ATTACHMENT 1

PROPOSED CHANGES
TO THE
FACILITY OPERATING LICENSE
AND THE
TECHNICAL SPECIFICATIONS
FOR
NORTH ANNA UNIT 1

VIRGINIA ELECTRIC AND POWER COMPANY

(1) Maximum Power Level

VEPCO is authorized to operate the North Anna Power Station, Unit No. 1, at reactor core power levels not in excess of 2893 megawatts (thermal). *

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. [__] are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Additional Conditions

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following the issuance of this amendment or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the license supported by a favorable evaluation by the Commission:

c. Virginia Electric and Power Company shall not operate the reactor in operational modes 1 and 2 with less that three reactor coolant pumps in operation.

d. VEPCO may use two (2) fuel assemblies containing fuel rods clad with an advanced zirconium base alloy cladding material as described in the licensee's submittals dated February 20, 1987 and September 30, 1988.

e. If Virginia Electric and Power Company plans to remove or to make significant changes in normal operation of equipment that controls the amount of radioactivity in effluents from the North Anna Station, the Commission shall be notified in writing regardless of whether the change affects the amount of radioactivity in the effluents.

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* The maximum reactor power level shall be limited to 2748 megawatts (thermal) which is 95% of RATED THERMAL POWER in accordance with the licensee's submittal dated January 28, 1992 (Serial No. 92-042) for the period of operation until the steam generator replacement.

ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE low head safety injection pump,

c. An OPERABLE flow path capable of transferring fluid to the Reactor Coolant System when taking suction from the refueling water storage tank on a safety injection signal or from the containment sump when suction is transferred during the recirculation phase of operation or from the discharge of the outside recirculation spray pump.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours. *

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

c. The provisions of Specifications 3.0.4 are not applicable to 3.5.2.a and 3.5.2.b for one hour following heatup above 324°F or prior to cooldown below 324°F.

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* Adherence to ACTION "a" shall require the following equipment OPERABILITY for the period of operation until steam generator replacement:

- With one low head safety injection pump inoperable, two centrifugal charging pumps (one in each subsystem) and their associated flow paths shall be OPERABLE or be in HOT STANDBY within the next 6 hours, and be in HOT SHUTDOWN within the next 6 hours.

ATTACHMENT 2

DISCUSSION OF PROPOSED CHANGE

VIRGINIA ELECTRIC AND POWER COMPANY

Discussion of Proposed Change

North Anna Power Station Unit 1 is currently involved in a mid-cycle steam generator inspection outage. An extensive eddy current inspection of the North Anna Unit 1 steam generator tubes is being performed using very conservative analysis guidelines and plugging criteria. As such, a substantially increased number of tubes are expected to be plugged.

The predictions of steam generator tube plugging required during this mid-cycle outage are such that the effects of increased RCS loop resistance on the large break Loss of Coolant Accident (LOCA) analysis would not permit full rated power operation for the remainder of Cycle 9 operation for North Anna Unit 1. Therefore, safety analyses and evaluations have been performed which support continued operation with an imposed reactor power restriction. The attached safety evaluation has been prepared to discuss the changes to the large break LOCA analysis and support this license amendment for the associated restriction in reactor power.

The proposed license change will limit maximum reactor power to 95% of RATED THERMAL POWER. Specifically, we request a change to Facility Operating License No. NPF-4 to the Virginia Electric and Power Company for North Anna Power Station Unit 1 to modify license condition 2.D.(1), Maximum Power Level, by adding a footnote which states that:

> * The maximum reactor core power level shall be limited to 2748 megawatts (thermal) which is 95% of RATED THERMAL POWER in accordance with the licensee's submittal dated January 28, 1992 (Serial No. 92-042) for the period of operation until the steam generator replacement.

In addition, an associated change to the Technical Specifications is required to accommodate the effects of the revised assumptions for the large break LOCA analysis. The proposed change to the Technical Specifications will impose more restrictive equipment operability requirements for the Emergency Core Cooling System (ECCS). Specifically, we request a change to Action Statement "a" of Specification 3.5.2, ECCS Subsystems - Tavg ≥350°F, by adding a footnote which states that:

> * Adherence to ACTION "a" shall require the following equipment OPERABILITY for the period of operation until steam generator replacement:
>
> - With one low head safety injection pump inoperable, two centrifugal charging pumps (one in each subsystem) and their associated flow paths shall be OPERABLE or be in HOT STANDBY within the next 6 hours, and be in HOT SHUTDOWN within the next 6 hours.

In effect, this proposed change will ensure that both low head safety injection pumps or one low head injection pump and two high head safety injection pumps remain operable during power operation. This change effectively maintains consistency between the Technical Specification Action Statements and the revised assumptions for the large break LOCA analysis.

The proposed changes are necessary to accommodate the expected increased steam generator tube plugging levels. The attached safety evaluation supports the above changes to the operating license and the Technical Specifications. The changes are required on an interim basis until the steam generator replacement in 1993, at which time it will no longer apply.

ATTACHMENT 3

SAFETY EVALUATION

VIRGINIA ELECTRIC AND POWER COMPANY

LARGE BREAK LOSS-OF-COOLANT ACCIDENT (UFSAR Section 15.4.1)

1.0 INTRODUCTION

North Anna Power Station Unit 1 is currently involved in a mid-cycle steam generator inspection outage. An extensive eddy current inspection of the North Anna Unit 1 steam generator tubes is being performed using very conservative analysis guidelines and plugging criteria. As such, a substantially increased number of tubes are expected to be plugged.

The physical consequences of extended SGTP are primarily (a) increased RCS loop resistance, resulting in a lower RCS flow rate, (b) decreased steam generator tube heat transfer area, resulting in lower steam generator outlet steam pressure, and (c) a decreased total RCS volume. The impact of these changes with respect to previously analyzed design conditions must be fully assessed for both normal operating and accident conditions. This assessment is performed following a steam generator inspection outage usually concurrent with a new reload safety evaluation. When required, revised safety analyses are performed and a Core Operating Limits Report (COLR) is prepared as required by Technical Specification 6.9.1.7.

In many cases, the incorporation of revised safety analyses into the North Anna design basis could be accomplished via Virginia Power processes employed to assess change per 10 CFR 50.59. However, based on current steam generator plugging projections, it is expected that the North Anna 1 Technical Specification RCS flow limit could be violated. This could potentially occur at average SGTP levels of approximately 20%. To address this concern a separate Technical Specification Amendment request to reduce the RCS total

flow rate limit by approximately 3% has been submitted for review and approval (1).

The Reference (1) package in combination with the existing Chapter 15 analyses and evaluations of plant system and component design support operation (within specific key core parameter limits) with up to 30% of the tubes plugged in any steam generator. Section 2.0 provides additional background information regarding this existing analysis basis.

The predictions of potential steam generator tube plugging during the current mid-cycle outage are such that the effects of increased RCS loop resistance on the large break LOCA analysis may not permit full rated power operation for the remainder of North Anna 1, Cycle 9. The existing large break LOCA analysis has obtained margin by taking credit for available Cycle 9 core characteristics but will not support 100% power operation with more than 30% SGTP. The large break LOCA analysis presented in Sections 3.0 through 5.0 of this evaluation extends this SGTP limit value to 35%, but with a reduced power level of 95% of rated thermal power.

2.0 ADDITIONAL EVALUATIONS AND BACKGROUND INFORMATION

There are a number of areas of plant design which are potentially impacted by operation with extended SGTP. This section presents background information relating to the key evaluations which Virginia Power has performed. Existing analyses for each design area have been evaluated for potential effects of extended SGTP. Specific limitations on such operation, where applicable, have been developed such that the results and conclusions

of existing design basis analyses will remain bounding for the proposed operation. The following major areas were evaluated:

- NSSS Systems and Components
- Balance of Plant Systems and Components
- NSSS Accident Analyses

For each of the above areas, the key aspects of the existing analysis basis supporting extended SGTP operation is discussed here.

Westinghouse Electric Corporation performed reviews of components and systems within their design responsibility to confirm that operation within the proposed conditions remains in compliance with the applicable codes and standards. It was concluded that all NSSS systems and components will remain within the bounds of existing design analysis results for operation with up to 40% of the tubes plugged in any steam generator.

The effect of extended SGTP operation upon balance of plant systems and components has been evaluated by Stone & Webster Engineering Corporation (SWEC). The evaluations concluded that effects of operation with extended SGTP will remain within the bounds of existing design analyses for operation with up to 37% average SGTP among the steam generators.

Virginia Power staff assessed the impact of extended SGTP operation upon NSSS accident analyses. The effects of reduced RCS flow associated with

extended SGTP upon most UFSAR Chapter 15 events has been evaluated in the Reference (1) amendment request. The remaining events requiring reanalysis, excluding small and large break LOCA, have assumed both reduced RCS flow rate and 40% average SGTP.

Additional events which are impacted by extended SGTP but are insensitive to reduced RCS flow have also been reanalyzed. These analyses have been implemented via the Virginia Power processes for assessing change per 10 CFR 50.59. These events (and the SGTP levels supported by the existing analyses) are:

EVENT	PLUGGING LIMIT
Small Break LOCA	35% SGTP in any SG
Boron Dilution at Power	40% average SGTP among SGs

Prior to restart of North Anna Unit 1, Cycle 9, an evaluation of the key core parameters and the actual final plugging will be performed to confirm that all applicable limits have been met. With the exception of large break LOCA, the existing analysis basis described in this section is valid for operation of North Anna Unit 1 at the rated thermal power of 2893 MWt with up to 35% SGTP in any steam generator. The current evaluation submits the evaluation of the large break LOCA accident with 35% SGTP. It, however, requires reduced power operation in order to achieve PCT results in compliance with the 10 CFR 50.46 acceptance criteria.

3.0 LARGE BREAK LOCA ACCIDENT DESCRIPTION

A reanalysis of the emergency core cooling system (ECCS) performance for the postulated large break loss of coolant accident (LOCA) has been performed in compliance with Appendix K to 10 CFR 50. The results of this reanalysis are presented here, and are in compliance with 10CFR50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Reactors." This analysis was performed with the NRC-approved version of the Westinghouse LOCA-ECCS evaluation model denoted as the 1981 model with BASH (2). The analytical techniques are in full compliance with 10CFR50, Appendix K.

