REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR: 8705260060 DOC. DATE: 87/05/13 NOTARIZED: NO DOCKET #
FACIL: 50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana & 05000315
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
MURLEY, T. E.	NRC - No Detailed Affiliation Given

SUBJECT: Requests review of util justification for Tech Spec interpretation re operability of ECCS subsystem. Safety analyses & response to NRC 870401 questions encl. NUREG 0737 requirements met. Response requested by 870615. Fee paid.

DISTRIBUTION CODE: A001D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 43
TITLE: OR Submittal: General Distribution

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD3-3 LA	1	0	PD3-3 PD	5	5
	WIGGINGTON, D	1	1			
INTERNAL:	ARM/DAF/LFMB	1	0	NRR/DEST/ADE	1	1
	NRR/DEST/ADS	1	1	NRR/DOEA/TSB	1	1
	NRR/PMAS/ILRB	1	1	OGC/HDS1	1	0
	REG FILE 01	1	1			
EXTERNAL:	EG&G BRUSKE, S	1	1	LPDR	1	1
	NRC PDR	1	1	NSIC	1	1

Rec'd w/check
057-0344

TOTAL NUMBER OF COPIES REQUIRED: LTTR 18 ENCL 15

INDIANA & MICHIGAN ELECTRIC COMPANY

P.O. BOX 16631
COLUMBUS, OHIO 43216

May 13, 1987
AEP:NRC:1024A

Donald C. Cook Nuclear Plant Unit Nos. 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
REQUEST FOR REVIEW OF SELECTED TECHNICAL INFORMATION REGARDING TECHNICAL SPECIFICATIONS FOR OPERATION WITH LIMITED SI OR RHR FLOW PATHS

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Attn: T. E. Murley

Dear Mr. Murley:

The purpose of this letter and its attachments is to provide justification to allow Technical Specifications (T/Ss) 3.5.2.e and 3.5.3.d for both units of the Donald C. Cook Nuclear Plant to be interpreted such that the flow path portion of one ECCS subsystem is considered operable when one safety injection pump or one residual heat removal pump can deliver flow to only two reactor coolant system loops while the other pump is aligned to deliver flow to four loops. This letter is to follow up on the program described in our previous submittal AEP:NRC:1024, dated March 16, 1987. This subject was also discussed in our letter AEP:NRC:0916Z, dated April 23, 1987, which included a draft version of Attachment 1 to this letter. We are asking that the interpretation be extended to any configuration (such as servicing an RHR heat exchanger) where flow from a single operating RHR or an SI pump can be delivered to only two RCS loops with the other pump aligned to deliver flow to four loops. This issue was discussed with members of the NRR staff and members of the Region III staff and was in part the subject of an Enforcement Conference held at Region III headquarters on January 21, 1987.

We believe that the attachments to this letter, along with the safety analysis in Attachment 4 to our previous submittal AEP:NRC:1024, cover the accident analyses related to closing the RHR or SI cross-ties on either unit.

Attachment 1 to this letter presents a safety analysis of the small-break loss-of-coolant accident (SBLOCA) for Unit 1 performed by Westinghouse Electric Corporation. The analysis assumes that one charging

REC'D W/CHECK
#057-0344

A001
1/1

pump delivers flow to four reactor coolant loops and one safety injection pump delivers flow to two loops, one of which spills out a four-inch break. The peak clad temperature resulting from this arrangement is 1427^{o}F, which is well below the 10 CFR 50.46 limit of 2200^{o}F. The peak clad temperature was calculated using a reference plant with a core representative of D. C. Cook Unit 1 with the NOTRUMP computer code. We believe this meets the requirements of NUREG-0737, Items II.K.3.30 and II.K.3.31.

Attachment 2 to this letter presents a safety analysis of the large-break loss-of-coolant accident (LBLOCA) for Unit 1, also performed by Westinghouse Electric Corporation. The analysis assumes that one residual heat removal pump delivers flow to two RCS loops and that one safety injection pump and one charging pump each deliver flow to four RCS loops. The peak clad temperature resulting from this arrangement is 1940^{o}F. The limiting case for LBLOCA is the maximum safeguards case with a calculated PCT of 2154^{o}F.

Attachment 3 to this letter presents a safety analysis of the LBLOCA for Unit 2 which was performed by Advanced Nuclear Fuels Corporation (formerly Exxon Nuclear Company). The analysis assumes that one residual heat removal pump delivers flow to two RCS loops, and that one centrifugal charging pump and one safety injection pump deliver flow to four RCS loops. The peak clad temperature resulting from this arrangement is 1988^{o}F. The previous limiting case for LBLOCA is the maximum safeguards case with a calculated PCT of 1823^{o}F.

Attachment 4 to this letter presents Table 2 of the Unit 2 SBLOCA analysis which was presented in our letter AEP:NRC:1024. A minor change has been made to correct the total fluid volume in core from 613.0 and 612.9 to 643.0 and 642.9 Ft^{3}.

Attachment 5 to this letter is a response to questions which Mr. Robert Jones of NRR asked us on April 1, 1987 concerning the SBLOCA analysis in our earlier submittal AEP:NRC:1024. Mr. Jones asked how the flow splits between the two SI lines in a SBLOCA when one line blows down to approximately atmospheric pressure. He also asked whether the given break size of 4 inches is the most limiting case for the reduced flow case, and whether charging and SI flow is adequate to match decay heat after refilling the core.

We believe that these analyses will resolve any NRC concerns with closing the cross-ties or performing any maintenance on the systems of concern downstream of the cross-ties.

We request that you inform us by June 15, 1987 if you have any objections or questions concerning the proposed interpretation of the T/Ss to allow isolation of two injection points in the RHR or SI systems. We are requesting a prompt response because until we receive a response, maintenance or testing of the safety injection or RHR systems may require unnecessary shutdown of either unit. If you have any questions, my staff is prepared to discuss this matter.

A check in the amount of $150.00 is enclosed with this letter for the NRC processing of the aforementioned request.

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
Vice President

cm

Attachments

cc: John E. Dolan
W. G. Smith, Jr. - Bridgman
R. C. Callen
G. Bruchmann
G. Charnoff
NRC Resident Inspector - Bridgman
A. B. Davis, Region III

ATTACHMENT 1 TO AEP:NRC:1024A

SMALL-BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS

UNIT 1

D.C. Cook Unit 1 Small Break LOCA Evaluation
ECCS Performance With HHSI Cross Tie Closure

BACKGROUND

There are two high head safety injection (HHSI) pumps in the D.C.Cook Unit 1 design. Each HHSI pump discharge line splits to deliver flow into two of the four cold legs. A cross-tie connects the two pump discharge lines enabling one pump to deliver flow to all four of the cold legs. The design basis small break Loss-of-coolant-accident (LOCA) analyses assume that high head safety injection flow delivery is available through all four lines.

American Electric Power Service Corporation has requested Westinghouse to evaluate the D.C. Cook Unit 1 Emergency Core Cooling System (ECCS) performance following a small break LOCA for a scenario in which the HHSI cross-tie line is closed during normal full power operation. The evaluation will serve as the basis for a change to the current LOCA design basis in order to allow full power operation of D.C. Cook Unit 1 with the HHSI system capable of injection to only two reactor coolant loops.

Closure of the cross-tie line results in the flow from one HHSI pump being delivered to only two loops. This results in a reduction in the amount of total safety injection flow delivered to the RCS during a LOCA event when the single failure of an emergency diesel generator to start following the loss of offsite power is considered. As a result of the diesel failure, one train of safety injection is lost.

The D.C. Cook Unit 1 licensing basis LOCA analyses consider both large and small break LOCA events. The large break LOCA result is not highly dependent on HHSI pump flow capability due to the rapid depressurization to the accumulator actuation pressure (600 psia) and the continued rapid depressurization to the Low Head Safety Injection (LHSI) pump actuation pressure (114.7 psia). Recovery from a large break LOCA event is governed by the availability of LHSI and accumulator delivery during the reflood phase of the transient, hence a reduction in the amount of total HHSI flow delivery will not affect the large break LOCA results. The small break LOCA result is highly dependent upon charging pump and HHSI pump flow delivery to the RCS, but is not dependent upon LHSI flow delivery. Small break LOCA's which result in the highest Peak Cladding Temperatures (PCT's) do not experience primary reactor coolant system depressurization to the LHSI delivery pressure, therefore the small break LOCA analysis considers

safety injection flow from the charging and HHSI pumps only. Hence, a change to the design basis in which the HHSI cross-tie line is assumed unavailable requires that only the small break LOCA results be considered.

In order to determine the effect of the cross-tie line closure on the plant response to a small break LOCA, Westinghouse performed an analysis on a reference plant similar in design to D.C. Cook Unit 1. The reference four loop plant used to determine the safety injection sensitivity is essentially identical to Cook 1 in vessel design and loop components. Table 1 provides a comparison of the basic vessel and components design of D.C.Cook Unit 1 with the reference plant design with a 15X15 Optimized Fuel Assembly (OFA) core. Table 1 shows that the plant designs are virtually identical except for the upper head bypass flow. The slight difference in the upper head bypass flow at such low flowrates is expected to have an insignificant effect on plant response to a small break LOCA. Table 2 provides a comparison of some of the important parameters influencing plant response to a small break LOCA for D.C.Cook Unit 1 and the reference plant design (again with a 15X15 OFA core). The major differences noted between the plants include the licensed core power and the licensed peaking factor. Both D.C.Cook Unit 1 and the modified reference plant operate with a 15X15 OFA fuel array. As noted, the licensed core power level of D. C. Cook Unit 1 is lower than the power level assumed in the modified reference plant used for this analysis by 2.64%. The total core power level influences the depth and duration of core uncovery. That is, the higher the power level, the deeper and longer the core will uncover. The reactor coolant system response to a small break LOCA, as calculated by the NOTRUMP code, demonstrates a rapid depressurization down to the steam generator secondary relief valve pressure. This condition represents a quasi-equilibrium pressure at which the primary system tends to stabilize prior to the venting of steam through the broken leg loop seal. Following loop seal venting in the broken loop, core boiloff exceeds the safety injection mass flow rate and a core uncovery transient results. Therefore, depth and duration of core uncovery prior to reaching the accumulator injection setpoint are dependent upon not only the initial power level, but also the difference between safety injection flow rate and the core boiloff flow rate. Since this analysis modelled the 15X15 OFA core, cross tie closed SI flow rates and a higher power level than that licensed for D. C. Cook Unit 1, regardless of other differences in the range of 100% power operating conditions, a plant specific analysis for the same break size would result in less limiting results than those presented herein.

