REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR: 8703310374 DOC. DATE: 87/03/23 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
DENTON, H. R.	Document Control Branch (Document Control Desk)

SUBJECT: Requests concurrence w/interpretation of Tech Specs 3.5.2.e & 3.5.3.d for both units re min number of flow paths from safety injection pumps & RHR pumps to reactor coolant sys. Related documentation encl. Fee paid.

DISTRIBUTION CODE: A046D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 36
TITLE: OR Submittal: TMI Action Plan Rqmt NUREG-0737 & NUREG-0660

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PWR-A ADTS	1	1	PWR-A EB	1	1
	PWR-A EICSB	2	2	PWR-A FOB	1	1
	PWR-A PD4 LA	1	0	PWR-A PD4 PD	5	5
	WIGGINGTON, D	1	1	PWR-A PSB	1	1
	PWR-A RSB	1	1			
INTERNAL:	ADM/LFMB	1	0	AEOD/PTB	1	1
	IE/DEPER DIR	1	1	IE/DEPER/EPB	3	3
	NRR BWR ADTS	1	1	NRR PAULSON, W.	1	1
	NRR PWR-A ADTS	1	1	NRR PWR-B ADTS	1	1
	NRR/DSRO EMRIT	1	1	OGC/HDS1	1	0
	REG FILE 01	1	1			
EXTERNAL:	LPDR	1	1	NRC PDR	1	1
	NSIC	1	1			

W/Check $150
#057-0346

TOTAL NUMBER OF COPIES REQUIRED: LTTR 30 ENCL 27

[illegible]

[illegible]
[illegible]
[illegible]
[illegible]
[illegible]
[illegible]
[illegible]

[illegible]
[illegible]
[illegible]
[illegible]

[illegible]
[illegible]

[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]		[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
			[illegible]	[illegible]	[illegible]
	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
			[illegible]	[illegible]	[illegible]
	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
			[illegible]	[illegible]	[illegible]

[illegible]

INDIANA & MICHIGAN ELECTRIC COMPANY

P.O. BOX 16631
COLUMBUS, OHIO 43216

March 23, 1987

AEP:NRC:1024

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 PROGRAM AND SELECTED TECHNICAL INFORMATION REGARDING TECHNICAL SPECIFICATIONS FOR OPERATION WITH LIMITED SI OR RHR FLOW PATHS

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

ATTN: Mr. H. R. Denton

Dear Mr. Denton:

The purpose of this letter is to request your concurrence with an interpretation of Technical Specifications (T/Ss) 3.5.2.e and 3.5.3.d for both units of the Donald C. Cook Nuclear Plant, concerning the minimum number of flow paths from the safety injection (SI) pumps and residual heat removal (RHR) pumps to the reactor coolant system.

T/S 3.5.2 states that two emergency core cooling system (ECCS) subsystems must be operable; it defines an operable ECCS subsystem as including one operable SI pump, one operable RHR pump, and associated flow paths. This T/S also allows the operator to remove one ECCS subsystem for up to 72 hours while in Modes 1, 2, or 3 while maintaining an operable flow path for the opposing subsystem. T/S 3.5.3 states that two ECCS subsystems must be operable; it defines an operable ECCS subsystem as including one operable RHR pump and associated flow paths. This T/S allows the operator to remove one ECCS subsystem for up to 72 hours while in Mode 4 while maintaining an operable flow path for the opposing subsystem.

We are proposing to interpret T/S 3.5.2 such that the flow path portion of one ECCS subsystem is considered operable when one SI pump or one RHR pump can deliver flow to only two reactor coolant system (RCS) loops while the other pump within the same subsystem is aligned to deliver flow to four loops. We are proposing to interpret T/S 3.5.3 such that the flow path portion of one ECCS subsystem is considered operable when an RHR pump can deliver flow to only two RCS loops. We are also asking that the interpretation be extended to any configuration (such as servicing an RHR heat exchanger) in which 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.

A046 w/check $150
1/1
#057-0346

We intend to justify this request by means of analyses which will show that in the case of a small- or large-break LOCA, the configurations we are requesting would provide sufficient flow to maintain peak clad temperature below the value stated in 10 CFR 50.46. The small-break LOCA analyses will show that one SI pump delivering flow to two RCS loops will provide sufficient cooling. The large-break LOCA analyses will show that the combination of (1) one RHR pump delivering flow to two RCS loops and (2) one SI pump delivering flow to four RCS loops will provide sufficient cooling. We are requesting your concurrence with an interpretation of these T/Ss that will allow us to operate in these configurations so that we can continue uninterrupted operation while performing maintenance and required testing on these systems.

An example of such maintenance and testing is work that requires the closing of SI or RHR cross-tie valves. The consequence of closing these valves is that SI or RHR flow would be delivered from the affected pumps to only two of the four RCS loops. Attachment 1 to this letter presents two flow diagrams for the Donald C. Cook Nuclear Plant. These diagrams (1-5104C-3 and 2-5104C-3) show the flow arrangements of the SI and RHR systems for Units 1 and 2, including the cross-tie valves IMO 270, 275, 314, and 324 for both units.

We believe that this is a generic issue. Several instances of operation with the RHR cross-tie closed and a single RHR pump operating have occurred in other Westinghouse plants, as cited in IE Information Notice 87-01. A copy of this Information Notice is included as Attachment 2 to this letter. We believe the analyses presented to justify our request will serve as a basis for solving the generic concern as applied to the Donald C. Cook Nuclear Plant.

This issue was discussed with members of the NRR staff and the Region III staff and was in part the subject of an Enforcement Conference held at Region III headquarters on January 21, 1987. As we stated at that conference, until very recently both units of the Donald C. Cook Nuclear Plant had been operated under the assumption that a single SI or RHR pump delivering flow into two separate loops constituted an operable flow path for that pump. This allowed performance of required maintenance and testing without having to bring the unit to cold shutdown. IE Inspection Report 86042, a copy of which is included in Attachment 3 to this letter, raised the issue of whether closing the SI cross-ties resulted in an acceptable degradation of safety according to the accident analysis. Two LERs, copies of which are also included in Attachment 3, were written on this subject.

In our detailed review of the LER resulting from the NRC Inspection Report on our SI pump cross-ties, we found that a similar situation existed with the RHR cross-ties as well. As a result, we requested an ASME code exemption so that we would be able to avoid testing certain valves while at power, since testing at power would provide an inoperable ECCS flow path during the time we were operating on a single pump and the RHR cross-ties were closed. The NRC responded to us in a letter dated December 19, 1986 from B. J. Youngblood to John E. Dolan (see Attachment 4). That letter, which granted temporary relief for testing the valves, also stated in part

that "The IMEC request for permanent relief for testing of these valves is considered premature in light of the test configuration analysis underway for D. C. Cook." We believe that being allowed to interpret the T/Ss in the manner indicated, based upon the information to be provided, will serve as the basis for closing the issues identified by the NRC.

Presented below is a historical and licensing discussion of the factors leading to the decision to make the request presented above, a discussion of the alternatives we have considered, and a summary of the action we would like the NRC staff to take in this matter.

Historical and Licensing Basis

Unit 1 of the Donald C. Cook Nuclear Plant is currently in the last stages of transition from fuel supplied by Advanced Nuclear Fuels Corporation (formerly Exxon Nuclear Company) to fuel supplied by Westinghouse Electric Corporation. Unit 1 currently contains 159 15 x 15 assemblies supplied by Westinghouse and 34 fuel assemblies supplied by Advanced Nuclear Fuels Corporation (ANFC). After the next reload (scheduled to begin at the end of May or early June 1987) the core will be fueled entirely with 15 x 15 fuel supplied by Westinghouse. Unit 2 is fueled with 17 x 17 fuel supplied by ANFC.

The latest SBLOCA analysis of record for Unit 1 is the analysis submitted to the NRC with our letter AEP:NRC:0860A, dated March 15, 1985 and approved by the NRC in their Safety Evaluation Report accompanying Amendment 84 for Unit 1 and Amendment 64 for Unit 2. The Unit 1 analysis was also determined to be acceptable for determining peak clad temperature (PCT) of the Unit 2 limiting break, and is referenced in XN-NF-85-28(P), Supplement 1, "D. C. Cook Unit 2, Cycle 6 Safety Analysis Report: Disposition of Standard Review Plan Chapter 15 Events."

In preparing for the January 21, 1987 Enforcement Conference, we contracted with Westinghouse to perform comparative evaluations using the NOTRUMP computer code to evaluate the effects of closing the safety injection pump cross-tie on Unit 2 with one safety injection pump operable. Westinghouse performed a SBLOCA evaluation using the NOTRUMP code because (1) it was approved for application to the Donald C. Cook Nuclear Plant (see May 24, 1985 letter from S. Varga, NRC, to John E. Dolan, AEPSC in Attachment 5) and (2) it was stated in that letter that "Future plant specific analyses performed for your plant by Westinghouse for reloads or Technical Specification amendments (those beyond 90 days of the date of this letter) should be calculated with the new code, NOTRUMP."

The Westinghouse SBLOCA evaluation for Unit 2 is based on a comparison of the Donald C. Cook Plant to a similar four-loop Westinghouse plant using the NOTRUMP computer code. NOTRUMP, rather than WFLASH, was used for the comparative evaluation in accordance with the May 24, 1985 letter mentioned above. A comparative evaluation (rather than a Cook-specific analysis) was performed because we were informed by Westinghouse that time did not permit the performance of Cook-specific analyses in preparation for the enforcement conference.

The report of that evaluation (which is included in Attachment 6) states:

> "The estimated PCT assuming HHSI [Safety Injection] cross-tie closure and the single failure of a diesel generator to start when all ECCS pumps are available was estimated to be approximately 1482°F. This indicates there is significant margin to the 2200°F limit of 10 CFR 50.46 for the operation of D. C. Cook Unit 2 at a licensed core power of 3411 MWt with the High Head Safety Injection System cross-tie connection closed."

We believe this statement supports our position that the plant can be within current regulatory limits with the safety injection cross-ties closed.

Alternatives

Before making this request, we considered several alternatives. These include (a) operation in compliance with the literal T/S requirements and no further request for NRC action, (b) system hardware modifications to allow compliance with the T/S requirements as currently interpreted, and (c) plant-specific analyses.

Alternative (a) was not desirable because it could require numerous plant shutdowns at regular intervals for required testing and/or for relatively minor maintenance. This would result in unnecessary thermal cycling of the system, would unnecessarily burden the plant staff, and would result in a loss of ability to provide continued power generation. Such shutdowns would last a week or longer and cost our ratepayers between $200,000 and $600,000 per day per unit in replacement power.

Alternative (b) was not desirable because of the extensive nature of the hardware changes which would be required to allow continued plant operation with the cross-ties closed, while still maintaining the active/passive failure requirements that are part of the plant design basis. These hardware changes, which could include both piping changes and addition of qualified motor-operated valves, would require extensive time to complete, and would involve extensive plant shutdowns.

