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AMERICAN ELECTRIC POWER

AEP:NRC:1024E

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 FOR UNIT 2

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

Attn: T. E. Murley

February 29, 1988

Dear Dr. Murley:

This letter is a follow up to our letter AEP:NRC:1024C, which was submitted on October 13, 1987. In that letter we provided a report, specific to Unit 1 of the Cook Nuclear Plant, which responded to questions raised by your staff during a telephone conversation on July 29, 1987. The questions involved potential shifts in the limiting size small-break LOCA due to closure of the safety injection system cross-tie valves. This letter transmits a similar response for Unit 2 of the Cook Nuclear Plant. Attachment 1 to this letter provides background information and a summary of the evaluations performed for Unit 2 in response to your staff's questions. Attachment 2 provides the response, which was prepared for us by Westinghouse Electric Corporation.

Attachment 3 contains two replacement pages for the Unit 1 report transmitted via AEP:NRC:1024C. The replacement pages contain corrections of minor errors which were detected in the Unit 1 report while we were reviewing the Unit 2 report. The changes are described in Attachment 3.

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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

dat

Attachments

cc: D. H. Williams, Jr.
W. G. Smith, Jr. - Bridgman
R. C. Callen
G. Bruchmann
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NRC Resident Inspector
A. B. Davis - Region III

ATTACHMENT 1 TO AEP:NRC:1024E

BACKGROUND INFORMATION ON CROSS-TIE ISSUES AND SUMMARY OF WESTINGHOUSE ELECTRIC CORPORATION EVALUATION OF SMALLER BREAK SIZES

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 redundant subsystem. The RHR and safety injection pump configuration at the Donald C. Cook Nuclear 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 two reactor coolant loops. The current small-break and large-break LOCA analyses for Cook Nuclear Plant 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 in which the Cook Nuclear Plant Units 1 and 2 were operated in Modes 1, 2 and 3 with the cross-tie valves closed. The valves were closed to allow maintenance and testing of 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 and 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 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 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. The investigation was performed to ensure that these smaller breaks will not become limiting with regard to peak clad temperature. The NRC staff was

concerned with smaller breaks, which might not depressurize the reactor coolant system below the accumulator discharge setpoint, when the SI cross-tie valves are closed. A response specific to Unit 1 of the Cook Nuclear Plant was prepared for us by Westinghouse, and submitted to you in our letter AEP:NRC:1024C dated October 13, 1987. A response specific to Unit 2 of the Cook Nuclear Plant is contained in Attachment 2 to this letter.

Since the July 29, 1987 discussion with your staff, we were informed by Westinghouse that the large-break LOCA evaluations that they performed to support closure of the RHR cross-tie valves may be inadequate in that the effect that closing the cross-tie valves has on containment long-term calculated pressure was not included in the review. As discussed previously with your staff, revised containment analyses will be performed by Westinghouse by June 1988 and will be transmitted to the NRC after undergoing appropriate internal review.

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 Cook Nuclear Plant Unit 2 SI cross-tie valves.

To determine the Cook Nuclear Plant Unit 2 core response to breaks smaller than 3 inches in diameter with the SI cross-tie valves closed, an examination was made of the small break LOCA transient response exhibited in the Westinghouse-designed PWRs for variations in break size, power level and safety injection flow rates. Studies performed from 1974 to the present indicated that even with substantial reductions in safety injection flow and no accumulator discharge, the smaller break LOCAs 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 Cook Nuclear Plant Unit 2 with the reduced SI flow, Westinghouse performed an evaluation of a 2-inch small-break LOCA. The results of the evaluation were that the peak clad temperature would be expected to be lower for the 2-inch break than for the larger breaks (3- and 4-inch) which were previously evaluated.

ATTACHMENT 2 TO AEP:NRC:1024E

WESTINGHOUSE ELECTRIC CORPORATION EVALUATION OF

SMALL BREAK SIZES FOR COOK NUCLEAR PLANT UNIT 2

LIMITING SMALL BREAK LOCA CONSIDERATIONS

FOR OPERATION OF DONALD C. COOK UNIT 2 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 Donald C. Cook Unit 2 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 Donald C. Cook Unit 2 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 calculated for the 4-inch case was determined to be greater than the PCT 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 Donald C. Cook Unit 2 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 Donald C. Cook Unit 2 cross-tie line.