SMALL BREAK LOCA ANALYSIS

A small break LOCA analysis was performed for the reference plant applying the limiting four-inch equivalent diameter cold leg break for the D.C. Cook Unit 1 licensing basis WFLASH analysis. The four inch break was chosen since it is also the limiting break size for the reference four loop plant when analyzed with the NOTRUMP evaluation model. Although, a reduction in safety injection can, in some instances, cause a shift in limiting break size towards the smaller breaks, there are indications that the four inch break will remain limiting. The reference four loop plant demonstrates core recovery on accumulator injection for all FSAR cases. This tends to mitigate the impact of the SI reduction on the duration of core uncovery, particularly for the smaller break sizes. The analysis assumed a safety injection flow rate representative of the HHSI flow configuration at D.C. Cook Unit 1 with the cross tie closed. Combined with Appendix K minimum SI assumptions, this results in one charging pump available to deliver flow through four lines and one HHSI pump available to deliver flow through two of four lines. The analysis was performed at 102% of the reference plant licensed core power assuming a fission product decay heat generation rate of 1.2 times the 1971 ANS Decay Heat values. The analysis also assumed the loss of offsite power and the single failure of a diesel to start. It was conservatively assumed that all safety injection flow delivered to the broken loop spilled out the break. As a result of the spilling assumption, safety injection flow from one charging pump is delivered to the three intact RCS loops and one HHSI pump delivers flow to one of the two remaining RCS loop delivery lines. The analysis was performed to determine the peak clad temperature using the D.C. Cook Unit 1 SI flow representative of the cross tie closure. The reference plant analysis was performed using the Westinghouse NRC approved small break LOCA ECCS evaluation model using the NOTRUMP code as described in WCAP-10054-P-A and WCAP-10079-P-A. NOTRUMP addresses all of the NRC concerns expressed in NUREG-0611 and meets the requirement of NUREG 0737 II.K.3.30.

The analysis resulted in a PCT of 1427°F, thereby illustrating that operation in the flow configuration in which one charging pump is available to deliver flow to four RCS loops and one HHSI pump is available to deliver flow to only two of four loops, does not violate the requirements of 10 CFR 50.46 and Appendix K.

The reference four loop plant FSAR analysis was performed at 102% of a licensed core power level of 3338 MWt with a core peaking factor of 2.32 and resulted in a PCT of 1427°F for the limiting four inch break. Table 5 provides the number of safety injection pumps, the number of lines available to deliver flow to the RCS and the spilling assumptions utilized in the FSAR analyses and this evaluation. The safety injection flow vs. pressure curve assumed in the D.C. Cook Unit 1 FSAR analysis is shown in figure 10. The safety injection flows for the proposed HHSI ECCS design basis operation with closure of the cross tie connection is also shown in figure 10. The basis for these flows is consistent with those conservative assumptions made for SI delivery for all FSAR analyses. That is: lowest resistance line spilling, 5% degradation in pump performance and no credit for reactor coolant pump seal injection and miniflow recirculation flow. Table 3 provides a comparison of D.C. Cook Unit 1 FSAR analysis assumptions to the assumptions used in the reference 412 plant with a D.C. Cook Unit 1 core analysis.

The four inch break for the NOTRUMP analysis employing SI flows used in the cross tie closed analysis is characterized by a rapid primary side depressurization to a pressure slightly above the steam generator secondary safety valve setpoint. Steam generator secondary side pressurization to the safety valve setpoint results from the loss of off-site power assumption. RCS inventory depletion results in steam venting through the pump suction leg loop seal in the broken loop at approximately 357 seconds. This permits steam generated in the core to exit through the break, resulting in continued RCS depressurization. Since the core boiloff rate exceeds the safety injection flow rate, core uncovery results. Accumulator injection, which reverses the net mass inventory depletion from the RCS, occurs when the RCS pressure decreases to approximately 600 psig. The safety injection pump performance interval of interest therefore extends from approximately 1000 psig to the accumulator actuation pressure.

Table 4 provides a comparison of the results for the D.C. Cook Unit 1 original FSAR analysis to the reference plant analyses with a 15X15 OFA core. The results reported for Cook Unit 1 reflect the original WFLASH analysis specific to D.C. Cook Unit 1. Note that these results are for the small break analysis performed to determine the effects of a reduction in Safety Injection Flow if the assumed HHSI miniflow is increased from 30 gpm to 60 gpm. The miniflow analysis was specific to D. C. Cook Unit 1 except for the assumed power level of 3411 MWt. This analysis was used to support the D.C. Cook Unit 1 license amendment for an increase in HHSI miniflow from 30 gpm to 60 gpm. All references to

the D.C. Cook Unit 1 current design basis SI flows assume the SI available with a HHSI miniflow of 60 gpm.

Figures typical of those presented in an FSAR showing plant response to a small break LOCA are also attached. Direct comparisons with the current D. C. Cook Unit 1 current FSAR small break analysis plots are not presented since the effect of the reduction in safety injection would be confounded by differences between the WFLASH and NOTRUMP small break evaluation models (as was concluded in the NRC Safety Evaluation Report for the "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study With The NOTRUMP Code"; WCAP-11145-P-A, WFLASH results are more limiting than results obtained using the NOTRUMP code).

The attached figures present the following results:

- Figure 1. RCS Depressurization Transient
- Figure 2. Broken Loop Steam Generator Secondary Pressure
- Figure 3. Intact Loop Steam Generator Secondary Pressure
- Figure 4. Core Mixture Level
- Figure 5. Pumped Safety Injection Flow Rate - Intact Loops
- Figure 6. Accumulator Flow Rate
- Figure 7. Total Liquid Break Flow Rate
- Figure 8. Peak Clad Temperature - Hot Rod
- Figure 9. Fuel Pellet Temperature - Hot Rod
- Figure 10. Safety Injection Flow Rate versus Pressure

Figure 10 illustrates the reduction in the safety injection flow rate assumed in the analysis presented herein when compared to the assumed in the current analysis docketed for D. C. Cook Unit 1.

CONCLUSION

The results presented here for the current limiting small break for D. C. Cook Unit 1 conservatively bound the effect of operation with the cross tie closed. The 1427°F peak clad temperature computed with the NOTRUMP evaluation model exhibit significant margin to the limits set forth in 10 CFR 50.46 and Appendix K. Therefore operation of D. C. Cook Unit 1 with the cross tie closed is justified.

TABLE 1

COMPONENT COMPARISON

D.C. Cook Units vs. Ref 412 with D. C. Cook Unit 1 Core

Components/Configuration	Cook Unit 1	Reference 412 w/D.C. Cook Unit 1 Core
VESSEL:		
Upper Support Plate	Top Hat	Top Hat
Barrel Baffle Conf.	Downflow	Downflow
Downcomer Shielding	Thermal Shield	Thermal Shield
Lower Support Plate	Curved	Curved
Fuel Array	15X15 OFA	15X15 OFA
Upper Head Spray Flow Percent	0.16%	0.21%
Upper Head Temp.	THOT	THOT
LOOP COMPONENTS:		
Pressurizer	1800 ft^3	1800 ft^3
Steam Generator	Model 51	Model 51
Pump Type	93A 6000 Hp	93A 6000 Hp

TABLE 2

PLANT CONDITIONS Actual vs. Modelled

PARAMETERS	COOK 1	REFERENCE 412 w/D.C. Cook Unit 1 Core
LICENSED POWER (MWt)	3250	3338
LICENSED PEAKING FACTOR	2.10	2.32
FUEL VOLUMETRIC HEAT GENERATION FOR TOTAL CORE (TOTAL KW/TOTAL FT^3 FUEL)	9415.10	9670.04
AVERAGE LINEAR HEAT GENERATION (KW/FT)	6.848	7.033
PEAK LINEAR HEAT GENERATION (KW/FT)	14.380	15.473
CORE FLUID VOLUME (FT^3)		
Rod Channel Fluid Volume	646.6	646.6
Core Thimble Tube Volume	65.52	65.52
Total Core Fluid Volume	712.12	712.12
THERMAL DESIGN FLOW (LBS/SEC.)	37666.7	36950.2
VESSEL EXIT TEMPERATURE (^{o}F)	599.3	608.0
VESSEL INLET TEMPERATURE (^{o}F)	536.3	542.4

TABLE 3

ANALYSIS ASSUMPTIONS

PARAMETERS	COOK 1 FSAR ANALYSIS	REFERENCE 412 w/D.C. Cook Unit 1 Core ANALYSES
LICENSED CORE POWER, 102% OF	3411 MWt	3338 MWt
PEAK LINEAR HEAT GENERATION	15.811 KW/FT	15.473 KW/FT
AVG. LINEAR HEAT GENERATION	7.187 KW/FT	7.033 KW/FT
TOTAL CORE PEAKING FACTOR, F_Q	2.32	2.32
FUEL TYPE	15X15 OFA WESTINGHOUSE	15X15 OFA WESTINGHOUSE
SMALL BREAK MODEL	WFLASH	NOTRUMP

TABLE 4

SMALL BREAK ANALYSIS RESULTS

RESULTS	COOK UNIT 1 FSAR ANAL.	REFERENCE 412 w/D.C. Cook Unit 1 Core FSAR ANAL.
SMALL BREAK MODEL	WFLASH	NOTRUMP
PEAK CLAD TEMPERATURE	1716°F	1427°F
PEAK CLAD TEMPERATURE LOCATION (FT.)	11.75	12.0
TIME OF PCT (SEC.)	823	947
REACTOR TRIP (SEC.)	17.50	1.46
CORE UNCOVERY TIME (SEC)	413.0	646.5
ACCUM. INJECTION (SEC.)	800.0	880.8
CORE RECOVERY TIME (SEC.)	1310	1143

Table 5

SAFETY INJECTION SYSTEM CONFIGURATION COMPARISON

	Pumps Available	Lines Available	Lines Spilling
D.C. COOK UNIT 1	1 CHARGING	INJ. TO 4 LINES	1 LINE SPILLS
ORIG. FSAR ANALYSIS	1 HHSI	INJ. TO 4 LINES	1 LINE SPILLS
REFERENCE 412 with	1 CHARGING	INJ. TO 4 LINES	1 LINE SPILLS
D.C. Cook Unit 1 Core	1 HHSI	INJ. TO 2 LINES	1 LINE SPILLS

