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SUBJECT: Forwards response to questions raised during 870729 telcon re small-break LOCA evaluation submitted to support operation of facilities w/safety injection cross-tie valves open.

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Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
SAFETY INJECTION CROSS-TIE: RESPONSE TO
NRC QUESTIONS ON SMALL-BREAK LOCA ANALYSES

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

Attn: T. E. Murley

October 13, 1987

Dear Dr. Murley:

The purpose of this letter is to transmit a response to questions raised by your staff during a telephone conversation on July 29, 1987. The questions were related to small-break LOCA evaluations which were submitted to support operation of D. C. Cook Units 1 and 2 with the safety injection cross-tie valves open. The response to your staff's question, which is specific to D. C. Cook Unit 1, was prepared by Westinghouse Electric Corporation (Westinghouse) and is attached to this letter; a response specific to D. C. Cook Unit 2 is currently being prepared and will be transmitted under separate cover.

Background

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 charging pump, safety injection (SI) pump, Residual Heat Removal (RHR) pump, RHR heat exchanger, and associated flow paths. This T/S 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. The RHR and safety injection pump configuration at the D. C. Cook Plant is such that any one pump can deliver flow to all four reactor coolant loops. This is accomplished by means of cross-tie valves. With the cross-tie valves closed, each pump can only supply flow to

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two reactor coolant loops. The current small-break and large-break LOCA analyses for D. C. Cook Unit 1 assume that the cross-tie valves in the SI and RHR lines are open. This requires that the cross-tie valves be open to satisfy the operable flow path requirements of T/S 3.5.2.e for Modes 1, 2, and 3.

In the past, there were instances where D. C. Cook Units 1 and 2 were operated in Modes 1, 2, or 3 with the cross-tie valves closed. The valves were closed to allow maintenance work to be performed on various system components. Because this operation was not in agreement with the existing safety analyses, it was the subject of an Enforcement Conference held at Region III headquarters on January 21, 1987.

Since some maintenance and testing work can only be performed on the RHR or SI systems in Modes 1, 2, or 3 with the cross-tie valves closed, we decided to pursue new analyses which would support two-loop injection. The new analyses were submitted in our letters AEP:NRC:1024, dated March 23, 1987, and AEP:NRC:1024A, dated May 13, 1987. The analyses presented in these letters supported closing of either the SI system cross-tie valves or the RHR system cross-tie valves, but not both. The new analyses involved large-break LOCA evaluations for the RHR cross-tie valves, and small-break LOCA evaluations for the SI cross-tie valves.

In a discussion with your staff on July 29, 1987 we were informed that the staff had no problems with the RHR (large-break) analyses but had remaining questions on the SI (small-break) analyses. The analyses we submitted in support of closing the SI cross-ties considered 3-inch and 4-inch small-break LOCAs. In the July 29, 1987 telephone conversation, your staff requested that we investigate break sizes smaller than 3 inches, to ensure that with the SI cross-ties closed, smaller breaks that do not depressurize the reactor coolant system below the accumulator setpoint will not become limiting with regard to peak clad temperature.

Summary of Westinghouse Evaluation of Effects of Smaller Size Breaks

The following paragraphs present the evaluation approach used by Westinghouse in determining the effect of the safety injection flow reduction resulting from closure of the D. C. Cook Unit 1 safety injection cross-tie valves.

To determine the D. C. Cook Unit 1 core response to breaks smaller than three inches in diameter with the SI cross-tie closed, an examination was made of the small LOCA transient response exhibited in the Westinghouse-design PWRs for variations in break size, power

level and safety injection flow rates. Historical information addressing plant response to a broad spectrum of break sizes for variations in the ratio of safety injection flow to core power was examined to conclude that with the evolution of LOCA technology, the basic system response predicted by the earliest LOCA models remains very similar to that predicted by the more sophisticated models used today. An investigation of the small-break LOCA studies performed from 1974 to the present indicated that even with substantial reductions in safety injection flow, smaller breaks that do not result in depressurization to the accumulator setpoint will not result in the most limiting peak clad temperatures.

To verify the conclusion predicted by the historical data, i.e., that the small breaks would not become limiting for D. C. Cook Unit 1 with the reduced SI flow, Westinghouse performed an evaluation for a 2-inch small-break LOCA. The results of the evaluation concluded that core uncovery should not result for the 2-inch break.

Based on their evaluations of historical small-break LOCA analyses and on the specific small-break LOCA evaluations performed in support of the cross-tie closure effort, Westinghouse was able to conclude that smaller breaks that do not result in depressurization to the accumulator setpoint will not result in the most limiting peak clad temperature.

Additional Studies

Since the July 29, 1987 discussion with your staff, we have been informed by Westinghouse that closing the RHR or SI cross-tie may impact the containment long term calculated pressure. We have undertaken a study to provide additional confidence that the containment LOCA response analyses are not significantly impacted by the reduction in flow associated with closure of the cross-tie valves. We will assess the significance of any impact prior to implementing the relief we are requesting. Should our study indicate that the present containment analyses may be significantly impacted, we will inform you and establish a schedule for resolution of the containment response-related items.

