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DENTON, H. R. Document Control Branch (Document Control Desk)

SUBJECT: Forwards info supporting applicability of WCAP-11145, "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study W/NOTRUMP Code," to facility, per BJ Youngblood 861222 request. Closeout of NUREG-0737 Item II. K. 3. 31 requested.

SEE REPTS.

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INDIANA & MICHIGAN ELECTRIC COMPANY
P.O. BOX 16631
COLUMBUS, OHIO 43216

April 23, 1987
AEP:NRC:0916Z

Donald C. Cook Nuclear Plant Unit No. 2
Docket No. 50-316
License No. DPR-74
APPLICABILITY OF GENERIC NOTRUMP ANALYSIS TO
COOK UNIT 2 WITH FUEL PROVIDED BY ADVANCED NUCLEAR
FUELS CORPORATION, AND CORRECTION TO AEP:NRC:1024

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

Dear Mr. Denton:

The purpose of this letter and its attachments is to present reasons why we believe the generic analysis in WCAP-11145, "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code," is applicable to Unit 2 of the Donald C. Cook Nuclear Plant. The core of the Unit 2 reactor currently contains fuel fabricated by Advanced Nuclear Fuels Corporation (ANFC). This justification was requested in the December 22, 1986 letter from Mr. B. J. Youngblood of your staff to our Mr. John E. Dolan.

Licensing Background

NUREG 0737 (Clarification of TMI Action Plan Requirements) Item II.K.3.30 required the nuclear industry to revise small break loss of coolant accident evaluation models. Item II.K.3.31 of NUREG 0737 required licensees to submit individual analyses, using the revised models dictated by Item II.K.3.30 to demonstrate compliance with the requirements of 10 CFR 50.46. In response to Section II.K.3.30, the Westinghouse Owners Group (WOG) developed the NOTRUMP code as the new licensing small break model. The NOTRUMP code was reviewed by the NRC and found acceptable for meeting the requirements of Item II.K.3.30. Although Item II.K.3.31 of NUREG 0737 required each licensee to submit a plant specific small break analysis, this requirement was later modified by Generic Letter 83-35, which allowed licensees to comply on a generic basis with Item II.K.3.31 by demonstrating that the previous small break evaluation model (which used the WFLASH code) gives conservative results when compared to analyses performed using the NOTRUMP code.

In response to this guidance, the WOG submitted WCAP-11145, which contains generic comparisons to WFLASH analyses for various plant types. To

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close out Item II.K.3.31 with respect to the Donald C. Cook Plant, we transmitted to you our letter AEP:NRC:0678X, dated July 30, 1986, which referenced the WCAP 11145 studies as applicable to the Cook Plant. By letter dated December 22, 1986, we were informed that the reference to WCAP 11145 was sufficient to close out item II.K.3.31 for D. C. Cook Unit 1 (which is fueled primarily with Westinghouse Electric Corp. fuel). For D. C. Cook Unit 2, however, which is fueled primarily with ANFC fuel, we were asked to provide justification as to why we believe the generic analysis presented in WCAP 11145 is applicable.

Summary of Justification

Our reasons for believing the NOTRUMP code is applicable to D. C. Cook Unit 2 is contained in Attachments 1 and 2 to this letter. Attachment 1 reviews previous small break analyses which were performed by Westinghouse to bound both of the D. C. Cook units. The attachment contains detailed comparisons of the fuel designs and heat generation characteristics for the units. These comparisons demonstrate that the differences between the fuel in the two units is small, and would not be expected to have a significant effect on the small break LOCA analysis results.

Attachment 2 to this letter discusses the results of generic plant NOTRUMP analyses Westinghouse Electric Corp. (Westinghouse) performed for 15 x 15 cores typical of D. C. Cook Unit 1, and 17 x 17 cores typical of D. C. Cook Unit 2. The NOTRUMP results for the two core types show very little difference in peak clad temperature following a small break LOCA.

We believe the information provided in the attachments to this letter justifies the use of the NOTRUMP code for D. C. Cook Unit 2. We therefore respectfully request that you close out NUREG 0737 Item II.K.3.31 for D. C. Cook Unit 2.

Other Licensing Issues

In addition to providing information related to the use of the NOTRUMP code for D. C. Cook Unit 2, this letter also corrects an error which was made in our letter AEP:NRC:1024, dated March 23, 1987. That letter transmitted Westinghouse NOTRUMP analyses for D. C. Cook Unit 2 which support operation with cross-tie valves in the high-head safety injection system in the closed position. A minor error related to the core fluid volume listed by Westinghouse was identified after the letter was submitted. Attachment 2 to this letter contains additional information on the error and the revised values for the core fluid volume.

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.

Sincerely,

M. P. Alexich
Vice President

MPA/WLZ/llj

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

Attachments

[illegible]

Attachment 1

to AEP:NRC:0916Z

Examination of Current Licensing Basis

For Unit 2 Small Break LOCA

Attachment 1

Examination of the current Cook Unit 2 licensing basis for the small break LOCA analysis supports the applicability of the NOTRUMP generic analysis to Unit 2 with Advanced Nuclear Fuels Corporation (ANFC) fuel.

The current Unit 2 small break LOCA licensing basis consists of two parts: the initial licensing analyses for a spectrum of small breaks, and an additional analysis for the most limiting break size to support an increase in the safety injection pump miniflow. The miniflow increase analysis for Unit 2 for the most limiting break was submitted to the NRC in our letter AEP:NRC:0860A and was supplemented with our letter AEP:NRC:0860B. These letters are furnished again here as Exhibits A and B to this Attachment. It can be seen in letter AEP:NRC:0860A that the actual analysis was performed for Cook Unit 1, with Westinghouse 15 x 15 "OFA" fuel design. The Unit 1 result that the change in peak clad temperature due to miniflow increase was acceptably small was used as justification for the same change for Unit 2. As part of our submittals, a detailed comparison of the fuel designs between the two units was presented (see Question and Response 6 in AEP:NRC:0860B). The information in Response 6 still applies except for the following modification in number of fuel assemblies:

	AEP:NRC:0860B	Now
Unit 1		
Westinghouse	80	159
Exxon (now ANFC)	113	34
Unit 2		
Westinghouse	29	1
Exxon (now ANFC)	164	192

Use of Unit 1 small break LOCA analyses to support requested Unit 2 licensing actions was found acceptable in the Safety Evaluation Report (SER) prepared by the NRC staff for Amendment 64 to the Unit 2 license. Applicable pages from this SER are included here as Exhibit C. This acceptability for Unit 2 was further cited in the SER for Amendment 84 to the Unit 1 license, included here as Exhibit D. (Unit 1 Amendment 84 granted safety-injection miniflow T/S changes identical to those granted to Unit 2 in Amendment 64).

More recently, ANFC performed a complete review of accidents cited in the NUREG-0800 Standard Review Plan Chapter 15. This appeared in report 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." This report was transmitted to you directly by ANFC in their letter RAC:069:85, dated October 15, 1985, and was referenced in our letter AEP:NRC:0916G, dated October 18, 1985. Pertinent pages from this report are pages 137 through 143.

In this report, the applicability of the above cited Unit 1 analysis to Unit 2 was proposed once again. In response, the SER for Amendment 82 to the Unit 2 license made no adverse findings in regard to this proposal. That same SER reviewed the ANFC large break analysis in detail, and concluded that the 10 CFR 50.46 criteria are satisfied with no qualification. Applicable pages from that SER appear as Exhibit E to this attachment.

Our review of the Unit 2 small break LOCA licensing basis has determined that the limiting small break LOCA analysis for D. C. Cook Unit 2 fueled with Exxon 17 x 17 fuel is a small break LOCA analysis for Unit 1 fueled with Westinghouse OFA 15 x 15 fuel.

For plants with Westinghouse fuel, the generic NOTRUMP analysis is accepted on the basis that the previous small break analysis method for the same conditions has been shown to be conservative by comparison. The current analysis for D. C. Cook Unit 1 fueled with Westinghouse 15 x 15 OFA fuel was approved on this basis in a letter from Mr. B. J. Youngblood of the NRC staff to Mr. John Dolan, Vice Chairman of AEP, dated December 22, 1986 (as discussed in the cover letter for this submittal).

It follows that the generic NOTRUMP analysis has demonstrated that the existing analysis for Cook Unit 2 is conservative because the approved D. C. Cook Unit 1 analysis has been found to be applicable by the NRC staff to D.C. Cook Unit 2 fueled with Exxon 17 x 17 fuel. Therefore, we conclude that the NOTRUMP generic analysis should be applicable to Cook Unit 2.

Exhibit A of

Attachment 1

to AEP:NRC:0916Z

March 15, 1984 Letter from M. P. Alexich (I&MECo)

to H. R. Denton (NRC) Regarding

Unit 2 Cycle 5

Reload Technical Specifications

B. Safety Injection Miniflow Line Modification

Background

The Indiana & Michigan Electric Company (IMECO) submitted a request to modify the piping geometry of the miniflow line for the D.C. Cook Unit No. 2 safety injection pumps by letters dated March 1 and 15, 1984. This modification has been previously performed for D.C. Cook Unit No. 1. As presently configured, the miniflow line for Unit 2 is comprised of both 1.5 inch and 0.75 inch diameter piping. The licensee has requested that it be allowed to replace the 0.75 inch diameter piping with 1.5 inch piping, thereby making the entire piping in system of one diameter. The purpose of this modification is based on economic and maintenance considerations. By maintaining both Units 1 and 2 as similar as possible, the licensee is able, in many cases, to apply one analysis to both units.

Increasing the miniflow line piping diameter doubles its flow rate from 30 gpm to 60 gpm. The increased flow is beneficial to the SI pump when operating in the shut-off configuration in that it reduces the temperature rise through the pump. This provides an added benefit of increased pump reliability by allowing smoother operation at reduced temperatures.

Increasing the miniflow coolant rate has a negative influence on ECCS performance in that it reduces the injected flow to the reactor coolant system. At runout conditions, the ECCS injection rate is decreased from 63.0 lbm/sec to 61.6 lbm/sec. At the other extreme, the ECC injection rate at 1314.7 psia is reduced from 19.0 lbm/sec to 16.1 lbm/sec. Since only the SI is influenced by the proposed hardware modification, the impact on large break LOCAs is

insignificant (total ECCS flow, not including accumulator injection, is reduced from 463.0 lbm/sec to 461.6 lbm/sec). This would have negligible impact on the calculated peak clad temperature for the large breaks.

For the limiting small break LOCA, however, IMECO has determined that the peak clad temperature would increase by about 87°F. This analysis was conservatively calculated for Unit 1, and was submitted by IMECO as applicable to Unit 2.

To demonstrate that the temperature increase for Unit 1 was applicable to Unit 2, IMECO had the reactor vendor (Westinghouse) confirm that the ECCS pump characteristics for both Unit 1 and Unit 2 are identical. Having anticipated the desirability to modify the geometry of the miniflow line for Unit 2 as well, the limiting small break LOCA for Unit 1 was analyzed at the Unit 2 power rating (3411 MWt versus 3250 MWt). In addition, the linear peak heat generation rate was analyzed at 16.67 kw/ft for Unit 1 (Unit 2 is rated at 12.88 kw/ft). Since the linear heat generation rate for Unit 1 is significantly greater than that for Unit 2, the calculated heat up rate would be conservative when applied to Unit 2.

The applicability of the Unit 1 calculation to Unit 2 was also based on comparison of the volumetric fuel heat generation rate for the total core. The volumetric heat generation rate for Unit 1 was

calculated at 9887 kw/ft^3 of fuel and for Unit 2 at 9835 kw/ft^3 of fuel. The total volume of coolant in the core was also calculated to be nearly identical (614.8 ft^3 and 613.0 ft^3 for Units 1 and 2, respectively). With respect to the remaining primary system coolant volume, both plants are identical.

The reduction of ECC injection by the SI pump resulted in an additional 6 inches of calculated core uncovery (5.5 ft versus 5.0 ft). This corresponded to a 10 second delay in coolant recovery of the core (838 versus 848 seconds). The consequential increase in peak clad temperature was 87°F. With the present calculated small break peak clad temperature of 1668°F, the Unit 2 core response for the limiting small break LOCA is expected to be less than 1750°F. This is well below the 2200°F licensing limit.

CONCLUSION OF THE MINIFLOW LINE REVIEW

We have reviewed the submittal by the Indiana & Michigan Electric Company to increase the pipe diameter of the miniflow line to the injection pumps. The acceptability of the miniflow line modification is based on the Unit 1 LOCA analysis and it's applicability to Unit 2. We find the analysis and applicability acceptable, and therefore find acceptable, the requested modification of increasing the cross sectional diameter of the miniflow line from 0.75 inch to 1.5 inch. We requested, however, that the SI pump flow characteristic be confirmed to be consistent with the analysis assumptions prior to full power operation. The licensee has agreed to perform this test prior to startup.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 82 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA AND MICHIGAN ELECTRIC COMPANY

DONALD C. COOK NUCLEAR PLANT UNIT NO. 2

DOCKET NO. 50-316

Introduction

By letters dated March 14 and March 27, 1986, the Indiana and Michigan Electric Company (the licensee), proposed changes to the Donald C. Cook Nuclear Plant, Unit No. 2. These changes are grouped and evaluated below by those changes related to the Unit 2 cycle 6 reload and Technical Specification changes.

Cycle 6 Reload

The reactor core for D. C. Cook Cycle 6 will contain 191 Exxon fuel assemblies and one Westinghouse fuel assembly each having a 17 x 17 fuel rod array. Eighty eight of the Exxon fuel assemblies are new. Cycle 6 burnup has been projected to be 17,790 Mwd/MTU at a core power of 3411 MWT.

The design characteristics of the Exxon fuel assemblies were reviewed and approved by the staff for the cycle 5 core. The Exxon fuel in the cycle 6 core will be of the same design as that of cycle 5. As additional confirmation of the integrity of Exxon fuel remote visual examinations of irradiated fuel were performed following cycle 4. No evidence of wear, fretting or other physical damage was noted for this fuel. The exposure for the examined fuel ranged from 16,440 MWD/MTU to 17,630 MWD/MTU.

In anticipation of steam generator tube plugging the cycle 6 safety analyses were performed with an average steam generator tube plugging of 10%. The effect of asymmetric tube plugging was considered in the analyses.

