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50-316 Donald C. Cook Nuclear Power Plant, Unit 2, Indiana & 05000316
AUTH.NAME AUTHOR AFFILIATION
ALEXICH,M.P. Indiana & Michigan Electric Co.
RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. Office of Nuclear Reactor Regulation, Director

SUBJECT: Forwards Westinghouse evaluation of impact of reduced auxiliary feedwater flow for Unit 1 & results of reanalysis performed for Unit 2.Feedwater sys acceptable for both units.

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INDIANA & MICHIGAN ELECTRIC COMPANY

P.O. BOX 16631
COLUMBUS, OHIO 43216

October 30, 1984
AEP:NRC:0300H

Donald C. Cook Nuclear Plant Unit Nos. 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
REVISED AUXILIARY FEEDWATER FLOW ANALYSES

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

Dear Mr. Denton:

Indiana & Michigan Electric Company (IMECo) letter No. AEP:NRC:0300C, dated November 3, 1980, provided information with regard to the flow design basis of the Donald C. Cook Nuclear Plant auxiliary feedwater system. It has recently been discovered, however, that an error was made in the input to the calculations supporting that earlier letter. The discrepancy has now been corrected, and the results of the revised flow calculations are expected to be transmitted to the NRC in the near future.

It is noted, however, that the revised calculations predict that for a feedwater line break in Unit 1 (for which the feedwater line break was not part of the original licensing basis), approximately 365 gallons per minute (gpm) of auxiliary feedwater flow could be delivered to the intact steam generators prior to operator action; the corresponding auxiliary feedwater flow value for Unit 2 is 375 gpm. Since these flow values are lower than those used in the Final Safety Analysis Report (FSAR) for Unit 2, we have contracted with Westinghouse Electric Corporation (W) to perform a reanalysis of the feedwater line break accident. W's evaluation of the impact of reduced auxiliary feedwater flow for Unit 1, and the results of the reanalysis performed for Unit 2, are presented in the Attachment to this letter.

Based on the reanalysis performed for Unit 2 with the LOFTRAN computer code, W has concluded that the plant meets the FSAR acceptance criteria assuming the auxiliary feedwater system delivers 375 gpm to the intact steam generators. More specifically, the analysis indicates that the Reactor Coolant System (RCS) and main steam system will remain below 110% of their respective design pressures, and that the RCS water level will not drop below the top of the core. Therefore, core integrity would be maintained and the radiological consequences would be only a small fraction of the 10 CFR 100 guidelines.

A001
1/1

The W evaluation performed for Unit 1 indicates that operator action (i.e., W action #1) at ten (10) minutes following a postulated feedwater line break would restore heat removal capability. This heat removal capability would be provided via an increase in auxiliary feedwater flow to approximately 460 gpm, assuming the availability of control air to the emergency leakoff valves. Operator actions taken in accordance with actions #1 and #2 of the W evaluation will ensure auxiliary feedwater flow well in excess of 460 gpm (approaching approximately 1200 gpm at the first safety valve setpoint). It is noted that although the feedwater line break is not part of the original Unit 1 licensing basis, the licensing basis does assume operator action at ten (10) minutes following a high energy line break for the purposes of maintaining containment integrity. With this action, W believes that the consequences of a Unit 1 feedwater line break would be bounded by the conclusions provided in the FSAR of plants similar to Cook Plant.

Additionally, it is noted that IMECo has taken the conservative step of temporarily reducing Cook Plant thermal power for both Units while the analysis and evaluation was being performed by W. The enclosed information from W indicates that the licensing basis for Unit 1 was not exceeded and, for Unit 2, the public health and safety would not be endangered by a return to full power operation, based upon an assumed auxiliary feedwater flow of 375 gpm to the intact steam generators following a postulated feedwater line break. We expect to return Unit 2 to full power operation upon concurrence of the staff of the Office of Nuclear Reactor Regulation. We believe the attached evaluation justifies the return of Unit 1 to full power operation.

This document has been prepared following Corporate procedures which incorporate a reasonable set of controls to ensure its accuracy and completeness prior to signature by the undersigned.

Very truly yours,

M. P. Alexich JBS 12/20/84
Vice President

MPA/dam

cc: John E. Dolan
W. G. Smith, Jr. - Bridgman
R. C. Callen
G. Charnoff
NRC Resident Inspector at Cook Plant - Bridgman
G. Bruchmann

bc: J. G. Feinstein/P. A. Barrett/H. Y. Fouad/D. A. Medek
H. N. Scherer, Jr.
J. J. Markowsky
T. O. Argenta
S. H. Steinhart/J. A. Kobyra
R. W. Jurgensen
J. B. Shinnock
R. F. Kroeger
T. P. Beilman - Bridgman
J. F. Stietzel - Bridgman
D. L. Wigginton - NRC
DC-N-6015.1
DC-N-6110

ATTACHMENT TO AEP:NRC:0300H
WESTINGHOUSE ELECTRIC CORPORATION EVALUATION AND ANALYSIS
DONALD C. COOK NUCLEAR PLANT UNIT NOS. 1 AND 2

To: W. G. Smith 1/4

Customer Reference No(s).