To determine the Donald C. Cook Unit 2 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 the trend predicted by the historical data, i.e. that the smaller breaks would not become limiting for Donald C. Cook Unit 2 with the reduced SI flow, an evaluation of the system mass inventory transient with the reduced safety injection flow was performed. The calculation required the determination of the RCS mass inventory depletion vs. time resulting from a two-inch break with the reduced safety injection flows. The depressurization transient for a representative four loop plant 2-inch break was used in conjunction with the safety injection mass flow rates for Cook Unit 2 when the HHSI cross-tie is assumed closed to determine the safety injection flow delivered throughout the 2-inch break LOCA transient. The system inventory transient was then used to determine the vessel mixture level throughout the duration of the LOCA transient to predict the depth and duration of core uncovery in the event of a 2-inch LOCA with the reduced safety injection flow. The core mixture level transient was then evaluated to determine the clad heat up expected to result from the core uncovery. Fuel rod clad heat up calculations performed for the two-inch diameter break, indicate that the Peak Clad Temperature is expected to lie in the range of 1060°F to 1200°F. The two-inch diameter break PCT result expected for a Cook Unit 2 specific LOCA analysis is lower than both the three-inch and four-inch break PCTs predicted in previous evaluations. Therefore, the four-inch diameter break LOCA is expected to remain the limiting break for Donald C. Cook Unit 2 with the safety injection flow reduction resulting from the HHSI cross-tie closure.

LIMITING SMALL BREAK LOCA CONSIDERATIONS

FOR OPERATION OF DONALD C. COOK UNIT 2 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 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 in the Donald C. Cook Unit 2 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 Donald 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). 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 Donald 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.

The small break LOCA analysis was performed for the reference plant applying the limiting four-inch equivalent diameter cold leg break for the Donald C. Cook Unit 2 licensing basis WFLASH

analysis. The sensitivity of the Cook 2 reference plant to large reductions in safety injection flow was then applied to determine an estimated PCT for a Cook Unit 2 4-inch break with the HHSI cross-tie closed. The evaluation of the 4-inch diameter break for Cook 2 with the HHSI cross-tie closed resulted in a PCT of 1482°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 evaluation 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 Donald C. Cook Unit 2 ECCS performance with the HHSI cross-tie closed for small break LOCA events smaller than three inches in diameter.

In determining the Donald C. Cook Unit 2 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 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 Donald C. Cook Unit 2 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 Donald C. Cook Unit 2 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 Donald C. Cook Unit 2 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 Donald C. Cook Unit 2. These comparisons were drawn to demonstrate the low probability of a shift in the Donald C. Cook Unit 2 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 Donald C. Cook Unit 2 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 four loop plant. The depressurization transient was then used in conjunction with the safety injection mass flow rates for Cook Unit 2 when the HHSI cross-tie is assumed closed to determine the safety injection flow delivered throughout the 2-inch break LOCA transient. The results of the mass inventory calculation indicated that some core uncovery would result in the event of a 2-inch diameter break LOCA. However, the evaluation of the depth and duration of the core uncovery indicates that the resultant clad heat up is not expected to be significant enough to result in a clad temperature greater than that determined for the four or the three-inch diameter breaks.

Limiting Small Break LOCAs: Historical Perspective

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 smaller break sizes can become limiting when less safety injection flow is available. However, the evaluation performed for Donald C. Cook Unit 2 indicates that the safety injection flow reduction resulting from closure of the HHSI cross-tie is not significant enough to cause a shift in the limiting break size. Comparison of the Cook 2 cross-tie closed case to the cases reviewed here, on the comparison of lb-mass/sec of safety injection flow per MWth of core power, and examining the limiting break sizes from the cases reviewed herein, leads to the conclusion that some core uncovery would result in the event of a two-inch break at Cook 2 with the cross-tie closed SI flows, however the resultant clad heat up would not be significant enough to result in the limiting peak clad temperature.

<u>WCAP-7422</u>

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 virtually identical to Donald C. Cook Unit 2 and was performed 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 flow rate 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 the 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 minimum 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 essentially the same as the Donald C. Cook Unit 2 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.

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 of 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 2. 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 5 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 2 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.