Alternative (c) was not desirable because it would involve expensive and time-consuming plant-specific analyses. We currently anticipate performing substantial new analyses to permit operating the Cook units at lower temperatures and pressures to extend steam generator lifetimes. However, we would prefer to approach the issue in a considered and deliberate manner. Since the large- and small-break LOCA analyses involve many aspects (both design and operational) of the plant, we believe it would be more effective and economical to review all aspects of the analyses and structure the effort to provide the maximum benefit. Until such analyses have been completed, we feel the comparative analysis approach as suggested in this letter is warranted.

Analyses to be Transmitted and Requested NRC Action

The analysis of record for both units assumes that SI and RHR flow is delivered to all four loops with a single pump of each subsystem operational. As mentioned above, we are enclosing or will transmit to you the following documents to support our request:

1. An evaluation of the consequences of two-loop injection on the small-break loss-of-coolant accident (SBLOCA) for Units 1 and 2. As mentioned above, the evaluation for Unit 2 has been completed and is included as Attachment 6 to this letter. A similar evaluation of the small-break analysis of record for Unit 1 is also under preparation by Westinghouse.

2. An evaluation of the consequences of two-loop injection on the large-break loss-of-coolant accident (LBLOCA) for Units 1 and 2. These evaluations are also under preparation by Westinghouse and Advanced Nuclear Fuel Corporation for Units 1 and 2 respectively and are scheduled for completion by mid-April 1987. We expect that the evaluations will show that two-loop injection using one RHR pump is acceptable, provided a single SI pump is delivering flow to four loops.

3. A justification that the generic analysis presented in WCAP-11145 is applicable to Unit 2 under the present operating conditions, as suggested in the letter dated December 22, 1986 from Mr. B. J. Youngblood of the NRC to Mr. J. E. Dolan of AEP. (See Attachment 5.) This evaluation is anticipated to be completed by mid-April 1987.

We believe that NRC acceptance of this interpretation will resolve the NRC-stated concerns with testing according to the ISI/IST ASME code as described in Attachment 4 and will allow closure of the open items as referenced in Attachment 5 regarding NUREG-0737, Item II.K.3.31.

In turn, we request that you inform us within two weeks of receipt of this letter whether the information presented here concerning our request and the technical information in Attachment 6 are adequate to permit you to reach a decision on this T/S interpretation by May 15, 1987. The reason we are requesting a prompt response is that until we receive a response, maintenance or testing of the emergency core cooling of the safety injection or RHR pump isolation valves may require unnecessary shutdown of both units. If you have any questions, my staff is prepared to discuss this matter by telephone or meet with you to provide further information.

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

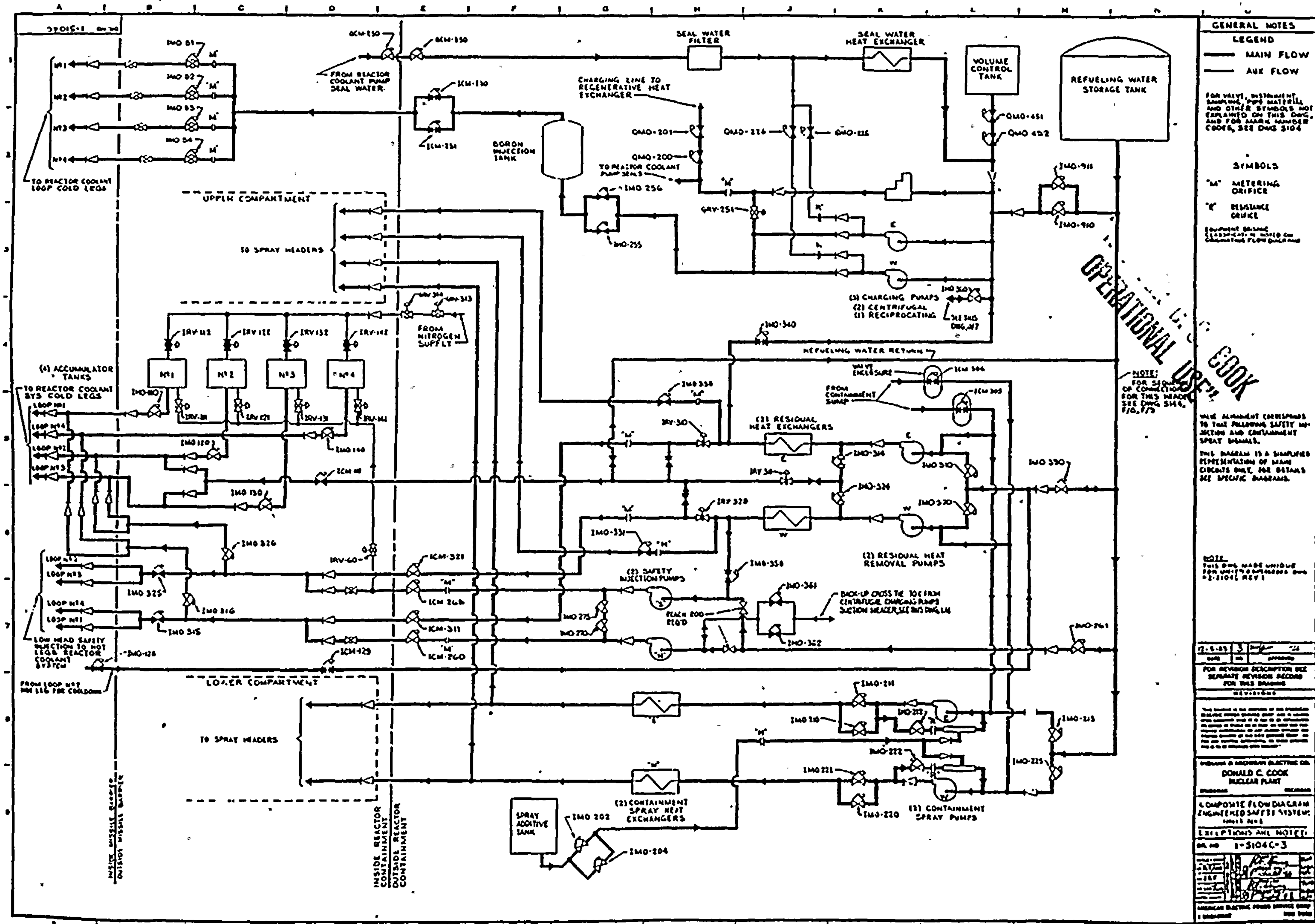
cm

Attachments

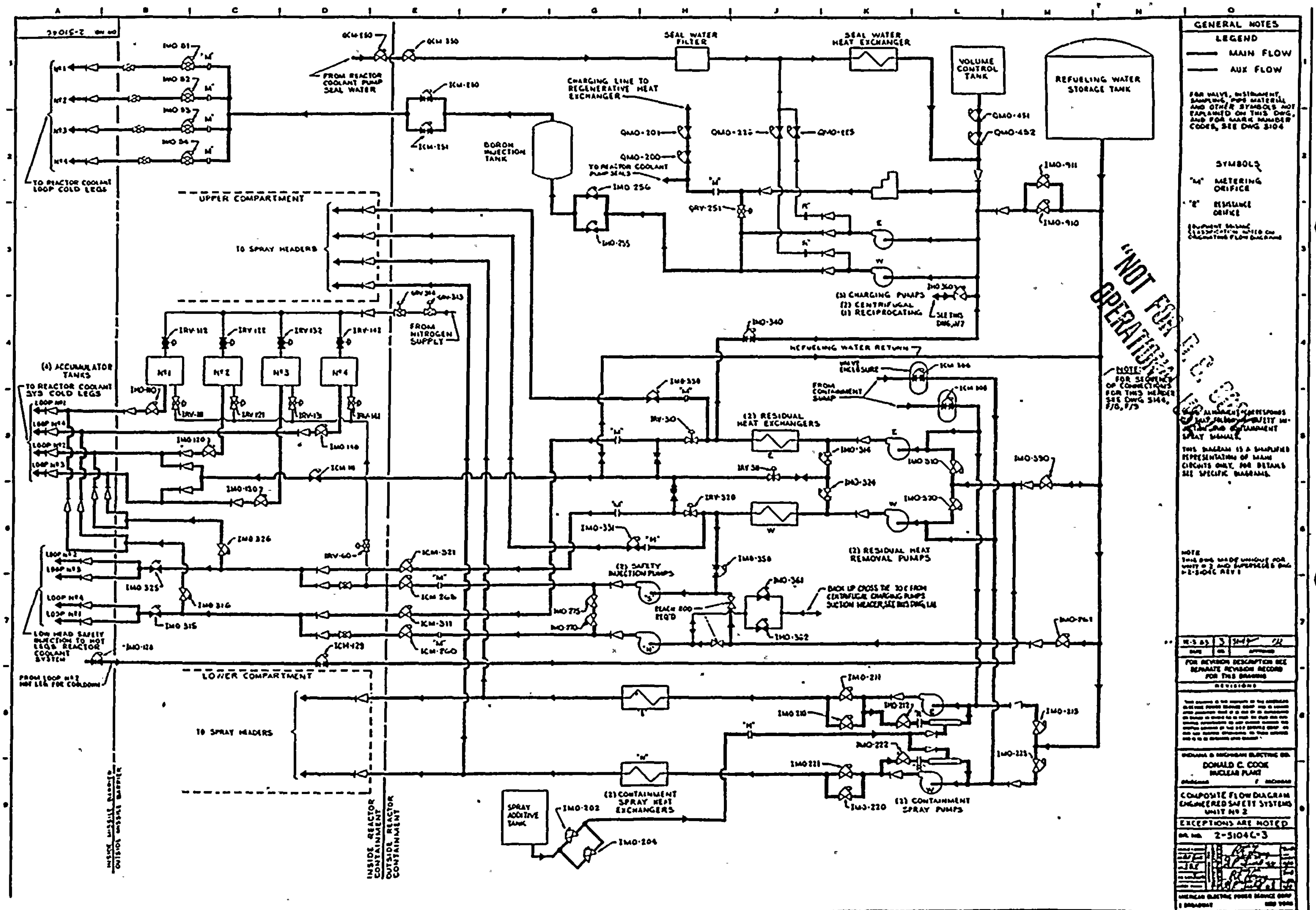
cc: John E. Dolan
W. G. Smith, Jr. - Bridgman
R. C. Callen
G. Bruchmann
G. Charnoff
NRC Resident Inspector - Bridgman (w/ attachments)
A. B. Davis, Region III (w/ attachments)