As required by Appendix K to 10CFR50, certain conservative assumptions were made for the LOCA-ECCS analysis. The assumptions pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA is assumed to occur, and include such items as the core peaking factors, the containment pressure, and the performance of the emergency core cooling system. Selection of input parameters for Appendix K analyses is made to represent a conservative configuration of the plant initial conditions. This was accomplished by assuming bounding input values for key parameters such as core power, FΔh, FQ, steam generator tube plugging and RCS flow. In general, the remaining key assumptions included in the current analysis are consistent with previous large break analyses performed by Virginia Power. Additional discussion of these analysis assumptions is provided in Section 4.0.

A LOCA is the result of a rupture of the reactor coolant system (RCS) piping or of any line connected to the system. The system boundaries

considered in the LOCA analysis are defined in the UFSAR. Sensitivity studies (5) have indicated that a double-ended cold-leg guillotine (DECLG) pipe break is limiting. Should a DECLG occur, rapid depressurization of the RCS occurs. The reactor trip signal subsequently occurs when the pressurizer low-pressure trip setpoint is reached. A safety injection system (SIS) signal is actuated when the appropriate setpoint is reached, activating the high-head safety injection pumps. The actuation and subsequent activation of the Emergency Core Cooling System, which occurs with the SIS signal, assumes the most limiting single-failure event. These countermeasures will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. No credit is taken in the analysis for the insertion of control rods to shut down the reactor.

2. Injection of borated water provides heat transfer from the core and limits the clad temperature increase.

Before the break occurs, the unit is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from fission product decay, hot internals and the vessel continue to be transferred to the reactor coolant system. At the beginning of the blowdown phase, the entire reactor coolant system contains subcooled liquid that transfers heat from the core by forced convection with some fully developed nucleate boiling. After the break develops, the time to DNB is calculated, consistent with Appendix K of 10CFR50. Thereafter, the core heat transfer is based on local conditions, with transition boiling and forced convection to steam as the major heat transfer mechanisms.

During the refill period, it is assumed that rod-to-rod radiation is the only core heat transfer mechanism. The heat transfer between the reactor coolant system and the secondary system may be in either direction, depending on the relative temperatures. For the case of continued heat addition to the secondary side, secondary-side pressure increases and the main safety valves may actuate to reduce the pressure. Makeup to the secondary side is automatically provided by the auxiliary feedwater system. Coincident with the safety injection signal, normal feedwater flow is stopped by closing the main feedwater control valves and tripping the main feedwater pumps. Emergency feedwater flow is initiated by starting the auxiliary feedwater pumps. The secondary-side flow aids in the reduction of RCS pressure. When the reactor coolant system depressurizes to 600 psia, the accumulators begin to inject borated water into the reactor coolant loops. The conservative assumption is then made that injected accumulator water bypasses the core and goes out through the break until the termination of bypass. This conservatism is again consistent with Appendix K of 10CFR50. In addition, the reactor coolant pumps are assumed to be tripped at the initiation of the accident, and effects of pump coastdown are included in the blowdown analysis.

The water injected by the accumulators cools the core, and subsequent operation of the low-head safety injection pumps supplies water for long-term cooling. When the refueling water storage tank (RWST) is nearly empty, the long-term cooling of the core is accomplished by switching to the recirculation mode of core cooling, in which the spilled borated water is drawn from the containment sump by the low-head safety injection pumps and returned to the reactor vessel.

The containment spray system and the recirculation spray system operate to return the containment environment to subatmospheric pressure.

4.0 LARGE BREAK LOCA ANALYSIS ASSUMPTIONS

As required by Appendix K of 10 CFR 50, certain conservative assumptions were made for the Large Break LOCA-ECCS analysis. The assumptions pertain to the condition of the reactor and associated safety system equipment at the time that the LOCA is assumed to occur, and include such items as the core peaking factors, core decay heat and the performance of the Emergency Core Cooling System. Tables 1 and 2 present the values assumed for several key parameters in this analysis. Assumptions and initial operating conditions which reflect the requirements of Appendix K to 10CFR50 have been used in this analysis. These assumptions include:

- The break is located in the cold leg between the pump discharge and the vessel inlet.

- The safety injection flow spills to containment back pressure in the broken loop. Safety injection occurs only in the intact loops cold legs.

- The accumulator in the broken loop also spills to containment.

- 120 percent of 1971 ANS decay heat is assumed following reactor trip.

- Initial power is 97% of the Technical Specifications rated thermal power of 2893 MWt, which includes 2% to account for the calorimetric uncertainty.

Several additional assumptions have been incorporated into the LBLOCA reanalysis described below to accommodate the effects of Unit 1 operation with extended SGTP. These changes are discussed here.

The analysis assumes that 35% of the tubes in each steam generator are plugged. This level of tube plugging is expected to bound that which is actually experienced at North Anna Unit 1. Since large break LOCA results are sensitive to SGTP, this assumption is necessary to demonstrate continued compliance with the 10 CFR 50.46 ECCS acceptance criteria. In conjunction with extended SGTP, a reduced RCS total flowrate of 264400 gpm has been assumed. This value bounds the expected RCS flow associated with 35% SGTP.

This analysis also assumes that reactor coolant system average temperature equals 586.8°F, the Technical Specifications nominal maximum allowed value. This bounds the actual Unit 1 nominal operating Tavg of 583°F and has been shown in Virginia Power sensitivities to produce conservative large break LOCA results.

The analysis assumed a reference cosine axial power distribution with a peak Heat Flux Hot Channel Factor, FQ(z), of 2.11 at 95% power (equivalent to a 2.00 limit at 100% power). This value, which is more restrictive than the existing analysis, was assumed to obtain additional analysis margin for operation with extended SGTP. Figure 1 illustrates the power shape assumed.

In addition, a peak Nuclear Enthalpy Hot Channel Factor, $F_N\Delta h$, of 1.573 at 95% was assumed. This is equivalent to the current Technical

Specifications limit of 1.55 at 100% power and has also been assumed to obtain acceptable results for operation with extended SGTP.

As required by Technical Specification 6.9.1.7, the Core Operating Limits Report (COLR) documents the applicable limit values of key core-related parameters for each reload core. The COLR will specify the appropriate limits which account for all design considerations, including large and small break LOCA effects.

As part of the safety evaluation to be performed by Virginia Power for restart and continued operation of North Anna 1, Cycle 9, a revised COLR will be issued. This safety evaluation, in conjunction with the COLR, will document acceptable limit values for key core parameters. Since the large break LOCA assumptions will impose the most restrictive requirements on the allowable FQ x K(Z) limit at each elevation, Z, the COLR for each reload core will document the appropriate limit.

To obtain additional margin, this analysis assumed the fuel rod temperature and internal pressure values associated with the North Anna 1, Cycle 9 burnup at shutdown on December 23, 1991. Using core design predictions and fuel performance data based on the PAD 3.4 thermal model (4), it was determined that the 10000 MWD/MTU accumulated cycle burnup correlated with 12000 MWD/MTU for the limiting fresh fuel assembly. These assumptions will bound the fuel characteristics for the remainder of North Anna 1, Cycle 9.

This analysis also modified the means of implementing the single failure assumption as compared with that in the existing analysis. Appendix K of 10 CFR 50 requires that the ECCS containment pressure analysis assume the operation of all pressure reducing equipment since minimum pressure is conservative. This is without regard for any assumed single failures, since operation of all such equipment is accomplished only by energizing all emergency equipment trains. In past large break LOCA analyses, the single failure requirement has typically been conservatively implemented by assuming that loss of offsite power occurs coincident with the LOCA and that one emergency diesel generator (EDG) fails to start. This has the effect of removing a single train of safety injection pumps from service, allowing flow from one high head and one low head safety injection pump. However, past analyses also have assumed, as required by Appendix K, that both trains of containment spray were operating. Westinghouse sensitivity studies (10) have demonstrated that the limiting single failure (within the required assumptions of Appendix K) is the assumption that one low head safety injection pump fails. This assumption, combined with Appendix K requirements, leaves flow available from two high head and one low head safety injection pump, and flow from both containment spray systems. Since the past single failure implementation was unnecessarily conservative and nonphysical, the assumption was changed to provide additional safety injection flow margin to help accommodate the effects of extended SGTP. The total assumed safety injection flowrate has been confirmed to be a conservative representation of actual system flow performance.

Even though the 1 low head/2 high head pump configuration represents the most limiting single failure combination, an additional restriction on ECCS

equipment operability requirements is being implemented for the remainder of North Anna 1, Cycle 9. This restriction (implemented as a footnote to T. S. 3.5.2 Action Statement "a") will require that 2 HHSI pumps and their associated flow paths be OPERABLE if an LHSI pump is out of service. This change ensures that ECCS equipment operability (while T.S. 3.5.2 Action Statement "a" is effective) is consistent with that assumed in the large break LOCA analysis.

Using these assumptions in the BASH ECCS evaluation model, it has been demonstrated that operation at a maximum reactor power of 2748 MWt with SGTP of up to 35% in any SG will comply with the 2200°F acceptance limit of 10 CFR 50.46.

5.0 ANALYSIS OF EFFECTS AND CONSEQUENCES

5.1 METHOD OF ANALYSIS

The large break LOCA is divided, for analytical purposes, into three phases: blowdown, refill and reflood. There are three distinct transients analyzed in each phase, including the thermal-hydraulic transient in the reactor coolant system, the pressure and temperature transient within the containment and the fuel clad temperature transient of the hottest fuel rod in the core. Based on these considerations, a system of interrelated computer codes has been developed for the analysis.