AWS-(162)

Westinghouse Reference No(s).
(Change Control or RFQ As Applicable)

WESTINGHOUSE

NUCLEAR SAFETY EVALUATION CHECK LIST

1) NUCLEAR PLANT(S) D. C. COOK UNIT 1

2) CHECK LIST APPLICABLE TO: Impact of Reduced AFW Flow on Hypothetical Feedline Break Transient at D. C. Cook Unit
(Subject of Change)

3) The written safety evaluation of the revised procedure, design change or modification required by 10CFR50.59 has been prepared to the extent required and is attached. If a safety evaluation is not required or is incomplete for any reason, explain on Page 2.

Parts A and B of this Safety Evaluation Check List are to be completed only on the basis of the safety evaluation performed.

CHECK LIST - PART A

(3.1) Yes ___ No _X_ A change to the plant as described in the FSAR?

(3.2) Yes ___ No _X_ A change to procedures as described in the FSAR?

(3.3) Yes ___ No _X_ A test or experiment not described in the FSAR?

(3.4) Yes ___ No _X_ A change to the plant technical specifications (Appendix A to the Operating License)?

4) CHECK LIST - PART B (Justification for Part B answers must be included on Page 2.)

(4.1) Yes ___ No _X_ Will the probability of an accident previously evaluated in the FSAR be increased?

(4.2) Yes ___ No _X_ Will the consequences of an accident previously evaluated in the FSAR be increased?

(4.3) Yes ___ No _X_ May the possibility of an accident which is different than any already evaluated in the FSAR be created?

(4.4) Yes ___ No _X_ Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

(4.5) Yes ___ No _X_ Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

(4.6) Yes ___ No _X_ May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?

(4.7) Yes ___ No _X_ Will the margin of safety as defined in the bases to any technical specification be reduced?

2/4

If the answers to any of the above questions are unknown, indicate under 5) REMARKS and explain below.

If the answer to any of the above questions in 4) cannot be answered in the negative, based on written safety evaluation, the change cannot be approved without an application for license amendment submitted to NRC pursuant to 10CFR50.90.

5) REMARKS:

The following summarizes the justification upon the written safety evaluation,(1) for answers given in Part B of the Safety Evaluation Check List:

See letter NS-RAT-PTA-84-111

(1) Reference to document(s) containing written safety evaluation: ______________________

FOR FSAR UPDATE

Section: __________ Page(s): __________ Table(s): __________ Figure(s): __________

Reason for/Description of Change:

NOT APPLICABLE

Prepared by (Nuclear Safety): Mark [signature] Date: 10/27/84

Coordinated with Engineer(s): [signature] Date: 10/27/84

Coordinating Group Manager(s): P.A.Loftus Date: 10/27/84

Nuclear Safety Group Manager: PA Loftus for MPO Date: 10/27/84

3/4

MS-OPLS-OPA-84-113
MS-RAT-PTA-84-111
OCTOBER 27, 1984

TO: D.P. DOMINICIS
A. SUDA

RE: Impact of Reduced AFW Flow on Feedline Break Transient for D. C. Cook Unit 1.

American Electric Power informed Westinghouse of an error in the auxiliary feedwater flow calculations for the feedline break analysis of D. C. Cook Unit 2. This error was found to apply to both units; however, Unit 1 does not have a feedline break analysis as part of its licensing basis and explicit calculations are not required. The following provides a safety evaluation of what action would be necessary to mitigate the consequences of a hypothetical feedline break at Unit 1.

The D.C. Cook Unit 1 design basis for operator action following a high energy line break is 10 minutes (cf. AEP-80-80). Thus credit for the operators to initiate the appropriate actions at 10 minutes is justified.

Should a feedline break occur at Unit 1 the reactor protection system would respond as follows:

> The reactor would trip on a steam generator low water level coincident with steam/feed water flow mismatch or steam generator low-low water level. When the steam generator low-low level is reached in one steam generator, the motor driven auxiliary feed pumps (MDP) start. The worst single failure for a feedline break is a failure of the MDP which feeds two intact steam generators. Following reactor trip and subsequent turbine trip, the low-low water level in a second steam generator would be reached and the turbine driven auxiliary feed pump (TDAFWP) is started. As the steam generators depressurize, the steam supply to the TDAFWP is reduced and eventually lost. Thus once the steam supply to the TDAFWP is lost, only a minimal amount of auxiliary feedwater flow would be available for heat removal.

Operator action should be performed to restore heat removal capability, i.e. 450 gpm to the intact steam generators. Available operator actions are as follows:

1. He can isolate AFW flow to the faulted SG and defeat the flow retention logic for the intact steam generator fed by the same MDP. This results in one MDP delivering 460 gpm to the intact steam generator. This action is similar to action required by other plants similar to D.C. Cook Unit 1 that have a feedline break analysis as part of their design basis.

2. He can restore steam supply to the TDAFWP. This can be accomplished by closing the main steamline isolation valve on the faulted steam generator or the MSIVs for the lines that connect to the TDAFWP.