Donald C. Cook Unit 2 could have a safety injection flow to core power ratio as low as 0.0100 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 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 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 Donald C. Cook Unit 2 for the safety injection flows available when 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 Donald C. Cook Unit 2 with the HHSI cross-tie line closed. Breaks greater than 1/2-inch in diameter were investigated for a four loop plant virtually identical in vessel design to Donald C. Cook Unit 2. 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 Donald C. Cook Unit 2, the mass inventory depletion resulting from a break smaller than 3-inches in diameter is not expected to result in the most severe core uncovery 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 2 HHSI cross-tie would not cause the limiting break size to shift.

As indicated previously, 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 Donald 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. To determine the expected effect of the Donald C. Cook Unit 2 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 Donald C. Cook Unit 2 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 a significant reduction in safety injection flow 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 Donald C. Cook Unit 2 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 Donald C. Cook Unit 2 with the safety injection flow reduction resulting from closure of the HHSI cross-tie line. American Electric Power submitted the results of a similar 3-inch break evaluation performed for Donald C. Cook Unit 1 to the NRC staff in AEP-NRC-1024A dated May 13th 1987. The response provided in attachment 5 to AEP-NRC-1024A has been demonstrated to be bounding for both Donald C. Cook Unit 1 and Unit 2.

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 Donald C. Cook Unit 2 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 Donald C. Cook Unit 2 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 Donald C. Cook Unit 2, 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 Donald C. Cook Unit 2 and could be used as boundary conditions. The Donald C. Cook Unit 2 safety injection flow rates available with the HHSI cross-tie closed were applied to the rate of depressurization as shown in WCAP-11145 to determine the safety injection flow available throughout the transient. The safety injection flow in conjunction with the break flow could then be combined to determine the RCS mass inventory depletion vs. time resulting from a two-inch diameter break with the reduced safety injection flows. The transient mass inventory was then used to determine the vessel mixture level during the 2-inch LOCA event to predict the depth and duration of core uncovery.

The core uncovery transient predicted to result in the event of a 2-inch diameter LOCA with the reduced safety injection resulting from closure of the cross-tie line indicates that a relatively long shallow core uncovery will result. The maximum depth of core exposure is not expected to exceed 1.7 ft. The total duration of uncovery was estimated to be 1575 seconds with the deepest uncovery occuring between 4250 and 4500 seconds following break initiation. Based on a comparison of various uncovery transient core steam flow rates, core decay heat levels and clad heat up rates, an estimate of the 2-inch PCT temperature range was made for the uncovery transient predicted for the 2-inch break with the reduced safety injection flow. Assuming the licensed core power level of 3411 MWth, a total core peaking factor of 2.32 and Westinghouse standard 17x17 fuel assemblies, the 2-inch break with the reduced safety injection flows is expected to result in a peak clad temperature between 1060^{o}F and 1200^{o}F. As a result, the 2-inch diameter break PCT is expected to remain less limiting than the 3-inch and 4-inch break PCTs determined for Cook Unit 2 with the reduced safety injection flow.

Conclusions:

An investigation of the ability of the Donald C. Cook Unit 2 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 are not expected to 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.

REFERENCES

1. WCAP-7422-L (Limited Distribution), PWR Systems Division, Core Engineering, "Topical Report Westinghouse PWR Core Behavior Following A Loss Of Coolant Accident", January 1970.

2. WCAP-8340, PWR Systems Division, Nuclear Safety, "Westinghouse Emergency Core Cooling System - Plant Sensitivity Studies", July 1974.

3. WCAP-9600, Nuclear Technology Division, Nuclear Safety, "Report On Small Break Accidents For Westinghouse NSSS System", June 1979.