D C COOK UNIT 1 (AEP) NOTRUMP SB LOCA - 412 STD
4-INCH COLD LEG BREAK - 15X15 OFA FUEL - X-TIE CLOSED

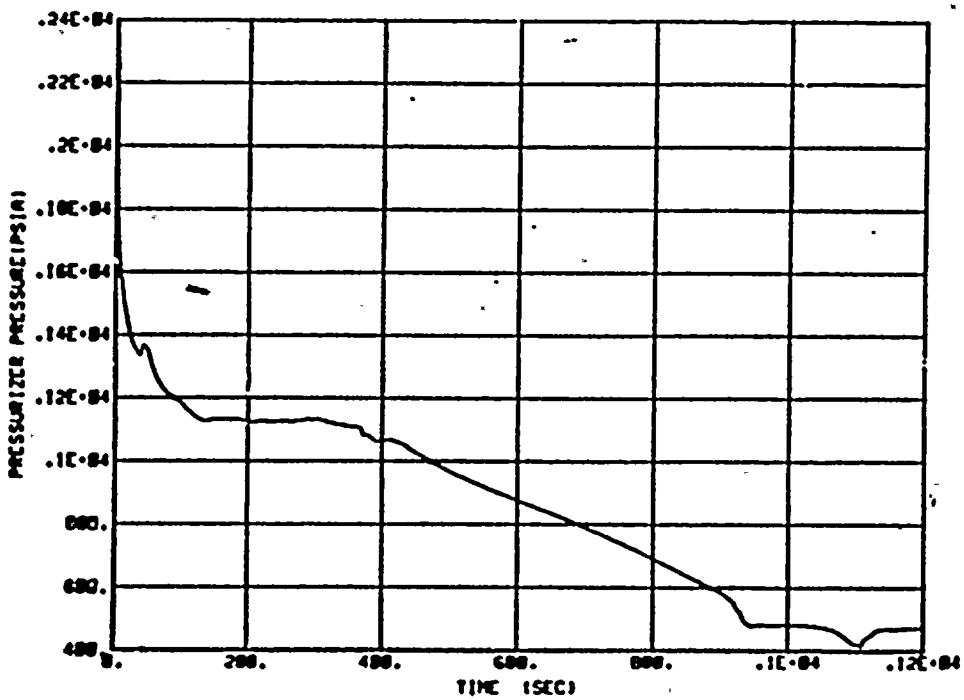


Figure 1

D C COOK UNIT 1 (AEP) NOTRUMP SB LOCA - 412 STD
4-INCH COLD LEG BREAK - 15X15 OFA FUEL - X-TIE CLOSED

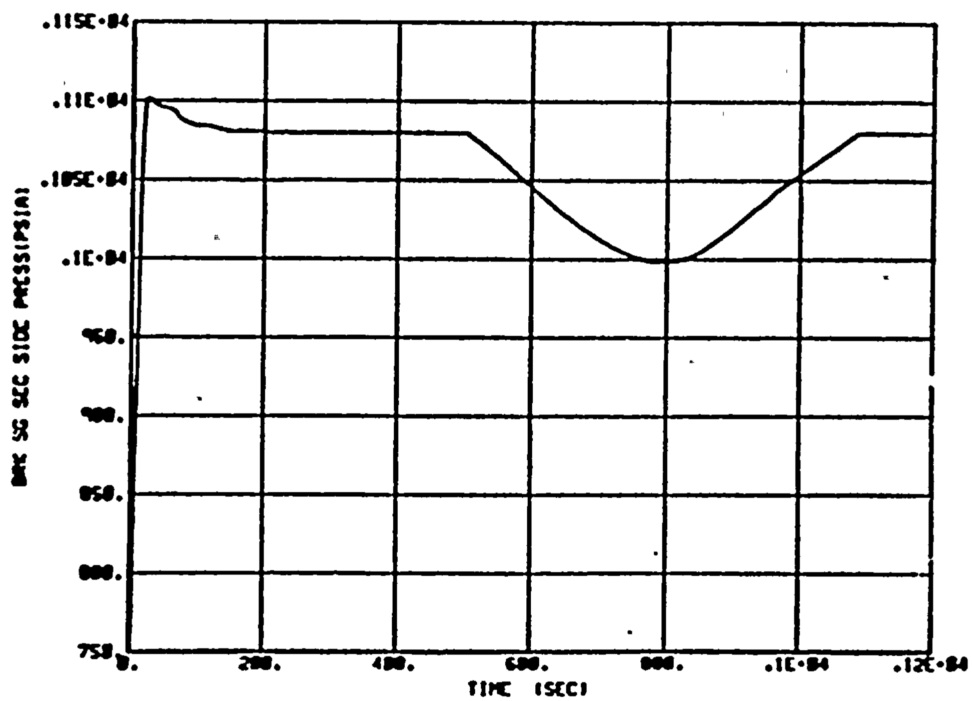


Figure 2

D C COOK UNIT 1 (AEP) NOTRUMP SB LOCA - 412 STD
4-INCH COLD LEG BREAK - 15X15 OFA FUEL - X-TIE CLOSED

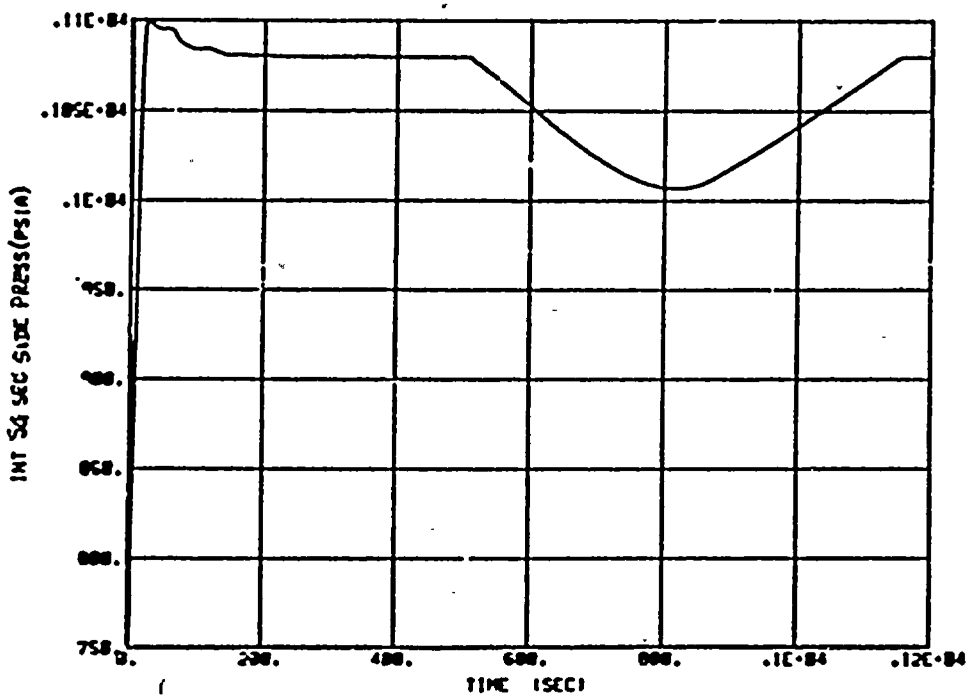


Figure 3

D C COOK UNIT 1 (AEP) NOTRUMP SB LOCA - 412 STD
4-INCH COLD LEG BREAK - 15X15 OFA FUEL - X-TIE CLOSED

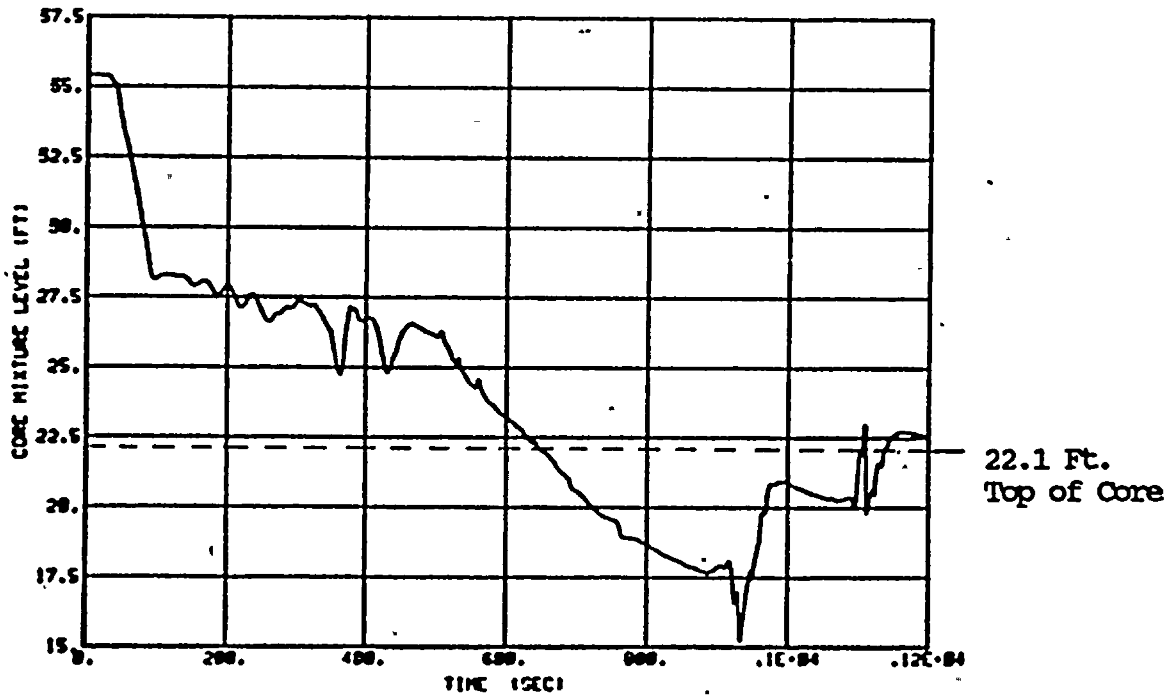


Figure 4

D C COOK UNIT 1 (AEP) NOTRUMP SB LOCA - 412 STD