CONCLUSION

With the assumption of action 1 above at 10 minutes, Westinghouse believes the feedline break consequences would be bounded by the conclusions (RCS pressure less than 110% of the design pressure, core integrity maintained, and dose releases are a small fraction of the 10 CFR Part 100 guidelines) provided in the FSAR of plants similiar to D. C. Cook since the D. C. Cook AFW system design assures AFW flow to the intact steam generators prior to the 10 minute operator action time. Thus provided operator action results in an adequate heat sink, Westinghouse sees no safety issue regarding the reduced AFW flow for a hypothetical feedline break.

M R Adler
M. R. ADLER
Plant Transient Analysis

Donald S Love
D. S. LOVE
Operating Plant Analysis

M. P. Osborne for
M. P. OSBORNE, Manager
Plant Transient Analysis

P. A. Loftus
P. A. Loftus, Manager
Operating Plant Analysis

NS-RAT-PTA-84-112

FROM: NS PTA
WIN: 284-5685
DATE: October 29, 1984
SUBJECT: D. C. COOK UNIT 2 (AMP) FEEDLINE BREAK REANALYSIS

TO:	D. P. Dominicis	MNC 4-01
	A. P. Suda	701- 25
CC:	A. L. Sterdis	MNC 4-01
	P. A. Loftus	MNC 4-09A
	J. L. Little	MNC 4-09A
	M. J. Parvin	Westinghouse Site Manager
	D. S. Love	MNC 4-09A
	D. C. Richardson	MNC 4-15

Please find attached the report documenting the feedline break reanalysis assuming 375 gpm of auxiliary feedwater flow to the 3 intact steam generators for D. C. Cook Unit 2. As shown in the attached, the FSAR acceptance criteria of no substantial overpressurization of the reactor coolant system and the core remains covered are met.

Should you have any questions the undersigned may be contacted.

This expedited effort was charged to shop order AKKP-485.

M. R. Adler

M. R. Adler
Plant Transient Analysis

Approved: MPOsborne
M. P. Osborne, Manager
Plant Transient Analysis

ATTACHMENT

BACKGROUND

The FSAR feedline break analysis for D. C. Cook Unit 2 was performed assuming 450 gpm of auxiliary feedwater (AFW) flow is delivered to the intact steam generators. The following demonstrates that the plant meets the FSAR acceptance criteria (described on page 14.2.8-2 of the FSAR) assuming the AFW system delivers 375 gpm to the intact steam generators.

METHOD OF ANALYSIS

This analysis was performed using the same methodology and assumptions as stated in the FSAR with the following exceptions.

The LOFTRAN[1] computer code was used. This code simulates the plant thermal kinetics, reactor coolant system (RCS) including natural circulation, pressurizer, steam generators, and feedwater system, and computes pertinent variables including the pressurizer pressure, pressurizer water level, and reactor coolant temperature.

Reactor trip and auxiliary feedwater initiation are assumed to occur on low-low water level in the faulted steam generator.

The most restrictive single failure in the AFW system is assumed. This is the loss of the motor driven auxiliary feedwater pump that normally would have supplied feedwater to two intact steam generators in one minute following reactor trip. The other motor driven auxiliary feedwater pump and the turbine driven auxiliary feedwater pump deliver a total of 375 gpm to the three intact steam generators within one minute following the reactor trip. An additional 280 seconds is assumed before the feed lines are purged and the relatively cold (120°F) auxiliary feedwater enters the intact steam generators.

RESULTS

Figures 1-7 illustrate the key plant parameters calculated following a feedline rupture. The calculated sequence of events are listed in Table 1. Results presented in Figures 4 and 6 show that pressures in the RCS and main steam system remain below 110% of the respective design pressures. Pressurizer pressure increase until reactor trip on low-low steam generator level. Pressure then decreases, due to the loss of heat input, until steamline isolation occurs on low steamline pressure in the faulted loop. Coolant expansion occurs due to reduced heat transfer capability in the steam generators; the pressurizer safety valves open to maintain primary coolant system pressure at an acceptable value.

The reactor core remains covered with water throughout the transient, as water relief due to thermal expansion is limited by the heat removal capability of the auxiliary feedwater system.

The assumed auxiliary feedwater flow rate is capable of removing all of the decay heat ~3200 seconds after reactor trip. After this time, core decay heat decreases below the auxiliary feedwater heat removal capacity and reactor coolant temperatures and pressure decrease.

<u>CONCLUSIONS</u>

Results of the analysis show that for the postulated feedline rupture, the assumed auxiliary feedwater system capacity (375 gpm) is adequate to remove decay heat and to prevent the water level in the RCS from dropping to the top of the core.