Date When Response is Needed

We believe the information in this letter is sufficient to close out your staff's concerns regarding the Unit 1 RHR and SI cross-tie analyses. Because certain maintenance can only be performed with the cross-tie valves closed, we would appreciate your acceptance of the analyses as soon as possible to avoid the possibility of our

having to shut the unit down for repair activities. Specifically, however, we note that in-service testing which must be performed for Unit 1 by December 13, 1987 will require the RHR cross-tie valves to be closed. If your staff's acceptance of our analyses is not granted, it will be necessary to shut down D. C. Cook Unit 1 in order to perform the required testing. Therefore, we request that the analyses be closed out by no later than November 15, 1987 so that appropriate planning for the in-service testing can be done.

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

Sincerely,

MPAlexich

M. P. Alexich
Vice President

cm

Attachment

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

Attachment to AEP:NRC:1024C

Westinghouse Electric Corporation
Evaluation of Smaller Break Sizes

Attachment to AEP:NRC:1024C

Westinghouse Electric Corporation
Evaluation of Smaller Break Sizes

LIMITING SMALL BREAK LOCA CONSIDERATIONS

FOR OPERATION OF D.C. COOK UNIT 1 WITH CLOSURE

OF THE HIGH HEAD SAFETY INJECTION CROSS-TIE

EXECUTIVE SUMMARY

An evaluation was performed to determine the effect of closure of the High Head Safety Injection (HHSI) Cross-Tie line on the D.C. Cook Unit 1 ECCS performance during a small break loss-of-coolant-accident (LOCA) which does not rely upon accumulator injection for recovery. The results indicate that the D.C. Cook Unit 1 ECCS performance with the HHSI Cross-Tie closed is effective in mitigating the consequences of small LOCAs which do not result in depressurization of the RCS to the accumulator injection setpoint.

The evaluation has shown that smaller breaks that do not result in depressurization of the RCS to the accumulator setpoint will not result in the most limiting Peak Clad Temperature (PCT). The 4-inch diameter cold leg break will remain the limiting break size with the HHSI Cross-Tie valves closed. Previous evaluations performed by Westinghouse have determined that the effects of the HHSI Cross-Tie closure will cause the PCT for the 3-inch and 4-inch breaks to increase. However, the PCT increase for the 4-inch case was determined to be greater than the increase for the 3-inch case. Therefore, it was concluded that the 4-inch diameter cold leg break will remain the limiting small break LOCA event for D.C. Cook Unit 1 with the safety injection flow reduction resulting from closure of the HHSI Cross-Tie line.

The following paragraphs present the evaluation approach used in determining the effect of the safety injection flow reduction resulting from closure of the D.C. Cook Unit 1 Cross-Tie line.

To determine the D.C. Cook Unit 1 core response to breaks smaller than three inches in diameter with the HHSI Cross-Tie closed, an examination was made of the small LOCA transient response exhibited in the Westinghouse design PWR for variations in break size, power level and safety injection flow rates. Historical information addressing plant response to a broad spectrum of break sizes for variations in the ratio of safety injection flow to core power was examined to conclude that with the evolution of LOCA technology, the basic system response predicted by the earliest LOCA models remains very similar to that predicted by the more sophisticated models used today. An investigation of the small break LOCA studies performed from 1974 to the present indicated that even with substantial reductions in safety injection flow, smaller breaks that do not result in depressurization to the accumulator setpoint will not result in the most limiting peak clad temperatures.

To verify that the trend predicted by the historical data, i.e. that the smaller breaks would not become limiting for D.C. Cook Unit 1 with the reduced SI flow, a system mass inventory calculation was performed. The calculation assumed a net mass inventory depletion and a depressurization transient for a 2-inch break analysis performed on a typical Westinghouse 4 loop plant. The depressurization transient was then used in conjunction with the safety injection mass flow rates for Cook Unit 1 when the HHSI Cross-Tie is assumed closed to determine the net safety injection flow delivered throughout the 2-inch break LOCA transient. The results of the mass inventory calculation indicated that the safety injection flow available when the HHSI Cross-Tie is closed would be sufficient to maintain the system mass inventory above the top of the core throughout the duration of the transient.

LIMITING SMALL BREAK LOCA CONSIDERATIONS

FOR OPERATION OF D.C. COOK UNIT 1 WITH CLOSURE

OF THE HIGH HEAD SAFETY INJECTION CROSS-TIE

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 in the D.C. Cook Unit 1 final safety analysis report (FSAR) assume that high head safety injection flow delivery is available through all four lines.

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 a single failure would result in the loss of flow from one of the pumps. The FSAR analyses assume the loss of one electrical safeguard emergency bus due to the failure of a diesel to start when offsite power 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). 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. 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 reduction in the amount of safety injection flow due to the cross-tie line closure on the plant response to a small break LOCA, Westinghouse performed a small break LOCA ECCS evaluation model 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.

The 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 of the 4-inch diameter break resulted in a PCT of 1427°F, thereby illustrating that the results of a small break LOCA with the flow equivalent to one charging pump delivering to four RCS loops and one HHSI pump delivering to only two of four loops, would meet the requirements of 10 CFR 50.46 and Appendix K.

The results of the small break LOCA analysis addressing the effect of closure of the HHSI Cross-Tie on the ability of the ECCS system to mitigate the consequences of a small rupture to the RCS piping was submitted to the NRC staff in support of an interpretation of the Technical Specifications which would allow the full power operation of the unit with the HHSI Cross-Tie closed.