4-INCH COLD LEG BREAK - 15X15 OFA FUEL - X-TIE CLOSED

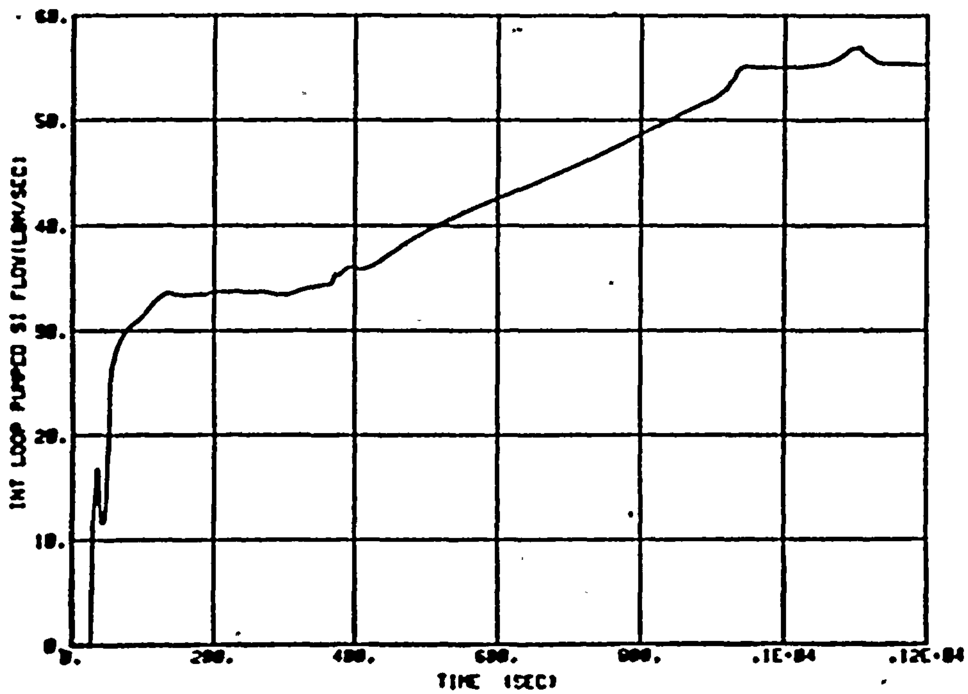


Figure 5

D C COOK UNIT 1 (AEP) NOTRUMP SB LOCA - 41Z STD
4-INCH COLD LEG BREAK - 15X15 OFA FUEL - X-TIE CLOSED

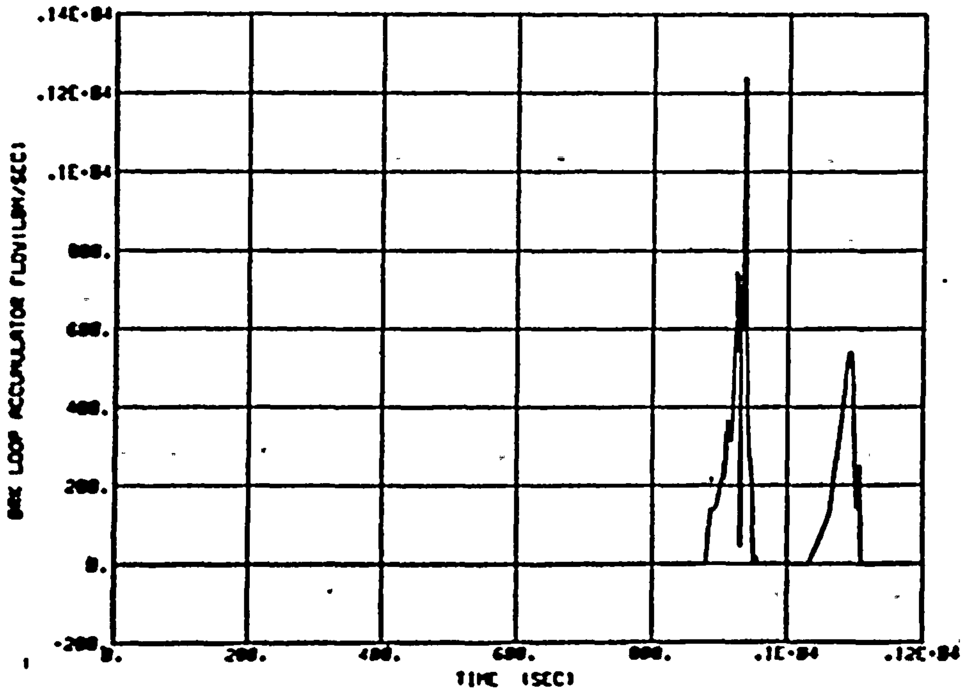


Figure 6

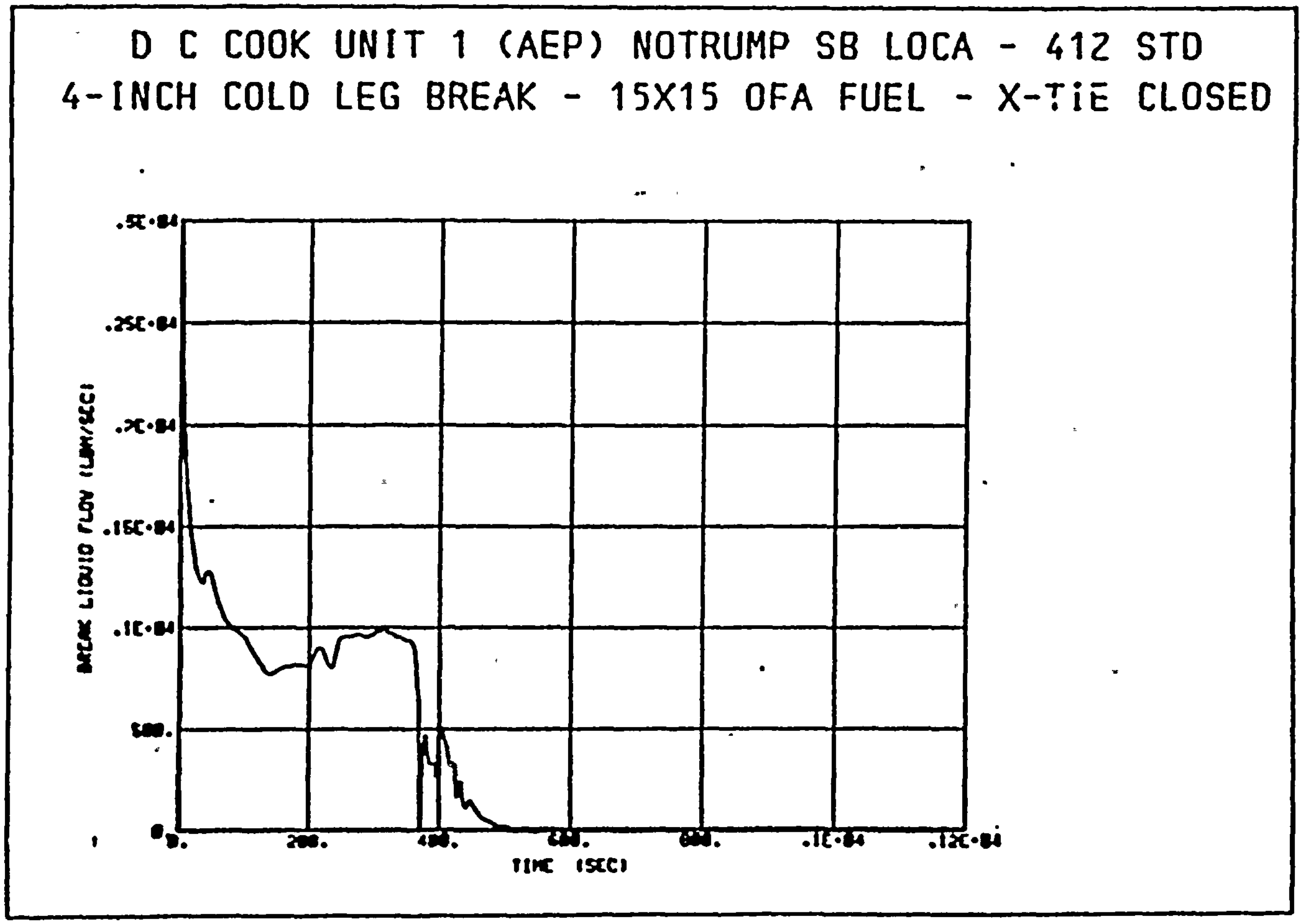

D C COOK UNIT 1 (AEP) NOTRUMP SB LOCA - 412 STD
4-INCH COLD LEG BREAK - 15X15 OFA FUEL - X-TIE CLOSED
.3E+04
.25E+04
.2E+04
.15E+04
.1E+04
500.
0.
BREAK LIQUID FLOW (LBM/SEC)
0.
200.
400.
600.
800.
.1E+04
.12E+04
TIME (SEC)

Figure 7

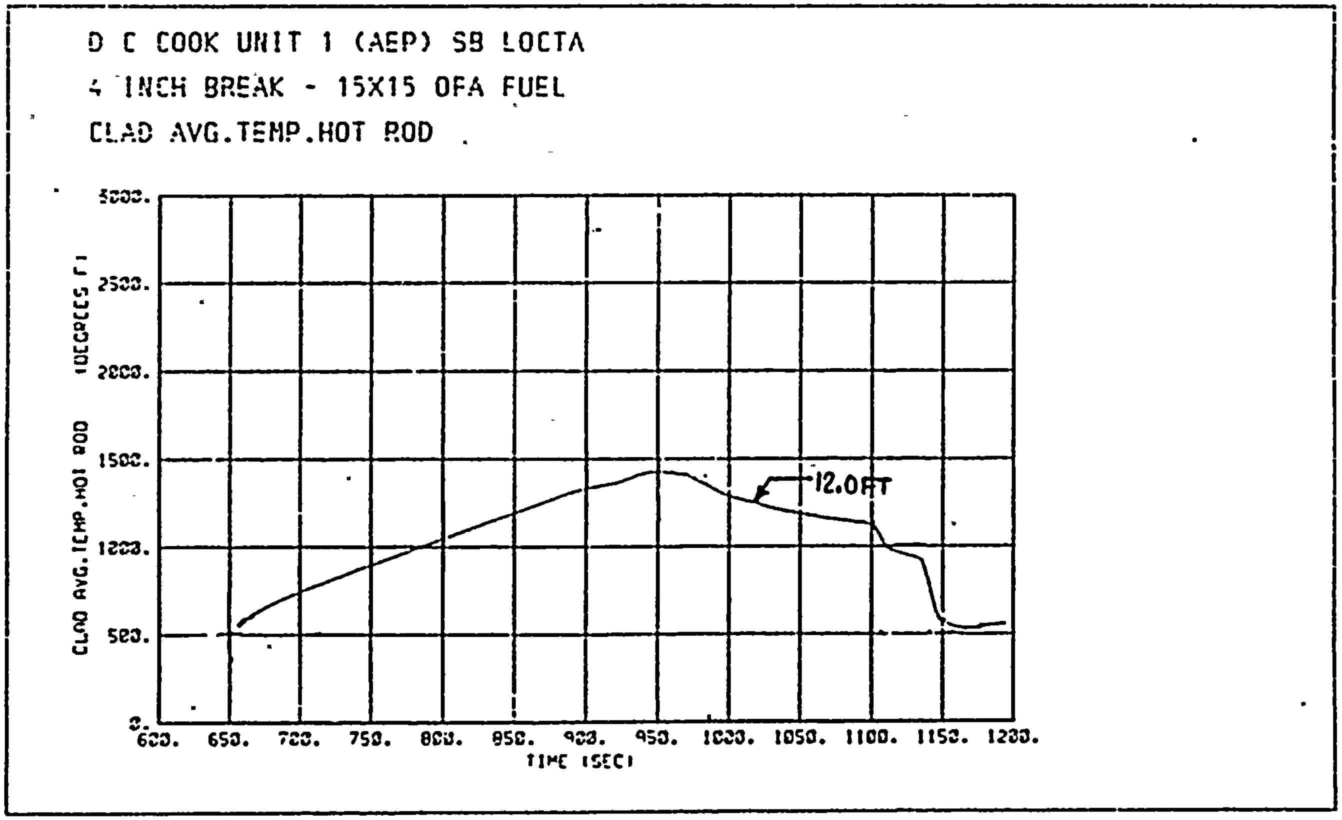

D C COOK UNIT 1 (AEP) SB LOCTA
4 INCH BREAK - 15X15 OFA FUEL
CLAD AVG.TEMP.HOT ROD
CLAD AVG.TEMP.HOT ROD (DEGREES F)
TIME (SEC)
12.0FT