Fuel Thermal-Mechanical Design

Cycle 5 contained a mixture of Exxon and Westinghouse fuel elements. Staff conclusions regarding the thermal-mechanical design of the cycle 5 core remain applicable to cycle 6. In particular cladding strain, external corrosion (oxidation), fuel rod internal pressure, and fuel rod pellet temperature were analyzed using the RODEX 2 code which has been approved by the NRC staff. The analytical results satisfied the acceptance criteria. Collapse of fuel cladding into a pre-existing axial gap produced by the differential pressure between the reactor coolant pressure and the internal fuel rod pressure was investigated. It was determined that the cladding would not collapse.

cycle result was determined to be bounding. The pressurizer pressure at beginning of cycle was adjusted to the maximum reactor system pressure location at the pump discharge and determined to be 2747.6 psia which is less than 110% of design. Regulatory Guide 1.77 recommends that the maximum pressure be less than that which would cause stresses to exceed "Service Limit C" of the ASME Code. "Service Limit C" corresponds to approximately 120% of design pressure. The licensee's pressure calculation is therefore acceptable.

The number of fuel rods experiencing DNBR less than the 1.17 design limit for EXXON fuel was calculated to determine the offsite dose consequences from the event. This was accomplished using the XCOBRA-IIIC code to determine the local power level at which the minimum DNBR would be 1.17. A flow penalty was included to account for non-symmetric effects on core flow from rod ejection. Using the calculated radial power distribution for the XTRAN calculations the fraction of the core with local power levels greater than that which would produce a DNBR of less 1.17 was determined. From this calculation 10.7% of the fuel was predicted to penetrate the DNBR limit and to fail. The offsite dose consequences from the event were calculated to be well within the exposure guidelines of 10CFR100 and are therefore acceptable.

Loss of Coolant Accidents

Large break LOCA/ECCS analysis were performed in 1982 (1, 2) to support operation of the D.C. Cook Unit 2 reactor at 3425 MWT with ENC fuel. The limiting break was identified as the 1.0 double ended cold leg guillotine (DECLG) break as developed in reference 1. The results of calculations with one and two LPS pumps operating were presented in reference 2 which indicated that a higher PCT occurred with two LPS pumps operation.

Reference 3 documents the results of LOCA/ECCS analysis performed in support of cycle 6 and future cycles with all ENC fuel at a thermal power rating of 3425 MWt, with up to 10% of the steam generator tubes plugged. Calculations were performed for the previously identified 1.0 DECLG break, with full ECCS flow. Three exposures using a center peaked axial power shape were studied to determine exposure dependence. The exposures range from 2 MWD/kg to 47 MWD/kg peak rod average burnup.

The axial dependence of the peaking factor limit is denoted K(Z) and is defined as (K(Z) = FQ(Z)/MAX FQ(z) where FQ(Z) is the maximum peaking factor allowed at any elevation Z. The topmost segment of the K(Z) curve is limited by the small break LOCA linear heat generation rate (LHGR) limits presented in the technical specifications.

Confirmation of the axial dependence is based on three power distributions: a center peak chopped cosine power distribution and two conservative top skewed power shapes as presented in Figure 3.2*. The power distributions are analyzed at the limiting exposure, 2 MWD/kg, where the peak stored energy occurs.

A summary of these results and the exposure study is presented in Table 3.5*. The normalized K(Z) curve verses core axial height is presented in Figure 2.1*. The calculations were performed using the EXEM/PWR LOCA/ECCS models, including

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* Reference 3, Exxon Report XN-NF-85-68(P) Revision 1.

fuel properties calculated at the start of the LOCA transient with the NRC generically approved RODEX2 code.(4). The quench time, quench velocity and CRF correlations in REFLEX and the heat transfer correlation in TOODEE2 are based on ENC's 17x17 Fuel Cooling Test Facility (FCTF) data (5, 6, 7). Reference 3 reflects the revisions in the correlations based on the FCTF data which are documented in References 6 and 7 and to reflect a 1.1 multiplier on the peak power parameter used in the FCTF correlations in the REFLEX code. This documents the NRC acceptance of the 1.1 multiplier as applied and agreed to with EXXON Nuclear Company, Inc.

The NRC finds that the analysis results and methods summarized above and as presented in reference 3 supports operation of the D.C. Cook Unit 2 reactor for cycle 6, and future cycles with ENC fuel, at a total power peaking factor limit (FQ) of 2.10.

Considering the results of the analysis as summarized in Table 3.5*, peak clad temperature is less than 2200°F, local oxidation is less than 17% and core wide metal-water reaction is less than 1.0%. Therefore we find that the criteria of 10CFR50.46 have been satisfied.

REFERENCES
Loss of Coolant Accidents

1. XN-XF-82-35, "Donald C. Cook Unit 2 LOCA/ECCS Analysis Using EXEM/PWR Large Break Results," Exxon Nuclear Company, Inc., Richland, WA 99352, April 1982.

2. XN-NF-82-35, Supplement 1, "Donald C. Cook Unit 2 Cycle 4 Limiting Break LOCA/ECCS Analysis Using EXEM/PWR," Exxon Nuclear Company, Inc., Richland, WA 99352, November 1982.

3. "Donald C. Cook Unit 2 Limiting Break LOCA/ECCS Analysis, Revision 1, 10% Steam Generator Tube Plugging, and K(z) Curve," XN-NF-85-68(P) Revision 1, Exxon Nuclear Company, Inc., Richland, WA 99352, April, 1986.

4. XN-NF-81-58(P)(A), Rev. 2, and Rev. 2 Supplements 1 and 2, "RODEX2: Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, Inc., Richaldn, WA 99352, February 1983.

5. XN-NF-85-16(P), Volume 2, PWR 17x17 Fuel Cooling Test Program Reflood Quench Carryover, and Heat Transfer Correlations," Exxon Nuclear Company, Inc., Richland, WA 99352, May 1985.

6. "PWR 17x17 Fuel Cooling Test Program Reflood Quench, Carryover, and Heat Transfer Correlations," XN-NF-86-16(P), Revision 1, and all supplements, Exxon Nuclear Company, Inc., Richland, WA 99352, January 1986.

7. "PWR 17x17 Fuel Cooling Test Program Sensitivity Studies," XN-NF-85-16(P), Volume 1, and all supplements, Exxon Nuclear Company, Inc., Richland, WA 93352, January 1986.

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* Reference 3, Exxon Report XN-NF-85-68(P) Revision 1.

ATTACHMENT 2

to AEP:NRC:0916Z

Detailed Evaluation of the Impact of Nuclear Fuel

Assembly Design on NOTRUMP Small Break

LOCA Analysis Results

Attachment 2

An evaluation of the impact of nuclear fuel assembly design on NOTRUMP small break LOCA analysis results can be established from work recently performed by Westinghouse at the request of American Electric Power. This work was done in support of an effort to change our licensing basis for operation of the safety injection system. Specifically, the work evaluated the acceptability of operating with the safety injection system cross-tie valves closed, such that each of the two safety injection pumps would only be capable of delivering flow to two Reactor Coolant System (RCS) cold legs. (The previous small break analyses required the cross-tie valves to be open, such that each pump would be capable of providing flow to all four RCS cold legs.) An evaluation to support operation of D.C. Cook Unit 2 with the safety injection cross-tie valves closed was submitted in our letter AEP:NRC:1024, dated March 23, 1987, which we have included as Exhibit B to this attachment. Included as Exhibit C to this attachment are excerpts from a letter from H. C. Walls of Westinghouse to our Mr. J. G. Feinstein, dated March 24, 1987 (Identifier AEP-87-205). This letter makes a correction to Table 2 of the Westinghouse evaluation submitted in AEP:NRC:1024. Specifically, core fluid volumes for the reference plant and D. C. Cook Unit 2 are modified. The revised values of core fluid volume have been cited in Table II to this attachment, which is presented below. This letter transmits an analogous evaluation for D.C. Cook Unit 1 as Exhibit A. We have transmitted the Unit 1 evaluation in draft form because it is currently undergoing corporate review. This Unit 1 evaluation will be the subject of a future letter which we anticipate will be transmitted to you in the near future.

Rather than perform D. C. Cook plant-specific analyses to justify operation with the cross-tie valves closed, Westinghouse took the approach of using evaluations based on NOTRUMP analysis results for a reference plant similar in design to the D. C. Cook Units. Additional information related to the decision to use the reference plant approach is provided in Exhibit B to this attachment.

Although the Westinghouse NOTRUMP analyses are not D. C. Cook Plant specific, use of Westinghouse's NOTRUMP model for the reference plant should be sufficient for examining the impact of different fuel designs on small break LOCA results. Westinghouse furnished reference plant analyses for both their 17 x 17 standard fuel design and their 15 x 15 "OFA" fuel design. Table 1 of this Attachment shows that the same component configurations were used in both analyses. Table 2 presents the plant conditions for both these analyses. For comparison, Table 2 also give some of those conditions that would represent a core of ANFC 17 x 17 fuel if it were to appear in the reference plant. For the case where one charging pump injects to four loops (with spilling to containment in one of those loops) and where one high head safety injection pump injects to two loops (with spilling in one of those loops), the 15 x 15 OFA results in a peak clad temperature of 1427°F. For the same case for 17 x 17 standard fuel*, the peak clad temperature is estimated at 1482°F**. Exhibit A of this attachment is the draft description of the Unit 1

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*No direct analysis was performed for the above cited flow injection configuration for 17 x 17 standard fuel, therefore this estimated result is based on extrapolation from analyses with other flow injections.

**The evaluations considered reference plants of identical power ratings. (Footnote Continued)

15 x 15 OFA work. The Unit 1 result quoted above appears in Table 4 of Exhibit A. Exhibit B of this attachment is the actual description of the Unit 2 17 x 17 standard work. The Unit 2 result quoted above appears toward the top of page 10 in the Westinghouse attachment contained in Exhibit B.

It can be seen that for the difference between the Westinghouse 15 x 15 OFA and 17 x 17 standard design, the peak clad temperature difference of 55°F is very small compared to the margin to the acceptance criterion of 2200°F. The difference between the ANFC 17 x 17 fuel design and the Westinghouse 17 x 17 standard fuel design is not greater than the difference between the two Westinghouse designs presented here. For illustration, certain parameters for the ANFC design are compared with those for the two Westinghouse designs in Table 2. Further comparisons among these designs can be seen in Exhibit B of Attachment 1 to this submittal. Therefore, any peak clad temperature difference between ANFC and Westinghouse fuel should be no more than on the order of the difference shown for the two Westinghouse designs, and the peak clad temperature for Exxon fuel will be similarly much less than the acceptance criterion. This conclusion is concurred with in the Westinghouse work for Unit 2 (Exhibit B) which states on page 2, "The difference in fuel pellet outer diameter, fuel rod outer diameter, and fuel rod pitch are small between the Exxon (ANFC) 17 x 17 fuel and the Westinghouse 17 x 17 standard

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(Footnote Continued)
They differed primarily in the core configuration (15 x 15, versus 17 x 17), which impacts parameters such as linear hear generation rate. They also differed slightly in assumed peaking factors: Unit 2 was analyzed at a peaking factor of 2.40 versus 2.32 for Unit 1. Westinghouse fuel in both Units is currently licensed for core peaking factors of 2.10. If identical values had been assumed for the peaking factor, the difference between the peak clad temperatures would most likely have been even less than the 55°F cited above.

fuel. Consequently, the effects of fuel parameter differences are expected to have only a small effect on the transient response."

Attachment 2

Table 1

Component Configuration

	Reference 412 Plant with 15 x 15 OFA	Reference 412 Plant with 17 x 17 STD
VESSEL:		
Upper Support Plate	Top Hat	Top Hat
Barrel-Baffle Configuration	Downflow	Downflow
Downcomer Shielding	Thermal Shield	Thermal Shield
Lower Support Plate	Curved	Curved
Fuel Array	15 x 15 OFA	17 x 17 STD
Upper Head Spray Flow Percent	0.21%	0.21%
Upper Head Temperature	T_{hot}	T_{hot}
LOOP COMPONENTS:		
Pressurizer	1800 ft^3	1800 ft^3
Steam Generator	Model 51	Model 51
Pump Type	93A; 6000 HP	93A; 6000 HP

Attachment 2

Table 2

Plant Conditions

	Reference 412 Plant 15 x 15 OFA	Reference 412 Plant 17 x 17 STD	Reference 412 Plant ANFC 17 x 17
Licensed Power (MWt)	3338	3338	3338
Licensed Peaking Factor	2.32	2.40	*
Fuel Volumetric Heat Generation for Total Core (Total kw/Total ft^3 fuel)	9670.04	9624.0	10902.6
Average Linear Heat Generation (kw/ft)	7.033	5.435	5.435
Total Fluid Volume in Core (ft^3) (excluding that in fuel assembly guide tubes)	646.6	642.9	679.6
Thermal Design Flow (lbs/sec)	36950.2	36916.7	*
Vessel Exit Temperature (^oF)	608.8	608.0	*
Vessel Inlet Temperature (^oF)	544.4	542.2	*

* No analysis performed, therefore no value generated for comparison.

Exhibit A of

Attachment 2

to AEP:NRC:916Z

Draft Version of Westinghouse

D. C. Cook Unit 1

Analysis Regarding Operation with

Safety Injection Cross-Tie Valves Closed

[illegible]

DRAFT

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 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 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

[illegible]

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 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 2.7%. 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 pump suction 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 initial

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 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

[illegible]

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. 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 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 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.