In response to concerns expressed to American Electric Power by the NRC staff, regarding the possible shift of the limiting break size, Westinghouse further examined the three-inch diameter break transient for the reference plant design analysis. The results of the investigation concluded that the integrated safety injection flow available prior to the onset of accumulator injection in the 3-inch diameter break would result in substantially less core uncovery than shown in the 4-inch break transient. American Electric Power submitted the results of the three-inch break evaluation to the NRC staff in AEP-NRC-1024A dated May 13th 1987.

On July 29th 1987, a telecon with AEP, Westinghouse and the NRC was held to further discuss the possible shift of the limiting break size. Because the three and four inch diameter breaks result in the depressurization of the RCS to the accumulator setpoint, recovery from the LOCA is not dependent on HHSI alone. The NRC expressed the concern that for breaks smaller than 3-inches in diameter the RCS may not depressurize to the accumulator injection setpoint without operator action and may therefore result in more limiting Peak Clad Temperatures.

INTRODUCTION

To address the NRC concerns regarding the possible limitations of small break LOCAs which do not rely on accumulator injection for recovery, American Electric Power requested Westinghouse to evaluate the D.C. Cook Unit 1 ECCS performance with the HHSI Cross-Tie closed for small break LOCA events smaller than three inches in diameter.

In determining the D.C. Cook Unit 1 core response to very small LOCA events, the cold leg break transient response in a typical four loop Westinghouse PWR was reviewed. The cold leg break location is assumed since this location has the propensity for the most severe core uncovery.

The response of a typical Westinghouse four loop plant to a small rupture in the the RCS cold leg piping is characterized by a complex sequence of events. At the onset of the break, the primary reactor coolant system will rapidly depressurize to the saturation pressure of the hot leg fluid. As flashing begins, voids collect and form in the high points of the system as the liquid inventory begins a top down drain and the rate of depressurization decreases. A reactor trip is generated by low pressurizer pressure. The loss of offsite power assumption results in the isolation of the steam generator secondary side which will then pressurize to the steam generator safety valve pressure. Some steam relief through the safety valves will occur. The primary pressure will depressurize to an equilibrium pressure above the steam generator secondary pressure. The primary equilibrium pressure plateau is governed by the steam generator secondary conditions which will determine the amount of primary to secondary heat transfer. The primary pressure will reach equilibrium where the primary to secondary heat transfer will compensate for the volume expansion due to safety injection and decay heat produced steam which is not removed by the break. The continued generation of decay heat produced steam results in steam flow from the core, through the hot legs and into the the steam generator U-tubes. This causes the depression of liquid in the downward portion of the steam generator U-tubes into the U-bend region of the pump suction leg to the point where the core produced steam forces the liquid seal sustained in the pump suction leg out the break.

The venting of steam through the pump suction leg loop seal allows the core decay heat produced steam to exit out the break. Primary pressure decreases after loop seal blowout because the volumetric removal by the break defined by the system pressure exceeds the volumetric addition due to safety injection and the decay heat produced phase change. As the pressure gradually decreases, mass flow through the break decreases and pumped safety injection flow increases.

For larger breaks, the primary RCS pressure will decrease to the point where the break mass flow will decrease below the mass flow input into the system by safety injection and accumulator injection flow and the system mass inventory will increase. Finally, the system will reach a stable equilibrium pressure wherein the mass flow through the break equals the mass flow input into the RCS.

For smaller breaks, the system will reach a stable equilibrium pressure above the accumulator nitrogen gas pressure. At this equilibrium pressure the volumetric removal out the break must equal the volume of safety injection fluid injected into the RCS, hence, the safety injection mass flow rate will exceed the mass flow through the break and result in a net mass inventory increase. The equilibrium pressure at which the safety injection mass flow rate begins to exceed the break mass flow rate will be lower for plants with reduced safety injection flow capabilities.

To determine the D.C. Cook Unit 1 core response to breaks smaller than three inches in diameter with the HHSI Cross-Tie closed, Westinghouse examined in depth, the small break LOCA transient response exhibited in the Westinghouse design PWR for variations in break size, power level and safety injection flow rates. In determining the D.C. Cook Unit 1 response to breaks smaller than 3-inches in diameter, historical information examining core response to variations in break size, core power and safety injection flow was introduced. This information was presented to establish that with the evolution of LOCA technology, the overall system response predicted by the earliest LOCA models remains unchanged from those predicted by the more sophisticated models used today. An investigation of the small break LOCA studies performed from 1974 to the present indicates that even with substantial reductions in safety injection flow, the smaller breaks do not result in the most limiting Peak Cladding Temperatures.

The applicability of the power levels, safety injection flow rates and plant configurations assumed in the historical background analyses was then established. The results of a comparison of the D.C. Cook Unit 1 plant configuration, power level and safety injection flow rates to the plant configurations, power levels and safety injection flow rates assumed in the analyses presented for historical background verified the applicability of the historical sensitivity studies to D.C. Cook Unit 1. These comparisons were drawn to demonstrate the low probability of a shift in the D.C. Cook Unit 1 limiting break size with the SI flow reduction resulting from the HHSI Cross-Tie closure.

Finally, to verify that the smaller breaks would not become limiting for D.C. Cook Unit 1 with the reduced SI flow, a system mass inventory calculation was performed. The calculation assumed a net mass inventory depletion and a depressurization transient for a 2-inch break analysis performed on a typical Westinghouse 4 loop plant. The depressurization transient was then used in conjunction with the safety injection mass flow rates for Cook Unit 1 when the HHSI Cross-Tie is assumed closed to determine the net safety injection flow delivered throughout the 2-inch break LOCA transient. The results of the mass inventory calculation indicated that the safety injection flow available when the HHSI Cross-Tie is closed would be sufficient to maintain the system mass inventory above the top of the core throughout the duration of the transient.