Attachment 1 to AEP:NRC:1024

Unit 1 and 2 ECCS Flow Diagrams



GENERAL NOTES
LEGEND
MAIN FLOW
AUX FLOW
SYMBOLS
"M" METERING ORIFICE
"R" RESISTANCE ORIFICE
SEAL WATER FILTER
SEAL WATER HEAT EXCHANGER
VOLUME CONTROL TANK
REFUELING WATER STORAGE TANK
FROM REACTOR COOLANT PUMP SEAL WATER
CHARGING LINE TO REGENERATIVE HEAT EXCHANGER
BORON INJECTION TANK
TO REACTOR COOLANT PUMP SEALS
TO REACTOR COOLANT LOOP COLD LEGS
UPPER COMPARTMENT
TO SPRAY HEADERS
(3) CHARGING PUMPS
(2) CENTRIFUGAL
(1) RECIPROCATING
FROM NITROGEN SUPPLY
(4) ACCUMULATOR TANKS
TO REACTOR COOLANT SYS COLD LEGS
REFUELING WATER RETURN
VALVE ENCLOSURE
FROM CONTAINMENT SUMP
(2) RESIDUAL HEAT EXCHANGERS
(2) RESIDUAL HEAT REMOVAL PUMPS
(2) SAFETY INJECTION PUMPS
LOW HEAD SAFETY INJECTION TO HOT LEGS REACTOR COOLANT SYSTEM
FROM LOOP NO 2 HOT LEG FOR COOLDOWN
LOWER COMPARTMENT
TO SPRAY HEADERS
(2) CONTAINMENT SPRAY HEAT EXCHANGERS
(2) CONTAINMENT SPRAY PUMPS
SPRAY ADDITIVE TANK
INSIDE REACTOR CONTAINMENT
OUTSIDE REACTOR CONTAINMENT
OPERATIONAL USE
COOK
DONALD C. COOK NUCLEAR PLANT
COMPOSITE FLOW DIAGRAM ENGINEERED SAFETY SYSTEMS
1-5104C-3



GENERAL NOTES
LEGEND
MAIN FLOW
AUX FLOW
SYMBOLS
"M" METERING ORIFICE
"R" RESISTANCE ORIFICE
THIS DIAGRAM IS A SIMPLIFIED REPRESENTATION OF MAIN CIRCUITS ONLY. FOR DETAILS SEE SPECIFIC DIAGRAMS.
SEAL WATER FILTER
SEAL WATER HEAT EXCHANGER
VOLUME CONTROL TANK
REFUELING WATER STORAGE TANK
FROM REACTOR COOLANT PUMP SEAL WATER
CHARGING LINE TO REGENERATIVE HEAT EXCHANGER
BORON INJECTION TANK
TO REACTOR COOLANT PUMP SEALS
TO REACTOR COOLANT LOOP COLD LEGS
UPPER COMPARTMENT
TO SPRAY HEADERS
(3) CHARGING PUMPS
(2) CENTRIFUGAL
(1) RECIPROCATING
FROM NITROGEN SUPPLY
(4) ACCUMULATOR TANKS
TO REACTOR COOLANT SYS COLD LEGS
REFUELING WATER RETURN
FROM CONTAINMENT SUMP
VALVE ENCLOSURE
(2) RESIDUAL HEAT EXCHANGERS
(2) RESIDUAL HEAT REMOVAL PUMPS
(2) SAFETY INJECTION PUMPS
LOW HEAD SAFETY INJECTION TO HOT LEGS REACTOR COOLANT SYSTEM
FROM LOOP NO 2 HOT LEG FOR COOLDOWN
LOWER COMPARTMENT
TO SPRAY HEADERS
SPRAY ADDITIVE TANK
(2) CONTAINMENT SPRAY HEAT EXCHANGERS
(2) CONTAINMENT SPRAY PUMPS
INSIDE REACTOR CONTAINMENT
OUTSIDE REACTOR CONTAINMENT
"NOT FOR OPERATION"
FOR REVISION DESCRIPTION SEE SEPARATE REVISION RECORD FOR THIS DRAWING
REVISIONS
DONALD C. COOK NUCLEAR PLANT
COMPOSITE FLOW DIAGRAM ENGINEERED SAFETY SYSTEMS UNIT NO 2
EXCEPTIONS ARE NOTED
2-5104C-3

D.C. Cook Unit 2 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 2 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 2 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 2 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 delivery 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 2 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 2. The reference four loop plant used to determine the safety injection sensitivity is essentially identical to Cook 2 in vessel design and loop components. Table 1 provides a comparison of the basic vessel and components design of D.C.Cook Unit 2 with the reference plant design. 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 2 and the reference plant design. The major differences noted between the plants include the licensed core power, the licensed peaking factor, and the fuel type.

Both D.C.Cook Unit 2 and the reference plant operate with a 17X17 fuel array. The current cycle of operation at D.C.Cook Unit 2 contains an EXXON fuel core with the exception of one Westinghouse fuel assembly, while the reference plant has a complete 17X17 Westinghouse standard fuel core. The difference in fuel pellet outer diameter, fuel rod outer diameter, and fuel rod pitch are small between the EXXON 17X17

fuel and the Westinghouse 17X17 standard fuel. Consequently, the effects of fuel parameter differences are expected to have only a small effect on the transient response.

The total core power level influences the depth and duration of core uncovery. The decay heat generation rate, which is a function of the total core power level, determines the rate of core boiloff steam production in conjunction with the safety injection flow rate following a small break LOCA. The reactor coolant system thermal-hydraulic response to a small break LOCA, as calculated by the NOTRUMP code, is then dependent upon the ratio of the core power to safety injection flow rate. Higher safety injection flow delivery rates provide more core subcooling to absorb decay heat as well as providing more mass addition to keep the core covered. Consequently, the effect of the difference in the total core power level between the plant designs is diminished due to the influence of the safety injection flow. The core power to safety injection flow ratio has been determined at a representative pressure of 1114.7, psia which approximately corresponds to the quasi-equilibrium pressure at which the primary system tends to stabilize prior to the venting of the pump suction leg loop seal. Following loop seal venting, core boiloff exceeds the safety injection mass flow rate and a core uncovery transient results.

The ratio of core power to the safety injection mass flow rate for the reference plant is 81.814 (MW/lbm/sec). For D.C. Cook Unit 2 in the current design basis configuration, one HHSI pump and one charging pump inject through four lines. The core power to safety injection mass flow rate is 65.345 (MW/lbm/sec). If a small break LOCA FSAR analysis were performed for D.C. Cook Unit 2 the results would be less limiting than the result of the FSAR NOTRUMP analysis performed for the reference plant. Hence the reference four loop plant used in the NOTRUMP analyses is expected to provide a representative thermal hydraulic response to a small break LOCA for variations in the safety injection flow for D.C. Cook Unit 2.

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 2 licensing basis WFLASH analysis. The analysis assumed a safety injection flow rate representative of the HHSI flow configuration at D.C. Cook Unit 2 during a recent License Event Report (LER) in which two charging pumps were available to deliver flow through four lines and one HHSI pump was available to deliver flow through two of four lines. D.C. Cook Unit 2 issued the LER for operation in Mode 1 (at 80% full power) with one HHSI pump removed from service and the cross tie line closed. Closure of the cross-tie line did not meet the design basis small break LOCA analysis assumption of HHSI flow delivery to all four loops. 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. In order to support the operation of D.C. Cook Unit 2 with two charging pumps and one HHSI, the loss of offsite power without the single failure of a diesel to start was assumed. It was conservatively assumed that all safety injection flow delivered to the broken loop spilled out the break. The analysis was performed to determine the effect on plant response for a change in the SI flow assumption, thereby establishing a safety injection sensitivity for a plant similar in design to D.C. Cook 2. 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 1132°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.

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 2.4 and resulted in a PCT of 1244°F for the limiting four inch break. The analysis performed to determine the SI sensitivity employed the same assumptions as the reference plant FSAR analysis with the exception of the safety injection flow delivery flowrates. 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 analyses and the evaluation. The safety injection flow vs. pressure curve assumed in the reference four loop plant FSAR analysis is shown in figure 11. This may be compared to the safety injection flow assumed for the sensitivity analysis. The safety injection flows for the proposed HHSI ECCS design basis operation with closure of the cross tie connection is also shown in figure 11. Table 3 provides a comparison of the reference plant analyses assumptions to the D.C. Cook Unit 2 FSAR analysis assumptions.

The four inch break for the NOTRUMP analysis employing SI flows used in the sensitivity 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 356 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 2 original FSAR analysis to the reference plant analyses. The results reported for Cook Unit 2 reflect the original WFLASH analysis specific

to D.C. Cook Unit 2. A footnote has been provided which reports the results of 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 and was determined to bound operation of D.C. Cook Unit 2 with the increase in HHSI miniflow. This analysis was used to support the D.C. Cook Unit 2 license amendment for an increase in HHSI miniflow from 30 gpm to 60 gpm. All references to the D.C. Cook Unit 2 current design basis SI flows assume the SI available with a HHSI miniflow of 60 gpm.

Figures for the reference plant analyses have been provided to illustrate the influence of the change in safety injection flow rate on the pertinent parameters identifying the plant response to the small break LOCA. Figure 1 shows the depressurization transient for the two reference plant analyses for the pressurizer pressure. Comparison of the two depressurization transients indicates that the change in the safety injection flow rates has very little influence on the system depressurization between the two analyses. Slight differences are apparent after the time of the pump suction loop seal clearing and are primarily due to the differences in the rate of core boiloff to safety injection flow delivery between the two cases. Figures 2 and 3 represent the broken loop and intact loop secondary side steam pressure response during the transient respectively. Comparison of the response for the two analyses shows that they are essentially identical prior to loop seal venting indicating that the amount of primary decay heat removal via the steam generator secondaries is approximately the same in the two cases. Figure 4 shows the core mixture level which illustrates the major effect of the change in the amount of total safety injection flow available in the two transients. A comparison of the uncovery transient of the two analyses shows that the reference FSAR case, which has less safety injection flow available than the SI sensitivity case prior to accumulator injection, experiences a deeper core uncovery of slightly longer duration. Figure 5 represents the pumped safety injection flow for the two cases. Figure 6 represents

the accumulator flow for the two cases. The time of initial accumulator injection is slightly delayed in the SI sensitivity case due to the delay in loop seal steam venting resulting from the increased safety injection mass flow rate. However, the initial delivery of accumulator water in conjunction with the higher safety injection flows is sufficient to initiate a rapid core recovery and a higher degree of subcooling which results in additional accumulator delivery earlier in the SI sensitivity case. Figure 7 provides a comparison of the total break flow for the two cases. In the SI sensitivity case a greater amount of safety injection flow to the intact loop will travel around the downcomer and flow into the broken loop cold leg causing a higher degree of subcooling. The change in enthalpy resulting from the subcooled fluid injection to the broken loop cold leg results in a slightly higher liquid break flow rate from 250 to 375 seconds. After loop seal clearing, the two cases show essentially identical break flow behavior.