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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

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(Includes corrections
from Exhibit C)

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	608.8
VESSEL INLET TEMPERATURE (^{o}F)	536.3	544.4

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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

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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	$1716^{o}F$	$1427^{o}F$
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

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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

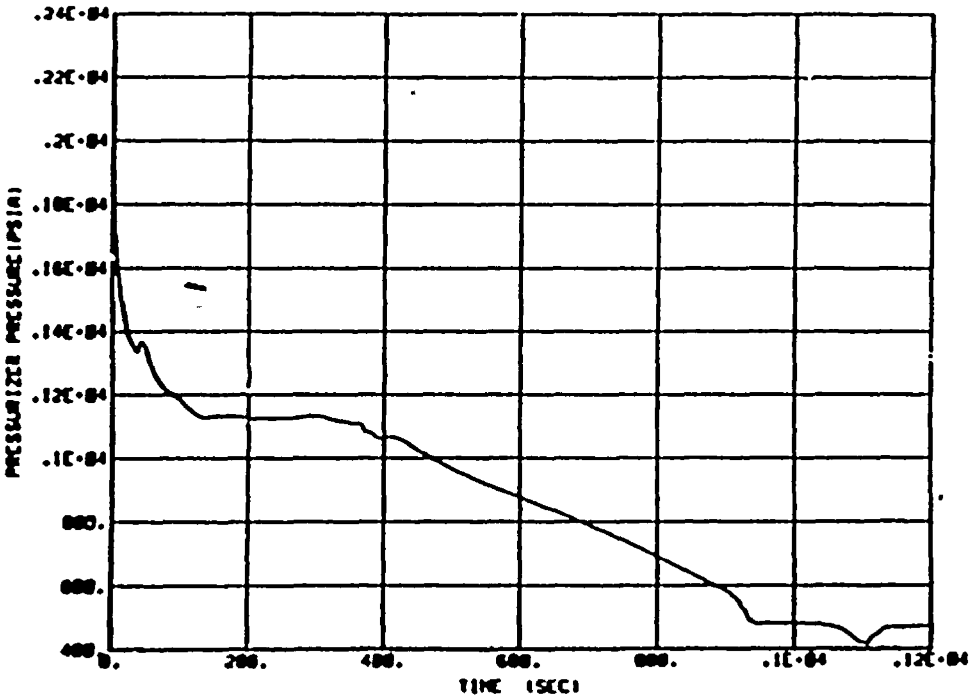

Figure 1

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

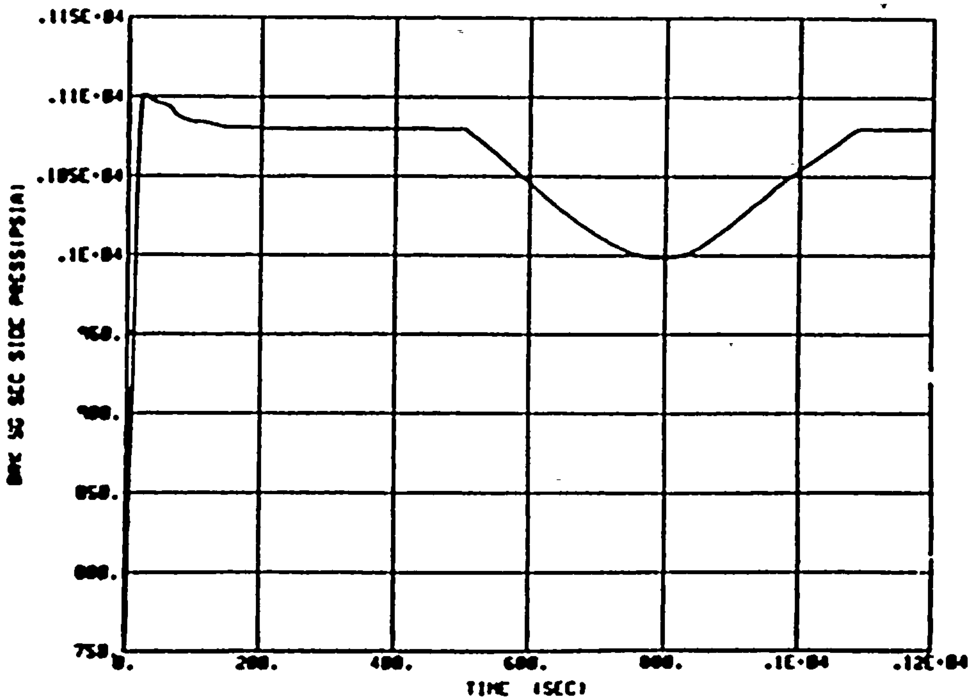

Figure 2

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