The description of the various aspects of the LOCA analysis methodology is given in WCAP-8339 (6). This document describes the major phenomena modelled, the interfaces among the computer codes and the features of the codes that ensure compliance with 10CFR50, Appendix K. The SATAN-VI, COCO,

WREFLOOD, BASH and LOCBART codes, which are used in the LOCA analysis, are described in detail in WCAP-8306 (7), WCAP-8326 (8), WCAP-8171 (9), and WCAP-10266 (3), respectively. BASH and LOCBART are described together in Reference (2). These codes assess whether sufficient heat transfer geometry and core amenability to cooling are preserved during the time spans applicable to the blowdown, refill and reflood phases of the LOCA. The SATAN-VI computer code analyzes the thermal-hydraulic transient of the reactor coolant system during blowdown, and the COCO computer code calculates the containment pressure transient during all three phases of the LOCA analysis. The thermal-hydraulic response of the reactor coolant system during refill is calculated by the WREFLOOD code; for the reflood phase, this response is calculated by the BASH code. Internal to the BASH code is the previously approved BART model, which is used to provide a mechanistic estimate of the heat transfer coefficient in the core during reflood.

SATAN-VI is used to determine the RCS pressure, enthalpy and density, as well as the mass and energy flow rates in the reactor coolant system and steam generator secondary, as a function of time during the blowdown phase of the LOCA. SATAN-VI also calculates the accumulator mass and pressure and the pipe break mass and energy flow rates that are assumed to be vented to the containment during blowdown. At the end of the blowdown, the mass and energy release rates during blowdown are transfered to the COCO code for use in the determination of the containment pressure response during this first phase of the LOCA. Additional SATAN-VI output data from the end of blowdown, including the core inlet flow rate and enthalpy, the core pressure and the core power decay transient are input to the LOCBART code.

With input from the SATAN-VI code at the end of blowdown, WREFLOOD is used to determine the vessel flooding rate, the coolant pressure and temperature and the quench of vessel metal mass during the refill phase of the LOCA (time period from end of blowdown to that time when flow enters the bottom of the core). WREFLOOD is also used to calculate the mass and energy flowrates assumed to be vented to the containment for refill and reflood phases. Since the mass flowrate to the containment depends on core pressure, which is a function of the containment backpressure, the WREFLOOD and COCO codes are interactively linked.

The COCO code, which is used throughout all three phases of the LOCA analysis, calculates the containment pressure. Input to COCO is obtained from the mass and energy flow rates assumed to be vented to the containment, as calculated by the SATAN-VI and WREFLOOD codes. In addition, conservatively chosen initial containment conditions and an assumed mode of operation for the containment cooling system are input to COCO. These initial containment conditions and assumed modes of operation are provided in Table 2.

Once the vessel has refilled to the bottom of the core, the reflood portion of the transient begins. Information is taken from the WREFLOOD code characterizing the thermal-hydraulic status of the vessel at this time as well as the containment backpressure transient as calculated by COCO and input into the BASH code. The BASH code is used to calculate the thermal-hydraulic simulation of the RCS for the reflood phase.

LOCBART is used throughout the analysis of the LOCA transient to calculate the fuel and clad temperature of the hottest rod in the core. Input to LOCBART consists of appropriate thermal-hydraulic outputs from SATAN-VI, WREFLOOD and BASH, and conservatively selected initial RCS operating conditions. These initial conditions are summarized in Table 1 and Figure 1. Using this information as boundary conditions, LOCBART computes the fluid conditions and heat transfer coefficient for the full length of the fuel rod by employing mechanistic models appropriate to the actual flow and heat transfer regimes. The axial power shape of Figure 1 assumed for LOCBART is a chopped cosine curve that has been historically used as the reference axial power shape for large break LOCA analyses. Verification that the cosine shape remains limiting is performed for each reload core.

5.2 RESULTS

Tables 1 and 2 and Figure 1 present the initial conditions and the modes of operation that were assumed in the analysis. The results of this analysis are tabulated in Tables 3 and 4 for a double ended guillotine break with a CD=0.4 discharge coefficient and 95% of rated thermal power. The mass and energy release for limiting cases are given in Tables 5 and 6. Prior Virginia Power and Westinghouse analyses employing the approved large break LOCA evaluation models have demonstrated that limiting PCT for North Anna is obtained for this case. Results for other typical cases (CD=0.6, CD=0.8) are consistently bounded by 150 - 200°F in PCT. The double ended guillotine break has been determined to be the limiting break size and location based on the sensitivity studies reported in Reference (5). The analysis resulted in a limiting peak clad temperature of 2140.8°F, a maximum local cladding

oxidation level of 7.22% and a total core metal-water reaction of less than 1%. The detailed results of the LOCA analysis are provided in Tables 3 through 6 and Figures 1 through 17. The attached figures show the following:

- Axial Power Shape - Figure 1 shows the cosine power shape used in this analysis.

- Core Mass Flow - Figure 2 shows the calculated core flow, both top and bottom.

- Core Pressure - Figure 3 shows the calculated pressure in the core.

- Accumulator Mass Flow - Figure 4 shows the calculated accumulator flow. The accumulator delivery during blowdown is discarded until the end of bypass is calculated. Accumulator flow, however, is established in the refill-reflood calculations. The accumulator flow assumed is the sum of that injected in the intact cold legs.

- Core Pressure Drop - Figure 5 shows the calculated core pressure drop. The core pressure drop is interpreted as the pressure immediately before entering the core inlet to the pressure just outside the core outlet.

- Break Mass Flow - Figure 6 shows the calculated flowrate out of the break. The flowrate out of the break is plotted as the sum of flow at both the pressure vessel end and the reactor coolant pump end of the guillotine break.

- Core Power - Figure 7 shows the core power transient calculated by the SATAN-VI code.

- Containment Wall Heat Transfer Coefficient - Figure 8 shows the containment wall heat transfer coefficient.

- Containment Pressure - Figure 9 shows the calculated pressure transient. The analysis of this pressure transient is based on the containment data, reflood mass and energy release, and accumulator flow to containment.

- Pumped ECCS Flow (Reflood) - Figure 10 shows the calculated flow of the emergency core cooling system.

- Core and Downcomer Water Levels - Figure 11 shows the reactor vessel downcomer and core water levels.

- Raw Flooding Rate Integral - Figure 12 shows the raw flooding rate integrals and smoothed line segment integrals used in the LOCBART calculations.

- Core Flooding Rate - Figure 13 shows the resulting line segment integrals from previous figures.

- Hot Rod Clad Average Temperature - Figure 14 shows the calculated hot-spot clad temperature transient and the clad temperature transient at the burst location. The peak clad temperature for the limiting discharge coefficient of 0.4 is 2140.8°F at 10.50 ft elevation in the core.

- Vapor Temperature - Figure 15 shows the calculated vapor temperature for the hot spot and burst locations.

- Hot Rod Heat Transfer Coefficient - Figure 16 shows the heat transfer coefficient at the hot spot location on the hottest rod.

- Hot Rod Mass Velocity - Figure 17 shows the mass velocity at the hot-spot location on the hottest fuel rod.

6.0 CONCLUSIONS

This large break LOCA analysis was performed for a double ended rupture of a reactor coolant pipe with CD=0.4, at 95% of the Technical Specification rated thermal power of 2893 MWt (2748 MWt), assuming the operating conditions specified in Tables 1 and 2. Based upon these results, the emergency core cooling system will meet the acceptance criteria as presented in 10CFR50.46 as follows:

1. The calculated peak fuel rod clad temperature is below the requirement of 2200°F.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1% of the total amount of Zircaloy in the reactor.

3. The clad temperature transient is terminated at a time when the core is still amenable to cooling. The localized cladding oxidation limits of 17% are not exceeded during or after quenching.

4. The core remains amenable to cooling during and after the break.

5. The core temperature is reduced and the long-term heat is removed for an extended period of time.

Table 1

INITIAL CORE CONDITIONS ASSUMED FOR THE
DOUBLE ENDED COLD LEG GUILLOTINE BREAK (DECLG)

Calculational Input	
Core Power (MWt), 97% of	2893
Peak Linear Power (Kw/ft) 97% of	11.35
Peak Heat Flux Hot Channel Factor, FQ(z)	2.11*
Peak Nuclear Enthalpy Hot Channel Factor, $F_N \Delta h$	1.573**
Accumulator Water Volume (ft³/accumulator)	1025
Reactor Vessel Upper Head Temperature	(T_{hot})

Limiting Fuel Region and Cycle	Cycle	Region
Unit 1	9***	All Regions

* Equivalent to a 100% power limit of 2.00
** Equivalent to a 100% power limit of 1.55
*** Analysis is only applicable to Cycle 9 from 10000 MWD/MTU to EOC.

Table 2

CONTAINMENT DATA

Net Free Volume (ft^3)	1.916×10^6
Initial Conditions[a]	
Pressure (psia)	9.608
Temperature (°F)	86.0
RWST Temperature (°F)	40.0
Outside Temperature (°F)	-10.0
Spray System[a]	
Number of Pumps Operating	2
Runout Flow Rate (per pump)	2000 gpm
Time in Which Spray is Effective	59 sec
Structural Heat Sinks[a]	
Thickness (in.)	Area (ft^2), with allowance for uncertainties
6 concrete	8,393
12 concrete	62,271
18 concrete	55,365
24 concrete	11,591
27 concrete	9,404
36 concrete	3,636
.375 steel, 54 concrete	22,039
.375 steel, 54 concrete	28,393
.500 steel, 30 concrete	25,673
26.4 concrete, .25 steel, 120concrete	12,110
.407 stainless steel	10,527
.371 steel	160,328
.882 steel	9,894
.059 steel	60,875

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a See UFSAR Section 6.3.3.12 for a detailed breakdown of the containment heat sinks and for justification of the other input parameters use to calcualte containment pressure.