These figures show that the change in the available safety injection water has a negligible impact on the depressurization transient, the secondary side pressure response and the break flow rate. The net effect of safety injection flow differences on small break LOCA response is in the depth and duration of the core uncovery transient. The effect of the reduction in SI flow on the plant response to a small break LOCA resulting from the proposed design basis change would therefore be to slightly increase the depth and duration of core uncovery. The increase in PCT resulting from the additional core uncovery can be estimated from the sensitivity obtained from the reference plant analyses.

The peak clad temperature and the peak fuel (pellet) temperature results for the two reference plant analyses are presented in figure 8 and figure 9 respectively. In figure 8 the clad average temperature for the hot rod of the reference plant FSAR analysis is compared to the hot rod clad average temperature of the SI sensitivity analysis. The

results confirm that the case with the higher power to SI flow ratio will result in a more limiting PCT. The clad average temperature in the hottest rod is determined assuming the power in the hottest rod to be equal to the Peak Linear Power (Peaking Factor * Avg. Linear Power). Clad heat up calculations specific to D.C. Cook 2 would result in slightly lower PCT's since the D.C. Cook 2 Peak Linear Power is limited by the large break LOCA F_Q. Figure 9 shows a comparison of the fuel pellet average temperature in the hottest fuel rod for the two cases. The pellet average temperature for both cases displays the same behavior throughout the transient as the corresponding clad average temperature with slightly higher temperature peaks than the clad average temperature.

Figure 10 provides a comparison of the D.C. Cook Unit 2 current design basis safety injection flow rates to the safety injection flow rates which would be assumed available in the small break LOCA analysis under the proposed design basis change. The effect of these safety injection flow changes on the small break LOCA peak cladding temperature will be determined from the safety injection flow sensitivity obtained from the reference plant analyses. In Figure 11 the reference plant analyses safety injection delivery rates are compared to the design basis change safety injection flow rates which consider the flow reduction from the cross tie closure. An evaluation of the effect of the proposed design basis change safety injection flows on a D.C. Cook Unit 2 small break LOCA analysis will be performed by establishing the sensitivity of peak clad temperature to variations in safety injection flow from the reference plant analyses.

The effect of the difference in licensed core power between the reference plant and D.C. Cook Unit 2 has been examined and has been shown to have only a small effect on the small break LOCA PCT. The total core power level in conjunction with the safety injection flow rate determines the rate of core boiloff steam production following a

small break LOCA. The reactor coolant system thermal-hydraulic response to a small break LOCA, as calculated by the NOTRUMP code, is then dependent upon the ratio of the core power to safety injection flow rate. This ratio for the reference plant at the representative pressure of 1114.7 psia for the design basis change SI flows has been determined to be 98.118 (MW/lbm/sec). At the D.C. Cook Unit 2 licensed core power level of 3411 MWt for the same safety injection flow, the ratio of power to SI flow is determined to be 100.26 (MW/lbm/sec). This indicates that any change in the core uncovery transient between the two cases would be negligible. In addition, the reference case analyses used to determine the sensitivity of PCT to SI flow changes were performed for a core peaking factor (F_Q) of 2.4. Hence, the hot rod clad heat up calculations were performed assuming a peak linear power higher than D.C. Cook Unit 2. The effect of the slightly higher power to SI flow ratio for D.C. Cook Unit 2 is compensated for in the hot rod clad heat up calculations. The direct sensitivity of PCT to SI flow changes obtained from the reference plant analyses is therefore applicable to D.C. Cook Unit 2 and the evaluated PCT will be representative of a D.C. Cook Unit 2 small break LOCA FSAR analysis.

A relationship can be established between the change in Peak Clad Temperature and the change in the total (integrated) safety injection flow delivered from break initiation until accumulator actuation for the two NOTRUMP analyses. This relationship may then be used to determine the Peak Clad Temperature which would result from a small break LOCA analysis with the safety injection delivery rates available with the HHSI cross tie line closed. The sensitivity established from the two analyses was determined to be

(Delta PCT)/(Delta Total SI Flow Delivered) = $-0.0162^{\circ}F/lbm$

Using this relationship and applying a conservative calculation of the integrated amount of safety injection flow which would be delivered in

a small break LOCA analysis when the HHSI cross tie is assumed closed, an FSAR grade NOTRUMP calculation for D.C.Cook Unit 2 is estimated to result in a PCT of approximately 1482°F. This estimate is based on the total amount of safety injection which would be available during the small break LOCA transient assuming that charging flow delivery from one pump is available to four lines with one line spilling to RCS pressure, and that HHSI flow delivery from one pump is available to only two of four lines with one line spilling to RCS pressure. The integrated amount of safety injection flow for the design basis change evaluation was calculated assuming the depressurization transient of the reference plant analysis. As shown in figure 1, a change in safety injection flow delivery (in the ranges being considered) has an insignificant effect on the depressurization of the RCS, hence it is appropriate to assume the depressurization of the reference plant analysis for the determination of the total safety injection delivered in a small break LOCA analysis applying the proposed design basis change SI flows. In this estimate the HHSI system cross-tie is assumed to be closed, but both HHSI pumps are assumed to be available to deliver flow to two cold legs. However, one train of safety injection flow is assumed to be lost due to the loss of off-site power and the failure of a diesel to start.

CONCLUSION

Applying the sensitivity to safety injection flow changes developed for a plant similar in design to D.C. Cook Unit 2, an estimate of the Peak Cladding Temperature which would be obtained from a small break LOCA analysis assuming a reduction in safety injection flow as a result of the HHSI system cross tie valve closure can be obtained. The estimated PCT assuming HHSI cross tie closure and the single failure of a diesel generator to start when all ECCS pumps are available was estimated to be approximately 1482°F. This indicates there is significant margin to the 2200°F limit of 10 CFR 50.46 for the operation of D.C. Cook Unit 2 at a licensed core power of 3411 MWt with the High Head Safety Injection System cross tie connection closed.

TABLE 1

COMPONENT COMPARISON
D.C. Cook Units vs. Reference 412

Components/Configuration	Cook Unit 2	Reference 412
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	17X17	17X17
Upper Head Spray Flow Percent	0.15%	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

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)	613.0	612.9
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

TABLE 3

ANALYSIS ASSUMPTIONS

PARAMETERS	COOK 2 FSAR ANALYSIS	REFERENCE 412 ANALYSES
LICENSED CORE POWER, 102% OF	3391 MWt	3338 MWt
PEAK LINEAR HEAT GENERATION	12.81 KW/FT	13.043 KW/FT
AVG. LINEAR HEAT GENERATION	5.521 KW/FT	5.435 KW/FT
TOTAL CORE PEAKING FACTOR, F_Q	2.32	2.40
FUEL TYPE	17X17 STD. WESTINGHOUSE	17X17 STD WESTINGHOUSE
SMALL BREAK MODEL	WFLASH	NOTRUMP

TABLE 4

SMALL BREAK ANALYSIS RESULTS

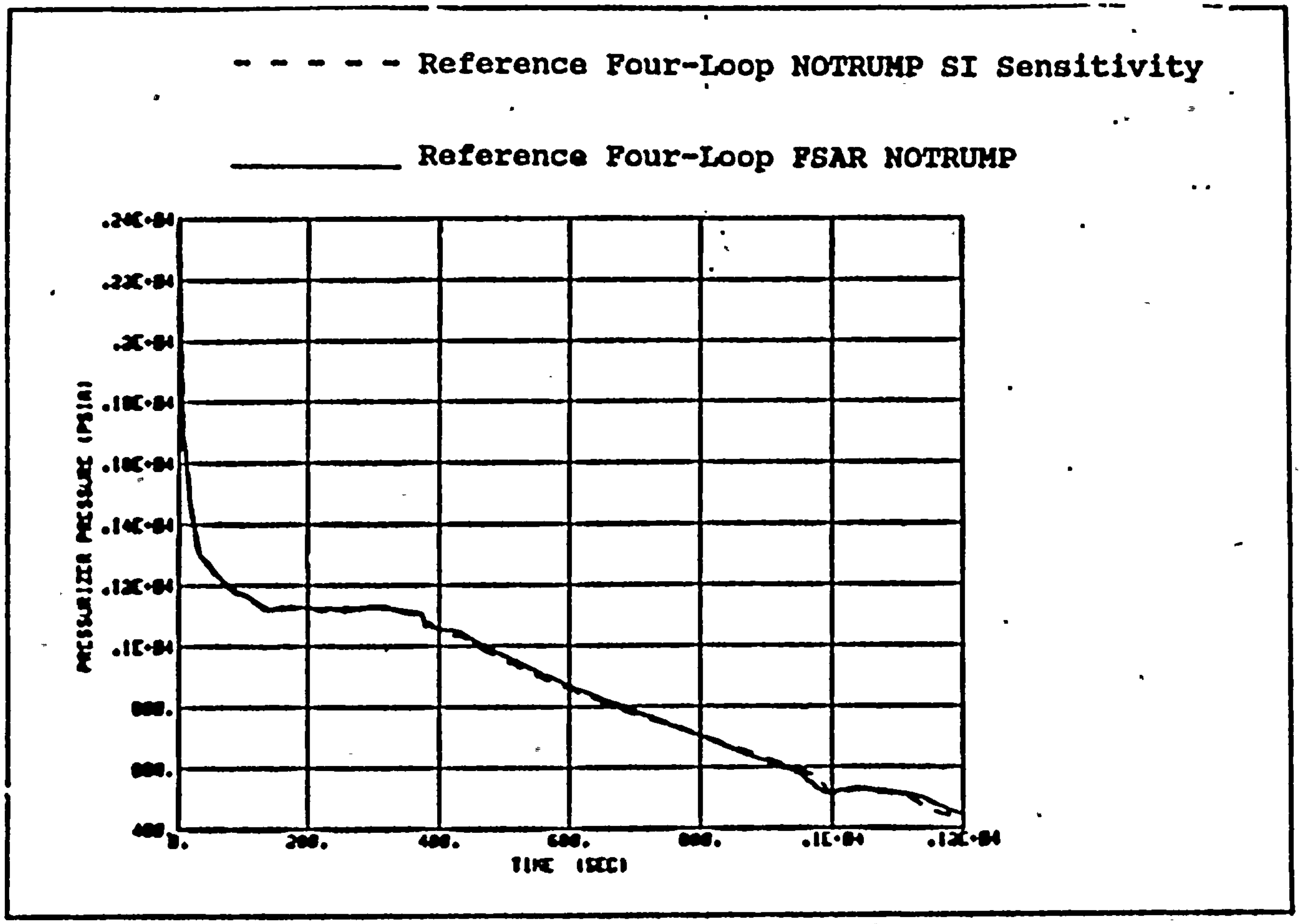
RESULTS	COOK UNIT 2 FSAR ANAL.	REFERENCE 412 FSAR ANAL.	REFERENCE 412 SI SENSITIVITY
SMALL BREAK MODEL	WFLASH	NOTRUMP	NOTRUMP
PEAK CLAD TEMPERATURE	1668°F*	1244°F	1132°F
PEAK CLAD TEMPERATURE LOCATION (FT.)	11.25	12.0	12.0
TIME OF PCT (SEC.)	913	991	1007
REACTOR TRIP (SEC.)	17.32	4.18	4.18
CORE UNCOVERY TIME (SEC)	413.0	704.0	709.5
ACCUM. INJECTION (SEC.)	875.0	919.0	932.0
CORE RECOVERY TIME (SEC.)	1650	1168	1168

* - The results of the small break LOCA analysis specific to D.C. Cook Unit 1 used to support the amendment to the D.C. Cook Unit 2 operating license for an increase in HHSI miniflow from 30 gpm to 60 gpm report a PCT of 1716°F.