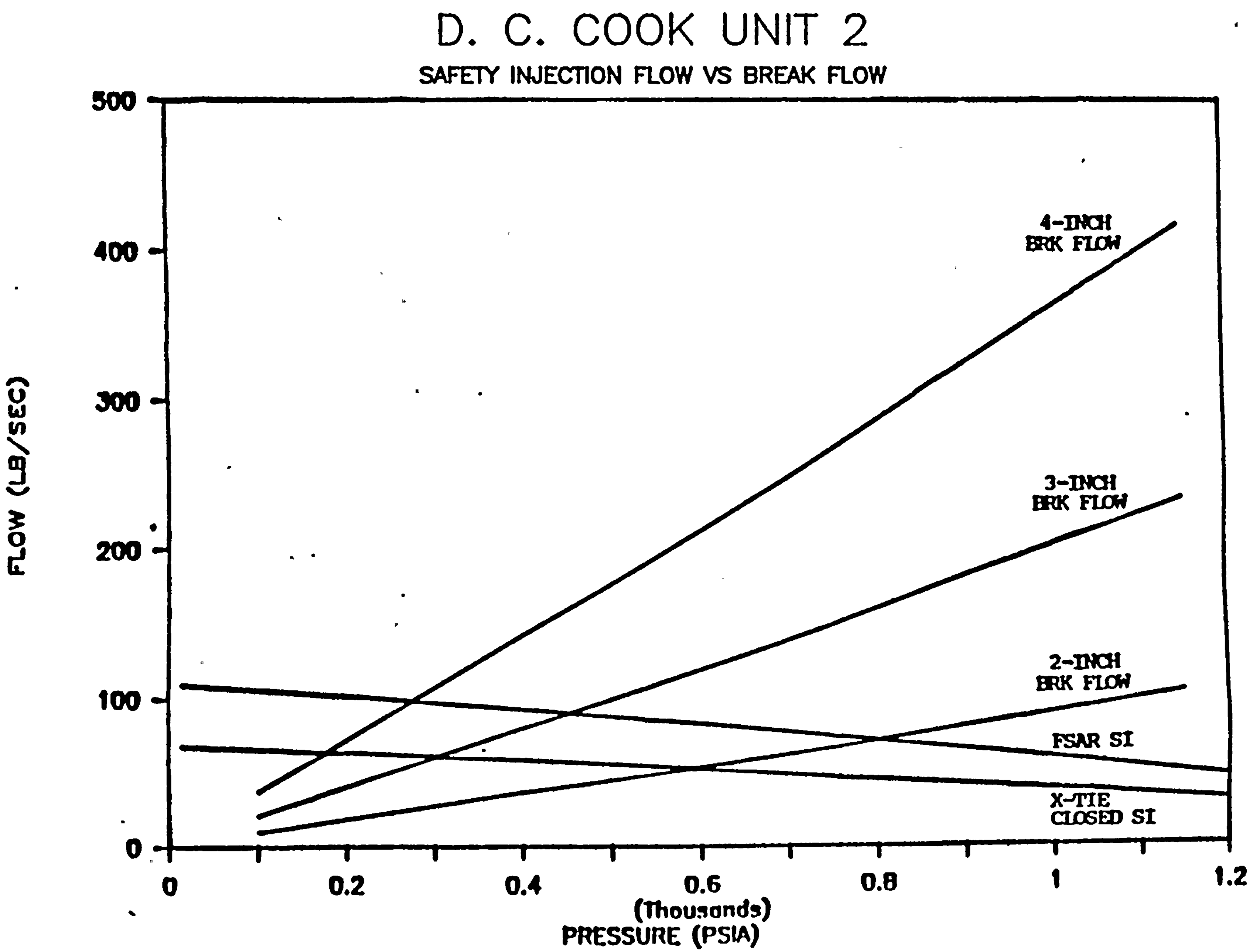
4. WCAP-10079-P-A, (Proprietary), Meyer, P.E., "NOTRUMP: A Nodal Transient Small Break General Network Code", August 1985.

5. WCAP-10054-P-A, (Proprietary), Lee, N., Tauche, W.D., et al., "Westinghouse Small Break ECCS Evaluation Model Using The NOTRUMP Code", August 1985.

6. WCAP-11145-P-A, (Proprietary), Rupprecht, S.D., Osterrieder, R.A., et al., "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study With The NOTRUMP Code", October 1986.

7. NUREG-0737, United States Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements", November 1980.

FIGURE 1


D. C. COOK UNIT 2
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)

ATTACHMENT 3 TO AEP:NRC:1024E

CORRECTED PAGES FOR UNIT 1 REPORT
TRANSMITTED IN LETTER NO. AEP:NRC:1024C

This attachment contains replacements for pages 8 and 12 of the Westinghouse Electric Corp. small break LOCA limiting break size evaluation for Unit 1 of the Cook Nuclear Plant. The report was originally transmitted via our letter AEP:NRC:1024C, on October 13, 1987. The replacement pages contain corrections of minor errors in the text.

On page 8, the word "minimum" replaced "maximum" on the last line of the third full paragraph. On page 12, "reference 6" was changed to "reference 5" in the fifth line of the first full paragraph. The changes are indicated by bars in the right hand margin.

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 minimum 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

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 5 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.

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