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

TABLE 1: TIME SEQUENCE OF EVENTS

EVENT	TIME, seconds
Feedline Rupture Occurs	10.0
Reactor trip on Low-Low steam generator water level	15.4
Rod motion and power lost to the reactor coolant pumps	17.4
Auxiliary feedwater is delivered to 3 intact steam generators	75.4
Low steamline pressure setpoint reached	149.4
Steamline isolation occurs	157.4
Pressurizer safety valve setpoint reached	612.0
Steam generator safety valve setpoint reached in intact steam generators reached	700.0
Core decay heat decreases to auxiliary feedwater heat removal capacity	~3200

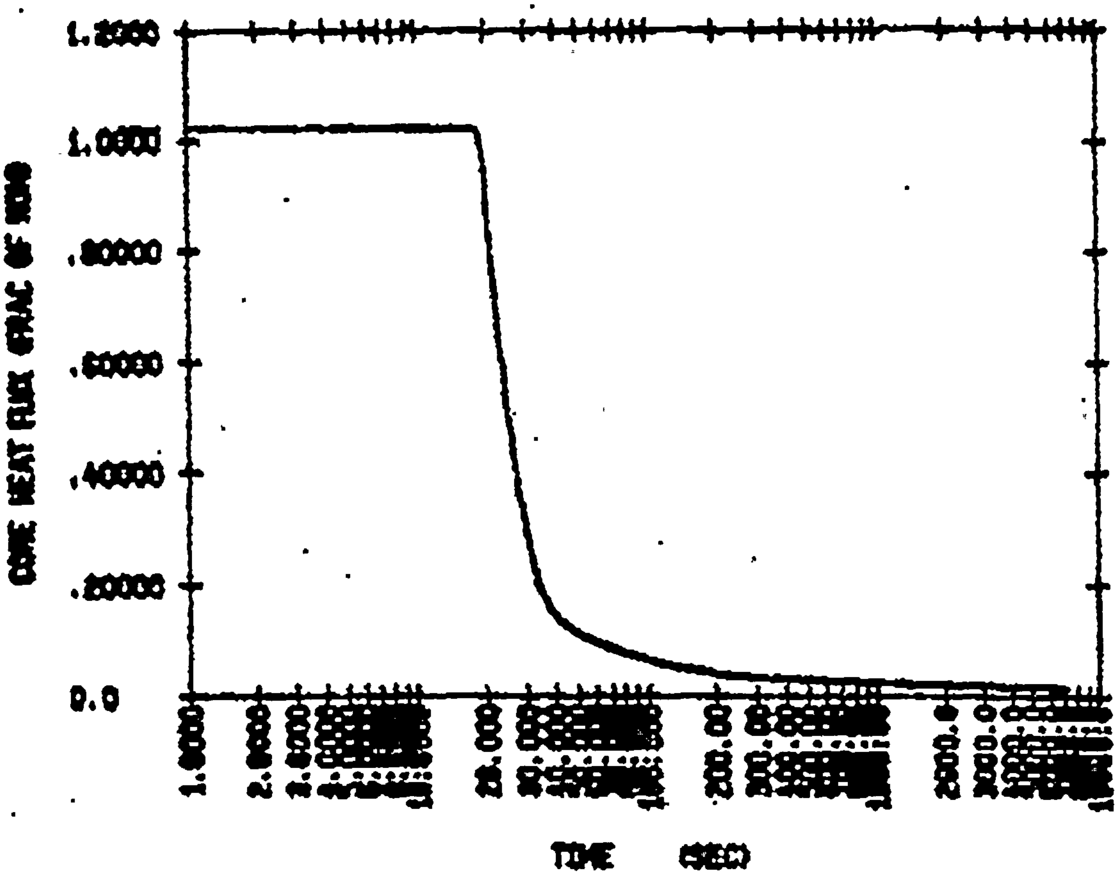

FIGURE 1: MAIN FEEDLINE RUPTURE ACCIDENT CORE HEAT FLUX VERSUS TIME

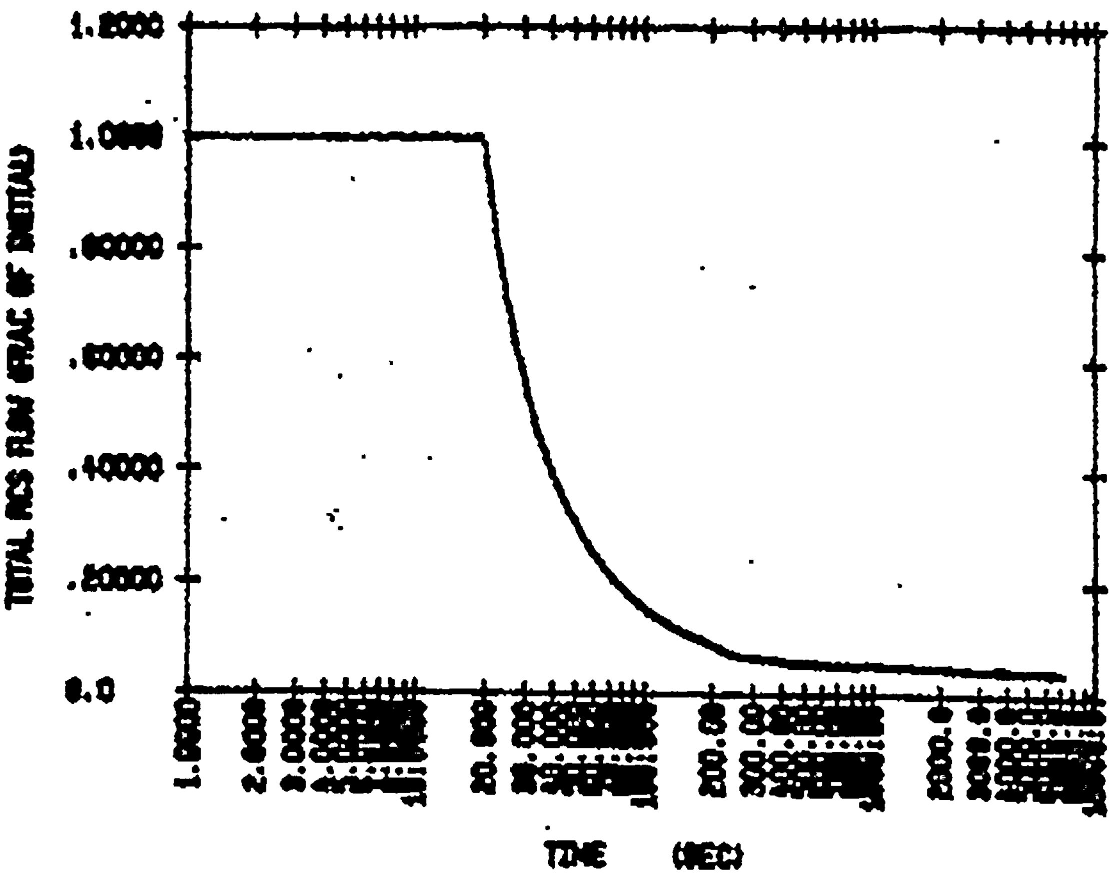

FIGURE 2: MAIN FEEDLINE RUPTURE ACCIDENT REACTOR COOLANT FLOW VERSUS TIME

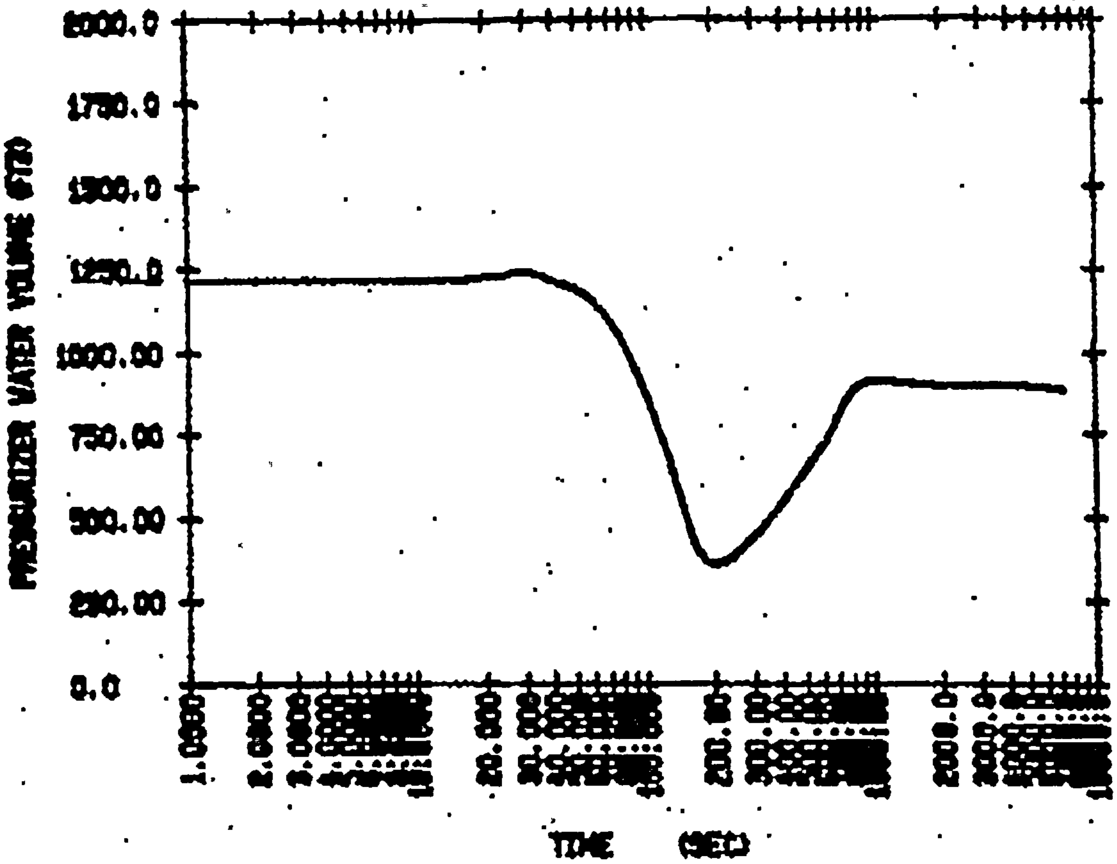

FIGURE 3: MAIN FEEDLINE RUPTURE ACCIDENT PRESSURIZER WATER VOLUME VERSUS TIME

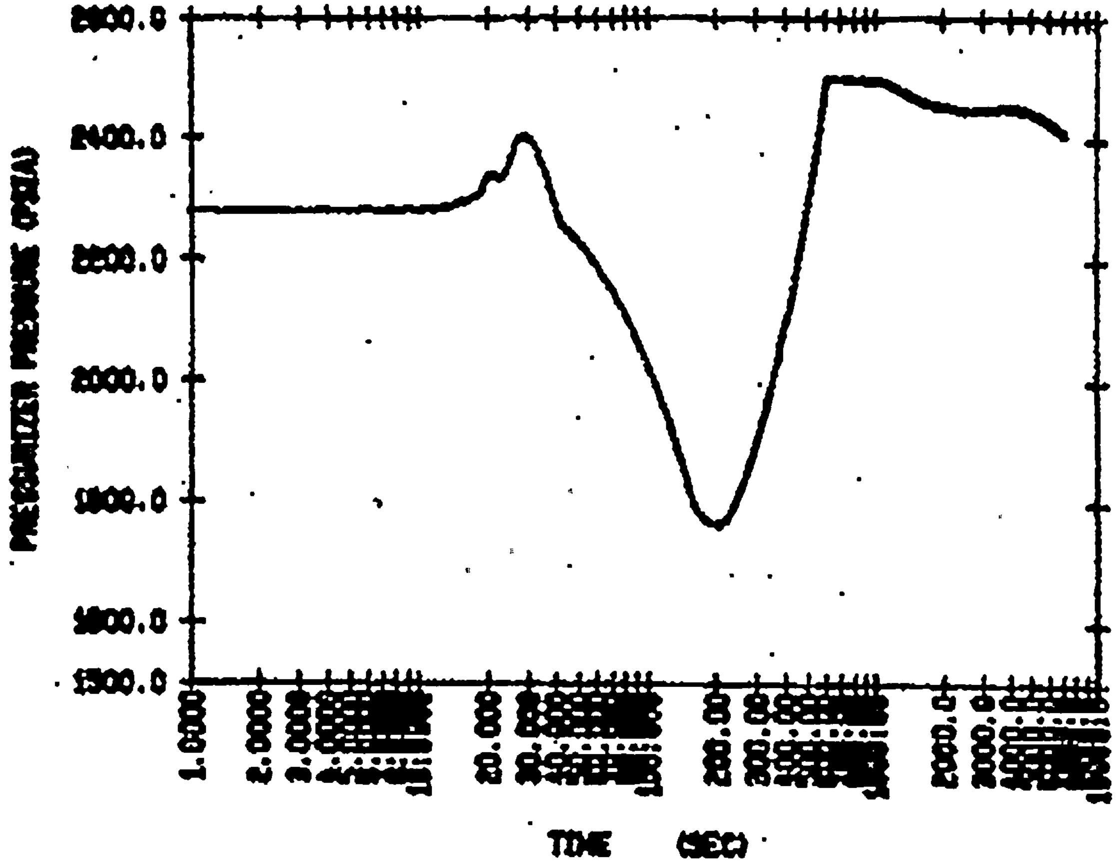

FIGURE 4: MAIN FEEDLINE RUPTURE ACCIDENT PRESSURIZER PRESSURE VERSUS TIME

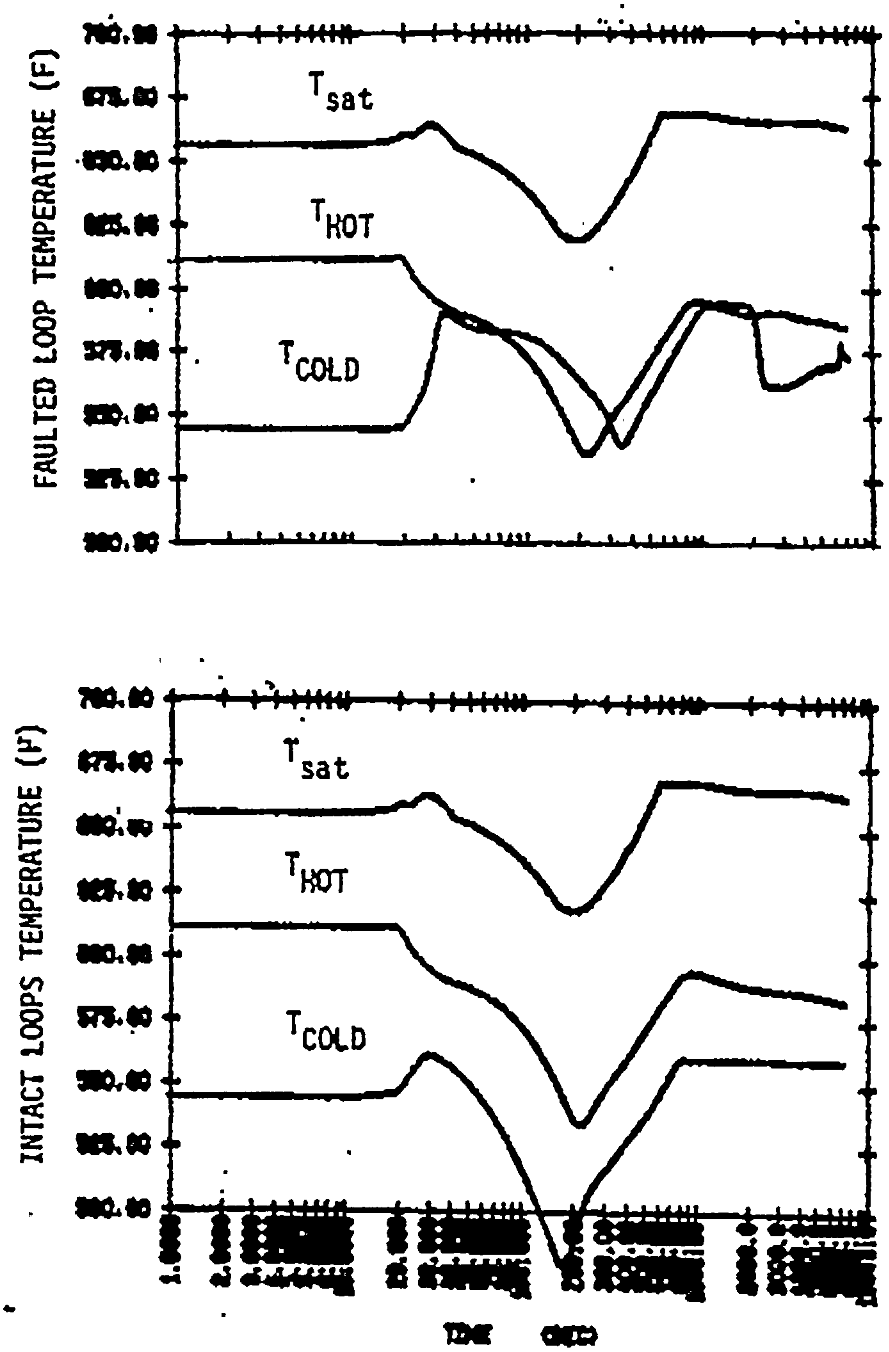

FIGURE 5: MAIN FEEDLINE RUPTURE ACCIDENT FAULTED AND INTACT LOOPS RCS TEMPERATURES VERSUS TIME