Table 3

TIME SEQUENCE OF EVENTS

Description of Parameters	DECLG (Cd=0.4) (seconds)
End of Bypass/ End of Blowdown (sec)	30.6685
Safety System Actions	
Reactor Trip (Sec)	0.549
Accumulator Injection (Sec)	13.6
SI Signal Generated (Sec)	3.8
Pump SI Starts (Sec)	30.8
Bottom of Core Recovery (sec)	44.65
Accumulator Empty (sec)	54.16

Table 4

RESULTS FOR DECLG

Description of Parameters	DECLG (Cd=0.4) (seconds)
Pe[illegible] Clad Temperature ([illegible])	2140.8
Peak Clad Location (ft)	10.50
Hot Rod Burst Data	
Location (ft)	6.00
Time (Sec)	51.63
Zr/H2O Results Data	
Local Maximum Reaction (%)	7.22
Location of Maximum (ft)	10.25
Total Reaction (%)	< 1.0

Table 5

REFLOOD MASS AND ENERGY RELEASES DECLG (CD=0.4)

Time (Sec)	Total Mass Flow Rate (lbm/sec)	Total Energy Flow Rate (10^5 Btu/sec)
44.6[illegible]	0.0	0.0
45[illegible]	[illegible]2	.00687
45.42[illegible]	[illegible]	.0068[illegible]
45.521	[illegible]	.0068[illegible]
45.721	[illegible]	.0067[illegible]
45.821	.5176	.00671
56.982	61.86	0.6939
75.582	117.18	0.9059
97.032	282.84	1.3010
120.782	296.24	1.2622
146.832	319.50	1.2341
220.082	350.41	1.2003

Table 6

BROKEN LOOP ACCUMULATOR FLOW TO CONTAINMENT DECLG (CD=0.4)

TIME (sec)	MASS FLOW RATE (lb/sec)
0.0	4096.76
1.01	3692.74
3.01	3157.39
5.01	2804.86
7.01	2546.44
10.01	2257.25
15.01	1928.86
20.01	1706.87
25.01	1550.55
29.01	1622.40

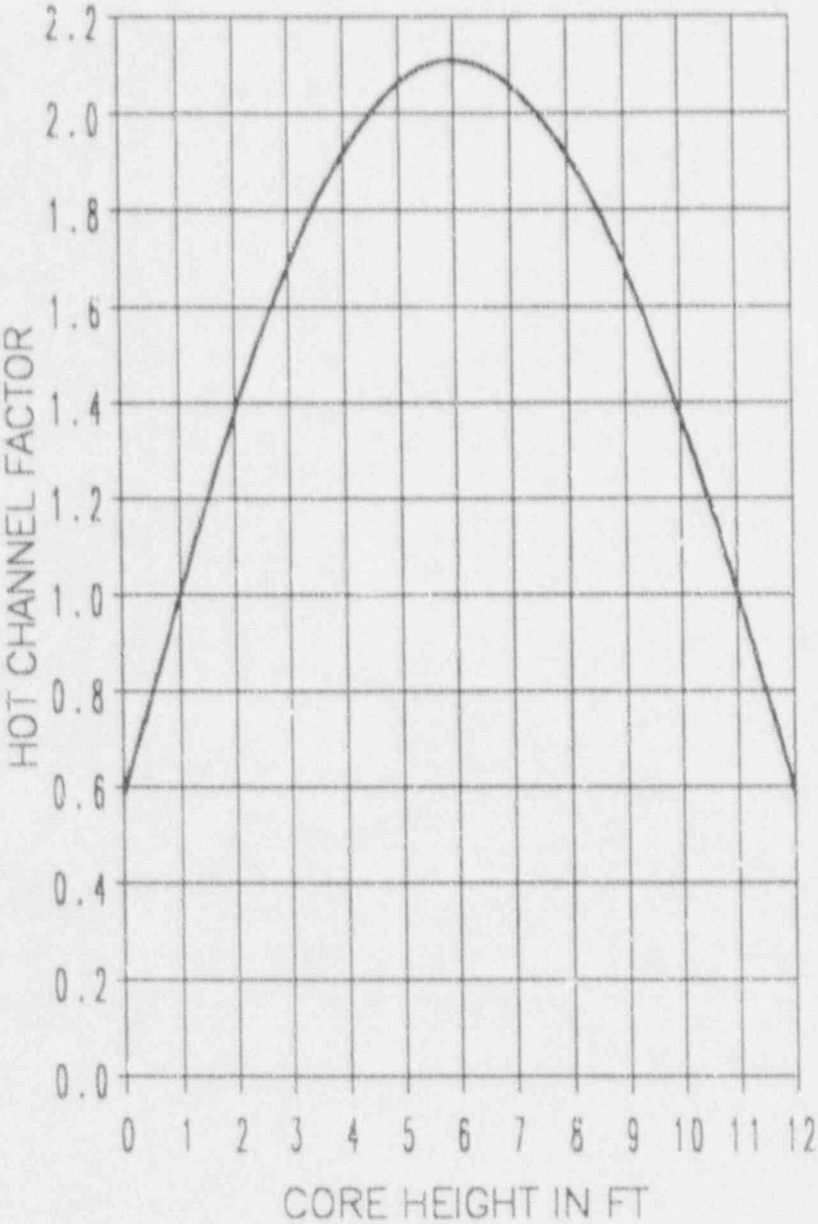
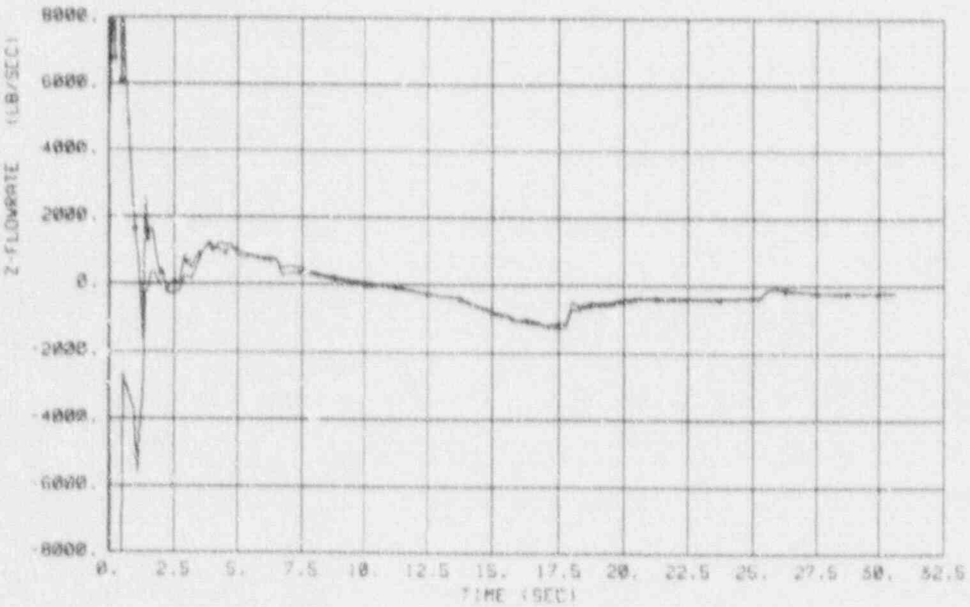
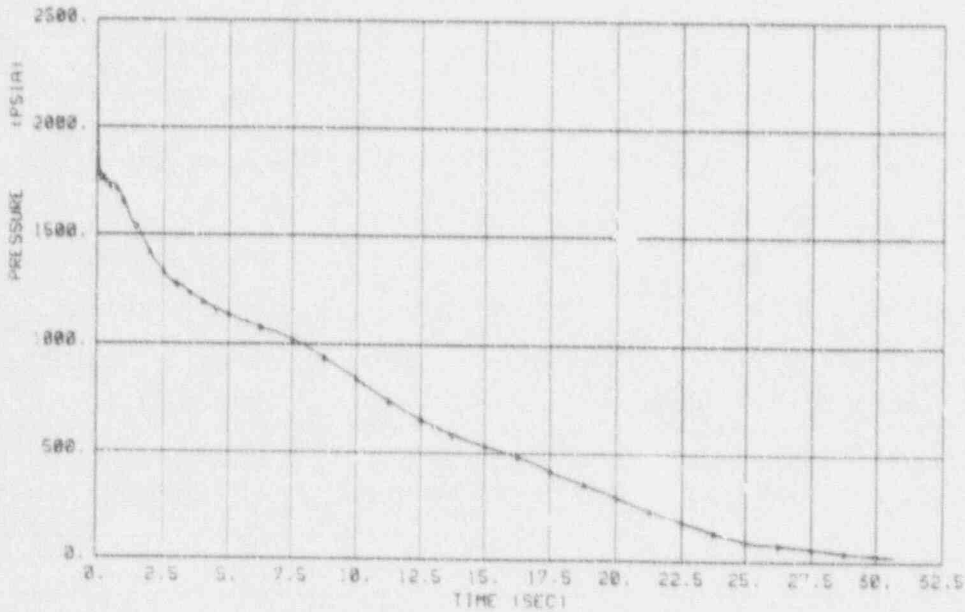


Figure 1 Axial Power Shape



01/17/25

Figure 2 Core Mass Flow (DECLG, CD=0.4)



01/17/25

Figure 3 Core Pressure (DECLG, CD=0.4)

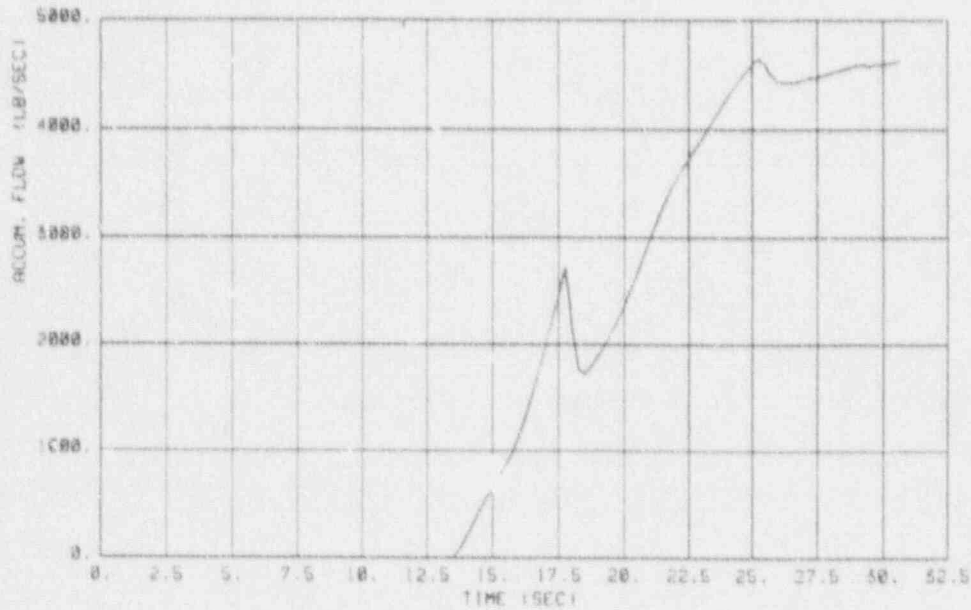


Figure 4 Accumulator Mass Flow (DECLG, CD=0.4)