Table 5

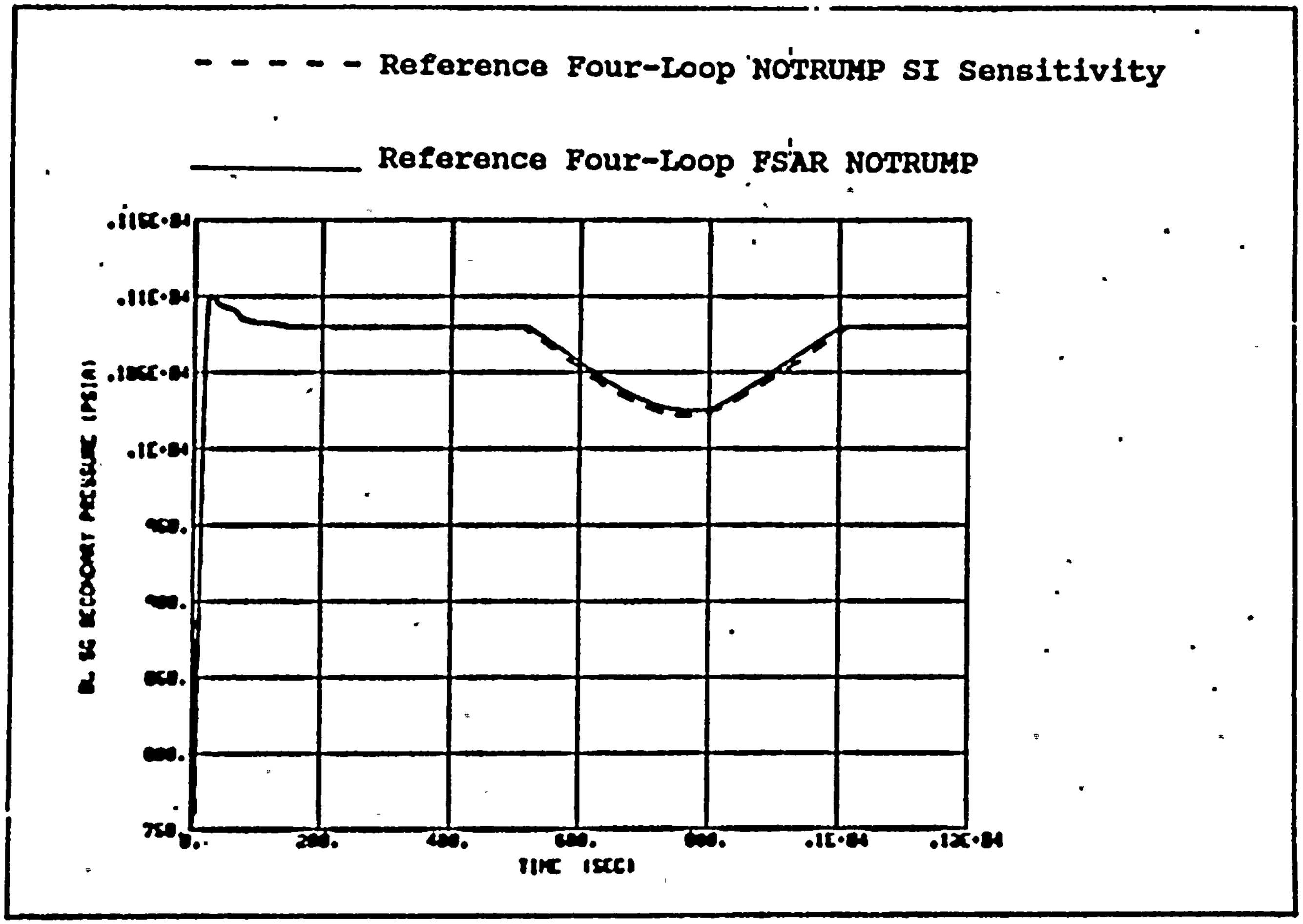
SAFETY INJECTION SYSTEM CONFIGURATION COMPARISON

	Pumps Available	Lines Available	Lines Spilling
REFERENCE PLANT	1 CHARGING	INJ. TO 4 LINES	1 LINE SPILLS
FSAR ANALYSIS	1 HHSI	INJ. TO 4 LINES	1 LINE SPILLS
D.C. COOK UNIT 2	1 CHARGING	INJ. TO 4 LINES	1 LINE SPILLS
ORIG. FSAR ANALYSIS	1 HHSI	INJ. TO 4 LINES	1 LINE SPILLS
REFERENCE PLANT SI	2 CHARGING	INJ. TO 4 LINES	1 LINE SPILLS
SENSITIVITY ANALYSIS	1 HHSI	INJ. TO 2 LINES	1 LINE SPILLS
D.C. COOK UNIT 2	1 CHARGING	INJ. TO 4 LINES	1 LINE SPILLS
PROPOSED DESIGN	1 HHSI	INJ. TO 2 LINES	1 LINE SPILLS
BASIS MODIFICATION			