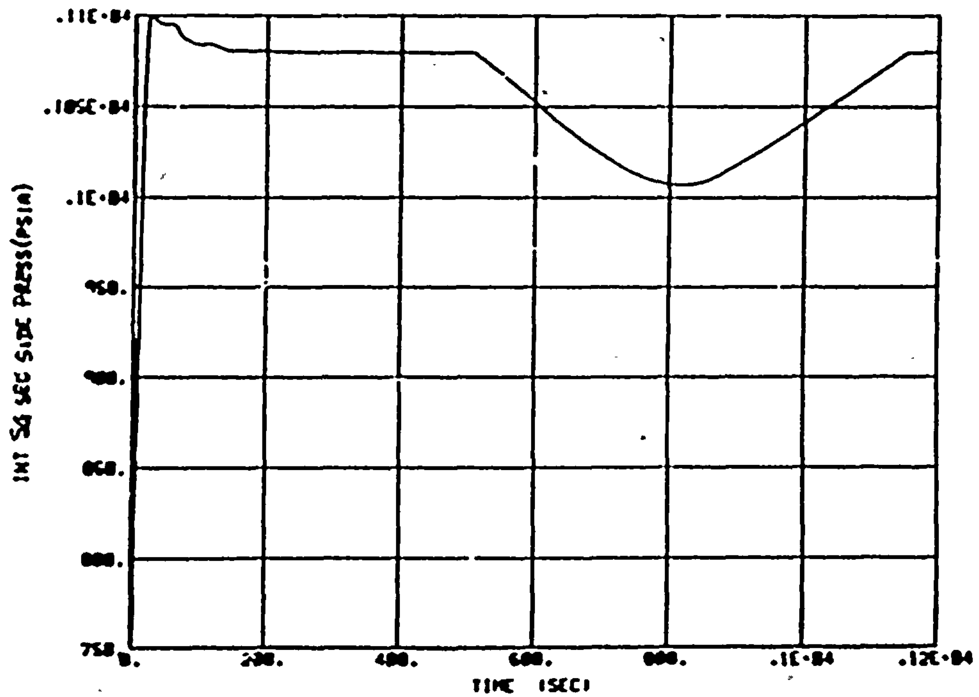

Figure 3

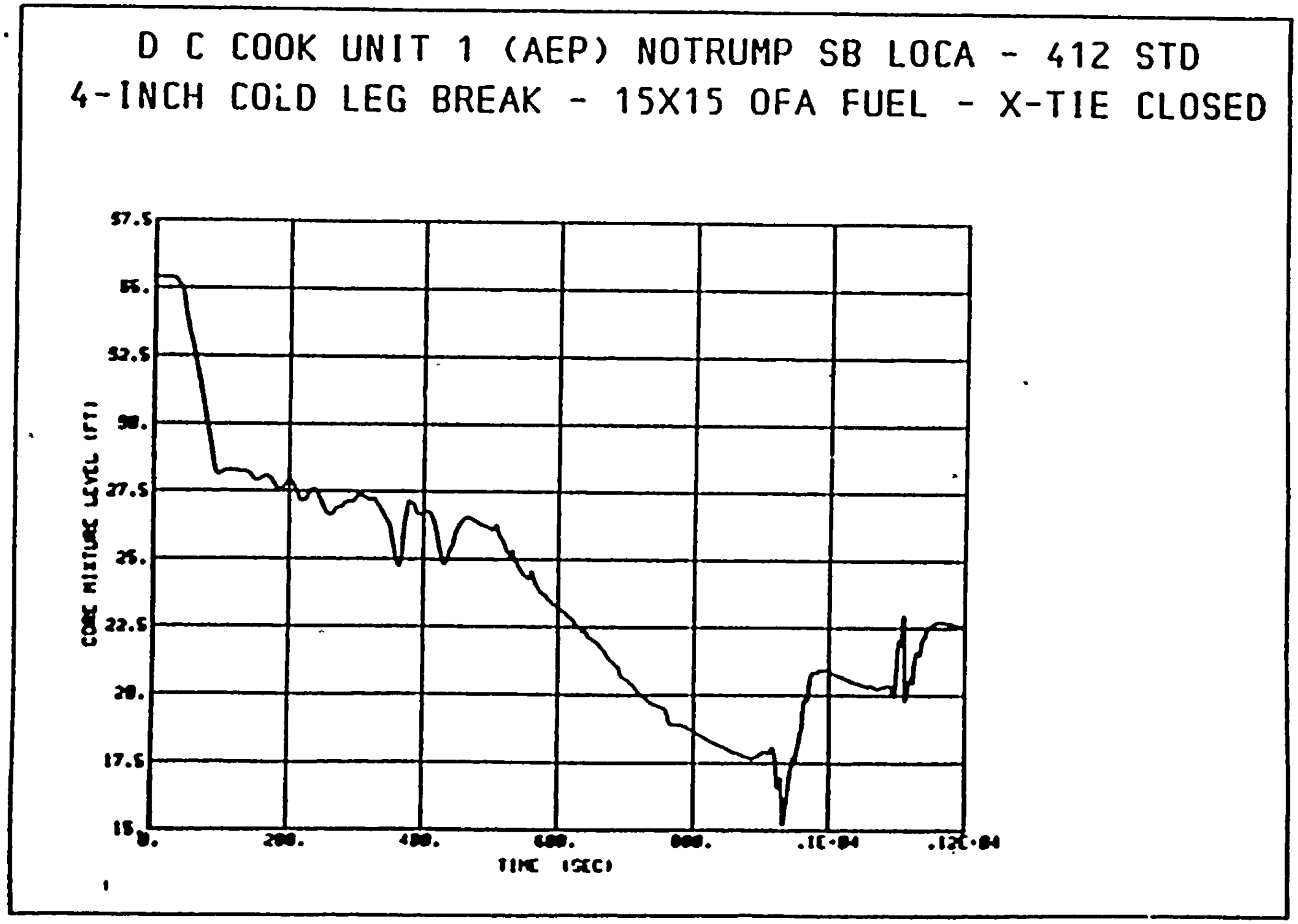

Figure 4

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

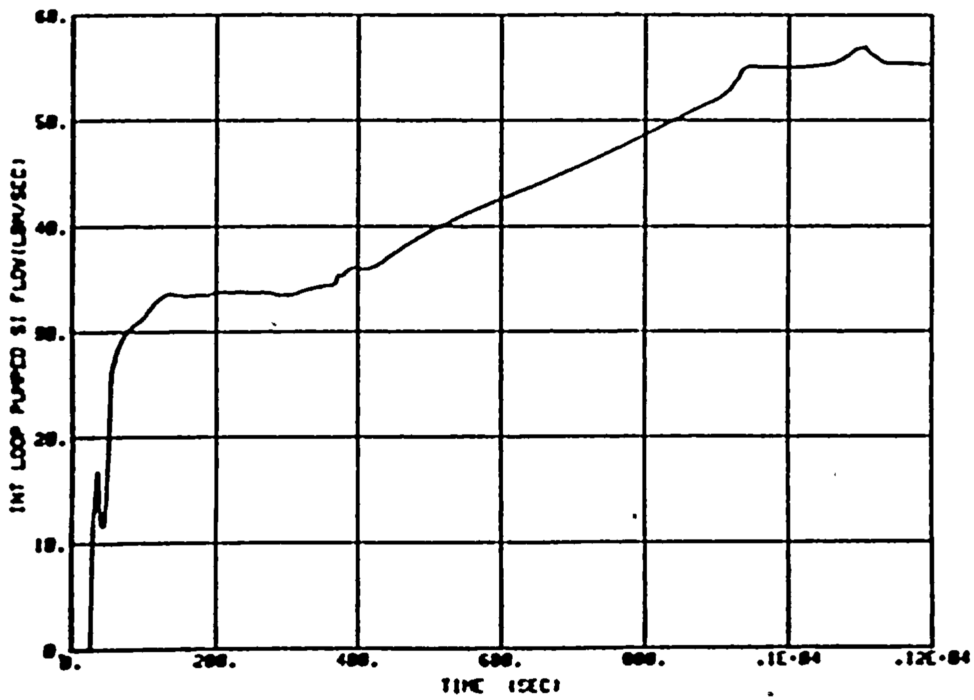

Figure 5

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

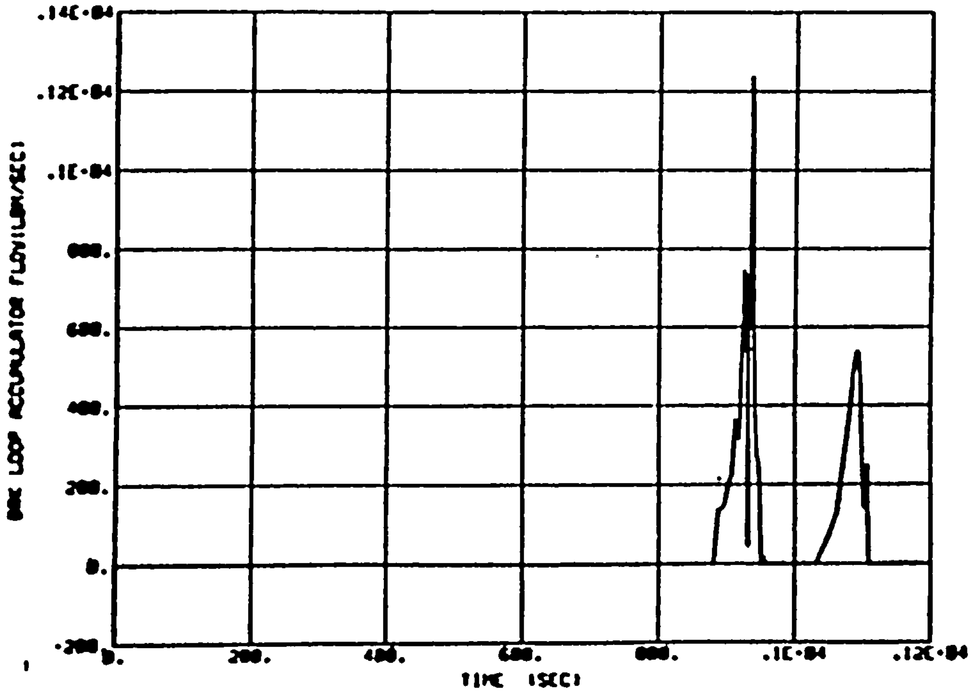

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Figure 6

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

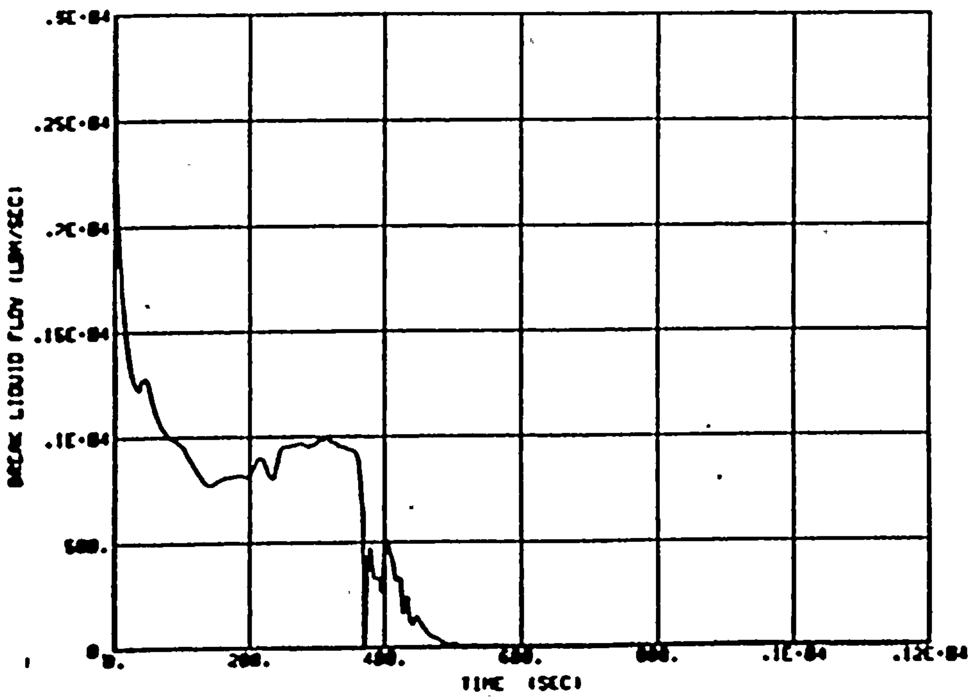
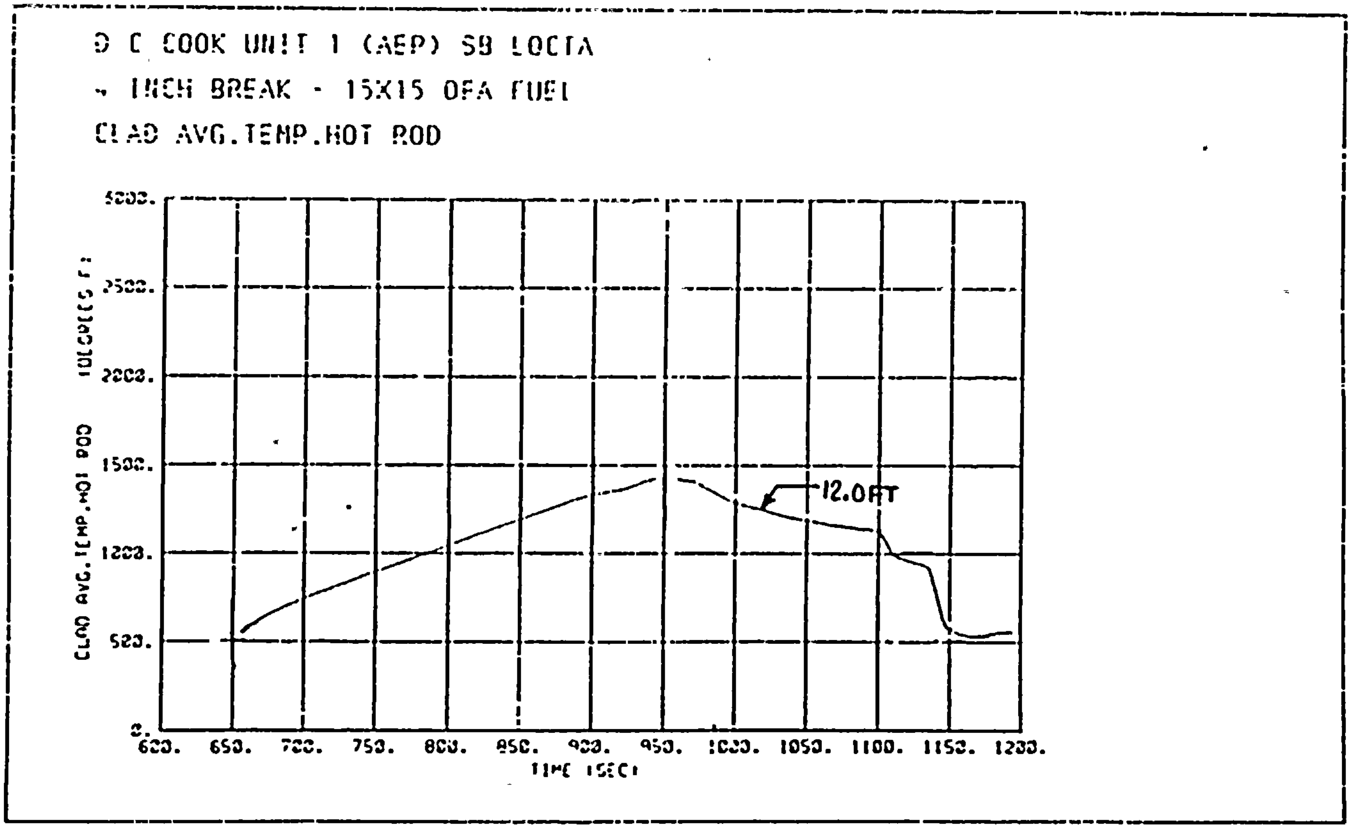


Figure 7



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)
3000.
2500.
2000.
1500.
1000.
500.
0.
600.
650.
700.
750.
800.
850.
900.
950.
1000.
1050.
1100.
1150.
1200.
TIME (SEC)
12.0FT

Figure 8

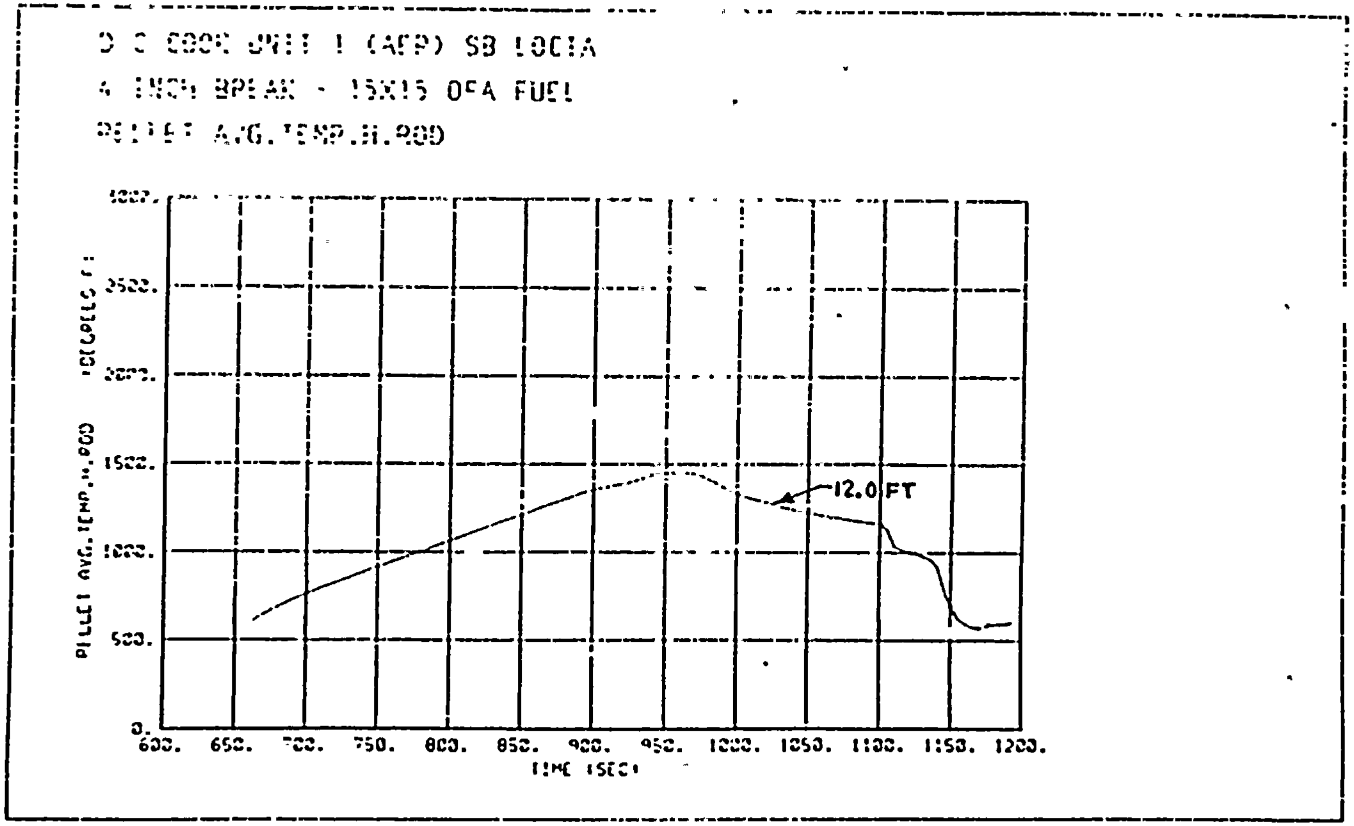


Figure 9

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

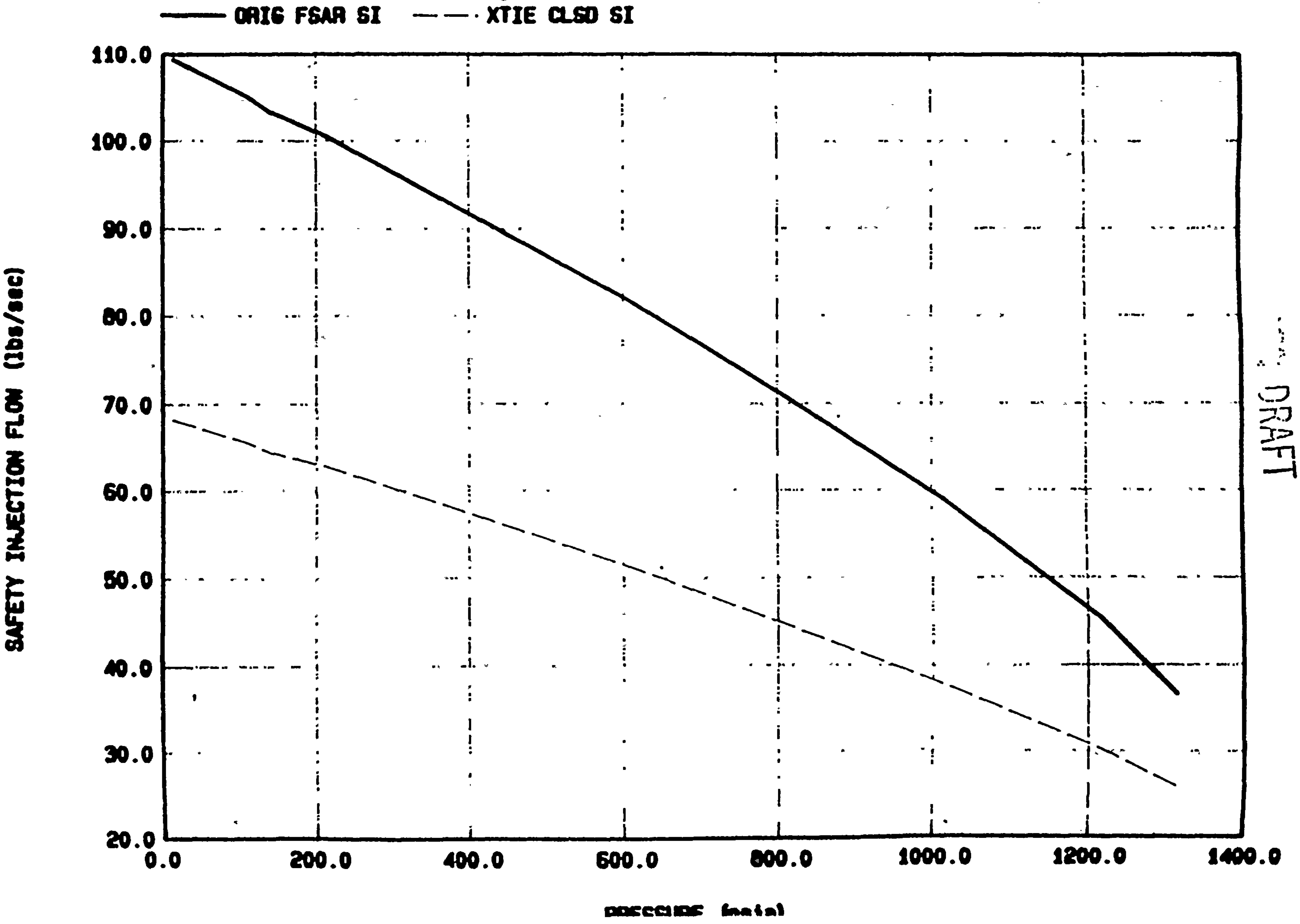


Exhibit B of

Attachment 2

to AEP:NRC:0916Z

Letter from M. P. Alexich (I&MECo.)