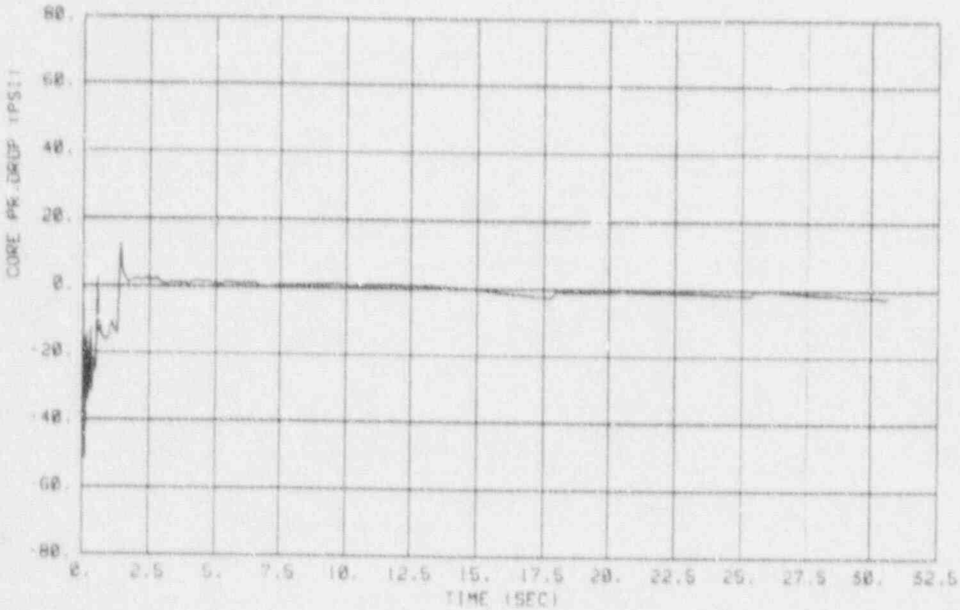


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Figure 5 Core Pressure Drop (DECLG, CD=0.4)

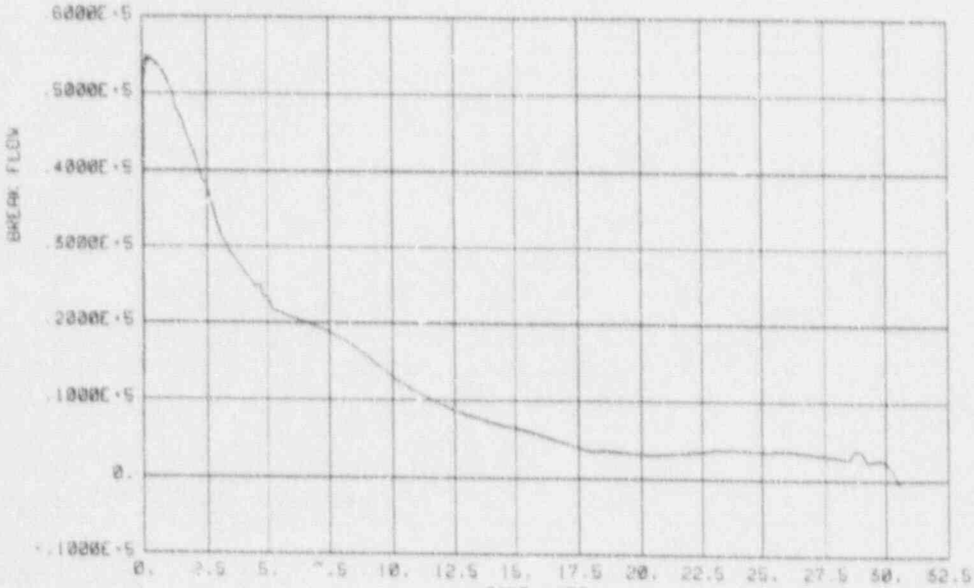

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Figure 6 Break Mass Flow (DECLG, CD=0.4)

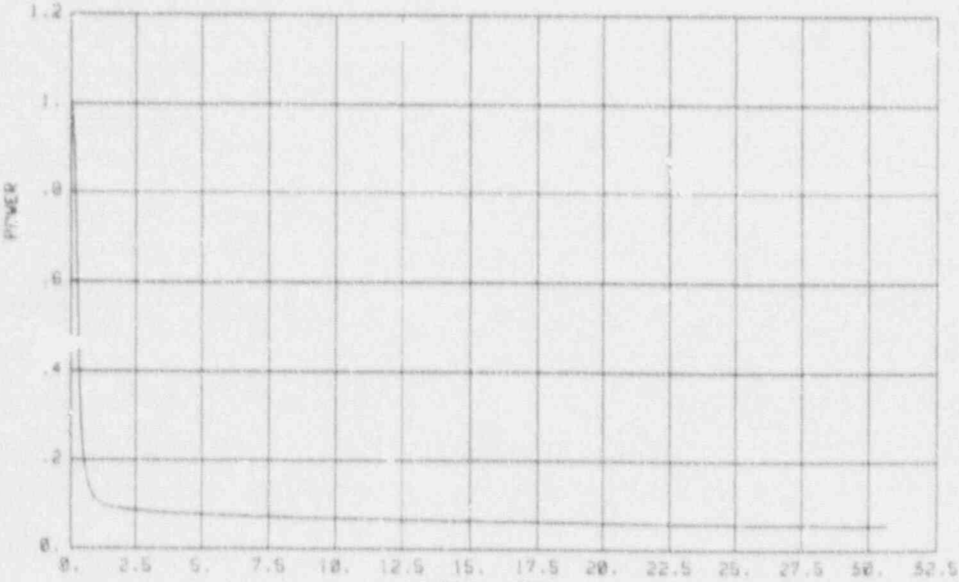

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Figure 7 Core Power (DECLG, CD=0.4)

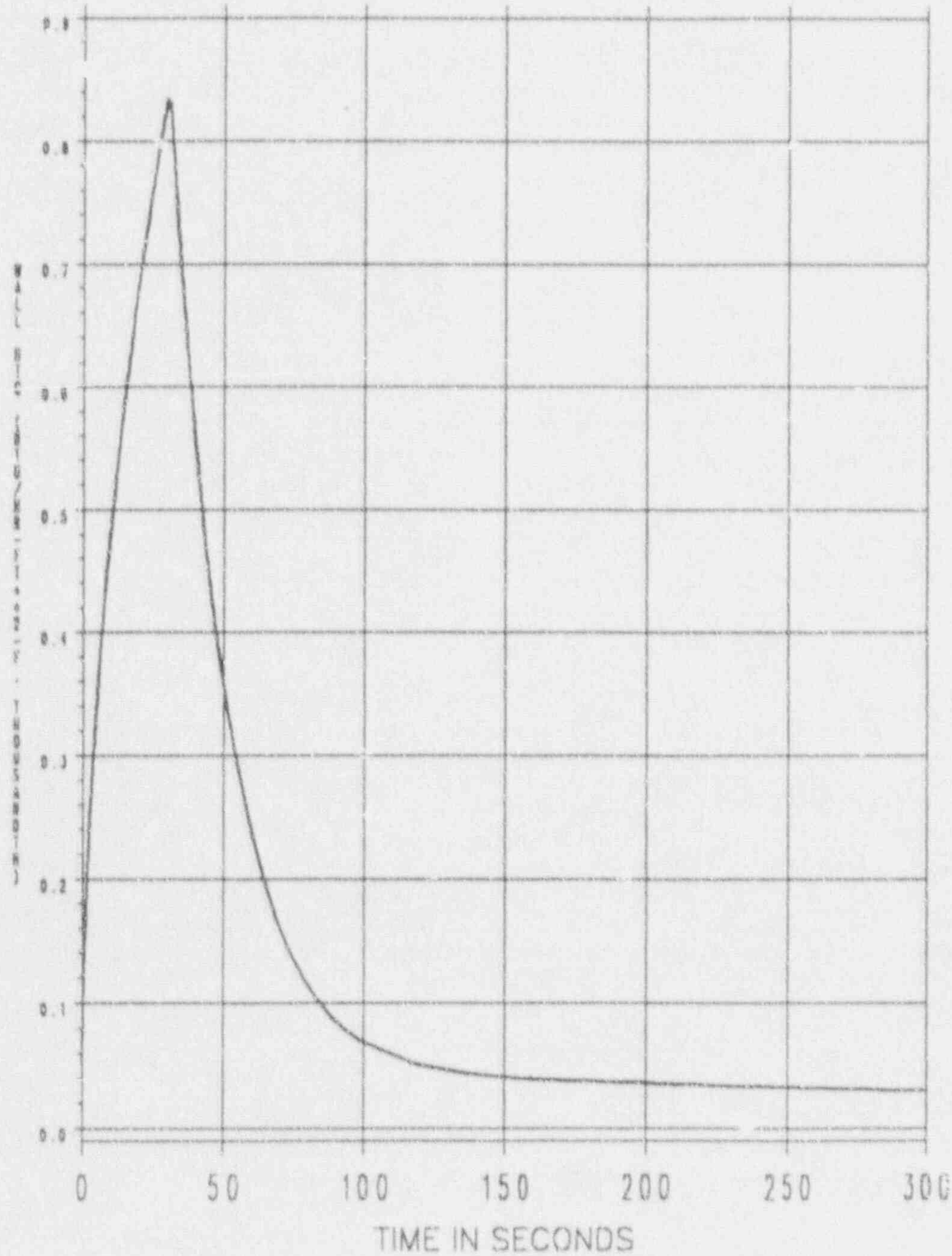

Figure 8 Containment Wall Heat Transfer Coefficient (DECLG, CD=0.4)