Figure 8

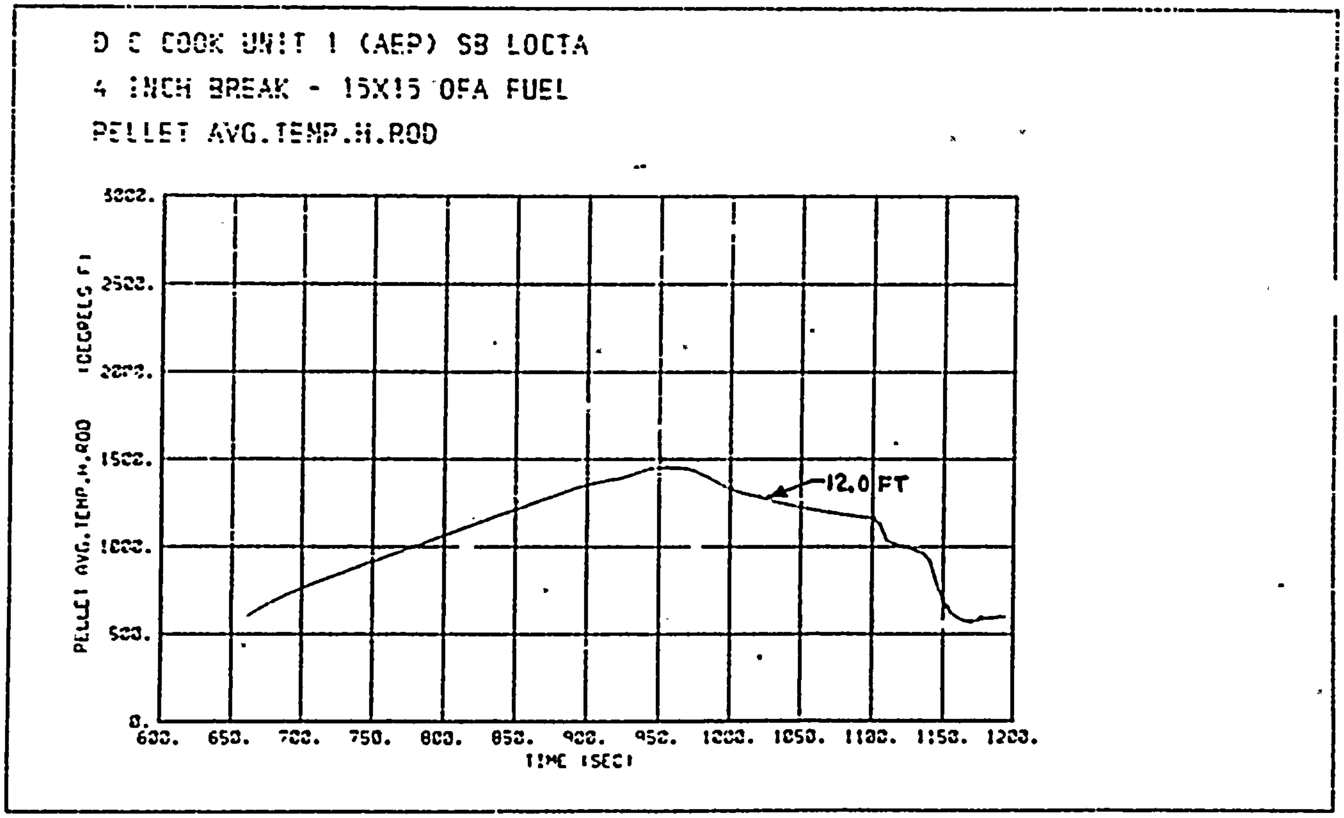
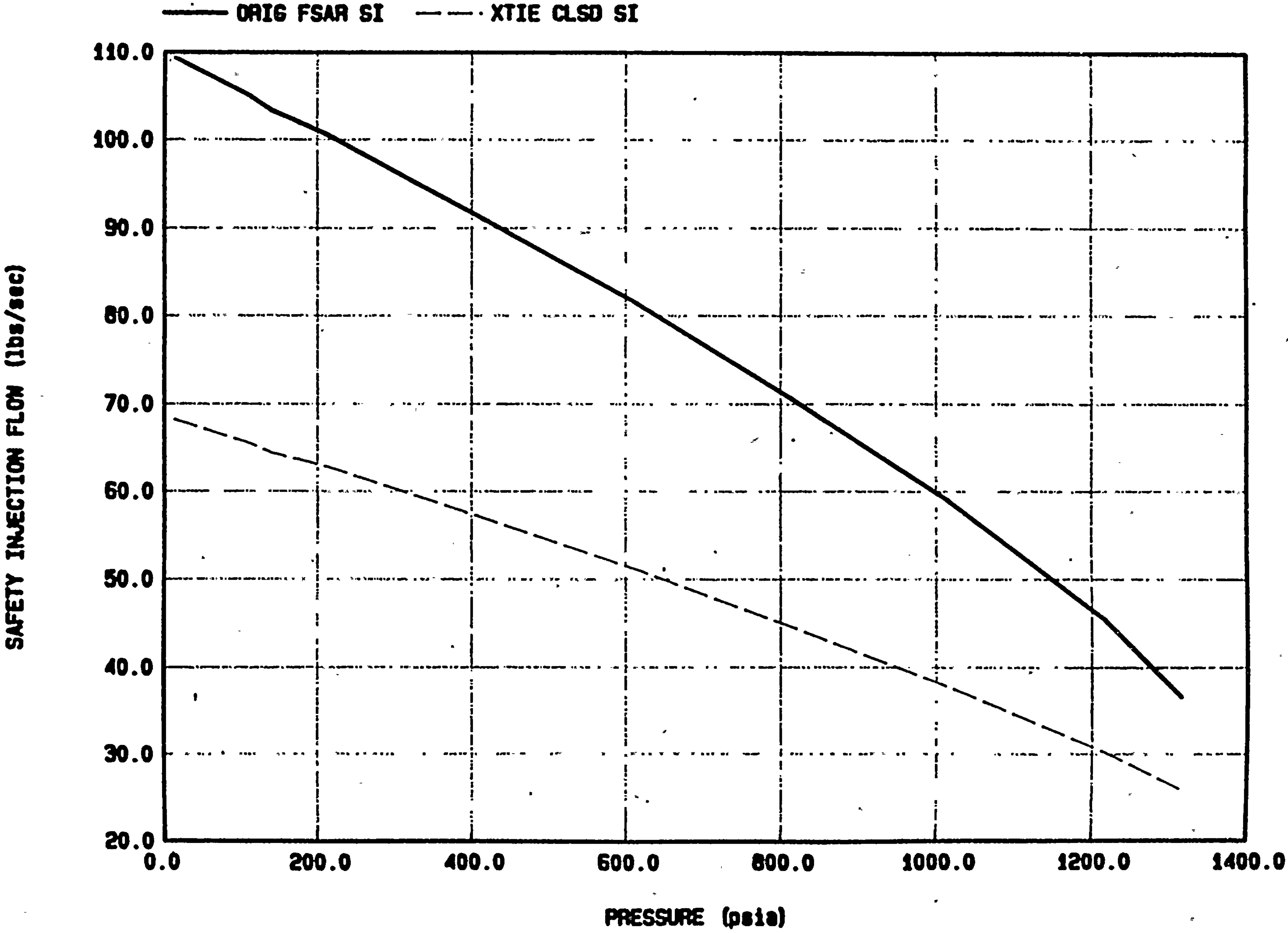

D C COOK UNIT 1 (AEP) SB LOCTA
4 INCH BREAK - 15X15 OFA FUEL
PELLET AVG.TEMP.H.ROD
12.0 FT
TIME (SEC)
PELLET AVG.TEMP.H.ROD (DEGREES F)

Figure 9

Figure 10
D. C. Cook Unit 1 SI Flows
Design Basis SI Flow vs X-Tie Closed SI



ATTACHMENT 2 TO AEP:NRC:1024A

LARGE-BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS

UNIT 1

D.C. Cook Unit 1 Large Break LOCA Evaluation:
ECCS Performance with Cross Tie Closure

BACKGROUND

The D.C. Cook Unit One safety injection system consists of two residual heat removal (RHR) pumps, two centrifugal charging (CCP) pumps and two high head safety injection (HHSI) pumps. Each HHSI pump discharge line splits to deliver flow into two of the four cold legs, and a cross tie connects the two pump discharge lines enabling one pump to deliver flow to all four cold legs. The RHR pump discharge piping is configured the same way as the HHSI pumps. The design basis large break Loss-of-Coolant Accident (LOCA) analyses assume that flow delivery is available through all four lines from each pump in the safety injection system.

American Electric Power Service Corporation has requested Westinghouse to evaluate the D.C. Cook Unit One Emergency Core Cooling System (ECCS) performance following a large break LOCA for a scenario in which the HHSI or RHR cross tie line is closed during normal full power operation. The evaluation will serve as the basis for a change to the current LOCA design basis and allow full power operation of D.C. Cook Unit One with either the RHR or the HHSI cross tie line closed.

Closure of the cross tie line results in the flow from one HHSI pump or one RHR pump being delivered to only two loops. This results in a reduction in the amount of total safety injection flow delivery to the core during a LOCA event when the single failure of an emergency diesel generator to start following the loss of off-site power is considered.

The D.C. Cook Unit One licensing basis LOCA analysis includes both large and small break LOCA events. The large break LOCA result is not highly dependent on HHSI pump flow capability due to the rapid system depressurization to the accumulator actuation pressure (600 psia) and the continued rapid depressurization below the RHR pump actuation pressure (114.7 psia). Recovery from a large break LOCA event is governed by the availability of pumped injection and accumulator water delivery during the reflood phase of the transient. Because the RHR pump delivers much more flow than the HHSI pump at low pressure, the reduction in the amount of total HHSI flow delivery due to cross tie closure will not affect large break LOCA calculated ECCS performance as significantly as the larger reduction caused by RHR pump cross tie closure.

The 10CFR50.46 analysis of the D.C. Cook Unit One large break LOCA is based upon 102% core power operation together with minimum safeguards and maximum safeguards safety injection (SI) flowrates. Minimum safeguards SI flowrates are predicted upon one charging pump, one high head safety injection pump and one low head safety injection RHR) pump operating; appropriate assumptions are made about pump head/flow characteristic curves and system resistance to minimize delivery to the reactor coolant system (RCS). Maximum safeguards SI flowrates are computed assuming all SI pumps present operate under pump head/flow characteristic and system resistance conditions which maximize post-LOCA flow delivery to the RCS.

In order to perform maintenance activities, it may at times be desirable to close the cross tie and isolate the delivery paths to the pairs of RCS cold legs. IE Information Notice 87-01 discusses this situation. Justification is provided herein demonstrating THAT for the large break LOCA event isolating two SI lines in order to perform maintenance activities is acceptable. The limiting case of closing the RHR pump cross tie is evaluated for impact at full power.

Full HHSI and CCP flow delivery will be available for the large break LOCA event even if two RHR delivery lines are unavailable due to cross tie closure.

EVALUATION

Consider the limiting scenario with minimum safeguards SI flow with two RHR delivery lines unavailable during a large break LOCA event, i.e., the broken cold leg is one of the two which is aligned to receive flow from the lone operative RHR pump. Under this scenario the pump can deliver only 180 lbs. per second to the RCS during the core reflood phase of the large break LOCA analysis rather than the 370 lbs. per second modeled in the BART analysis presented in the Cook Unit One FSAR. Pumped safety injection is actuated during the latter part of the blowdown of the initial RCS inventory; it supplements the accumulators in providing the water necessary to refill the reactor vessel lower plenum and then the vessel downcomer. Review of the large break LOCA spectrum, including the limiting case break (C_D=0.6 DECLG) of minimum safeguards Cook Unit One cases establishes that the vessel downcomer fills completely during accumulator injection in all cases.

Once the accumulators have emptied, the pumped safety injection must supply the water needed to maintain downcomer level during core reflood. The CCP and HHSI pumps taken together supply about 110 lbs. per second to replenish downcomer inventory; together with the RHR pump output which reaches the RCS through the one available delivery line, a total pumped injection rate of 290 lbs. per second is available in the minimum safeguards condition with the RHR cross tie closed.