to H. R. Denton (NRC) Regarding

Operation of D. C. Cook Unit 2

with Safety Injection Cross-Tie Valves

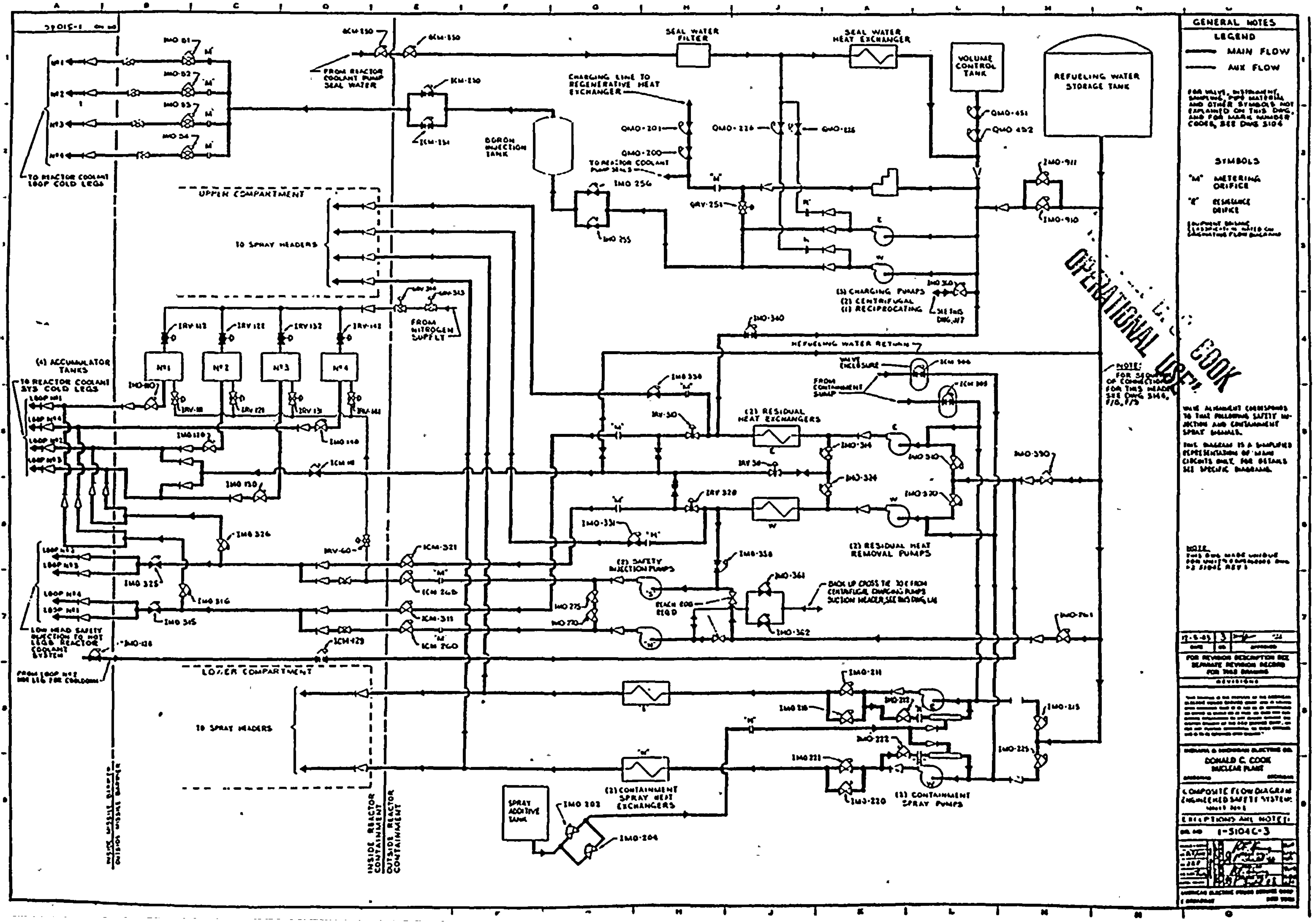
Closed (Previously Submitted in

Letter AEP:NRC:1024,

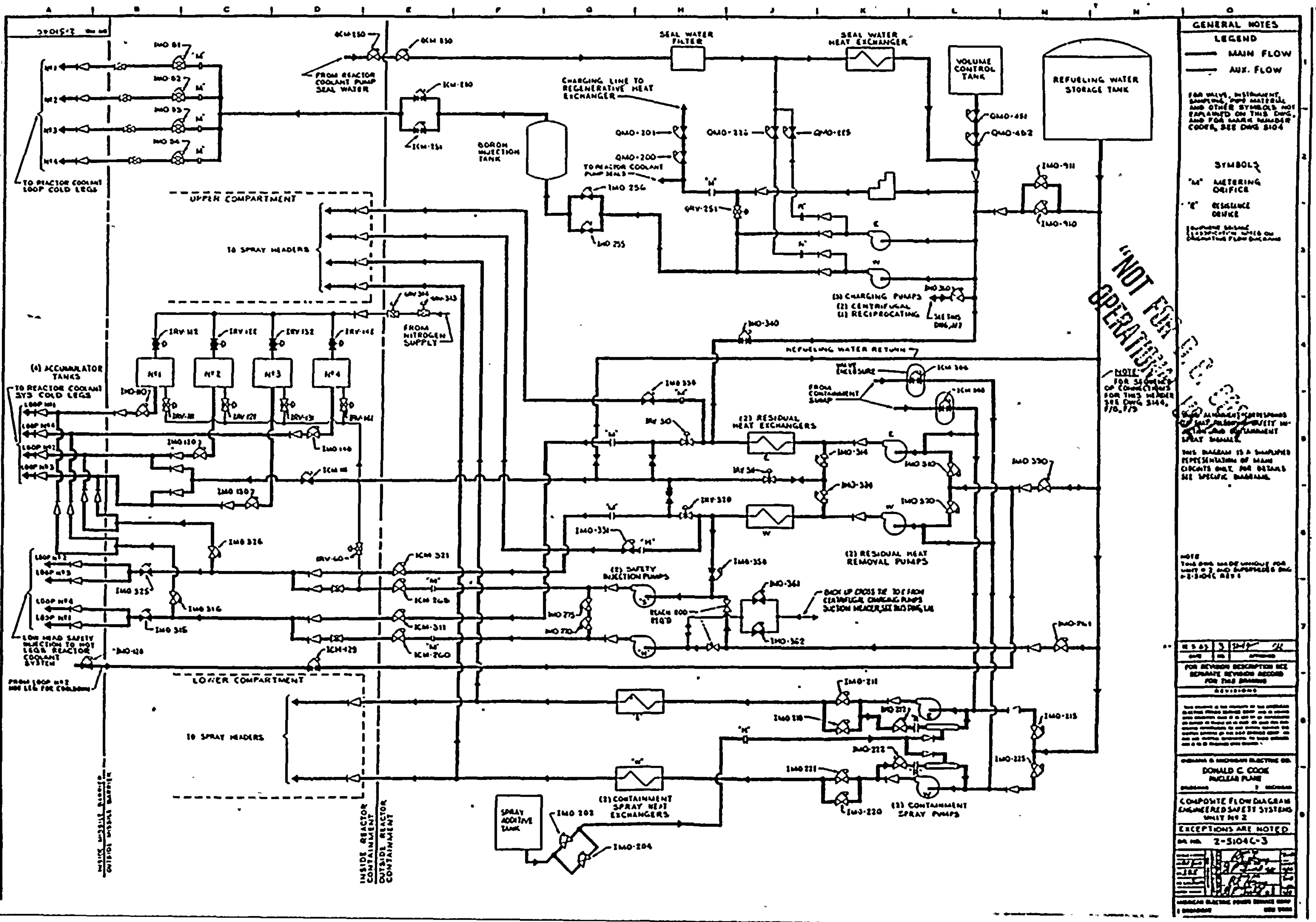
Dated March 23, 1987)

Attachment 1 to AEP:NRC:1024

Unit 1 and 2 ECCS Flow Diagrams



GENERAL NOTES
LEGEND
MAIN FLOW
AUX FLOW
SYMBOLS
"M" METERING 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 LOOP COLD LEGS
UPPER COMPARTMENT
TO SPRAY HEADERS
FROM NITROGEN SUPPLY
(3) CHARGING PUMPS
(2) CENTRIFUGAL
(1) RECIPROCATING
(4) ACCUMULATOR TANKS
TO REACTOR COOLANT SYS COLD LEGS
REFUELING WATER RETURN
(2) RESIDUAL HEAT EXCHANGERS
(2) RESIDUAL HEAT REMOVAL PUMPS
(2) SAFETY INJECTION PUMPS
LOW HEAD SAFETY INJECTION TO HOT LEGS REACTOR COOLANT SYSTEM
LOWER COMPARTMENT
TO SPRAY HEADERS
(2) CONTAINMENT SPRAY HEAT EXCHANGERS
(2) CONTAINMENT SPRAY PUMPS
SPRAY ADDITIVE TANK
INSIDE REACTOR CONTAINMENT
OUTSIDE REACTOR CONTAINMENT
D.C. COOK OPERATIONAL USE
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
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
(2) CONTAINMENT SPRAY HEAT EXCHANGERS
(2) CONTAINMENT SPRAY PUMPS
SPRAY ADDITIVE TANK
INSIDE REACTOR CONTAINMENT
OUTSIDE REACTOR CONTAINMENT
"NOT FOR OPERATIONS"
FOR REVISION DESCRIPTION SEE SEPARATE REVISION RECORD FOR THIS DRAWING
DONALD C. COOK NUCLEAR PLANT
COMPOSITE FLOW DIAGRAM ENGINEERED SAFETY SYSTEMS UNIT NO 2
EXCEPTIONS ARE NOTED
2-5104C-3

Attachment 2 to AEP:NRC:1024

Information Notice 87-01

Attachment 3 to AEP:NRC:1024

I.E. Inspection Report 86042

Unit 1 and Unit 2 LERs

September 15, 1986

1150 hrs. Licensee Problem Assessment Group (PAG) determined that this Condition Report involved events which are 10 CFR 50.72 and 10 CFR 50.73 reportable.

1400 hrs. 10 CFR 50.72 Four-Hour Report Notification was made.

4. Root Causes

The proximate cause of the event was lack of recognition by licensee personnel that the HPSI cross-tie valves are embodied in the meaning of "flow path" for that system. Several other factors contributed.

First, the Technical Specifications provide no clear meaning of the term "flow path" as it applies to the SI system. This is also true for other systems, but not all of them and in some cases, the meaning is clear.

Second, an insufficiently challenging Job Order review and approval process occurred in this case. This was voluntary maintenance, the leak which was repaired was small, and presented no threat to operability of the valve or other nearby components. Several NRC licensed Senior Reactor Operators (SROs) had opportunities to question or challenge the proposed maintenance activity as inconsistent with plant MODE, however, none did so.

The licensee lacked experience in setting up a "clearance" to work on the cross-tie valves. Neither valve has required maintenance for several years, if ever. Despite the fact that the activity involved both a novel and a non-mandatory isolation of safety related equipment, the licensee did not appear to perform anything other than a routine review and approval process.

The licensee performs on average, from 4,000 to 5,000 "clearances" (primarily for maintenance) each year. Many are repetitive; some are unique. Applicable administrative control procedures currently provide no special conditions or other criteria for handling uncommon clearances. The Shift Supervisor (licensed SRO) is ultimately responsible for approving all removal from and return to service of safety-related (and other) equipment.

Training considerations are addressed in Paragraph 7, below.

5. Event Significance

Overall safety significance is small. The margin of safety to fuel damage was reduced but not by more than about 25 percent, for a small spectrum of small break Loss of Coolant Accidents (LOCA's), which are the accidents for which SI pumps provide significant contribution.

[illegible]

The licensee and Westinghouse analyzed the subject 1 pump/2 cold leg configuration for the limiting break size (4-inch) and determined an average 5.84 percent flow reduction, for ECCS flow to the reactor, would occur in the critical small break LOCA, as the RCS depressurized from 1000 psi to 600 psi. The analysis credited flow from the two operable centrifugal charging pumps via alternate paths. At 600 psi, additional flow becomes available from the accumulators. Previous analyses had established a 21 degree Fahrenheit increase in peak clad temperature for each percent reduction in flow. Thus, a peak clad temperature (PCT) increase of about 123 degrees Fahrenheit could have resulted under these circumstances. The PCT could then be calculated to be 1791 degrees Fahrenheit with this configuration, rather than the 1668 degrees Fahrenheit originally calculated in the FSAR for breaks in this spectrum.

This constitutes about a 23 percent reduction in the 532 degree margin of safety to ECCS Rulemaking PCT of 2200 degrees Fahrenheit.

The event involved a violation of a Technical Specification Limiting Condition for Operation for an Emergency Core Cooling System which resulted in a reduction of the systems design safety margin. The system configuration that resulted in the violation was reviewed in advance by several licensed senior operators and persisted through two shift turnovers without disclosure. The improper cross-tie configuration was not identified by control room operators because they did not understand safety injection design basis due to a lack of specific system design basis training.

6. Reporting

Reporting requirements were met. A four-hour telephone notification pursuant to 10 CFR 50.72(b)(2)(iii) was made on September 15, 1986 about two hours and 10 minutes after the determination that the circumstances were reportable. A written Licensee Event Report (LER 316/86026, Rev. 0) pursuant to 10 CFR 50.73(a)(2)(v) was submitted on October 10, 1986 within the required 30 days.

7. Training

A review of the training program and discussions with shift personnel was performed focusing on the training and qualifications of personnel involved in this event. Licensed personnel were knowledgeable concerning SI system design and concerning administrative control processes for removal and return of safety equipment for maintenance. The effectiveness of the training process was deficient; however, in that factual knowledge about SI system design, administrative processes, and the Technical Specifications, were not integrated effectively. The design bases were not adequately understood.

[illegible]

8. Conclusion

As shown above, the licensee operated Unit 2 in MODE 1 for about 20 hours with the safety injection system in a condition not considered in the design bases and contrary to Technical Specification requirements. The primary cause was failure to recognize the significance of the SI cross-tie valves. This was contributed to by a lack of specificity in Technical Specifications, a lack of challenge in the review and approval process of non-routine clearances and ineffectively integrated training information.

9. Exit Interview

The inspectors met with licensee representatives (denoted in Paragraph 1) on November 7, 1986 to discussed the scope and findings of the inspection report. This exit was performed in conjunction with the exit for Inspection Report No. 315/86035; 316/86035. The inspector asked those in attendance whether they considered any of the items discussed to contain information exempt from disclosure. No items were identified.