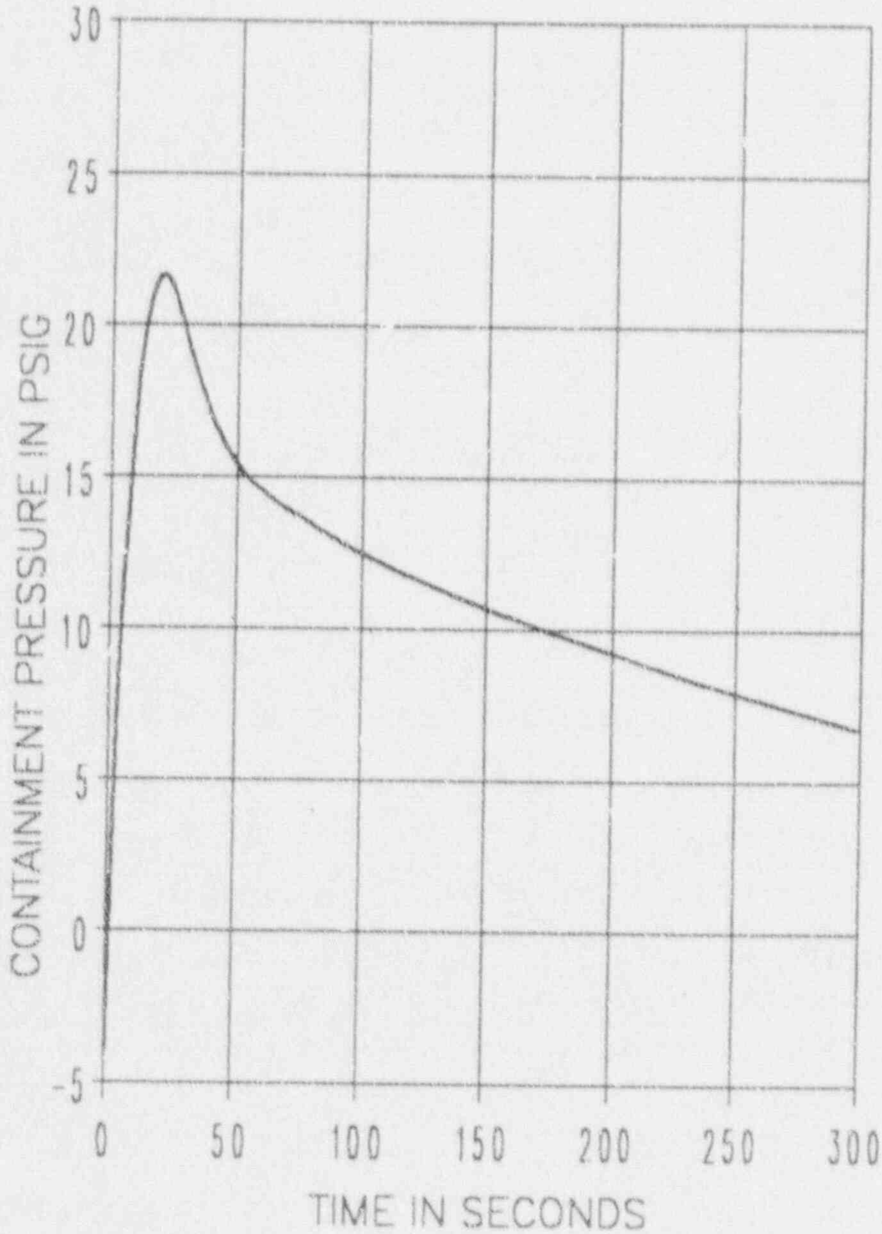

Figure 9 Containment Pressure (DECLG, CD=0.4)

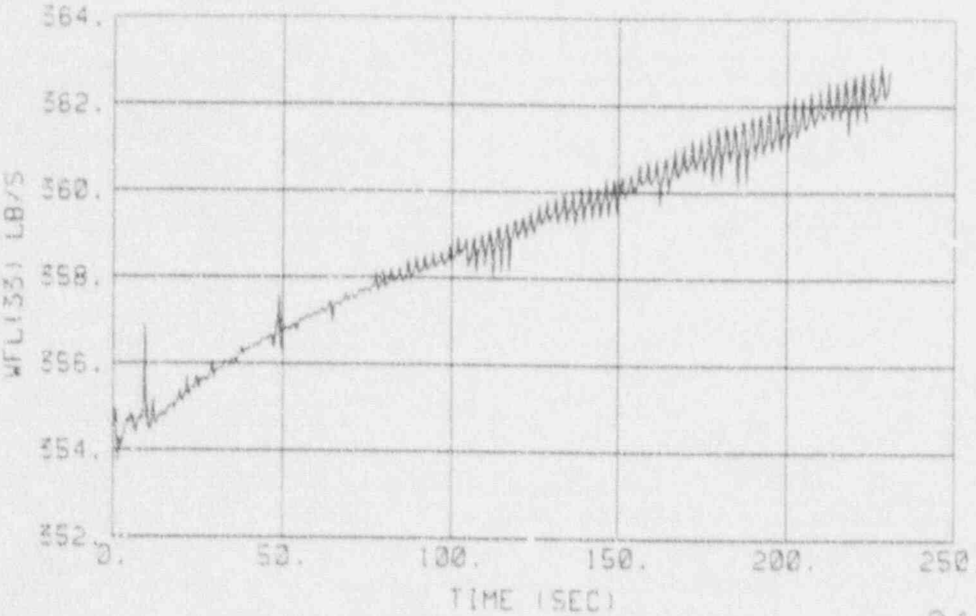

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Figure 10 Pumped ECCS Flow (DECLG, CD=0.4)

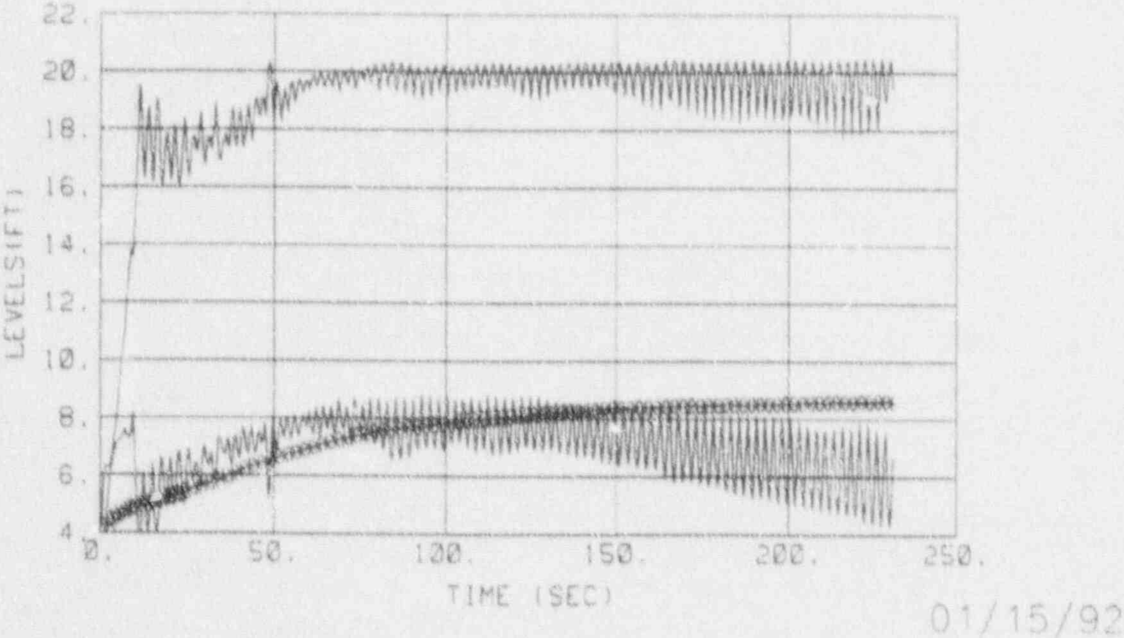

Figure 11 Core and Downcomer Water Levels (DECLG, CD=0.4)

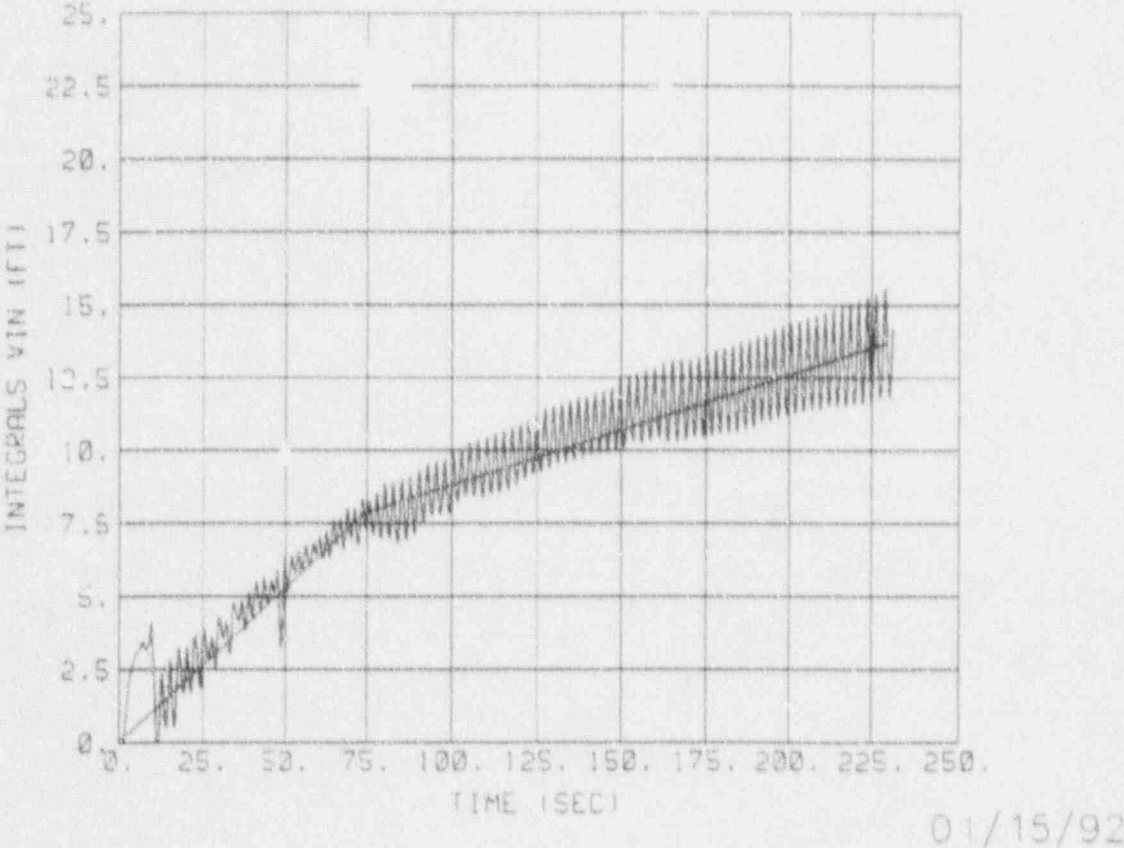

Figure 12 Raw Flooding Rate Integral (DECLG, CD=0.4)