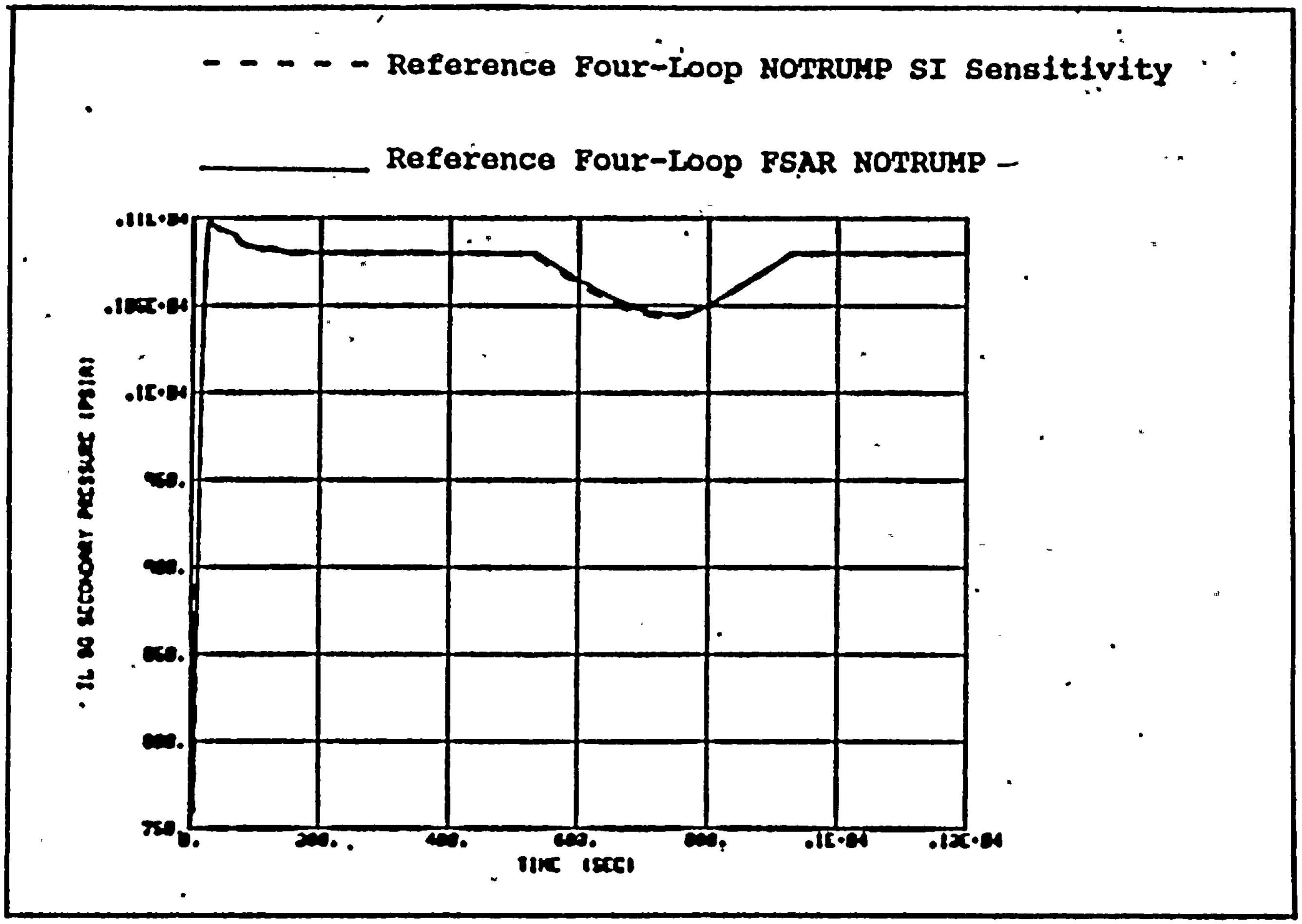
REFERENCE 4-LOOP NOTRUMP FSAR SI FLOWS vs. SI SENSITIVITY FLOWS

Figure 1



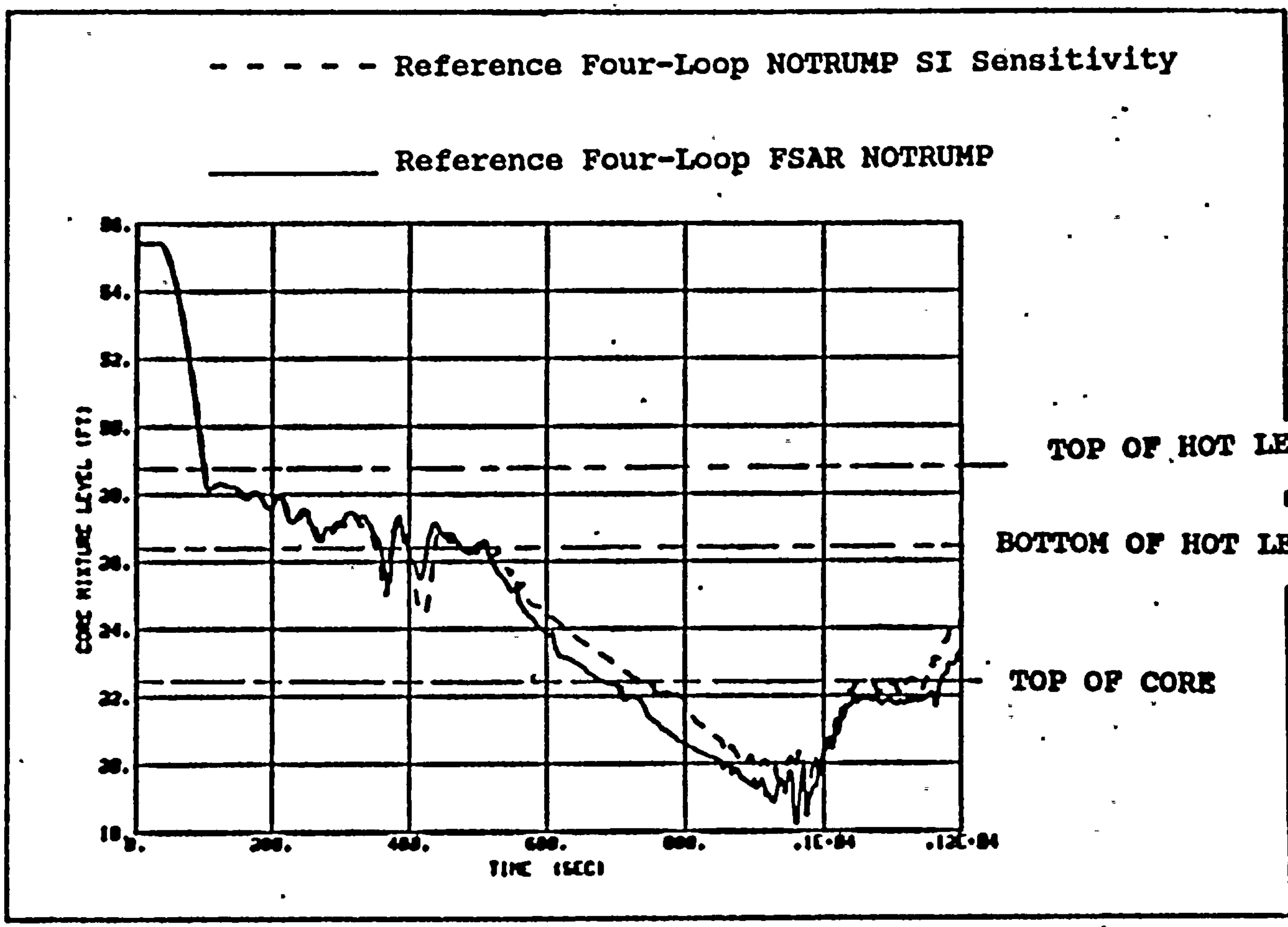
REFERENCE 4-LOOP NOTRUMP FSAR SI FLOWS vs. SI SENSITIVITY FLOWS

Figure 2



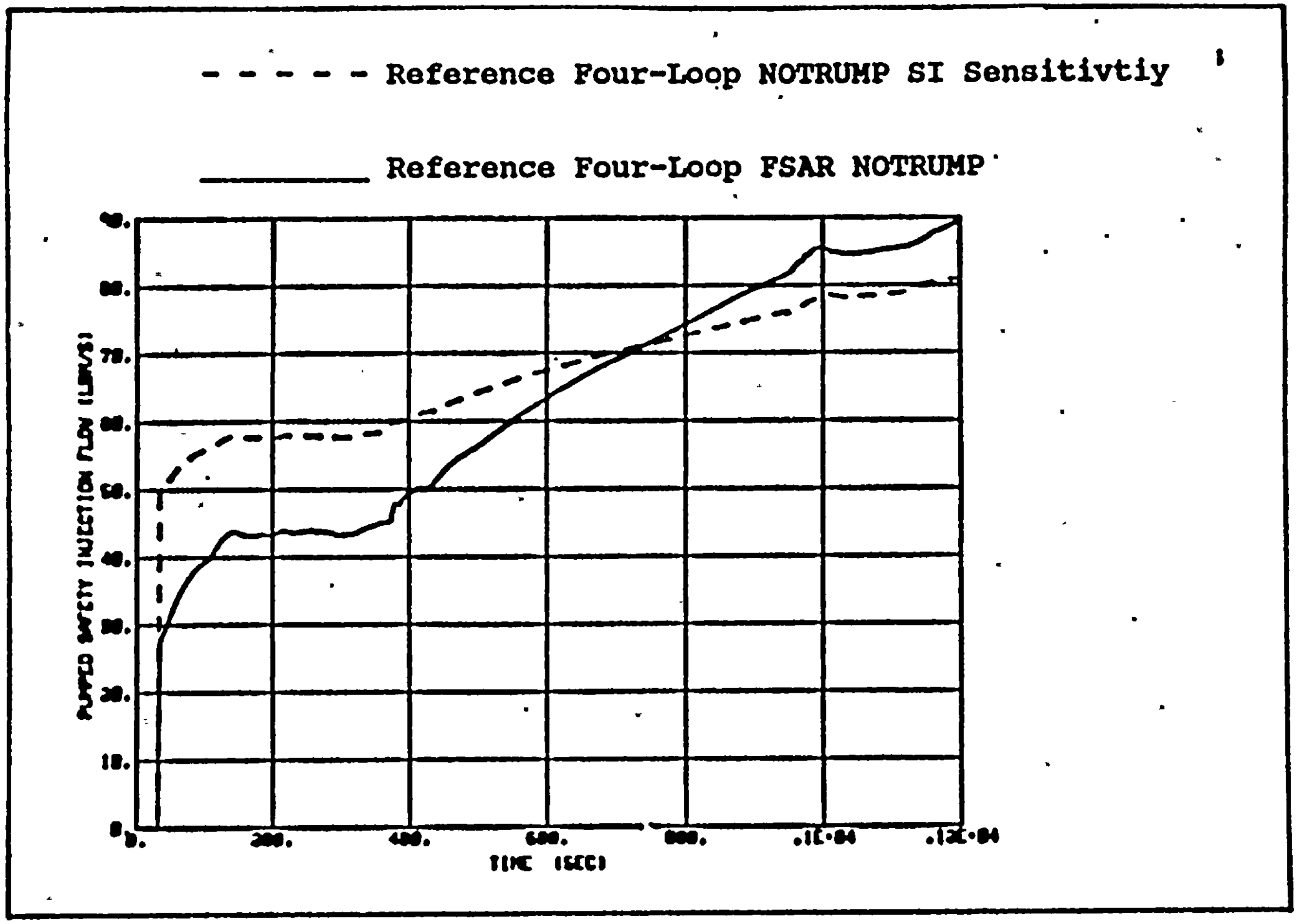
REFERENCE 4-LOOP NOTRUMP FSAR SI FLOWS vs. SI SENSITIVITY FLOWS

Figure 3



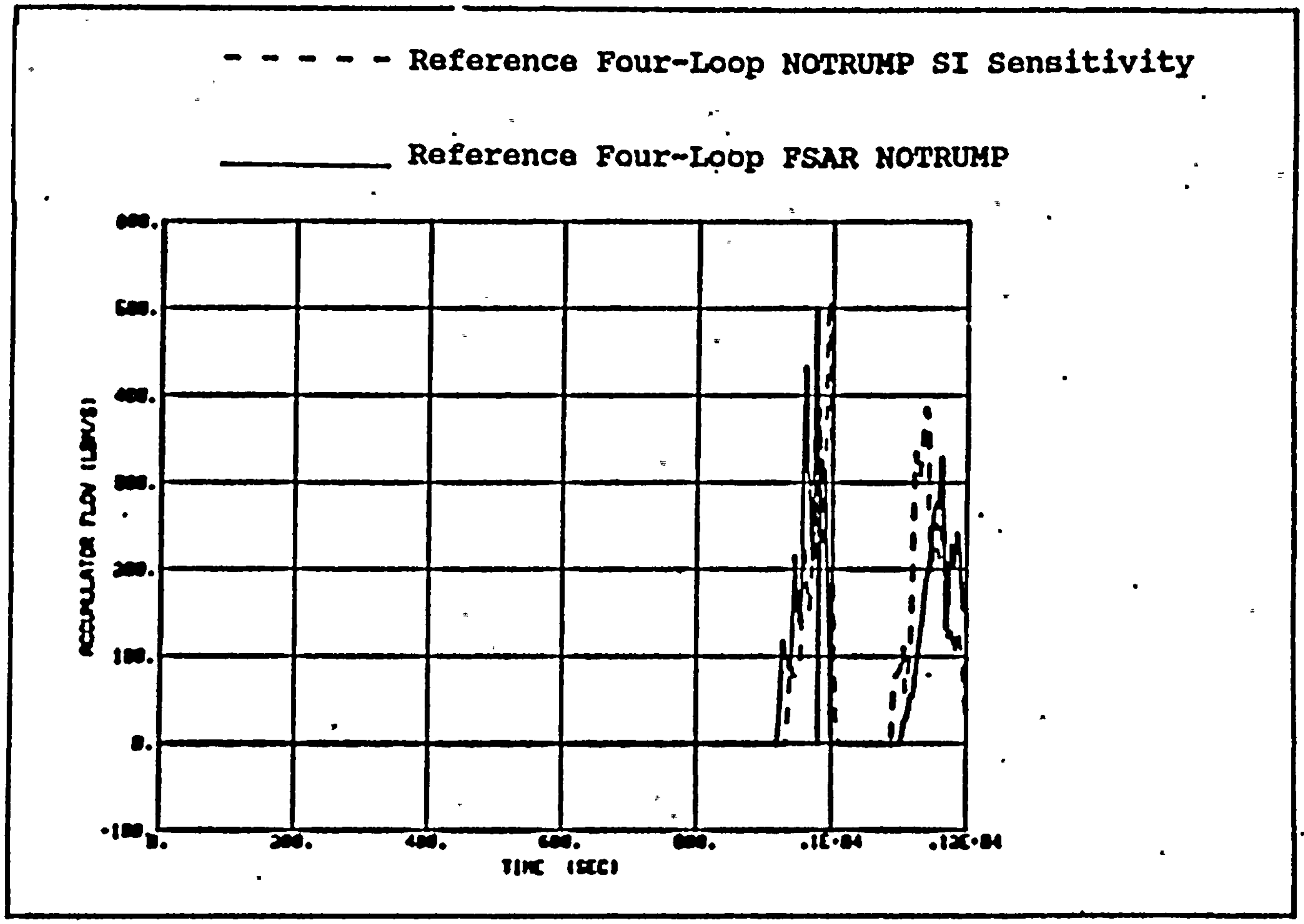
REFERENCE 4-LOOP NOTRUMP FSAR SI FLOWS vs. SI SENSITIVITY FLOWS

Figure 4



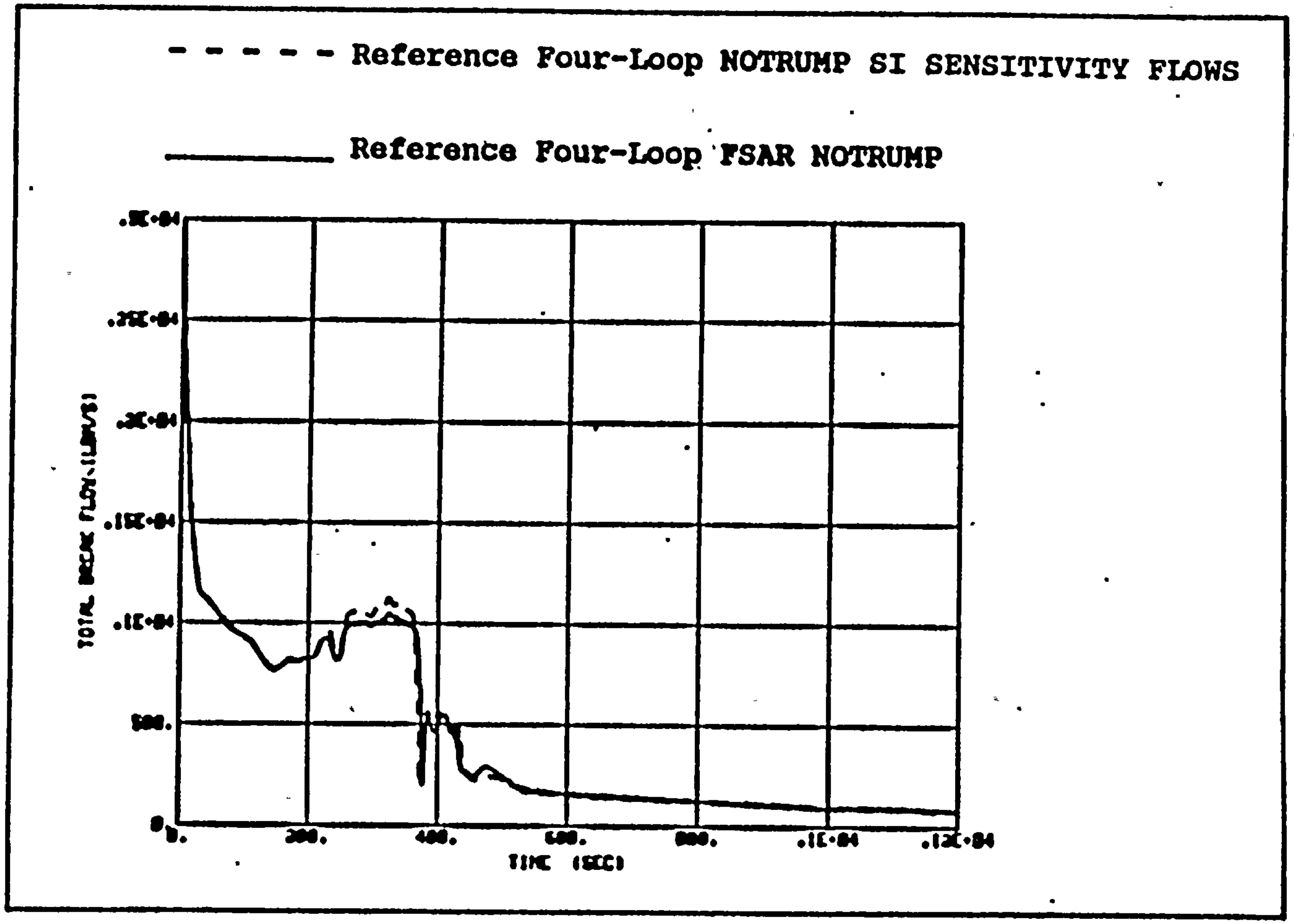
REFERENCE 4-LOOP NOTRUMP FSAR SI FLOWS vs. SI SENSITIVITY FLOWS

Figure 5



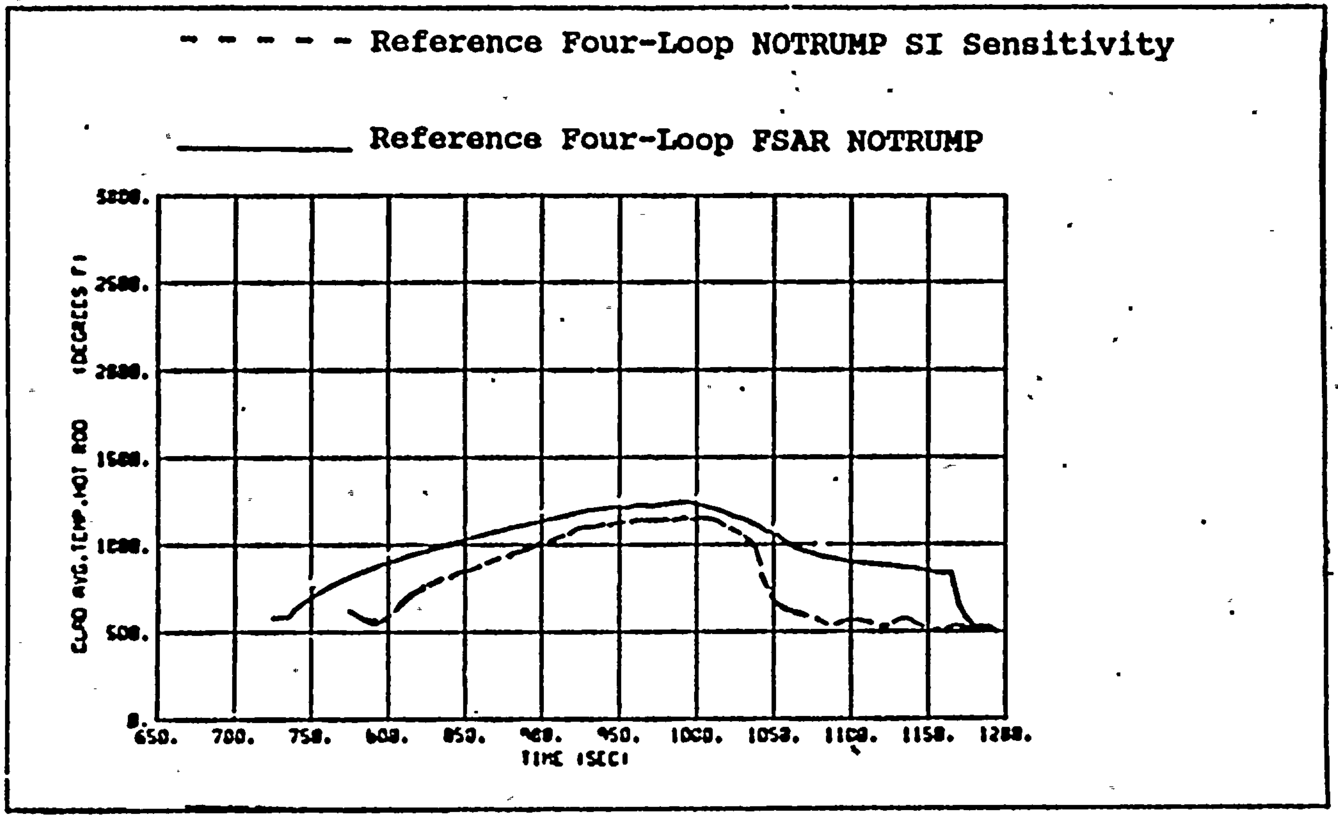
REFERENCE 4-LOOP NOTRUMP FSAR SI FLOWS vs. SI SENSITIVITY FLOWS

Figure 6



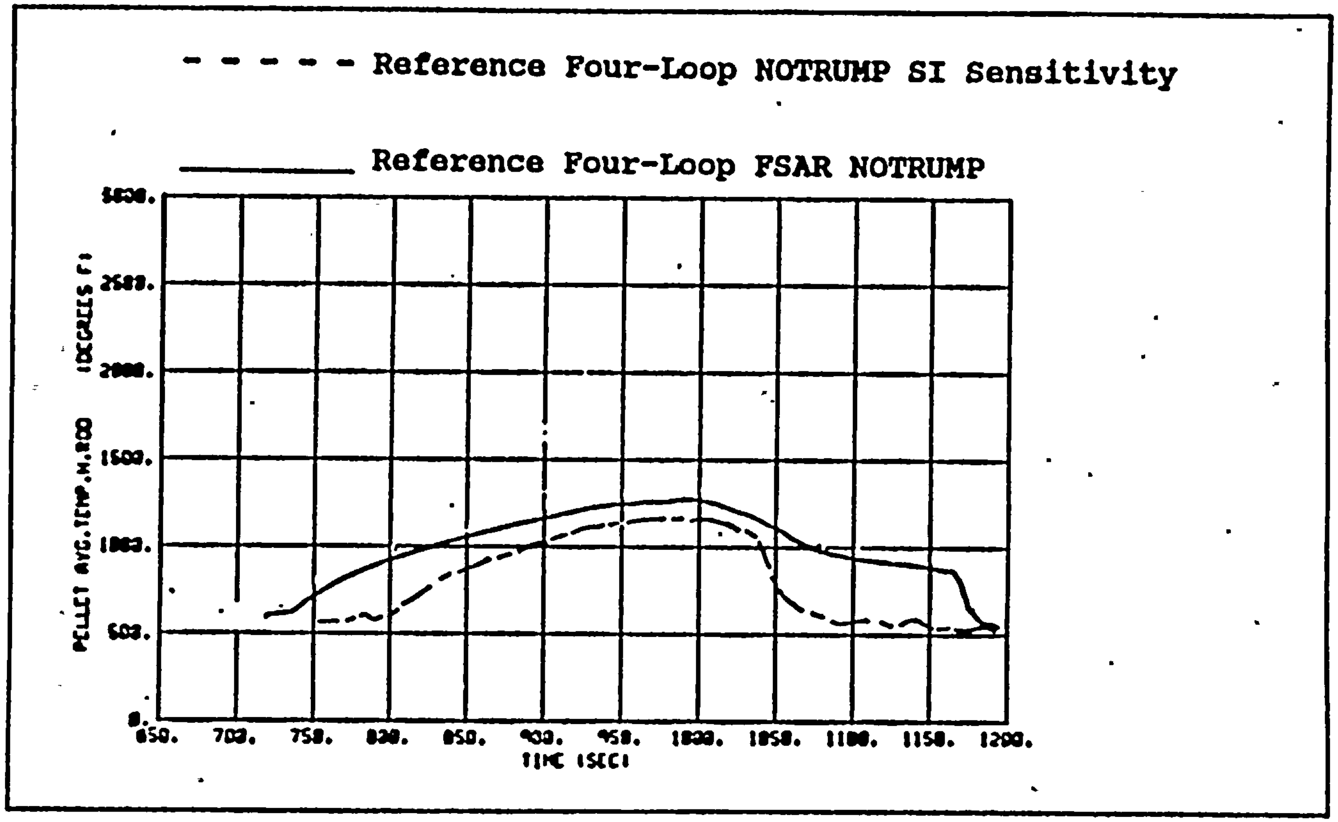
REFERENCE 4-LOOP NOTRUMP FSAR SI FLOWS vs. SI SENSITIVITY FLOWS

Figure 7



REFERENCE 4-LOOP NOTRUMP FSAR SI FLOWS vs. SI SENSITIVITY FLOWS

Figure 8



REFERENCE 4-LOOP NOTRUMP FSAR SI FLOWS vs. SI SENSITIVITY FLOWS

Figure 9

Figure 10
D. C. COOK UNIT 2 SI FLOWS
DESIGN BASIS SI FLOW VS X-TIE CLOSED SI

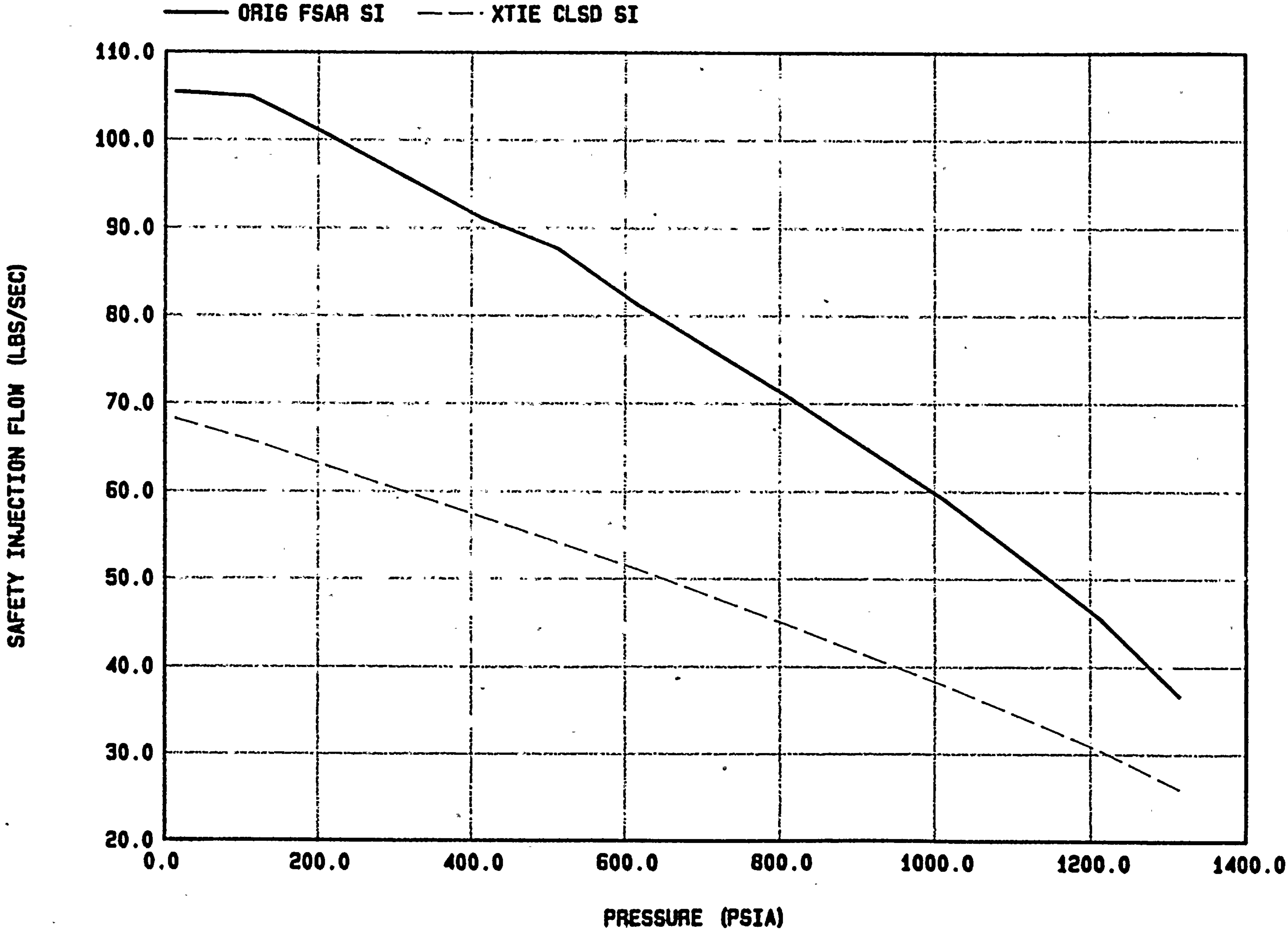
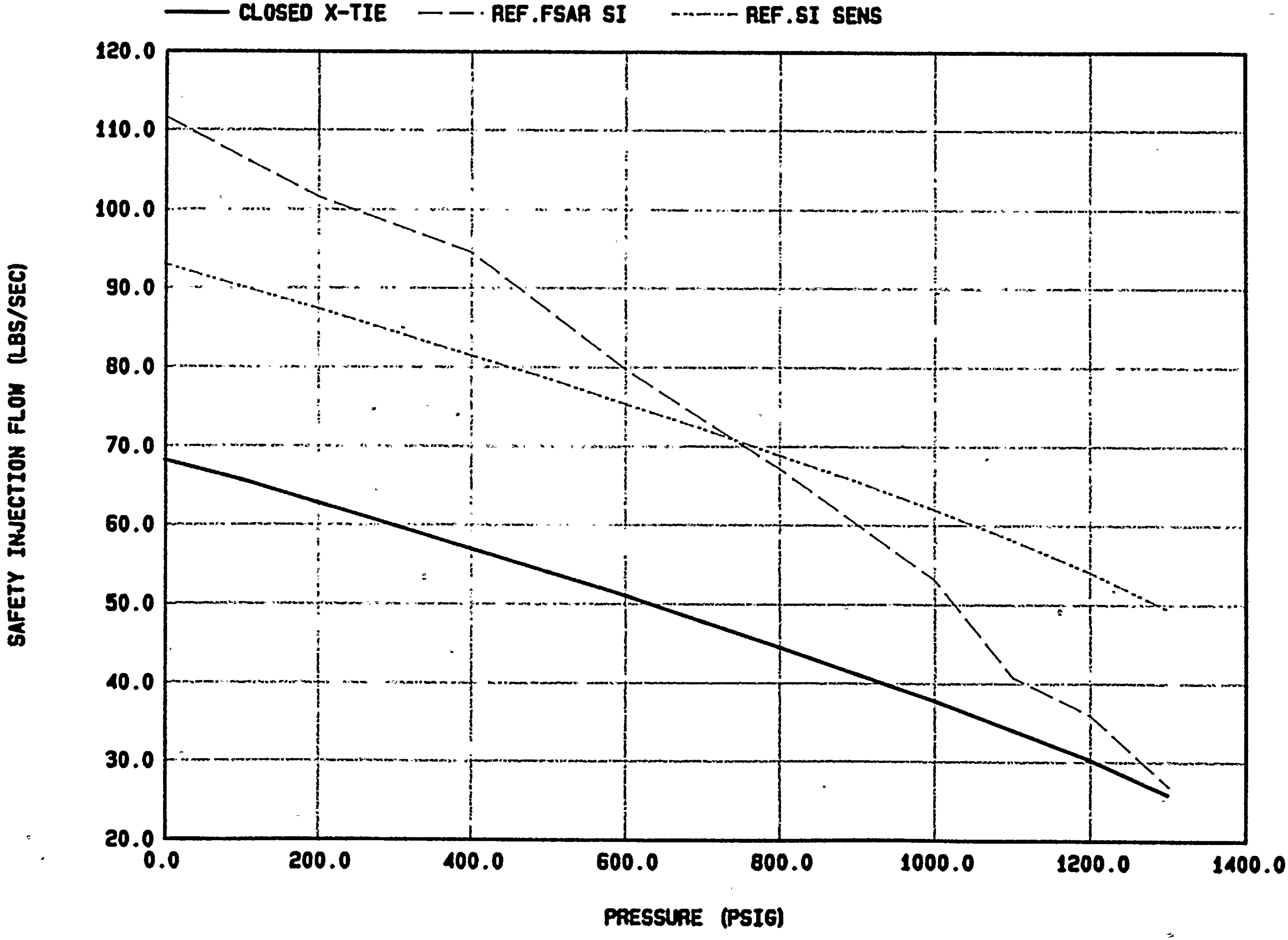


Figure 11
SAFETY INJECTION FLOW COMPARISON
REF. ANALYSIS SI VS. X-TIE CLOSE SI



Attachment 2 to AEP:NRC:1024