Attachment 6 to AEP:NRC:1024

Westinghouse Small-Break LOCA Evaluation
for Unit 2

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

[illegible]

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 (^oF)	606.4	608.0
VESSEL INLET TEMPERATURE (^oF)	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

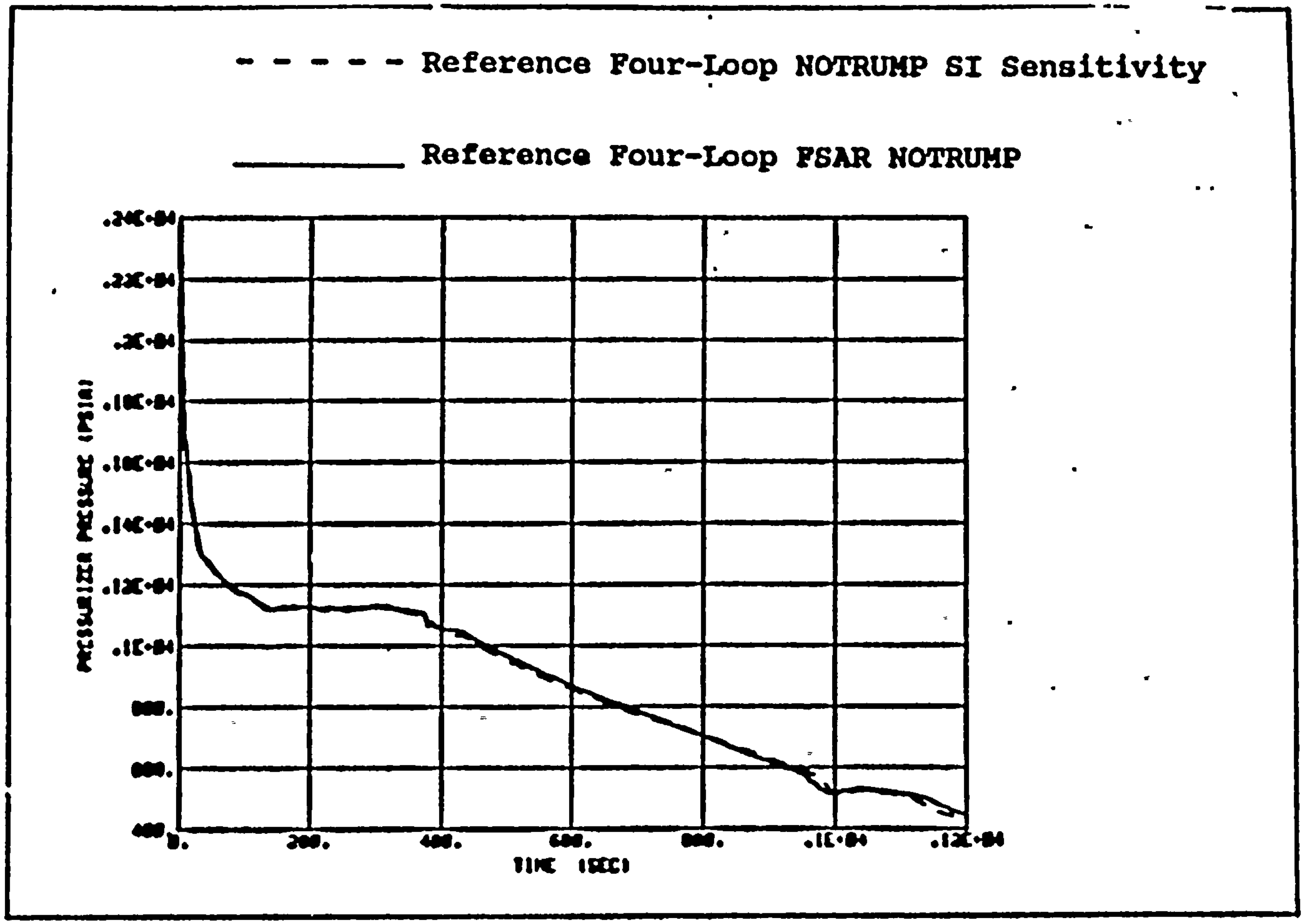
RESULTS	COOK UNIT 2 FSAR ANAL.	REFERENCE 412 FSAR ANAL.	REFERENCE 412 SI SENSITIVITY
SMALL BREAK MODEL	WFLASH	NOTRUMP	NOTRUMP
PEAK CLAD TEMPERATURE	1668°F*	1244°F	1132°F
PEAK CLAD TEMPERATURE LOCATION (FT.)	11.25	12.0	12.0
TIME OF PCT (SEC.)	913	991	1007
REACTOR TRIP (SEC.)	17.32	4.18	4.18
CORE UNCOVERY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

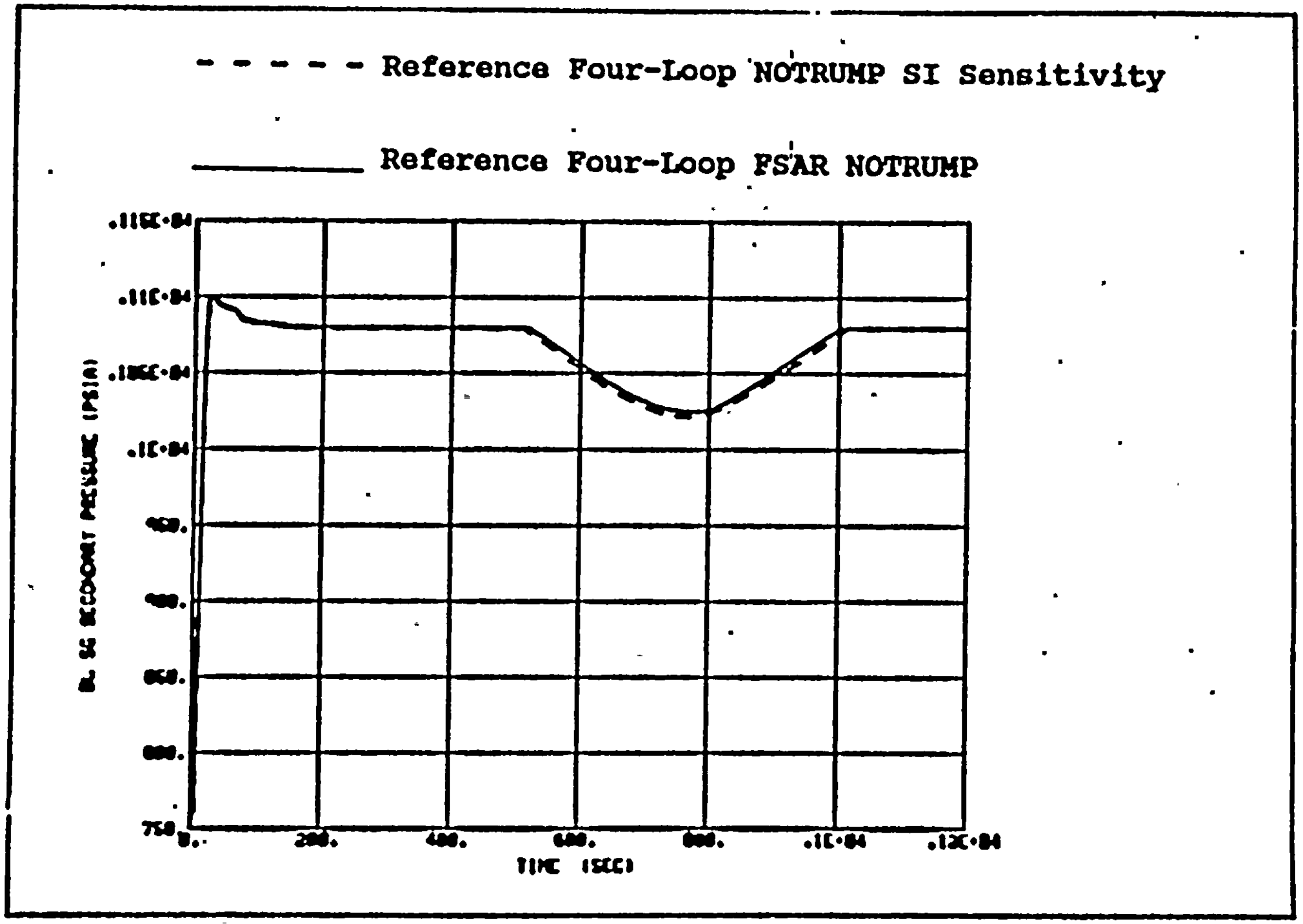
SAFETY INJECTION SYSTEM CONFIGURATION COMPARISON

	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			



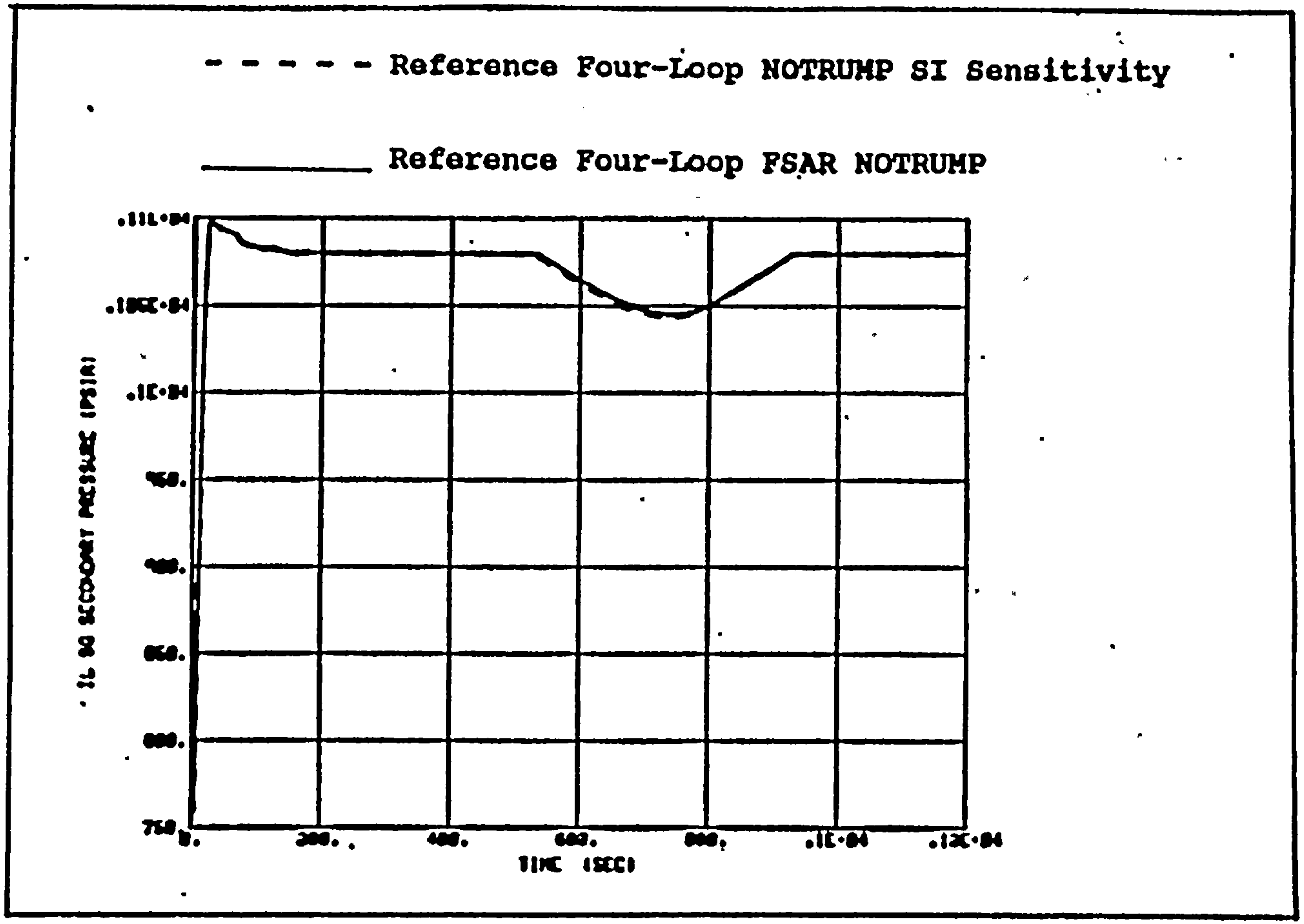
REFERENCE 4-LOOP NOTRUMP FSAR SI FLOWS vs. SI SENSITIVITY FLOWS

Figure 1



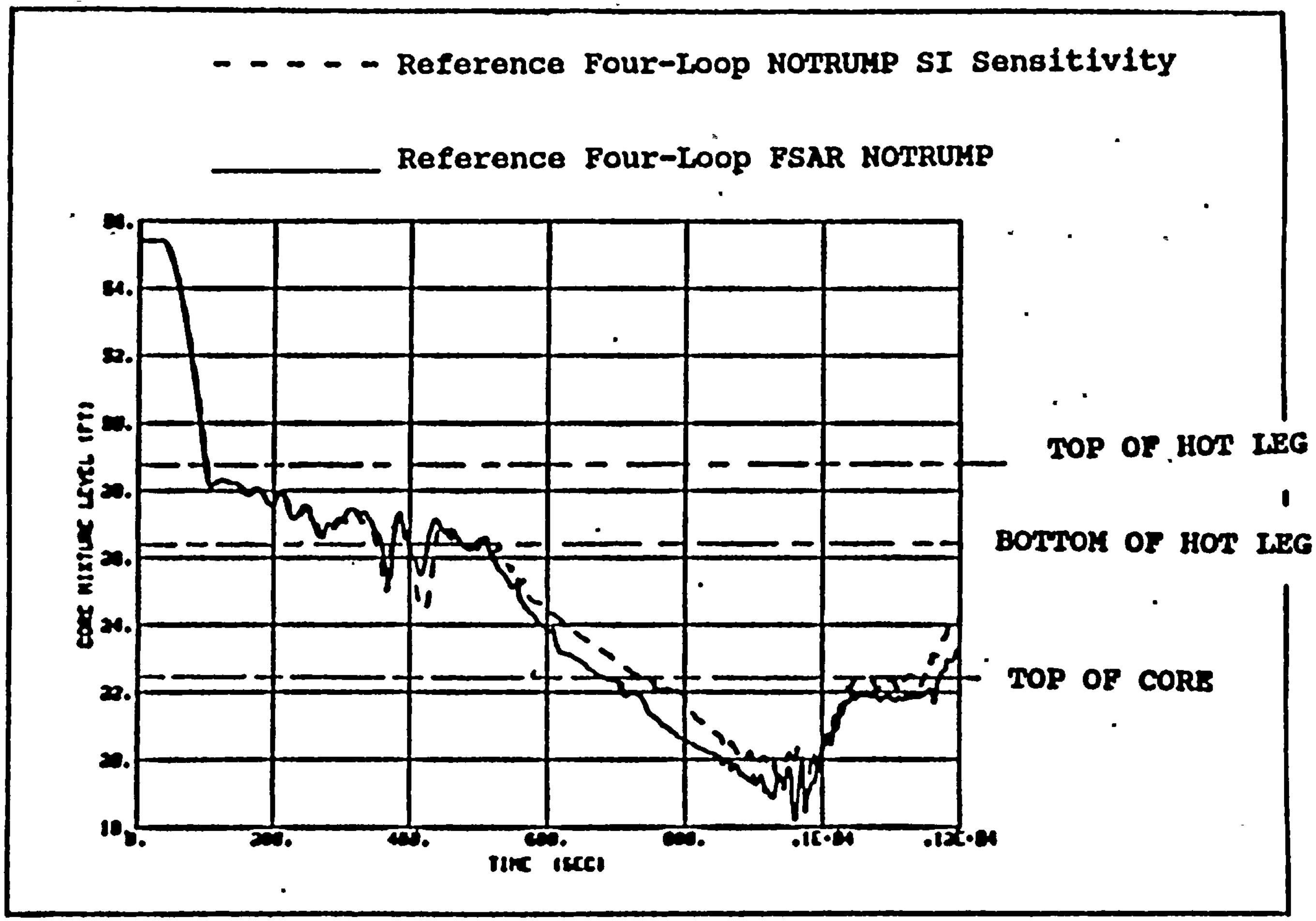
REFERENCE 4-LOOP NOTRUMP FSAR SI FLOWS vs. SI SENSITIVITY FLOWS.