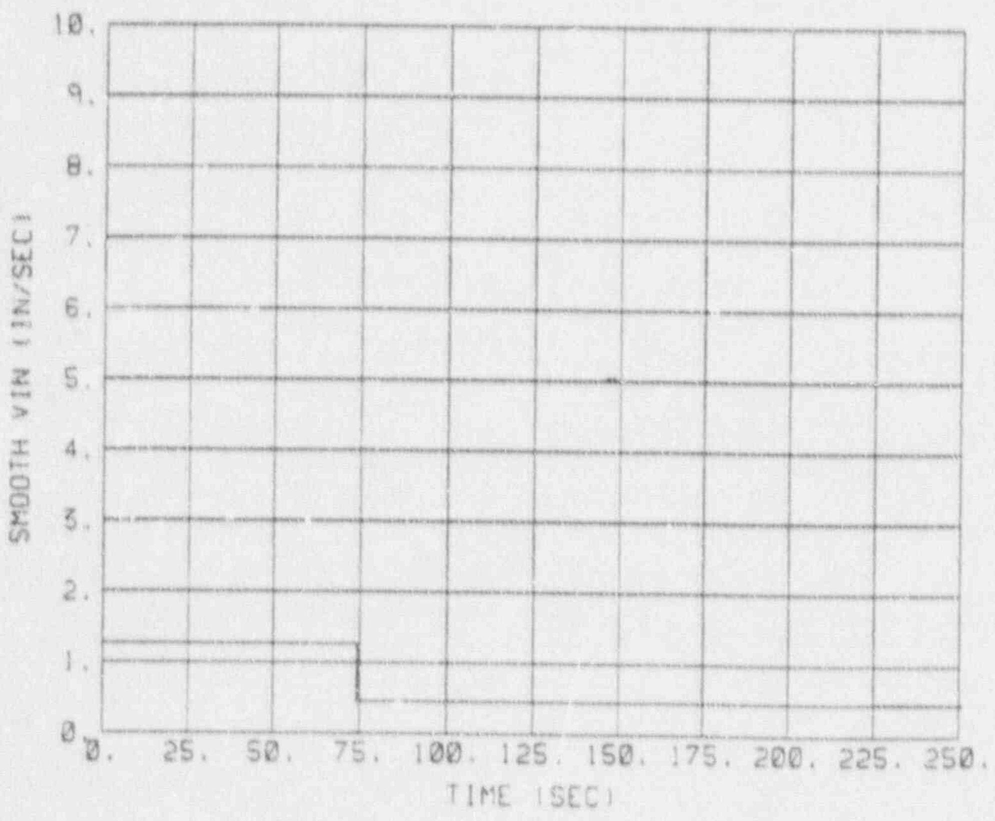

Figure 13 Core Flooding Rate (DECLG, CD=0.4)

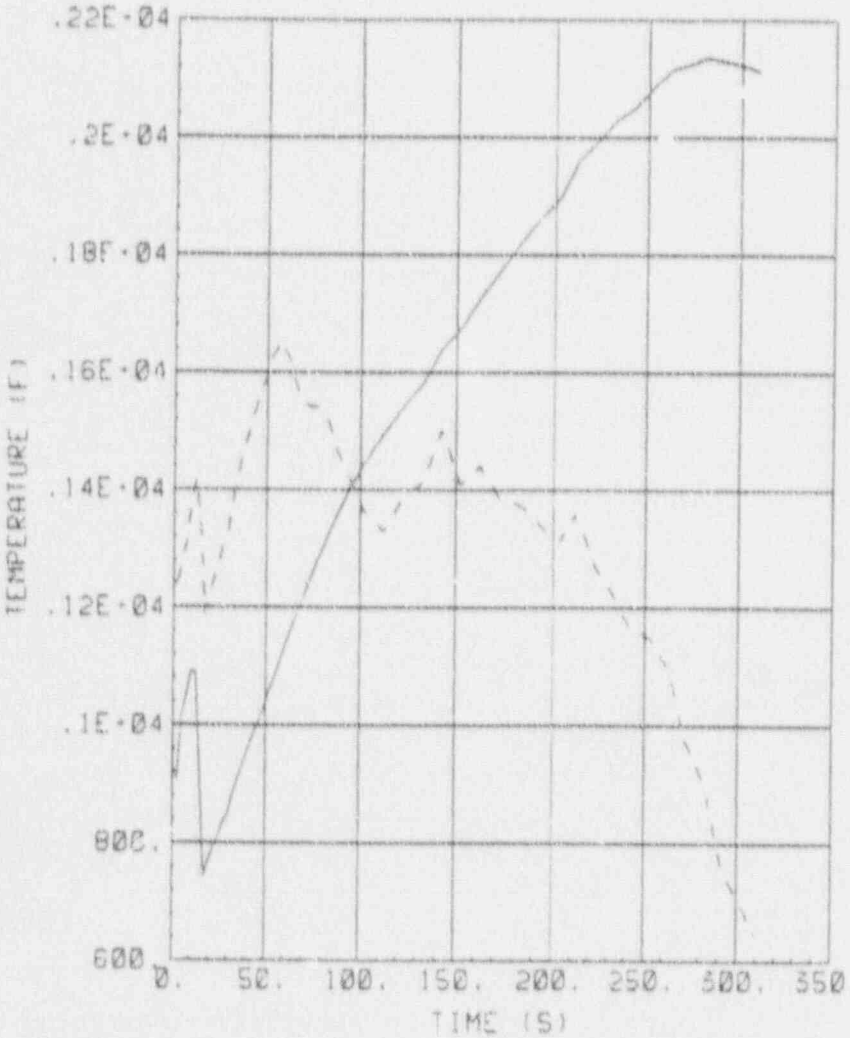

Figure 14 Hot Rod Clad Average Temperature (DECLG, CD=0.4)

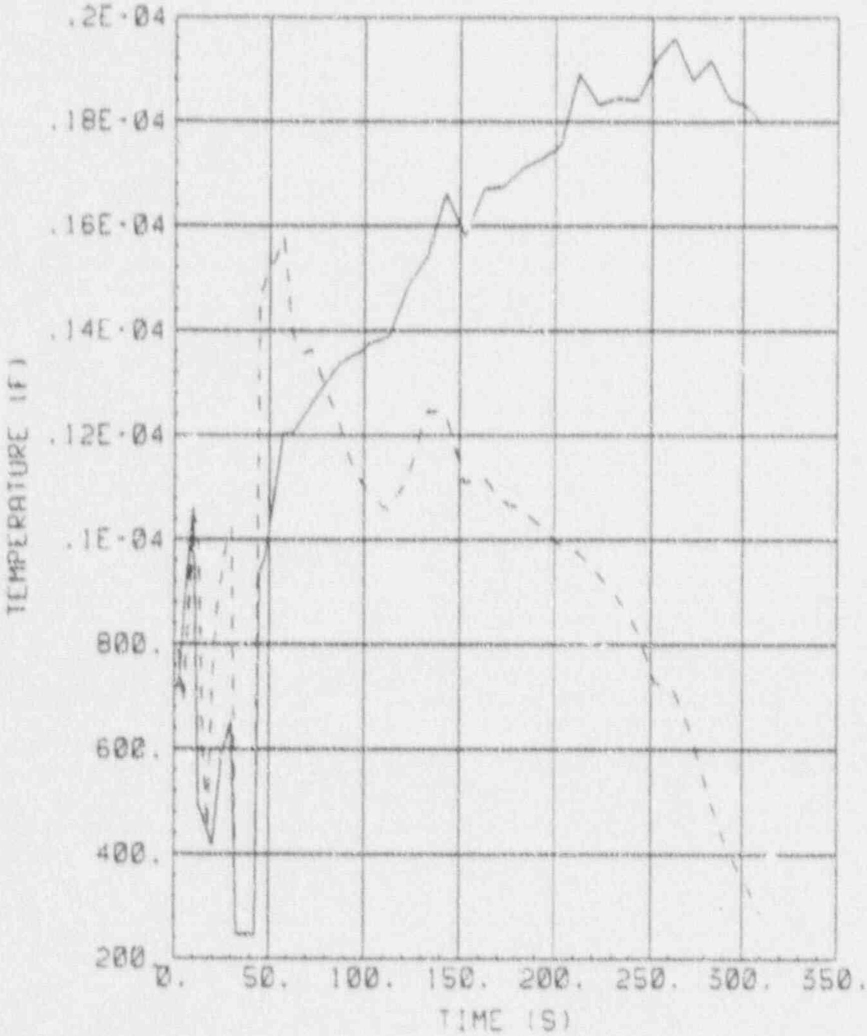

Figure 15 Vapor Temperature in Hot Assembly (DECLG, CD=0.4)