Information Notice 87-01

Bill H

JAN 1 4 1987

cc: M. P. Alexich
T. O. Argenta
S. J. Brewer
J. M. Cleveland
C. A. Erikson
J. G. Feinstein
R. F. Kroeger
J. J. Markowsky
J. B. Shinnock
D. H. Williams, Jr.
John E. Dolan

SSINS No.: 6835
IN 87-01

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

January 6, 1987

JAN 15 1987

IE INFORMATION NOTICE NO. 87-01: RHR VALVE MISALIGNMENT CAUSES DEGRADATION OF ECCS IN PWRS

Addressees:

All pressurized-water reactor facilities holding an operating license or a construction permit.

Purpose:

This information notice is provided as notification of a potentially significant problem pertaining to residual heat removal (RHR) valve alignment in the low-pressure emergency core cooling system (ECCS). It is expected that recipients will review the information for applicability to their facilities and consider actions, if appropriate, to preclude a similar problem from occurring at their facilities. However, suggestions contained in this information notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

In late May 1985, plant operators at Callaway isolated the RHR crossover line (by closing the normally open valves X1 or X2 in Figure 1 on the next page) to perform an operability test of train A of the low-pressure ECCS. This action would allow the B train to feed only two reactor coolant system (RCS) loops. When an NRC inspector questioned the advisability of this configuration, the licensee requested technical assistance from Westinghouse. Westinghouse indicated that the licensing bases for the ECCS analysis assume that all four RCS cold legs are being supplied water from at least one RHR pump. Isolation of the crossover line to place the A train in the test condition violated this analysis assumption. At this facility, however, the degraded configuration was never in existence for longer than 1 hour.

Subsequently, Byron Unit 1 identified numerous occasions in 1985 when the RHR system would have been capable of injection to only two RCS loops.

8612300151

No specific action or written response is required by this information notice. If you have any questions about this matter, please contact the Regional Administrator of the appropriate regional office or this office.

for Edward L. Jordan, Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement

Technical Contact: Mary S. Wegner, IE
(301) 492-4511

Attachment:
List of Recently Issued IE Information Notices

Attachment 3 to AEP:NRC:1024

I.E. Inspection Report 86042

Unit 1 and Unit 2 LERs