<u>Limiting Small Break LOCAs: Historical Perspective</u>

Correct assessment of the consequences of "smaller" (2-inch equivalent diameter or less) small break loss-of-coolant-accidents (LOCAs) can be accomplished through review of previous Westinghouse analyses of various reference

plants. Small break LOCAs refer to small ruptures of the primary reactor coolant system (RCS) which result in a net depletion of the liquid inventory and which could cause the fuel rod cladding to heat up. However, for significant cladding heat up to occur the break location, break size, decay heat rate, power distribution, and magnitude of the safety injection flow rate must interactively result in an inner vessel mixture level transient which would uncover the top of the core. The magnitude of the cladding heat up is a function of the amount of core uncovery, the decay heat distribution, the decay heat rate, and the steam flow rate past the exposed portion of the fuel rods. The limiting break size for small ruptures in the RCS is that size break that would result in the highest cladding temperature.

This section's review will identify the limiting break size for the various plant analyses used in the assessment of core response to very small LOCAs. The calculated limiting break size has evolved with time as greater experimental evidence and more advanced analytical methods have become available. However, the fundamental behavior and response of the calculated peak cladding temperature remains basically unchanged. Historically it has been shown that small ruptures in the RCS piping which do not result in depressurization of the RCS to the accumulator injection setpoint result in less limiting Peak Clad Temperatures than larger breaks where recovery is dependent upon accumulator injection.

The analyses reviewed indicate that the limiting break size is smaller when less safety injection flow is available. However, comparing the Cook Unit 1 Cross-Tie closed case to the cases to be reviewed here, on the comparison basis of lb-mass/sec of safety injection per MWth of core power, and examining the limiting break sizes from the cases reviewed herein, leads to the conclusion that Cook Unit 1 Cross-Tie closed situation still has its limiting break size in the range considered in previous Cook Unit 1 small break LOCA analyses.

WCAP-7422

Analyses were performed for a typical Westinghouse designed four loop plant and presented in WCAP-7422-L (Reference 1), which demonstrated the adequacy of the emergency core cooling system (ECCS) to terminate core exposure and limit the temperature rise of the fuel rods during a LOCA. The analyses assumed a vessel design identical to D.C. Cook Unit 1 and was performed for standard Westinghouse 15X15 fuel assuming a core power level of 2931.2 MWt. The analyses were performed for a wide spectrum of break sizes and examined various combinations of safety injection pump availability. The limiting ratio of the safety injection mass flow rate at the steam generator safety valve setpoint to the core power ratio was 0.0147 lbm/sec/MWt for the analyses presented in WCAP-7422-L. The safety injection flow rate used in determining this ratio was a pumped mass flowrate

of 43 lbm/sec. For small breaks, the report showed that the high head safety injection system is capable of maintaining adequate system liquid inventory to preclude significant fuel rod cladding heat up.

Ruptures of very small cross sections (up to about the equivalent of a 3/8-inch connecting pipe) will cause expulsion of coolant at a rate which can be accommodated by the charging pumps well before the core would become uncovered. WCAP-7422-L (Reference 1) also found that for larger leaks (up to about 1/2-inch), the charging pumps would maintain an operational level of water in the pressurizer, which would permit the operator to execute an orderly shutdown. The resultant loss of liquid inventory from a larger break would cause reactor to trip and the initiation of safety injection flow supplementing the charging flow.

WCAP-7422-L (Reference 1) examined break sizes of 1, 2, 3, 4, and 6 inches in equivalent diameters with various combinations of safety injection pump availability. Three safety injection flow performance curves were analyzed which varied from maximum flow capacity to a minimum flow capacity case assuming a pumped injection flow of 43 lbm/sec at a pressure of 1114.7 psia. The maximum flow capacity analyzed was 136 lbm/sec at a pressure of 1114.7 psia. The RCS pressure of 1114.7 psia is characteristic of the steam generator safety valve setpoint pressure which was chosen as representative for the determination of the SI flow to core power ratio.

WCAP-7422-L (Reference 1) found that the limiting break location was in the cold leg. Furthermore, WCAP-7422-L (Reference 1) found that for the range of safety injection examined, the cladding temperature did not increase above the normal operating condition temperature for all break sizes up to and including the 4-inch equivalent diameter break in the cold leg. The 6-inch equivalent diameter cold leg break hot spot (point of maximum clad temperature at break initiation) was uncovered for a short period of time for the minimum injection case, but remained covered for the full injection case. The 6-inch break analysis resulted in a calculated peak clad temperature of 1550°F. From the results of the analyses, it was concluded that a break in the range of 3-inches to 4-inches in equivalent diameter defined the break size for which no clad heat up would occur with the maximum safety injection flow capability.