During the core reflood phase of the Cook Unit One C_D=0.6 DECLG minimum safeguards case, much of the pumped injection to RCS spills from the full downcomer out the break. Review of the predicted core reflood transient verifies that the 290 lbs/second available under minimum safeguards conditions with two RHR lines presumed unavailable is adequate to maintain the downcomer water level full. Therefore, the only impact the pumped flow reduction due to two RHR lines being unavailable exerts on the Cook Unit One large break LOCA performance is the delay incurred in vessel refilling while the accumulators are injecting. Reduction in total pumped injection flow delays slightly the bottom of core recovery, the time at which the water level reaches the bottom of the fuel during the large break LOCA transient. This delay in turn extends the time interval during which adiabatic heatup of fuel rods must be computed per 10CFR50 Appendix K. Also, the pumped injection flow reduction will slightly delay the filling of the downcomer, giving a slightly diminished head of water at any point in time as the downcomer fills.

The peak clad temperature (PCT) impact of these delays is identified as the increase in temperature which occurs while they remain in effect; it is based upon the rate of adiabatic heatup at the end of lower plenum refill. For the scenario in which only one RHR pump delivery line is available to the RCS, RHR flow into the RCS will be virtually zero as long as the accumulators are delivering water. Assuming no RHR pump injection during blowdown and lower plenum refill, bottom of core recovery (BOCREC) will require an additional 0.2 seconds based upon the rate of fill prevalent at BOCREC time. Since the adiabatic heatup rate is 30°F/second at BOCREC, a 6°F increase in calculated PCT will be incurred. Considering also the impact of the slower downcomer fill, an overall penalty of approximately 10°F in calculated PCT is all that occurs in the limiting Cook Unit One minimum safeguards large break LOCA FSAR case under the closed cross tie scenario. The established PCT for

this case is 1937oF, so a large PCT margin remains to the regulatory limit of 2200°F.

The C_D=0.6 DECLG large break LOCA with maximum safeguards is the identified limiting case for Cook Unit One, at a calculated PCT of 2154°F. Since no failure of safety injection equipment is the basis for the maximum safeguards LOCA scenario, each RHR (or HHSI) pump delivers into two of the four cold legs whether the cross tie is closed or open. The flow delivery remains as analyzed, so calculated PCT for this limiting case is unaffected by the cross tie assumption.

CONCLUSION

Closing of the cross tie in either the RHR pump or HHSI pump delivery system does not affect the calculated PCT for the limiting (maximum safeguards) large break LOCA scenario. The estimated PCT assuming RHR cross tie closure and the single failure of a diesel generator to start for the limiting C_D=0.6 DECLG break is estimated to be 1947°F. This indicates that margin remains to the 2200°F limit of 10CFR50.46 for the operation of D.C. Cook Unit One at its licensed core power during such time as either the High Head Safety Injection System cross tie connection or the RHR system cross tie connection (but NOT both simultaneously) is closed.

ATTACHMENT 3 TO AEP:NRC:1024A

LARGE-BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS

UNIT 2

ADVANCED NUCLEAR FUELS CORPORATION

?00 108th AVENUE NE, PO BOX 90777, BELLEVUE, WA 98009-0777
?06 453-4300

April 14, 1987
ANF/AEP-0559

Mr. R. B. Bennett, Engineer
Nuclear Materials & Fuel Management
Indiana & Michigan Electric Company
c/o American Electric Power Service Corp.
One Riverside Plaza
Columbus, OH 43216-6631

Ref.: (1) IE Information Notice 87-01, "RHR Valve Misalignment Causes Degradation of ECCS in PWRs," dated January 6, 1987

(2) Letter, G.N. Ward (ANF) to R. Bennett (AEP), dated November 14, 1986 (GNW:123:86)

(3) XN-NF-85-68(P), Rev. 1, "Donald C. Cook Unit 2 Limiting Break LOCA/ECCS Analysis, 10% Steam Generator Tube Plugging, and K(Z) Curve," Exxon Nuclear Company, Inc., April 1986

Dear Mr. Bennett:

Enclosed please find ANF's response to the issue of two point RHR injection, identified in Reference 1, for D.C. Cook Unit 2. The results of the evaluation show that the D.C. Cook 2 nuclear reactor satisfies the criteria specified by 10 CFR 50.46(b) for the case of RHR isolation of two injection points. The results of the analysis indicated an overall increase in peak cladding temperature as shown below relative to the previous analyses.[2,3]

	Base Case (Ref. 2)	Two Point RHR Injection Case
BOC K(Z)	1823°F	1813°F
EOC K(Z)	1790°F*	1988°F

* Note: This value is lower than that reported in Reference 3 due to corrections made in the TOODEE2 code which were reported in Ref. 2.

If you have any questions, please feel free to call our Mr. Jerry Holm, telephone 509-375-8142.

Sincerely,

H. G. Shaw
Contract Administrator

gf

Enclosure

cc: J.M. Cleveland
D.H. Malin
V. VanderBurg
R. Vasey
J.S. Holm (ANF)

Enclosure

D.C. COOK UNIT 2 LOCA/ECCS ANALYSIS WITH DEGRADED ECCS DUE TO TWO POINT RHR INJECTION

Ref.:

(1) Updated FSAR Figure 14.3.2-1 dated July 1982, with 5% degradation

(2) XN-NF-85-68(P), Rev. 1, "Donald C. Cook Unit 2 Limiting Break LOCA/ECCS Analysis, 10% Steam Generator Tube Plugging and K(Z) Curve," Exxon Nuclear Company, Inc., April 1986

(3) Letter, G.N. Ward (ANF) to Mr. J.E. Lyons (Office of Nuclear Reactor Regulation), dated November 18, 1986 (GNW:124:86)

INTRODUCTION

To support operation while one RHR train is out of service with two RCS injection points isolated, AEP requested Advanced Nuclear Fuels Corporation (ANF) to investigate the effect of RHR isolation of two injection points on the Large Break Loss of Coolant Accident (LOCA) transient for D.C. Cook Unit 2 nuclear reactor. Isolation of two injection points includes isolation of the crossover line between the two RHR pumps and isolation of one RHR pump. This results in a situation where only one RHR pump is available for injection into two cold legs, one of which is an intact loop and the other is the broken loop.

The RHR isolation of two injection points case was evaluated with ANF's EXEM/PWR Evaluation Models to demonstrate conformance to 10 CFR 50.46 criteria.

ASSUMPTIONS

The total RHR, SI, and charging pump flow rate tables input to RELAP4 for this analysis were estimated from a previous single failure analysis (loss of 1 RHR pump) where flow was from 1 RHR pump to 3 intact loops[1] and 1 broken loop. For the two point RHR injection case, the lower total RHR pump flow rate would allow the RHR pump to force more flow out of each of the two points than to each of the four points for the single RHR failure case. The new pump operating point would therefore be at a higher head than the single failure case on its operating curve, with more than half of the total pump flow rate of the single failure case. ANF conservatively assumed the total RHR flow rate table input to RELAP4 to be 1/2 of the single failure case values as a function of system or containment back pressure. The RHR flow rate at zero psig system pressure for the two point RHR injection case is 272.7 lb/sec. It was also assumed that only one charging pump was operating with a maximum flow rate of 35 lb/sec and one SI pump with a maximum flow rate of 56 lb/sec.

Since the primary system pressure is at approximately 150 psia at the end-of-bypass, an estimate of the depressurization of the primary system was included in the calculation of accumulator, SI, charging pump, and RHR flow rates to the intact loop during the refill period. The flow rates from the SI, charging pump, and RHR systems to the intact loop are less than the flow rates to the broken loop for the two point RHR injection case until several seconds into the refill period when the primary system and containment pressures equilibrate.

The change in RHR flow rate to the intact and broken loops was the only difference between the two point RHR injection and the full ECCS flow analysis.[(2)] The limiting break in the break spectrum can be identified by the break size with the highest fuel rod stored energy at the end-of-bypass. The change in RHR flow rate does not affect the system blowdown behavior or the fuel rod stored energy at the end-of-bypass. Therefore, the 1.0 DECLG remains the limiting break size.

RESULTS

The reduced total RHR flow rate for the two point RHR injection case resulted in a slightly slower rate of filling the lower plenum during the refill period. The beginning-of-core-recovery time increased from 40.31 sec to 40.48 sec. The accumulators emptied at approximately the same time as in the full ECCS flow analysis.[(2)] In both cases, the downcomer is filled to approximately the same level above the cold leg lip with accumulator flow. However, the two point RHR injection case resulted in less condensation at the ECCS injection point of steam flowing around the intact loops. The reduced condensation caused more steam to flow out the broken loop stub from the intact loops. The larger steam flow required a larger pressure drop, resulting in a higher system pressure for the two point RHR injection case than for the full ECCS flow case.[(2)] Although the reduced RHR flow to the intact loop caused some reduction in the downcomer liquid level compared to the full ECCS flow case, the higher absolute system pressure caused a net increase in reflood rate of up to 4% after the time the accumulators and lines emptied (~57 sec) due to increased steam density and a higher pressure drop from the core to the containment through the broken loop steam generator line. Since the full ECCS flow analysis conservatively modeled flows to the containment by including broken loop accumulator, RHR and SI system flows but not including break flows (steam flow out either side of the break), only the change in the broken loop RHR flow rate was included in calculating a revised containment pressure for the two point RHR injection case. The reduced RHR flow to the containment resulted in a small increase in containment pressure compared to the full ECCS flow case (less condensation of steam in the containment). The higher system pressure had an accompanying higher saturation temperature, which also had an effect on fuel rod plenum temperature and pressure during reflood which can affect the timing and location of fuel rod cladding rupture.

The results of the full ECCS flow licensing analysis are reported in Ref. 2 for both BOC and EOC power shapes. The results in Ref. 2 were revised in response to the TOODEE2 errors discussed in Ref. 3. The peak cladding temperature for the BOC power shape was 1823°F and 1790°F for the EOC power shape. Both the BOC and EOC shapes were considered in the two point RHR injection analysis. For the BOC case, the increase in reflood rate resulted in a 10°F reduction in PCT. The PCT was predicted to be 1813°F as compared to the full ECCS flow case of 1823°F. The location of cladding rupture and time of rupture were unchanged for the BOC shape. For the EOC shape, a change in the timing and location of cladding rupture was observed. A node lower in the core was observed to rupture for the two point RHR injection case. This node was within a few degrees of rupture in the full ECCS flow case, but was turned over prior to rupturing. The ruptured node in the full ECCS flow case became the PCT node in the two point RHR injection case and caused the PCT to occur later in time. This resulted in an increase in PCT to 1988°F as compared to a PCT of 1790°F for the full ECCS flow analysis. Thus, the EOC shape became the limiting shape for the two point RHR injection case as opposed to the BOC shape being the limiting shape for the full ECCS flow case.[2] Plots of cladding temperature versus time during the refill and reflood period are shown in Figures 1 through 4 for both the BOC and EOC shapes and for the full ECCS flow and the two point RHR injection cases.

For a condition when the SI and RHR cross ties are closed simultaneously such that two point SI injection (1 intact loop and 1 broken loop) occurs as well as two point RHR injection, a small decrease in SI flow to the intact side and a small increase in SI flow to the broken loop side would be observed relative to the two point RHR injection case. Since the SI flow rate is already a small portion of the total ECCS flow, a small change in the SI flow rate would not be expected to significantly alter the results of the two point RHR injection analysis and would not preclude meeting the acceptance criteria of 10 CFR 50.46(b).

CONCLUSIONS

The results of the evaluation demonstrate that the plant will operate in conformance with 10 CFR 50.46(b) criteria with RHR injection to two points (1 intact loop and 1 broken loop) and SI injection to 4 points (3 intact loops and 1 broken loop).

