COPY

AMERICAN ELECTRIC POWER *Service Corporation*

AEP

2 Broadway, New York, N. Y. 10004
(212) 422-4800

March 28, 1978

50-315
316

Mr. George W. Knighton, Chief
Environmental Projects Branch No. 1
Division of Site Safety
and Environmental Analysis
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Knighton:

Would you please remove Robert S. Hunter from your mailing list and add Robert W. Jurgensen in his place. Mr. Jurgensen is Chief Nuclear Engineer, Nuclear Engineering Division.

Very truly yours,

P. L. Bonomo for
R. W. Jurgensen
Chief Nuclear Engineer

RWJ:clb

780970238

8007
ES
1/0
ADD SCHWENCER
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NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

DOCKET NUMBER 50-315/316

FILE NUMBER

NRC DISTRIBUTION FOR PART 50 DOCKET MATERIAL

TO: Mr. Edson G. Case

FROM: Westinghouse Elect Corp
Pittsburgh, Pa. 15230
M. H. Judkis

DATE OF DOCUMENT 03/06/78

DATE RECEIVED 05/04/78

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DESCRIPTION

Revised 5/18/78 J.W.

1p

ENCLOSURE

Forwarding Amend 81 to the Final Safety Analysis Rept consisting of revisions to the FSAR to reflect the steam line break protection changes....

5p

PLANT NAME: DONALD C COOK UNITS 1 & 2
jcm 05/05/78

Dist Per M. Mlynczak 5/4/78

70 ENCL

FOR ACTION/INFORMATION

	SAFETY			ENVIRONMENTAL
/	ASSIGNED AD: LTR	VASSALLO		ASSIGNED AD: V. MOORE (LTR)
/	BRANCH CHIEF: LTR	KNIEL		BRANCH CHIEF:
/	PROJECT MANAGER:	MLYNCZAK		PROJECT MANAGER:
/	LIC. ASST: LTR	J. LEE		LIC. ASST:
				B. HARLESS

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	P. COLLINS		NOVAK		BUTLER		
	HOUSTON		ROSZTOCZY		GRIMES		SITE ANALYSIS
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AEW-7095

Westinghouse Electric Corporation   Power Systems

PWR Systems Division

Box 355
Pittsburgh Pennsylvania 15230

March 6, 1978
S.0. AMP-460

Mr. Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, Maryland 20014

AMERICAN ELECTRIC POWER PROJECT
Donald C. Cook FSAR Amendment 81

Dear Mr. Case:

Enclosed please find 70 copies of FSAR Amendment 81 applicable to the Donald C. Cook Docket 50-315 and 50-316. The legal papers for this amendment are being forwarded by American Electric Power Service Corporation by a separate cover letter.

This amendment revises the FSAR to reflect the steam line break protection changes.

Very truly yours,

M. H. Judkis, Manager
American Electric Power Projects

M. Oper/je
Attachment

cc: R. W. Jurgensen 1L, 250A
R. S. Hunter 1L
R. F. Hering 1L
S. H. Horowitz 1L
S. J. Milioti 1L
J. G. Feinstein 1L

7812500022

Revised 5/18/78
Charles J. Wilhelm

DONALD C. COOK
NUCLEAR PLANT
AMENDMENT 81
INSTRUCTION SHEET

REMOVE	INSERT
FRONT/BACK	FRONT/BACK
14.2.5-1/14.2.5-2	14.2.5-1/14.2.5-2
14.2.5-3/14.2.5-4	14.2.5-3/14.2.5-3

Supersed per AMDT # 81 to FSAR 50-315
Dtd 3/6/78 MCN 7812500022

14.2.5 RUPTURE OF A STEAM PIPE

1. DISCUSSION OF ACCIDENT

A rupture of a steam pipe results in an uncontrolled steam release from a steam generator. The steam release results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the Reactor Coolant System causes a reduction of coolant temperature and pressure. In the presence of a negative coolant temperature coefficient, the cooldown results in a reduction of core shutdown margin. If the most reactive RCCA is assumed stuck in its fully withdrawn position, there is an increased possibility that the core will become critical and return to power. A return to power following a steam pipe rupture is a potential problem mainly because of the high hot channel factors which exist when the most reactive assembly is assumed stuck in its fully withdrawn position. Assuming the most pessimistic combination of circumstances which could lead to power generation following a steam line break, the core is ultimately shut down by boric acid delivered by the Emergency Core Cooling System.