In the 6-inch cold leg break case, the cladding temperature excursion was terminated by accumulator injection flow. Decreasing the safety injection flow decreased the RCS minimum liquid volume. The 4-inch cold leg break showed a higher minimum liquid volume than the 6-inch break case, even for the reduction in safety injection flow. The RCS liquid volume depletion transient was also reversed by accumulator injection flow. The 3-inch cold leg break case, which was also reversed by accumulator injection flow, showed a minimum liquid volume

which was substantially higher than the 4-inch break case and much higher than the 6-inch break case. The 2-inch cold leg break liquid inventory depletion transients reversed prior to accumulator injection and showed minimum liquid volumes very much higher than the 3-inch break case, the 4-inch break case, or the 6-inch break case. The 1-inch cold leg break cases showed very little liquid volume depletion. The minimum liquid volume in the 1-inch, 2-inch and 3-inch breaks would not have resulted in uncovery of the core even without the frothing of the mixture due to decay heat.

WCAP-8340

WCAP-8340 (Reference 2) presented the results of loss-of-coolant accident (LOCA) sensitivity analyses for typical Westinghouse designed two-loop, three-loop, and four loop plants. The report presented the results of small break LOCA analyses for 2-inch, 3-inch, 4-inch, 6-inch, and 8-inch equivalent diameter breaks, along with a 0.5-square foot, and a 1.0-square foot break in the cold leg, and an 8-inch equivalent diameter break in the hot leg for the four-loop plant analyses.

The four loop plant design analyzed in WCAP-8340 is the essentially the same as the D.C. Cook Unit 1 design with the exception of a slight variation in vessel design. The vessel analyzed in WCAP-8340 has neutron pads to provide protection from neutron flux while Cook's vessel design incorporates a thermal shield to provide this function. In addition, the WCAP-8340 plant has a flat lower support plate where Cook's lower support plate is curved. The change in plant response to a small break LOCA afforded to these differences is very small.

The four loop reference plant safety injection flow to core power ratio determined at the steam generator safety valve setpoint pressure is .016 lbm/sec/MWt. WCAP-8340 (Reference 2) found that the 8-inch cold leg break resulted in the highest calculated peak cladding temperature. The calculated peak cladding temperature for the 2-inch and 3-inch cold leg breaks were substantially lower. Increases in the cladding temperature transient due to core uncovery were reversed by accumulator injection for the 4-inch, 6-inch, 8-inch, 0.5-square foot, and 1.0-square foot equivalent diameter cold leg breaks while the temperature transients for the 2-inch and 3-inch equivalent diameter cold leg breaks were reversed prior to accumulator injection.

WCAP-9600

Following the accident at the Three Mile Island nuclear power plant, WCAP-9600 (reference 3) was compiled to provide a description of analytic methods and expected system behavior for

a range or postulated small break LOCAs in the Westinghouse designed nuclear steam supply system. The report examined the complete range of break sizes which would exhibit all of the expected transient characteristics in terms of core heat removal mechanisms, RCS mixture level transients, core uncovery tendencies, and long term recovery methods.

As noted in WCAP-9600 (reference 3), the RCS response to a small break LOCA will vary with the high pressure safety injection delivery characteristics. For breaks less than 3/8-inch equivalent diameter, the normal charging system will maintain system liquid inventory. For breaks greater than 3/8-inch equivalent diameter, but less than approximately 1-inch equivalent diameter, the RCS pressure will stabilize above the steam generator safety valve setpoint with minimum safeguards safety injection flow available. For breaks greater than 3/8-inch equivalent diameter, but less than approximately 1-inch equivalent diameter, RCS repressurization will occur when full safeguards safety injection flow is available. For breaks greater than 1-inch equivalent diameter, but less than approximately 2-inch equivalent diameter, the RCS pressure will stabilize below the steam generator safety valve setpoint, but above the accumulator injection setpoint. For breaks greater than 2-inch equivalent diameter, RCS depressurization below the accumulator injection setpoint occurs.

WCAP-9600 (reference 3) showed that with the possible exception of the larger small break sizes, the range of break sizes resulting in the most severe core uncovery is determined by the high pressure safety injection system characteristics, and flow assumptions because the RCS pressure tends to reach equilibrium pressure where the pumped safety injection flow equals the subcooled or saturated liquid break flow, when the steam generator serves as a heat sink.

Although the break size limits and localized phenomena may vary somewhat from plant to plant, the characteristic transient behavior presented in WCAP-9600 (reference 3) is generally applicable to the response of all plants.

For breaks greater than 1-inch equivalent diameter, the RCS depressurizes generating an automatic reactor trip and safety injection signal on low pressurizer pressure. If the break is incapable of removing all of the decay heat, the RCS pressure will temporarily reach equilibrium above the steam generator safety valve setpoints, assuming no steam dump capability is available, to provide a primary to secondary decay heat removal mechanism. As the RCS liquid inventory is depleted, voids are formed and the RCS begins to drain down. The rate of system drain is determined by the net loss of liquid inventory.

Depending upon the location of the break in the system, WCAP-9600 (reference 3) showed that draining of the RCS may

partially uncover the reactor core. For breaks in the hot leg, decay heat produced steam may exit through the break resulting in continued depressurization until a volumetric balance is achieved. No core uncovery is predicted for breaks in the hot leg. For breaks in the cold leg (limiting location), the core may partially uncover in order to create a vent path for steam to exit through the break as liquid is cleared from the pump suction leg loop seal. The core uncovery resulting from the venting of core decay heat produced steam through the loop seal does not typically result in the limiting PCT because as the loop seal clears, the core will undergo a rapid recovery. Following the loop seal core uncovery and subsequent core recovery, the decay heat induced boiling of the vessel liquid inventory may result in another core uncovery. WCAP-9600 (reference 3) showed that the potential for uncovering the core after clearing the loop seal is determined by the break size, the decay heat boiloff rate, and the safety injection mass flow rate. The break size governs the rate of primary mass inventory depletion in conjunction with the safety injection mass flow rate. The decay heat boiloff rate governs the depressurization rate in conjunction with the break flow. As soon as the break flow becomes predominantly steam, continued depressurization occurs. As the RCS depressurizes, the safety injection flow rate increases. When safety injection flow exceeds the break mass flow rate, net liquid inventory loss is reversed.