$F_Q = 2.1$

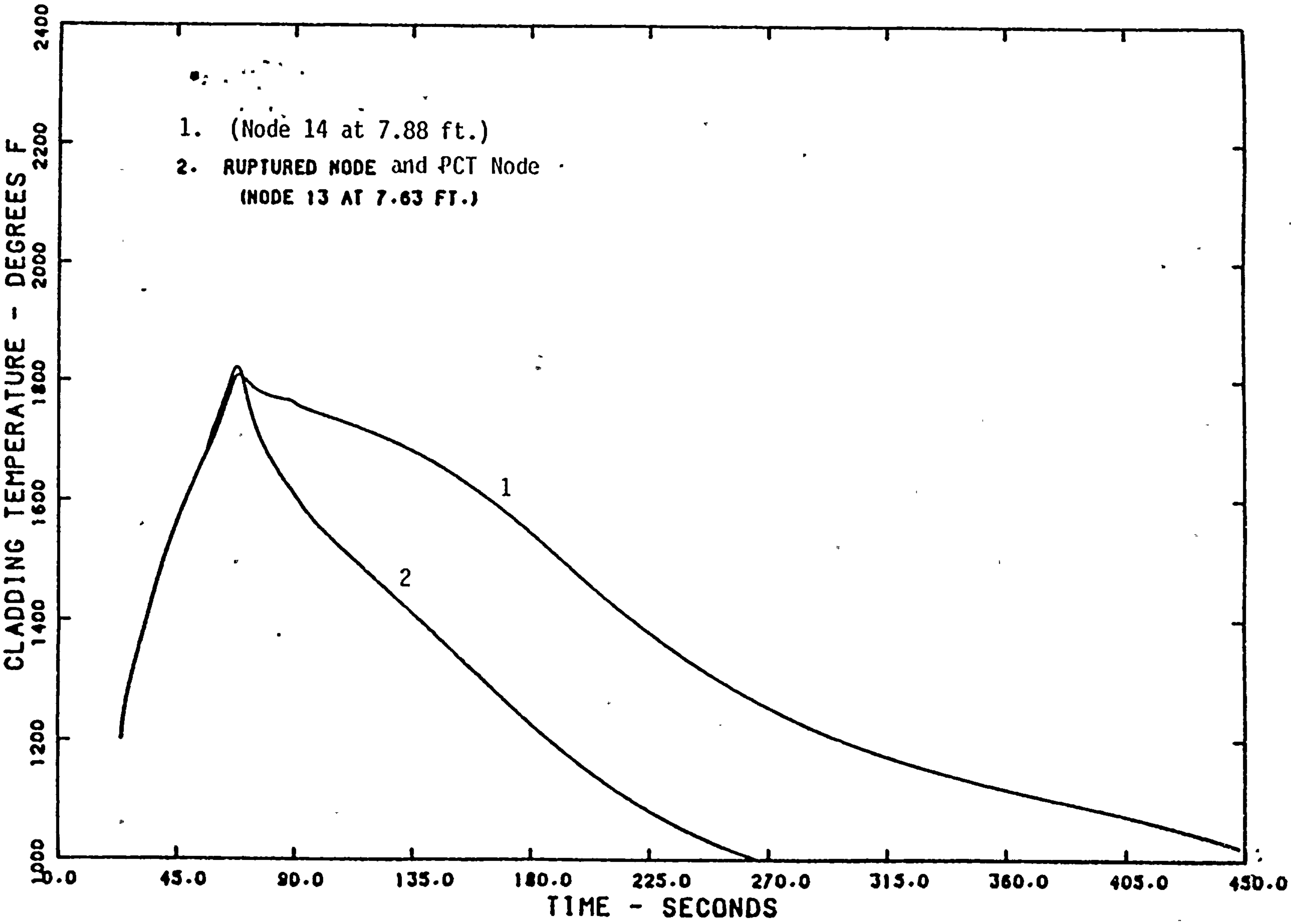


Figure 1 D.C. Cook Unit 2, BOC Shape, Full ECCS Flow Case

$F_Q = 2.1$

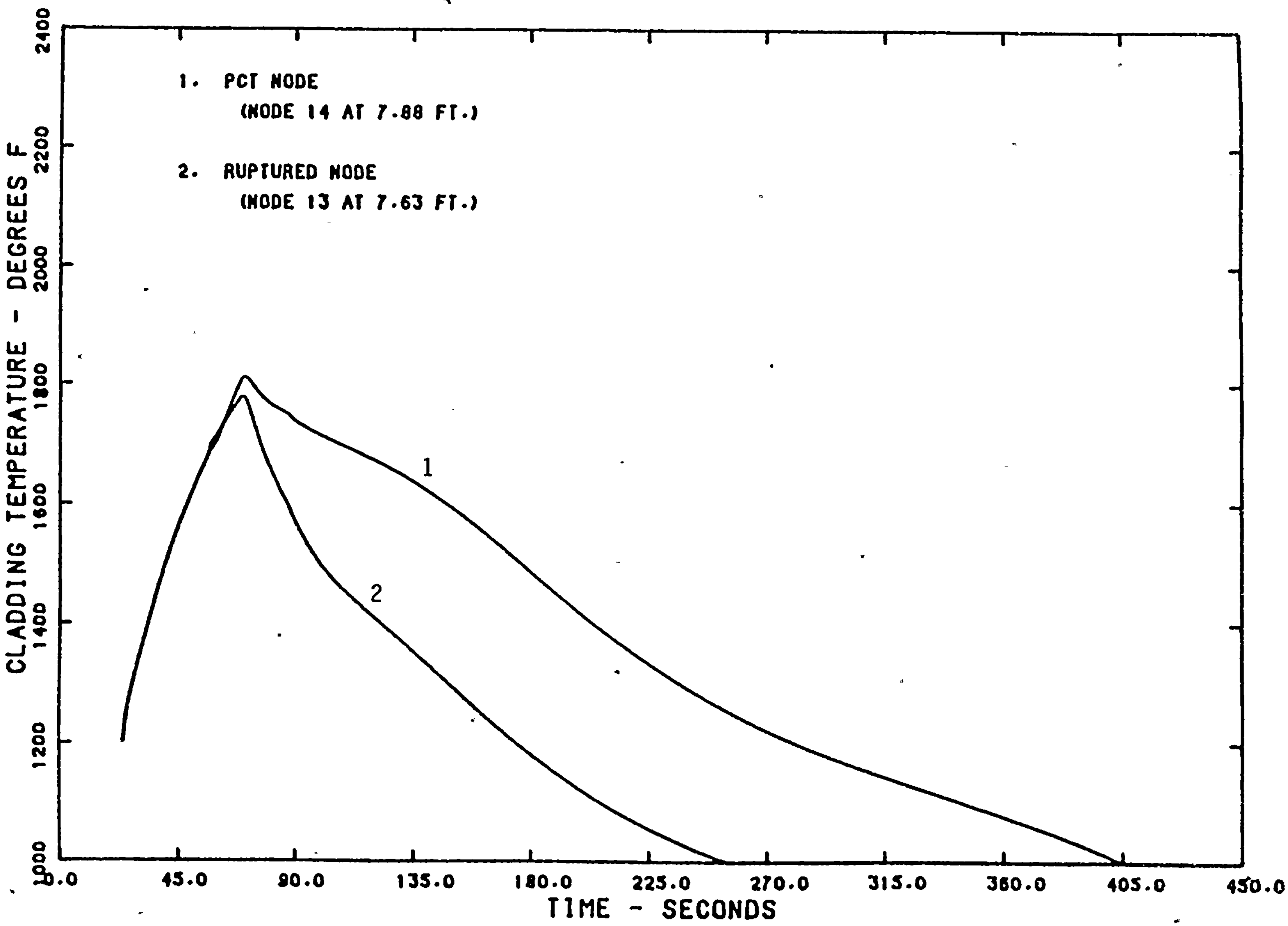


Figure 2 D.C. Cook Unit 2, BOC Shape, Two Point RHR Injection Case

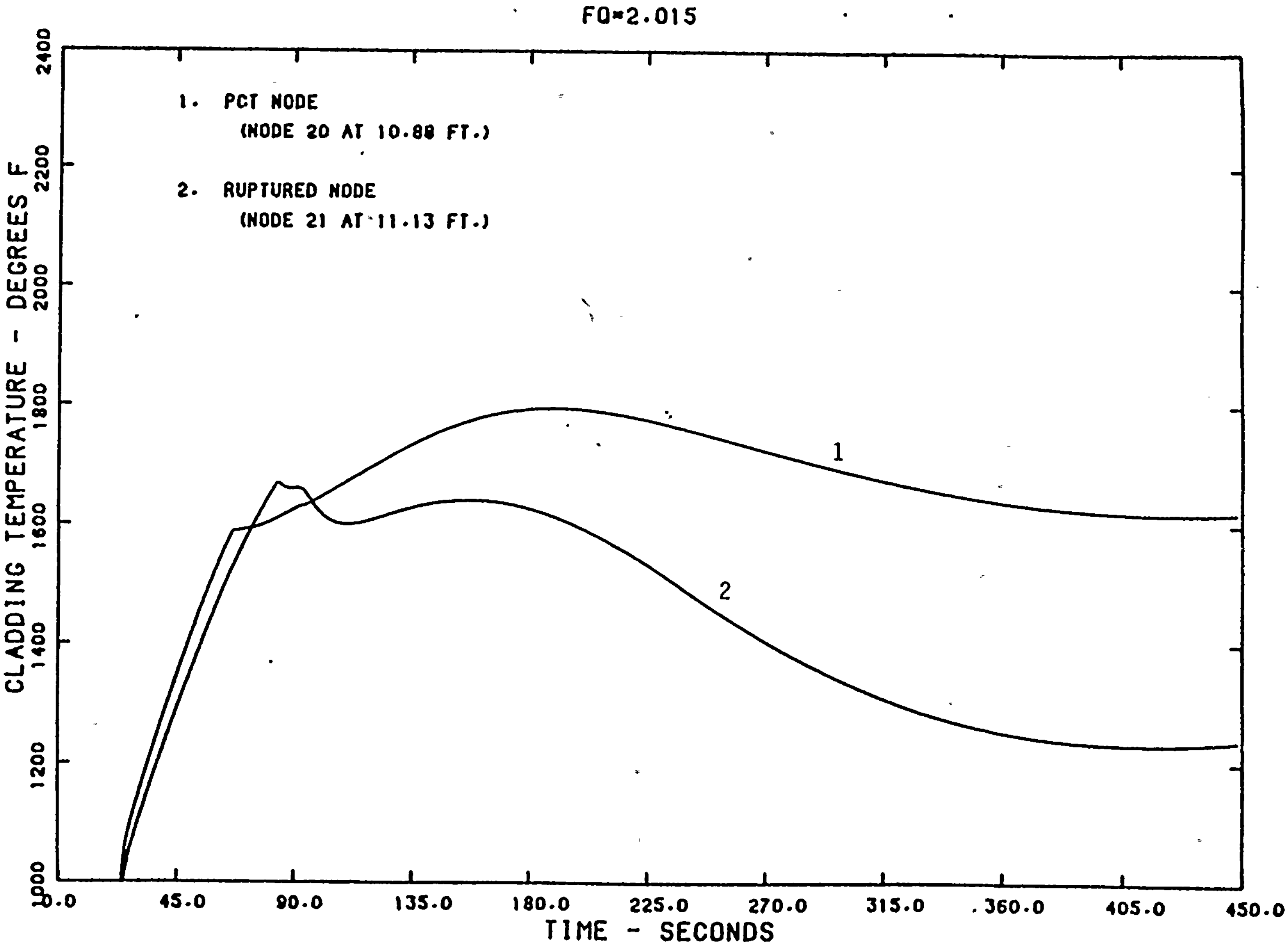


Figure 3 D.C. Cook Unit 2, EOC Shape, Full ECCS Flow Case

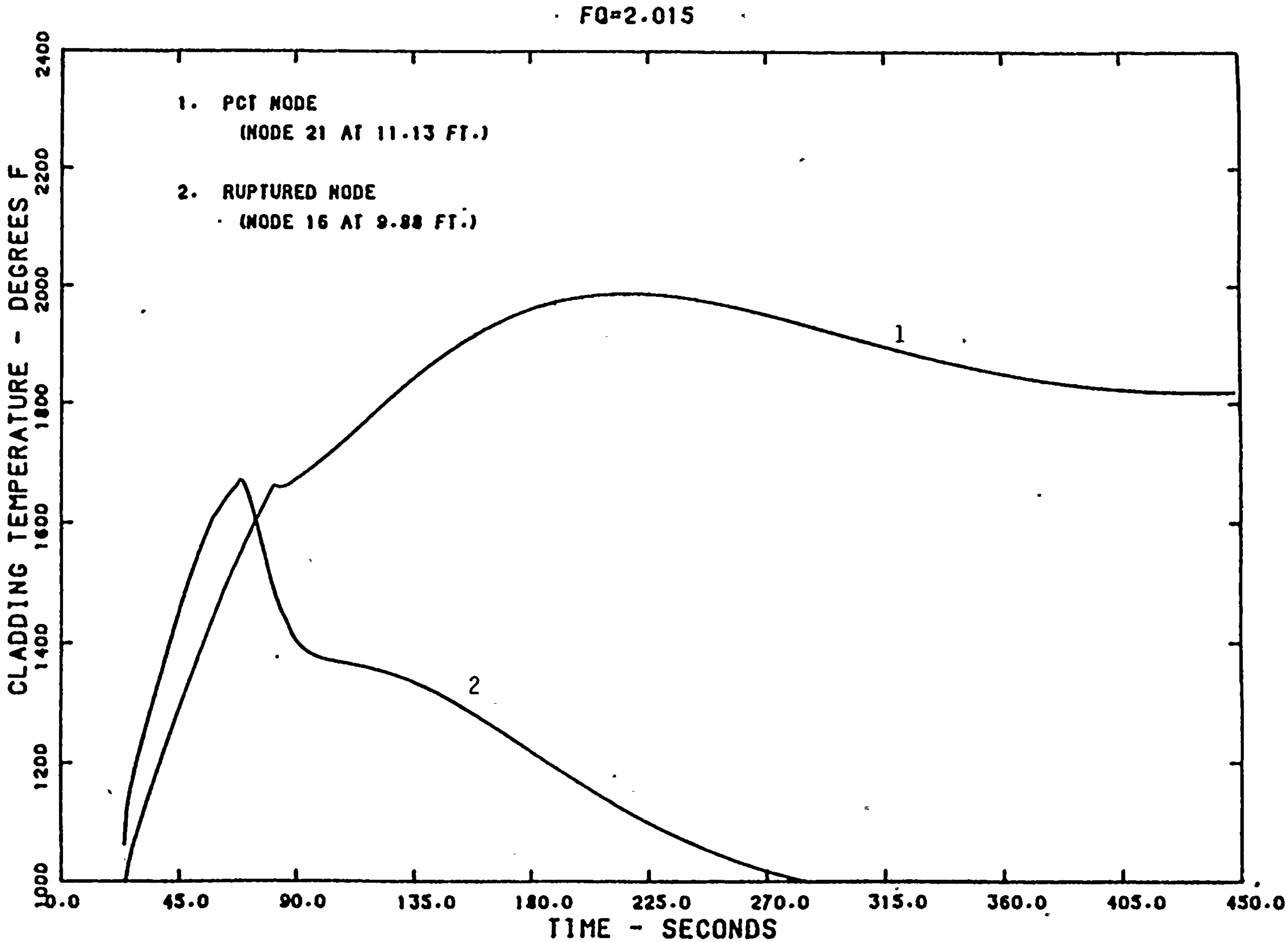


Figure 4 D.C. Cook Unit 2, EOC Shape, Two Point RHR Injection Case

ATTACHMENT 4 TO AEP:NRC:1024A

TABLE 2 TO SMALL-BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS

UNIT 2

TABLE 2

PLANT CONDITIONS

PARAMETERS	COOK 2	REFERENCE 412
LICENSED POWER (MWt)	3411	3338
LICENSED PEAKING FACTOR	2.10	2.40
FUEL VOLUMETRIC HEAT GENERATION FOR TOTAL CORE (TOTAL KW/TOTAL FT^3 FUEL)	9834.46	9624.0
AVERAGE LINEAR HEAT GENERATION (KW/FT)	5.554	5.435
PEAK LINEAR HEAT GENERATION (KW/FT)	11.663	13.043
TOTAL FLUID VOLUME IN CORE* (FT^3)	643.0	642.9
THERMAL DESIGN FLOW (LBS/SEC.)	37388.9	36916.7
VESSEL EXIT TEMPERATURE (^{o}F)	606.4	608.0
VESSEL INLET TEMPERATURE (^{o}F)	541.3	542.2

* - Core Fluid Volume Does Not Include Core Thimble Tube Volume

ATTACHMENT 5 TO AEP:NRC:1024A

RESPONSE TO MR. ROBERT JONES' QUESTIONS ON
SMALL-BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS

Question: How does the flow split between the two SI lines when one line blows down to approximately atmospheric pressure?

Response: The SBLOCA analysis assumes that SI is injected into one RCS loop while the other loop spills to the containment atmosphere through the 4-inch hole created in the SBLOCA. Both lines of SI will supply approximately the same amount of water to the RCS, because the entire RCS is at approximately the same pressure even though the break is in one loop. The back pressure on both lines is approximately the same. Since it is assumed that one line spills out the break, one half of the SI flow is then available to refill and cool the core.

Question: Is the 4-inch equivalent diameter cold leg break still the limiting case?

Response: As stated in Attachment 1 several arguments support analyzing the 4-inch break for the cross-tie closed for D. C. Cook Unit 1:

1. The previous D. C. Cook Unit 1 WFLASH analysis and the reference four-loop plant NOTRUMP analysis both exhibit 4-inch break as limiting.

2. Documented sensitivities demonstrate that reductions of this magnitude in safety injection do not affect the reactor coolant system (RCS) depressurization transient for a fixed break size. Also, the reference plant exhibits core recovery for FSAR spectrum break sizes on accumulator injection. Therefore, the small break penalty due to lower SI is restricted to that portion of the transient when RCS cold leg pressure remains above the 600 psia accumulator pressure. Based on conservative estimates, the increase in peak clad temperature due to the decrease in safety injection would not be large enough to shift the limiting break size to the 3-inch break.

Question: The analysis describes a situation where the boiloff exceeds the SI flow rate until the accumulators inject. After the accumulator injection, at what point does SI flow rate match or exceed the boiloff?