Figure 2



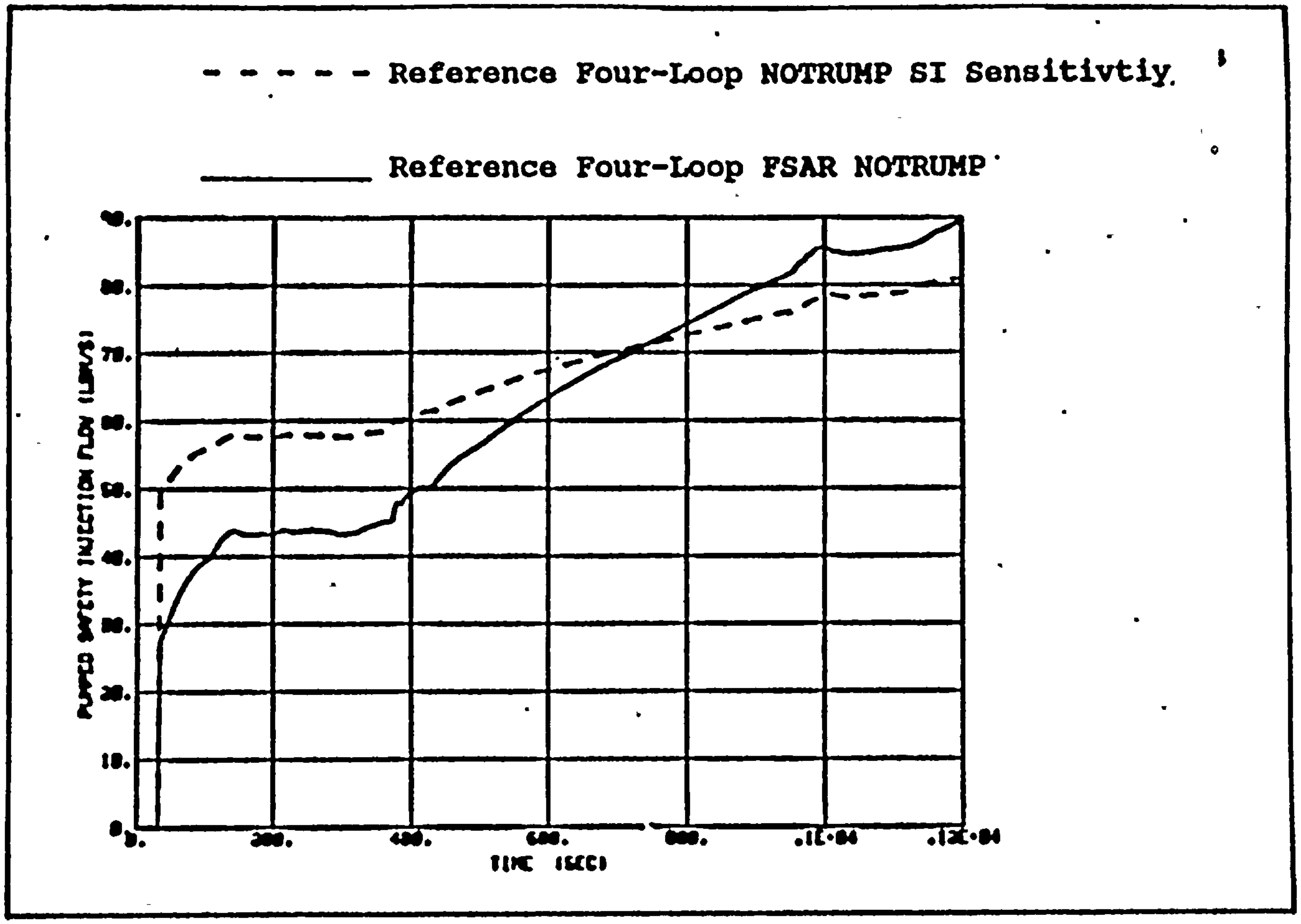
REFERENCE 4-LOOP NOTRUMP FSAR SI FLOWS vs. SI SENSITIVITY FLOWS

Figure 3



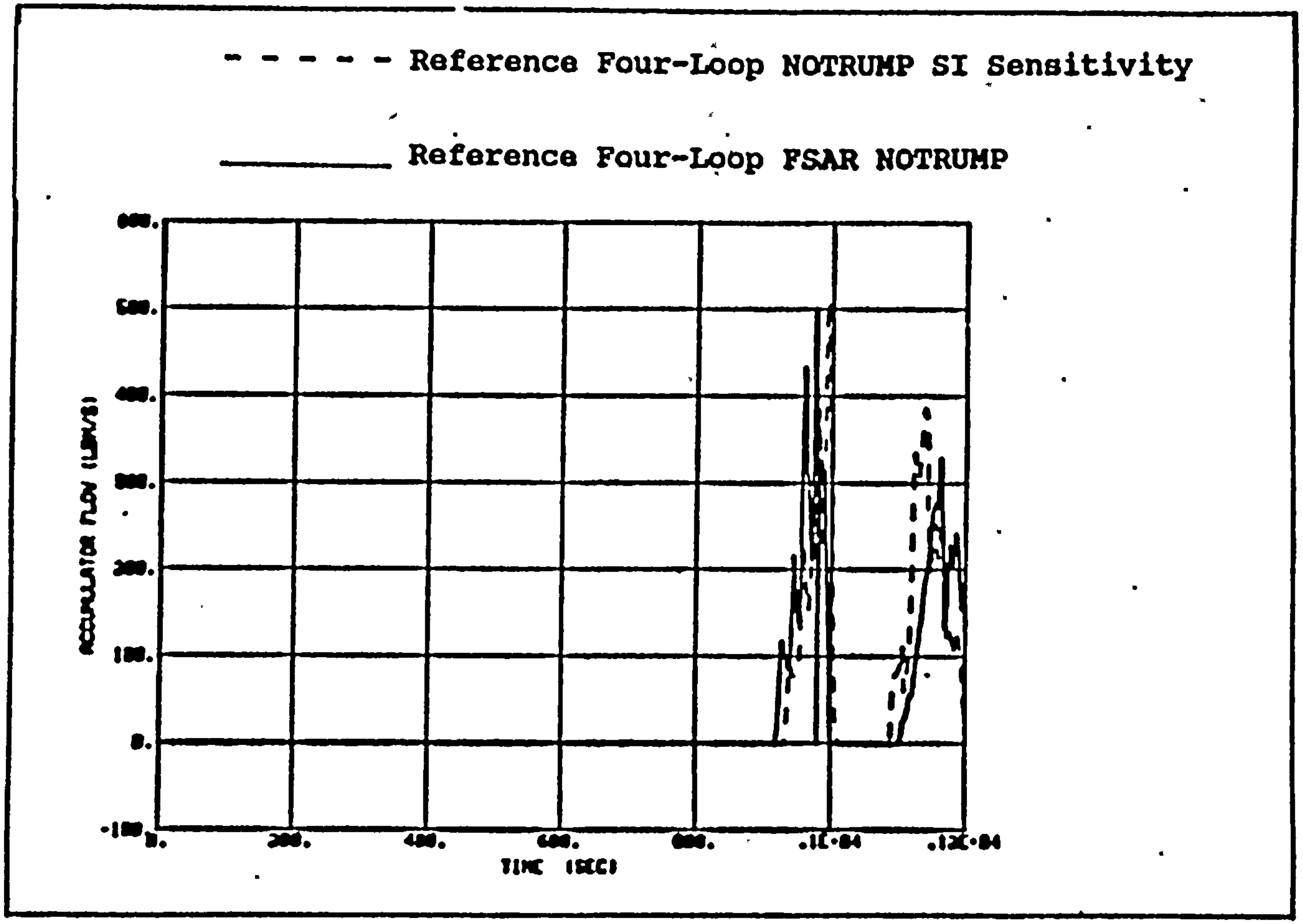
REFERENCE 4-LOOP NOTRUMP FSAR SI FLOWS vs. SI SENSITIVITY FLOWS

Figure 4



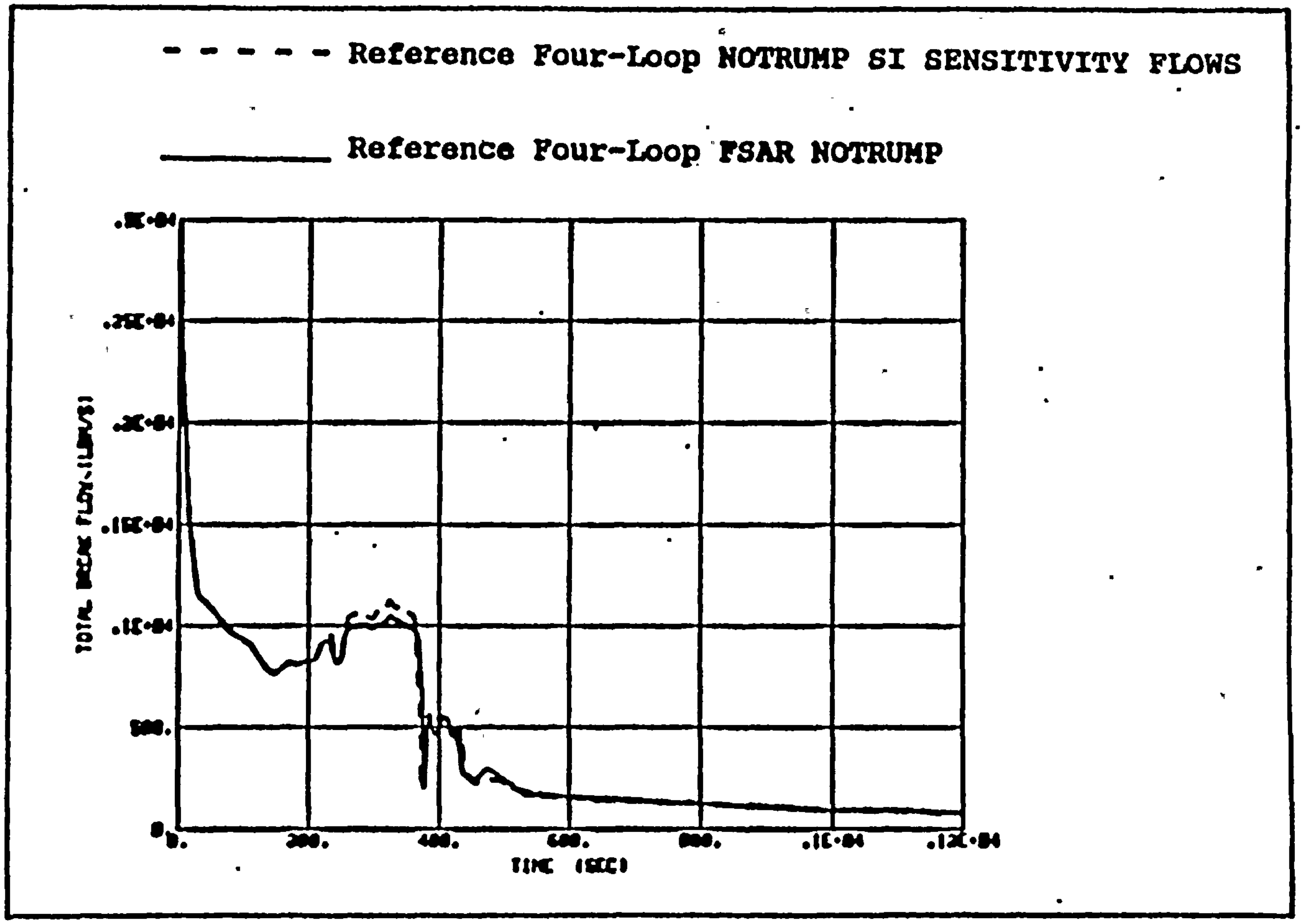
REFERENCE 4-LOOP NOTRUMP FSAR SI FLOWS vs. SI SENSITIVITY FLOWS

Figure 5



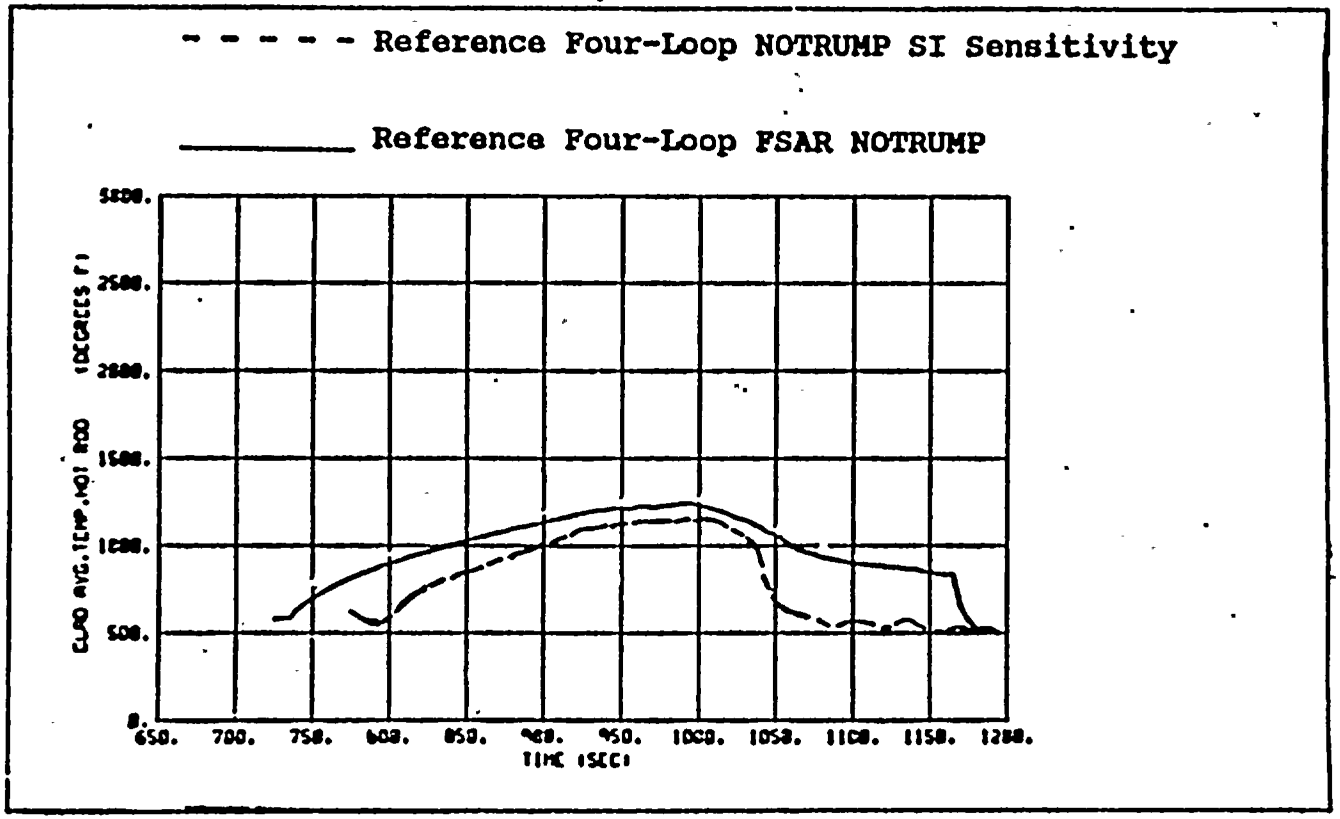
REFERENCE 4-LOOP NOTRUMP FSAR SI FLOWS vs. SI SENSITIVITY FLOWS

Figure 6



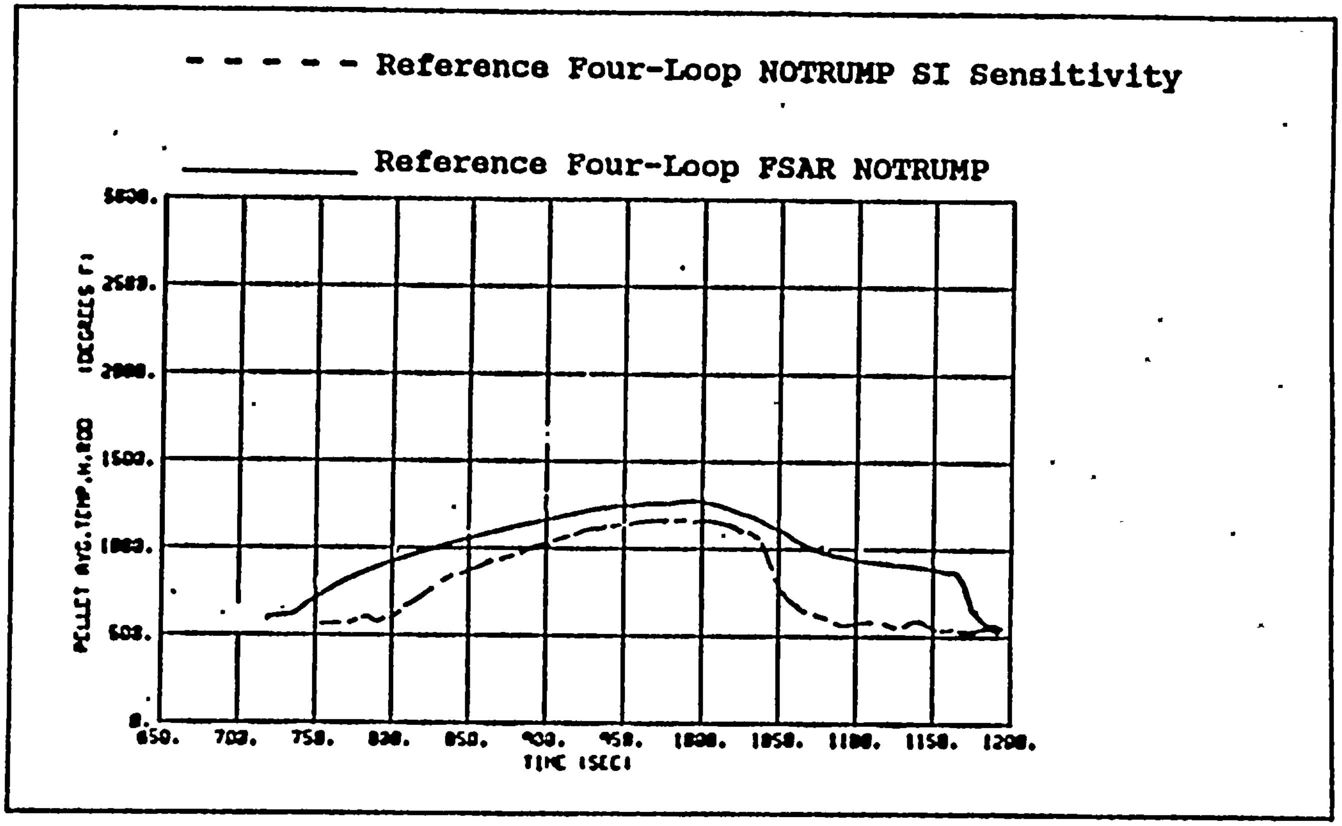
REFERENCE 4-LOOP NOTRUMP FSAR SI FLOWS vs. SI SENSITIVITY FLOWS

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

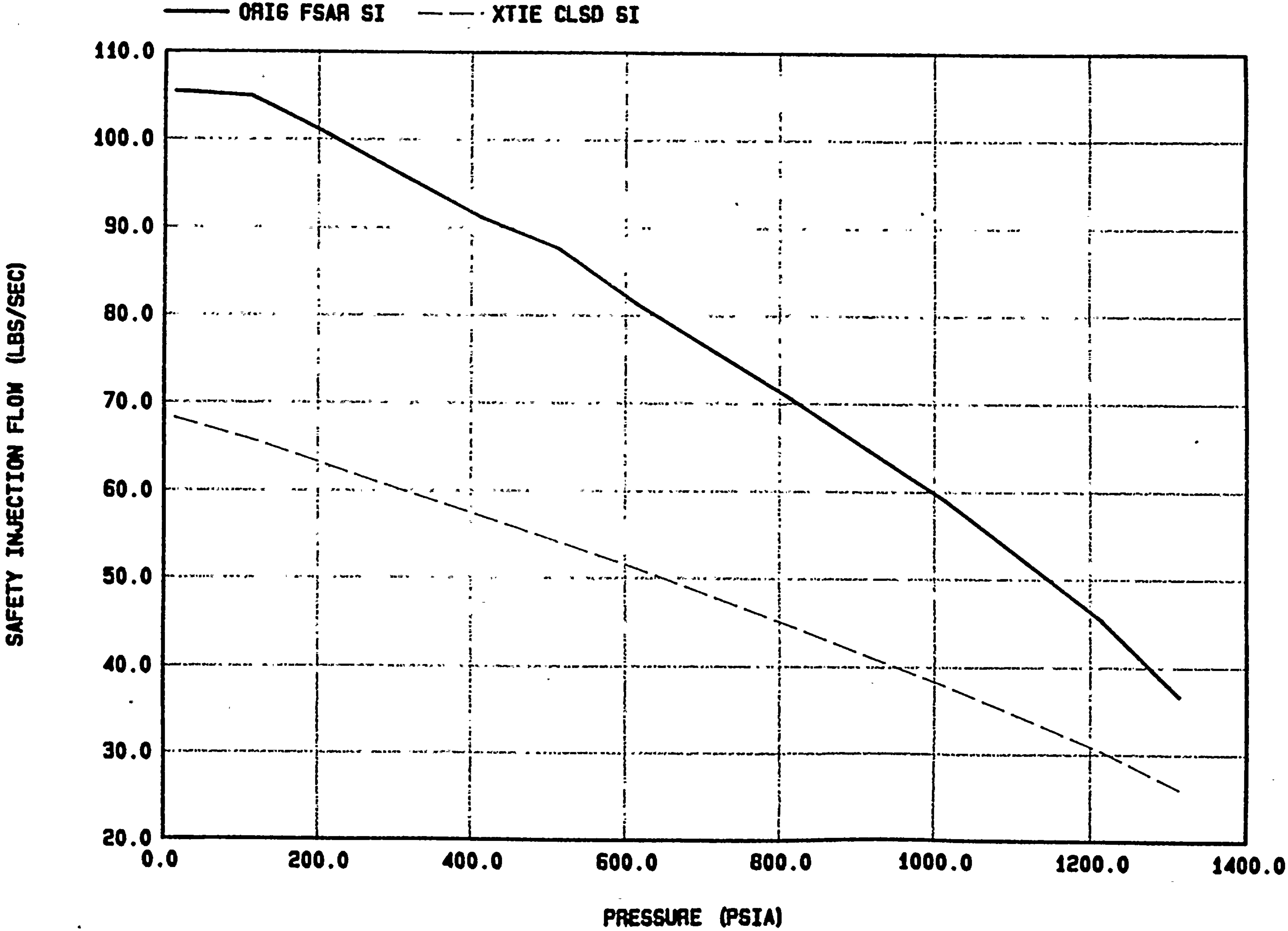


Figure 11
SAFETY INJECTION FLOW COMPARISON
REF. ANALYSIS SI VS. X-TIE CLOSE SI

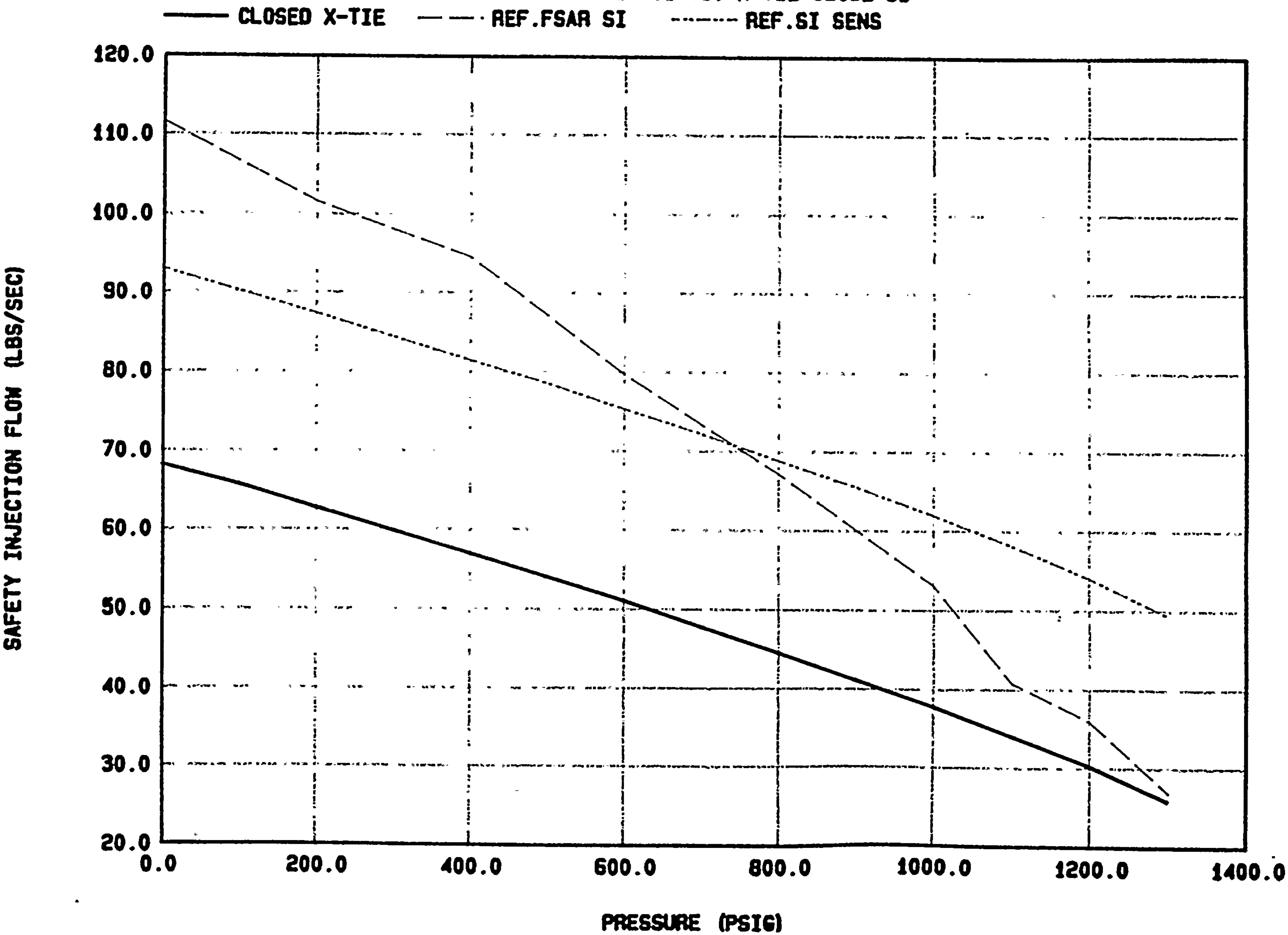


Exhibit C of

Attachment 2

to AEP:NRC:0916Z

Westinghouse Electric Corp.

Letter AEP-87-200

Related to Our Submittal AEP:NRC:1024

Westinghouse Electric Corporation

Power Systems

Nuclear Technology
Systems Division

Box 355
Pittsburgh Pennsylvania 15230-0355

AEP-87-205

March 24, 1987

NS-OPLS-OPL-II-87-062
REF: AEP-87-158 dated
March 5, 1987

Mr. J. G. Feinstein, Manager
Nuclear Safety & Licensing Section
American Electric Power Service Corporation
One Riverside Plaza
Columbus, Ohio 43216

Attention: W. G. Harvey

AMERICAN ELECTRIC POWER SERVICE CORPORATION
D. C. COOK UNIT
ECCS PERFORMANCE/CROSS-TIE CLOSURE

Dear Mr. Feinstein:

The above reference transmitted the final reports which examined both the effect that closure of the High Head Safety Injection (HHSI) cross-tie valve would have on the Emergency Core Cooling System (ECCS) performance in the event of a small break LOCA (Unit 2) as well as the effect on the ECCS following a Large Break LOCA in which the HHSI or Residual Heat Removal cross-tie was closed during normal full power operation (Unit 1).

As a result of telecons between Westinghouse and AEP (Ted Zimmerman) on March 23rd and March 24th it was identified that there was an error in the values reported for the core fluid volume for the reference plant and Cook Unit 2. The core fluid volume values have been corrected in the attached tables and should be incorporated into the final cross-tie closure reports prior to transmittal to the NRC.

Please find attached a revised Table 2 for both the D. C. Cook Unit 1 and Unit 2 cross-tie closure reports.

If you have any questions, please do not hesitate to contact us.

Very truly yours,

Casimir C. Swirat for

H. C. Walls, Project Manager
Mid-America Area
U. S. Nuclear Projects

D. L. Cecchett/dmr
Attachment

ATTACHMENT TO AEP-87-205

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 (°F)	599.3	608.8
VESSEL INLET TEMPERATURE (°F)	536.3	544.4

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 (^oF)	606.4	608.0
VESSEL INLET TEMPERATURE (^oF)	541.3	542.2

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