The fundamental response to a small break LOCA in a Westinghouse nuclear steam supply system was discussed in WCAP-9600 (reference 3) for a typical four-loop plant similar in design to D.C.Cook Unit 1. The transient response predicted for the four loop plant design analyzed illustrates the same behavior seen in the previous four loop plant analyses discussed.

In addition, WCAP-9600 (reference 3) discussed the behavior of the RCS response for breaks in various locations with varying amounts of safety injection flow in a three-loop plant for breaks ranging in size from 2-inch to 6-inch equivalent diameter. WCAP-9600 (reference 3) found that breaks greater than 2-inch equivalent diameter result in the limiting peak cladding temperature. In these studies, the ratio of safety injection mass flow rate at the steam generator safety valve setpoint to core power was varied from 0.011 lbm/sec/MWth to 0.023 lbm/sec/MWth. WCAP-9600 (reference 3) also found that a reduction in safety injection flow to core power ratio in cold leg breaks increased the potential for core uncovery after a steam vent path had been cleared through the loop seal. In these studies, the 3-inch equivalent diameter cold leg break with minimum safeguards was found to be limiting with a peak clad temperature of 1708°F, which is typical of three-loop plants. The 2-inch equivalent diameter cold leg break with minimum safeguards resulted in a peak clad temperature of 1003°F. In these studies, a significant reduction (approximately 50%) in the safety injection mass flow rate resulted in a peak cladding temperature of 2169°F for the 3-inch

break, while the peak cladding temperature for the 2-inch break increased only to 1088°F. Even though the safety injection flows were reduced significantly, the 2-inch break was small enough to not uncover the core significantly.

CURRENT EVALUATION MODEL STUDIES - WCAP-11145

Following the incident at Three Mile Island Unit 2, Westinghouse and the Westinghouse Owners Group developed a more advanced small break LOCA analysis model (references 4 and 5), which was approved in May 1985. Small break analysis studies were performed with the new evaluation model in reference 6 for 2-inch, 3-inch, 4-inch, 5-inch, and 6-inch equivalent diameter cold leg breaks for a typical Westinghouse designed 3411 MWth four-loop plant with a ratio of safety injection mass flow rate at the steam generator safety valve setpoint to core power of 0.013 lbm/sec/MWth. In those analyses, relatively low peak cladding temperatures were calculated even with conservative Appendix K to 10 CFR 50.46 assumptions. The 3-inch break with a peak cladding temperature of 1342°F was slightly more limiting than the 4-inch break with 1287°F or the 5-inch break with 1249°F. The analysis of the 2-inch break did not result in core uncovery.

In NUREG-0737 (reference 7), section II.K.3.31, the NRC required reanalysis with the approved model. In generic letter 83-35, the NRC relaxed the requirement for reanalysis, provided that the previous evaluation model (WFLASH) could be shown to be conservative. Analyses were performed and reported in WCAP-11145 (reference 6) to demonstrate, in general, that the NOTRUMP evaluation model calculates lower peak cladding temperatures than did the previous WFLASH evaluation model. A typical four-loop plant similar in design to D.C.Cook unit 1 with a ratio of safety injection mass flow rate at the steam generator safety valve setpoint to core power of 0.014 lbm/sec/MWth was analyzed in WCAP-11145 (reference 6) for 2-inch, 3-inch, 4-inch, and 6-inch cold leg breaks. A 4-inch equivalent diameter break was also analyzed for a rupture in the pump suction leg and hot leg. The 4-inch cold leg break was limiting with a peak cladding temperature of 1253°F. The cladding temperature rise during the uncovery of the core was reversed by accumulator injection flow. The 3-inch break calculated a peak cladding temperature of 1154°F which did not require accumulator injection to reverse the temperature rise. Again in these analyses the 2-inch break did not result in uncovery of the core and stabilized above the accumulator injection set pressure.

SMALL BREAK LOCA LIMITING BREAK SIZE EVALUATION

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, then, is dependent upon the ratio of the safety injection mass flow rate to core power. The historical perspective establishes the effect of large variations in the safety injection mass flow rate to core power ratios on the system inventory, vessel mixture level, and clad heat up. For Safety Injection Flow to Core Power ratios ranging from 0.011 lbm/sec/MWt to 0.023 lbm/sec/MWt the RCS response to a range of break sizes was provided.

D.C. Cook Unit 1 could have a safety injection flow to core power ratio as low as 0.0105 lbm/sec/MWt when the HHSI Cross-Tie is closed and one HHSI pump fails to deliver flow. This represents an SI Flow to Power ratio slightly lower than the 0.011 to 0.023 lbm/sec/MWt range examined in WCAP-9600 (reference 3). The studies performed in WCAP-9600 showed that with substantial reductions in safety injection flow, the smaller breaks did not result in the most limiting peak clad temperatures. For the variations of safety injection flow examined in WCAP-9600, the core response to the spectrum of break sizes analyzed exhibited the same behavior trends. Therefore, the magnitude of the effect of reducing the safety injection flow rates can be interpreted from the results and applied for small additional reductions in safety injection flow. Consequently, the results established by Westinghouse for SI flow to core power ratios ranging from 0.011 lbm/sec/MWth to 0.023 lbm/sec/MWth in WCAP-9600 (reference 3) can be shown to be representative of D.C. Cook Unit 1 for the safety injection flows available with the HHSI Cross-Tie is closed.