Response: At 1200 sec, the core boiloff is about 95 lbm/sec compared with 56 lbm/sec of SI. Additional reference plant cases indicate that at 1600 sec there is 82 lbm/sec of core boiloff; however, SI at the reduced rates is delivering only 58 lbm/sec of flow. (Note: Reference plant cases indicate minimal core re-uncovery due to intermittent accumulator injection; however, for FSAR cases it is standard to run the transient until SI flow exceeds break flow.) For this reduced SI case, the transient would have to be run long enough to allow system depressurization to the low-head SI pump pressure, in order for the SI to exceed core boiloff.

DISTRIBUTION:
Docket Files
DWigginton
PKreutzer
PDIII-3 r/f

May 13, 1987

DOCKET NO(S). 50-315/316
Mr. John Dolan, Vice President
Indiana aad Michigan Electric Company
c/o American Electric Power Service Corporation
1 Riverside Plaza
Columbus, OH 43216

SUBJECT: DONALD C. COOK NUCLEAR PLANTS

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

- [ ] Notice of Receipt of Application, dated ________.
- [ ] Draft/Final Environmental Statement, dated ________.
- [ ] Notice of Availability of Draft/Final Environmental Statement, dated ________.
- [ ] Safety Evaluation Report, or Supplement No.________ dated ________.
- [ ] Environmental Assessment and Finding of No Significant Impact, dated ________.
- [ ] Notice of Consideration of Issuance of Facility Operating License or Amendment to Facility Operating License, dated ________.
- [X] Bi-Weekly Notice; Applications and Amendments to Operating Licenses Involving No Significant Hazards Considerations, dated May 6, 87 [*see page(s)*] ________.
- [ ] Exemption, dated ________.
- [ ] Construction Permit No. CPPR-________, Amendment No. ________ dated ________.
- [ ] Facility Operating License No. ________, Amendment No. ________ dated ________.
- [ ] Order Extending Construction Completion Date, dated ________.
- [ ] Monthly Operating Report for ________ transmitted by letter dated ________.
- [ ] Annual/Semi-Annual Report- ________ ________ transmitted by letter dated ________.

Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc: See next page

OFFICE	PDIII-3 PK						
SURNAME	PKreutzer						
DATE	05/13/87						

NRC FORM 318 (10/80) NRCM 0240

OFFICIAL RECORD COPY

Mr. John Dolan
Indiana and Michigan Electric Company

Donald C. Cook Nuclear Plant

cc:
Mr. M. P. Alexich
Vice President
Nuclear Operations
American Electric Power Service
Corporation
1 Riverside Plaza
Columbus, Ohio 43215

Attorney General
Department of Attorney General
525 West Ottawa Street
Lansing, Michigan 48913

Township Supervisor
Lake Township Hall
Post Office Box 818
Bridgeman, Michigan 49106

W. G. Smith, Jr., Plant Manager
Donald C. Cook Nuclear Plant
Post Office Box 458
Bridgman, Michigan 49106

U.S. Nuclear Regulatory Commission
Resident Inspectors Office
7700 Red Arrow Highway
Stevensville, Michigan 49127

Gerald Charnoff, Esquire
Shaw, Pittman, Potts and Trowbridge
2300 N Street, N.W.
Washington, DC 20037

Mayor, City of Bridgeman
Post Office Box 366
Bridgeman, Michigan 49106

Special Assistant to the Governor
Room 1 - State Capitol
Lansing, Michigan 48909

Nuclear Facilities and Environmental
Monitoring Section Office
Division of Radiological Health
Department of Public Health
3500 N. Logan Street
Post Office Box 30035
Lansing, Michigan 48909

The Honorable John E. Grotberg
United States House of Representatives
Washington, DC 20515

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

J. Feinstein
American Electric Power
Service Corporation
1 Riverside Plaza
Columbus, Ohio 43216