The analysis of a steam pipe rupture is performed to demonstrate that:

1) Assuming a stuck assembly, with or without offsite power, and assuming a single failure in the engineered safety features there is no consequential damage to the primary system and the core remains in place and intact.

2) Energy release to the containment from the worst steam pipe break does not cause failure of the containment structure.

3) There will be no return to criticality after reactor trip, for a break equivalent to the spurious opening, with failure to close, of the largest of any single steam dump, relief or safety valve.

The following systems provide the necessary protection against a steam pipe rupture:

1) Safety Injection System actuation from any of the following:

   a) One out of three coincident low pressurizer pressure and low pressurizer level signals.

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

   c) High steam line flow in two out of four main steam lines (one out of two per line), in coincidence with either low Reactor Coolant System average temperature (two out of four loops) or low main steam line pressure (two out of four lines).

   d) Two out of three high containment pressure signals.

2) The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the Safety Injection Signal.

3) Redundant isolation of the main feedwater lines: Sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action which will close the main feedwater valves, a safety injection signal will rapidly close all feedwater control valves, trip the main feedwater pumps, and close the feedwater pump discharge valves.

4) Trip of the fast acting steam line stop valves (designed to close in less than 5 seconds) on:

a. High steam flow in any two steam lines in coincidence with either low Reactor Coolant System average temperature or low steam line pressure.

b. High containment pressure.

Each steam line has a fast-closing stop valve capable of stopping flow in either direction. These four valves prevent blowdown of more than one steam generator for any break location even if one valve fails to close. For example, in the case of a break upstream of the stop valve in one line, closure of any three stop valves will prevent blowdown of the other steam generators. In particular, the arrangement precludes blowdown of more than one steam generator inside the containment and thus prevents structural damage to the containment. In addition each main steam line incorporates a 16 inch diameter venturi type flow restrictor which is located inside the containment. These components serve to limit the rate of release of steam for an outside break.

2. METHOD OF ANALYSIS

The analysis of the steam pipe rupture has been performed to determine:

1) The core heat flux and Reactor Coolant System temperature and pressure resulting from the cooldown following the steam line break. A full plant digital computer simulation has been used.

2) The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital-computer calculation has been used to determine if DNB occurs for the core conditions computed in (1) above.

3) The containment pressure response to a large area break.

The following conditions were assumed to exist at the time of a steam break accident:

1) A 1.6% end of life shut down margin at no-load, equilibrium Xenon conditions, and with the most reactive assembly stuck in its fully withdrawn position: Operation of the control rod banks during core burnup is restricted in such a way that addition of positive reactivity in a steam break accident will not lead to a more adverse condition than the case analyzed.

2) The negative moderator coefficient of reactivity corresponding to the end of life rodded core with the most reactive assembly in the fully withdrawn position. The variation of the coefficient with temperature and pressure has been included. The k_{eff} versus coolant temperature corresponding to the negative moderator temperature coefficient used is shown in Figure 14.2.5-1. In computing the power generation following a steam line break, the local reactivity feedback from the high neutron flux in the region of the core near the stuck RCC assembly has been included in the overall reactivity balance. The local reactivity feedback is composed of the doppler reactivity from the high fuel temperatures near the stuck RCC assembly and moderator feedback from the high water enthalpy near the stuck assembly. The effect of power generation in the core on the total core reactivity is shown in Figure 14.2.5-2.

3) Minimum capability for injection of high concentration boric acid solution corresponding to the most restrictive single failure in the Safety Injection System. The injection curve used is shown on Figure 14.2.5-3. This corresponds to the flow delivered by one high head centrifugal charging pump delivering its full contents to the cold leg header. Low concentration boric acid (2000 ppm) must be swept from the injection lines downstream of the Boron Injection Tank isolation valves prior to the delivery of high concentration boric acid (20,000 ppm) to the main coolant loops. This effect has been allowed for in the analysis.