WCAP-7422 (reference 1) found that breaks up to 1/2-inch in diameter are not a concern since the charging pumps could maintain an operational level of water in the pressurizer, permitting the operator to execute an orderly shutdown. Those findings remain valid for D.C. Cook Unit 1 with the HHSI cross-tie line closed. Breaks greater than 1/2-inch in diameter were investigated for a four loop plant identical in vessel design to D.C. Cook Unit 1. The results reported in WCAP-7422 show that when the safety injection mass flow rate to core power ratio is lowered, the minimum liquid volume response did not result in a more limiting response for the 1-inch and 2-inch cases. The results indicate that the SI flow reduction did not adversely affect the ability to recover from the smaller diameter breaks. These results have been examined and indicate that in the case of D.C. Cook Unit 1, the mass inventory depletion resulting from breaks smaller than 3-inches in diameter would not become limiting when the HHSI Cross-Tie is closed.

The effect of significant reductions in safety injection flow were analyzed in WCAP-9600 (reference 3). These studies also concluded that even with significant reductions in safety injection flow, breaks greater than 2-inches in diameter result

in the limiting peak clad temperature. The break spectrum, ranging from 2-inch to 6-inch diameter breaks, was analyzed assuming reductions in SI flow of almost 50%. The results indicated that the limiting 3-inch diameter break remained limiting with the SI flow reduction with a 461^{o}F increase in PCT. The 2-inch diameter break PCT increased only 85^{o}F with the SI flow reduction. These results also indicate that the SI flow reduction resulting from closure of the Cook Unit 1 HHSI cross-tie would not cause the limiting break size to shift.

As indicated previously, Westinghouse performed a 4-inch equivalent diameter cold leg small break LOCA ECCS analysis on a reference plant similar in design to D.C. Cook Unit 1 with the safety injection flow representative of D.C. Cook Unit 1 with the HHSI cross-tie closed. To determine the expected effect of the D.C. Cook Unit 1 SI flow reduction on the 3-inch diameter break, an evaluation was performed. The pressure response of the reference plant licensing basis 3-inch diameter break LOCA transient was used to determine the integrated safety injection flow which would be available in the event of a 3-inch LOCA for D.C. Cook Unit 1 when the cross-tie is closed. This assumed that the depressurization transient is relatively unaffected by small differences in the safety injection flow. Comparison of the reference plant licensing basis 4-inch diameter depressurization transient to the reference plant 4-inch break depressurization transient which assumed the Cook Unit 1 Cross-Tie closure SI flows confirmed that the depressurization transient is essentially unaffected by the SI flow reduction. An existing sensitivity study of integrated SI flow to core power ratio specific to the reference plant was then used to conservatively estimate the peak cladding temperature that would result for a 3-inch break in D.C. Cook Unit 1 with the cross-tie closed. The results indicated that the peak cladding temperature for the 3-inch equivalent diameter break would indeed increase, but with a significantly smaller magnitude than the 4-inch diameter break. These results confirmed that the 4-inch diameter cold leg break would remain more limiting than the 3-inch diameter break for D.C. Cook Unit 1 with the safety injection flow reduction resulting from closure of the HHSI Cross-Tie line. American Electric Power submitted the results of the 3-inch break evaluation to the NRC staff in AEP-NRC-1024A dated May 13th 1987.

In both the 3-inch and 4-inch break transients for the reference plant with the HHSI cross-tie open, the RCS depressurized to the accumulator injection setpoint. Immediately following accumulator injection, the core mixture level increased and the cladding temperature excursion was reversed. As stated previously, in the event of a small break LOCA, the RCS pressure tends to reach equilibrium where pumped safety injection flow equals the break flow. Figure 1 shows a conservative estimate of the RCS pressure equilibration point

for D.C. Cook Unit 1 in the event of a 4-inch, 3-inch and 2-inch diameter break for both the current licensing basis safety injection flow and the safety injection flow available when the HHSI cross-tie is closed. The break flow rates are conservatively estimated for saturated steam. The 2-inch diameter break is expected to equilibrate at or above the D.C. Cook Unit 1 minimum accumulator setpoint pressure of 600 psia when the HHSI cross-tie is closed. Therefore, recovery from RCS inventory depletion transient for the 2-inch diameter break should be dependent upon safety injection flow delivery alone.

To determine if the safety injection flow available with the cross-tie closed is capable of mitigating the consequences of a 2-inch break at D.C. Cook Unit 1, a system mass inventory calculation was performed. The calculation assumed that the depressurization transient and the break flow rates calculated for the 2-inch break in a typical Westinghouse 4-loop plant represented in WCAP-11145 (reference 6) would be representative of the response in D.C. Cook Unit 1 and could be used as boundary conditions. The D.C. Cook Unit 1 safety injection flow rates available with the HHSI cross-tie closed were used to calculate the integrated safety injection flow rate. The integrated safety injection flow was combined with the integrated break flow rate and the initial RCS mass inventory to determine a transient RCS mass inventory. The transient mass inventory was used with the existing reference plant analysis. A comparison of the mass inventory lost out the break to the total safety injection flow delivered determined that the net RCS mass inventory maintained throughout the 2-inch break transient is nearly equal to the initial reactor vessel mass inventory. Therefore, little if any core uncovery is expected to result in the event of a 2-inch break for D.C. Cook Unit 1 even with the reduced safety injection flow. As a result, the 2-inch diameter break PCT will remain less limiting than the 3-inch and 4-inch break PCTs determined for Cook Unit 1 with the reduced safety injection flow.