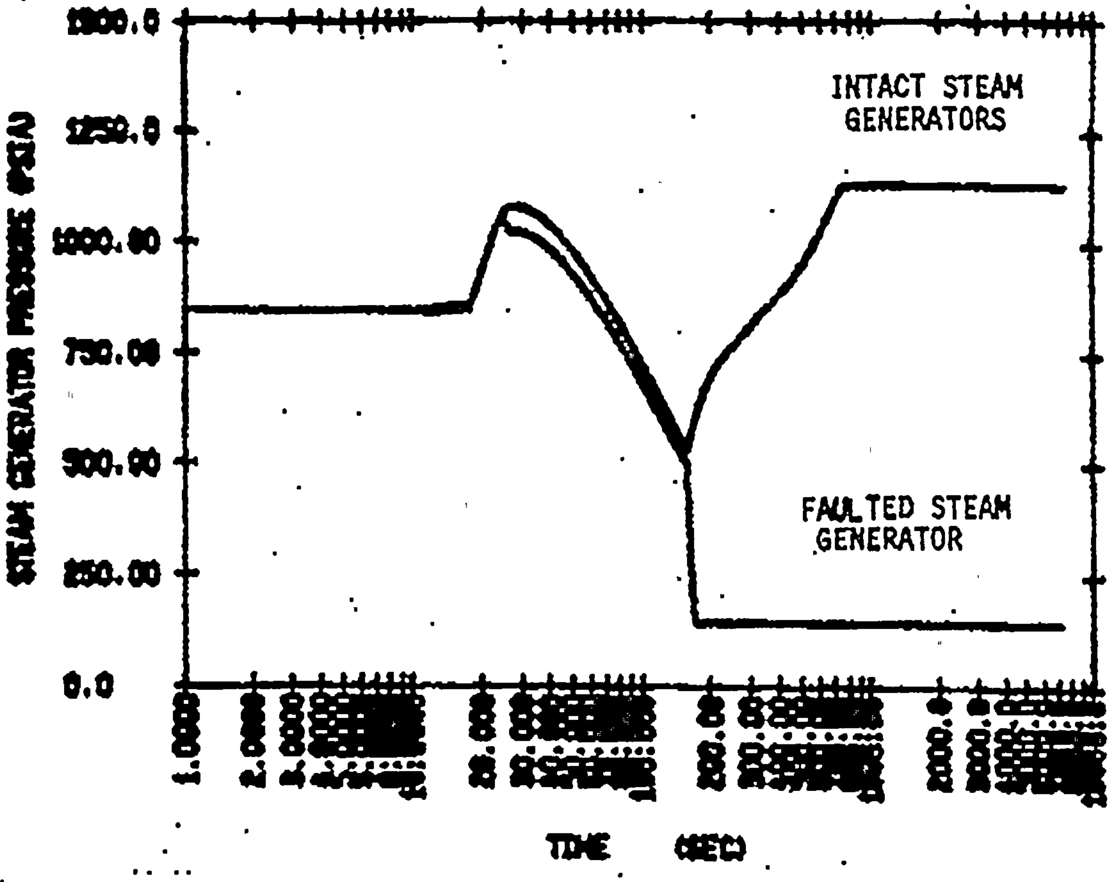

FIGURE 6: MAIN FEEDLINE RUPTURE ACCIDENT STEAM GENERATOR PRESSURE VERSUS TIME

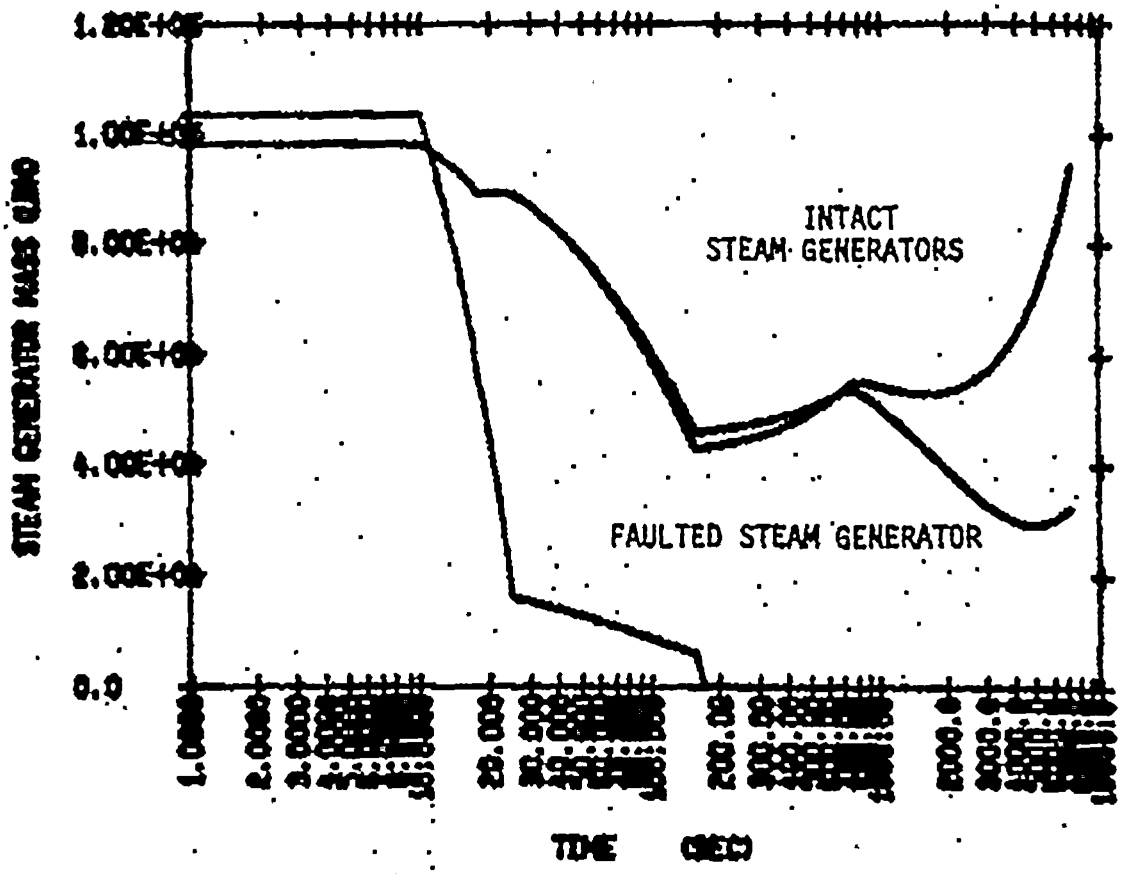

FIGURE 7: MAIN FEEDLINE RUPTURE ACCIDENT STEAM GENERATOR MASS VERSUS TIME

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October 25, 1984

DOCKET NO(S). 50-315
and 50-316

Mr. John Dolan, Vice President
Indiana and Michigan Electric Company
c/o American Electric Power
Service Corporation
1 Riverside Plaza
Columbus, Ohio 43215

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

The following documents concerning our review of the subject facility are transmitted for your information.

☐ Notice of Receipt of Application, dated ______.

☐ Draft/Final Environmental Statment, dated ______.

☐ Notice of Availability of Draft/Final Environmental Statement, dated ______.

☐ Safety Evaluation Report, or Supplement No. ______, dated ______.

☐ Notice of Hearing on Application for Construction Permit, dated ______.

☐ Notice of Consideration of Issuance of Facility Operating License, dated ______.

☐ Monthly Notice; Applications and Amendments to Operating Licenses Involving no Significant Hazards Considerations, dated ______.

☐ Application and Safety Analysis Report, Volume ______.

☐ Amendment No. ______ to Application/SAR dated ______.

☐ Construction Permit No. CPPR- ______, Amendment No. ______ dated ______.

☐ Facility Operating License No. ______, Amendment No. ______, dated ______.

☐ Order Extending Construction Completion Date, dated ______.

XX Other *(Specify)* Monthly Notice covering period through October 24, 1984. Expiration date for hearing requests and comments November 26, 1984.

Division of Licensing
Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc: w/enclosure

OFFICE	ORB#1:DL						
SURNAME	CParrish/ts						
DATE	10/26/84						

NRC FORM 318 (1/84) NRCM 0240