Conclusions:

An investigation of the ability of the D.C. Cook Unit 1 ECCS system to mitigate the consequences of small RCS piping ruptures was performed to determine if LOCAs which do not result in accumulator injection will result in the most limiting PCTs due to the Safety injection flow reduction resulting from closure of the HHSI Cross-Tie line. The evaluation has shown that smaller breaks that do not result in depressurization to the accumulator setpoint will not result in the most limiting Peak Clad Temperature. The 4-inch diameter cold leg break will remain the limiting break size with the HHSI Cross-Tie valves closed.

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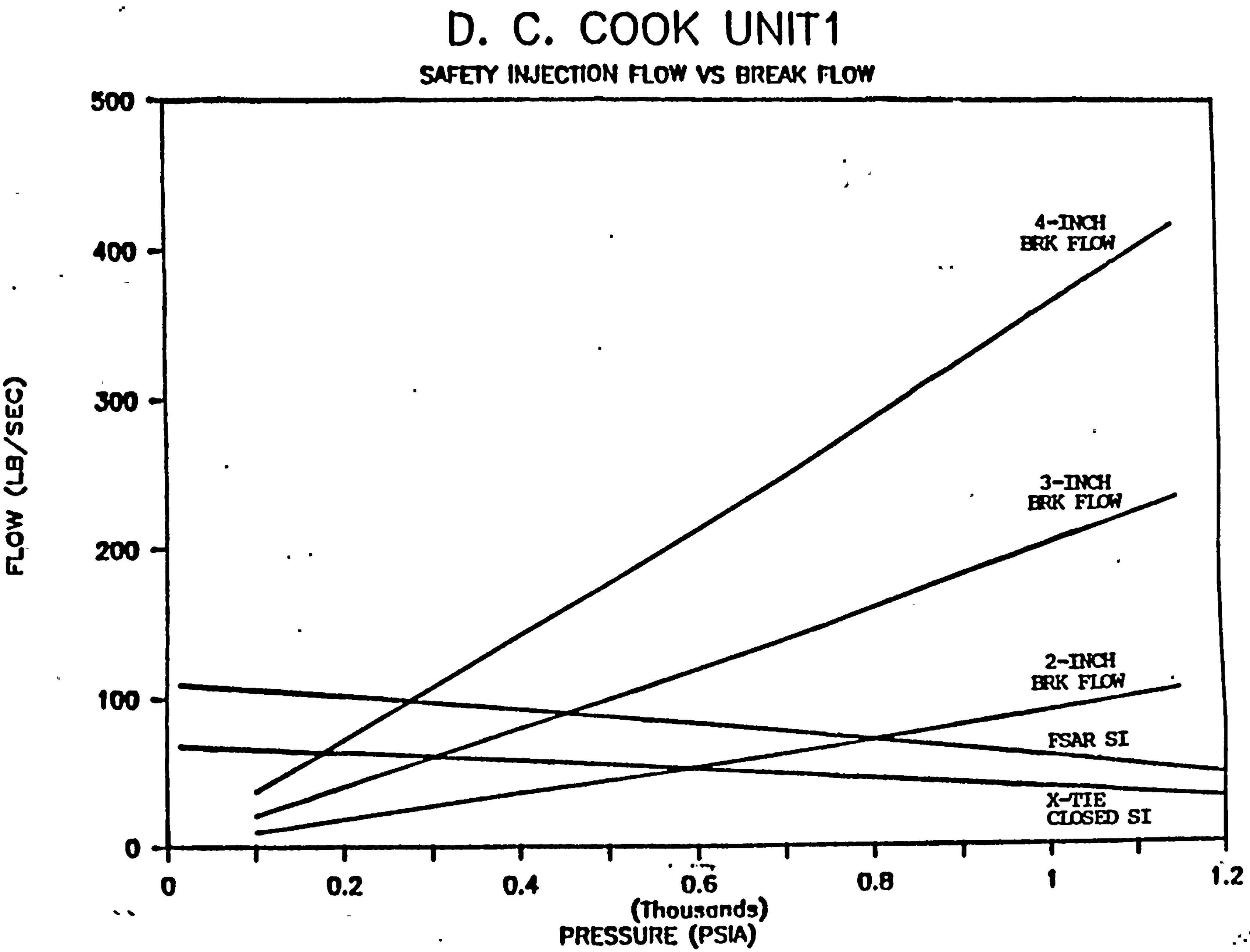
FIGURE 1


D. C. COOK UNIT1
SAFETY INJECTION FLOW VS BREAK FLOW
500
400
300
200
100
0
FLOW (LB/SEC)
4-INCH
BRK FLOW
3-INCH
BRK FLOW
2-INCH
BRK FLOW
FSAR SI
X-TIE
CLOSED SI
0
0.2
0.4
0.6
0.8
1
1.2
(Thousands)
PRESSURE (PSIA)