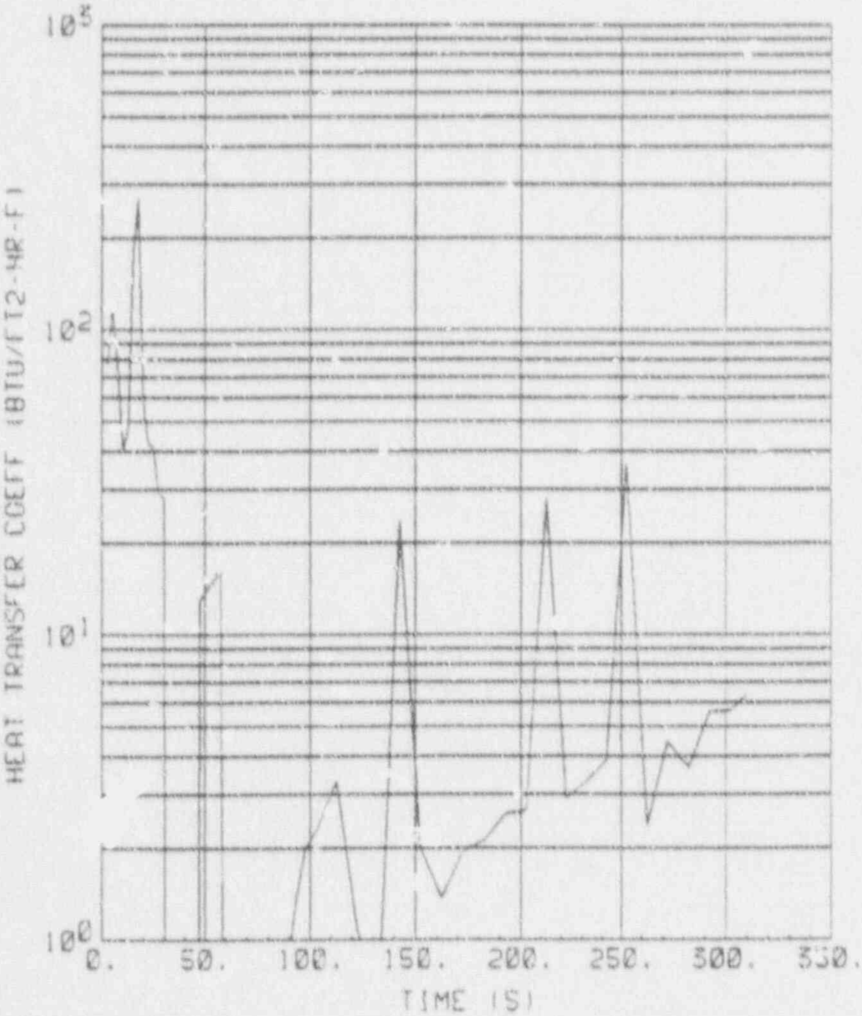

Figure 16 Hot Rod Heat Transfer Coeffici nt (DECLG, CD=0.4)

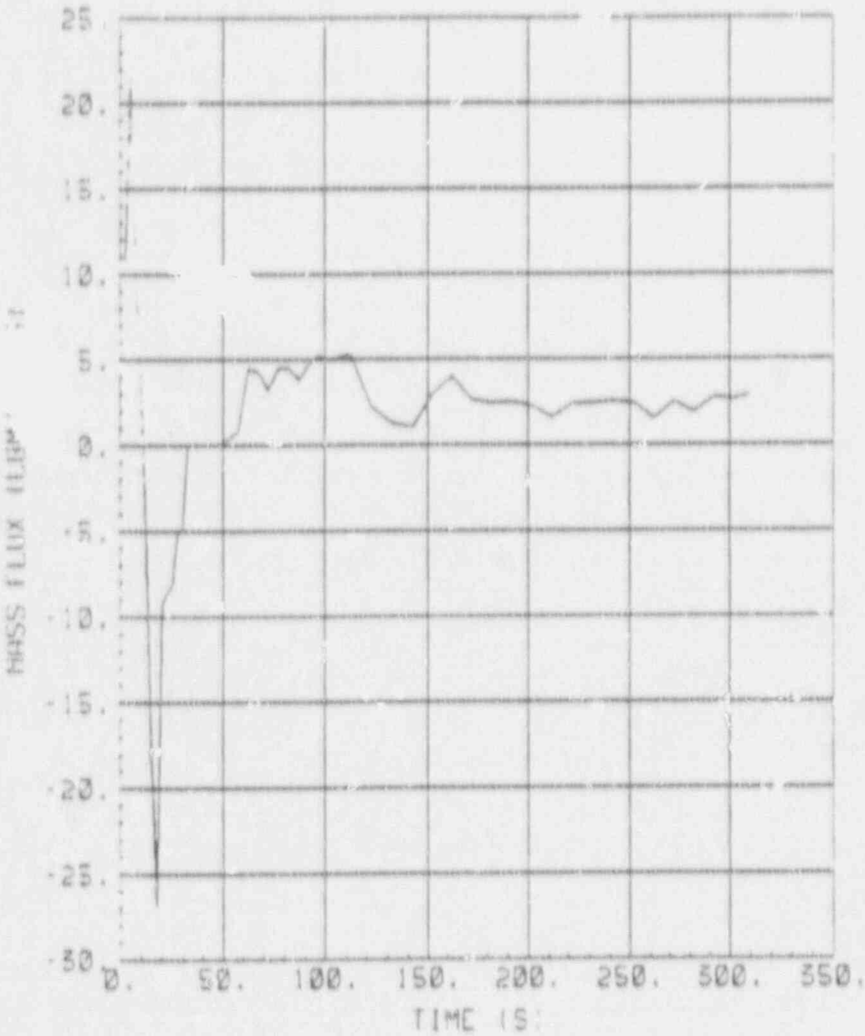

Figure 17 Hot Rod Mass Flux (DECLG, CD=0.4)

REFERENCES

(1) Letter from W. L. Stewart (Virginia Power) to U.S. NRC, "North Anna Power Station Unit 1 Proposed Technical Specification Change - Reduced Minimum RCS Flow Rate Limit to Support Increased Steam Generator Tube Plugging Level," Serial No. 92-018, January 8, 1992.

(2) WCAP-10266-P-A, Rev. 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," March 1987.

(3) WCAP-10444-P-A, Addendum 2, "Vantage 5H Fuel Assembly," April 1988.

(4) WCAP-8720, and Addendum 2, "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations," October 1976.

(5) WCAP-8356, "Westinghouse ECCS Plant Sensitivity Studies," July 1974.

(6) WCAP-8339, "Westingnouse EC' Evaluation Model-Summary," July 1974.

(7) WCAP-8306, "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant," June 1974.

(8) WCAP-8326, "Containment Pressure Analysis Code (COCO)," June 1974.

(9) WCAP-8171, "Calculational Model for Core Reflooding After a Loss-of-Coolant Accident (WREFLOOD Code)," June 1974.

(10) WCAP-8471-P-A, "The Westinghouse ECCS Evaluation Model: Supplementary Information," April 1975.

ATTACHMENT 4

10 CFR 50.92
NO SIGNIFICANT HAZARDS CONSIDERATION
EVALUATION

VIRGINIA ELECTRIC AND POWER COMPANY

10 CFR 50.92
No Significant Hazards Consideration Evaluation

In accordance with the requirements of 10 CFR 50.91(a), the proposed change to the North Anna Power Station Unit 1 Facility Operating License has been evaluated against the criteria described in 10 CFR 50.92 and it has been determined that the proposed amendment to the operating license involves no significant hazards consideration. The basis for this determination is as follows:

North Anna Power Station Unit 1 is currently involved in a mid-cycle steam generator inspection outage. An extensive eddy current inspection of the North Anna Unit 1 steam generator tubes is being performed using very conservative analysis guidelines and plugging criteria. As such, a substantially increased number of tubes are expected to be plugged.

The predictions of potential steam generator tube plugging during the current mid-cycle outage are such that the effects of increased RCS loop resistance on the large break LOCA analysis would not permit full rated power operation for the remainder of North Anna Unit 1, Cycle 9. The existing large break LOCA analysis has obtained margin by taking credit for available Cycle 9 core characteristics and will not support 100% power operation with more than 30% steam generator tube plugging. The large break LOCA analysis presented in Sections 3.0 through 5.0 of the attached safety evaluation extends this steam generator tube plugging limit value to 35%, but with a reduced power level of 95% of rated thermal power. At this reduced power level, all analyses meet the requirements of 10 CFR 50.46 and Appendix K to 10 CFR Part 50.

Because the large break LOCA presents the limiting considerations for core power and total core power peaking, it was necessary to reduce the maximum core power level to 2748 megawatts (thermal) and the maximum allowable Hot Channel Peaking Factor (Fq) to 2.11 at the core mid-plane. The change to the power level is proposed as a modification to license condition 2.D.(1), Maximum Power Level, by adding a footnote limiting maximum reactor power to 2748 megawatts (thermal) until steam generator replacement is accomplished.

In addition, an associated change to the Technical Specifications is required to accommodate the effects of the revised assumptions for the large break LOCA analysis. The proposed change to the Technical Specifications will impose more restrictive equipment operability requirements for the Emergency Core Cooling System (ECCS). This is accomplished by modifying the Action Statement "a" of Specification 3.5.2 to ensure that both low head safety injection pumps or one low head injection pump and two high head safety injection pumps remain operable during power operation. This change effectively maintains consistency between the Technical Specification Action Statements and the revised assumptions for the large break LOCA analysis.

Further, a revised K(Z) surveillance function and a reduced Enthalpy Rise Hot Channel Factor were utilized to provide additional analysis margin. With these changes, the analysis supports power operation at up to 95% of rated thermal power for North Anna Unit 1 for the remainder of Cycle 9. Changes to the peaking factor and K(Z) surveillance function will be accomplished via the Technical Specifications Core Operating Limits Report (COLR).

The large break LOCA analysis assumed uniform steam generator tube plugging of 35% which supports operation with peak steam generator tube plugging levels up to 35%. With the exception of the parameters described above, which will be incorporated via the proposed license change and the forthcoming COLR, all analysis parameters were equivalent to, or conservative with respect to, those assumed in the existing analyses. All analysis parameters are expected to be conservative with respect to actual plant conditions for the remainder of North Anna Unit 1 Cycle 9.

Virginia Electric and Power Company has reviewed the proposed license condition change relative to operation of North Anna Unit 1 with increased steam generator tube plugging and determined that the proposed change does not involve a significant hazards consideration as defined in 10 CFR 50.92. The basis for this determination is that this change:

1. Does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The impact of the increased level of steam generator tube plugging (up to 35% peak) with a maximum reactor power of 95% on the large break LOCA was analyzed. The analysis demonstrated that operation with increased steam generator tube plugging will not result in more severe consequences than those of the currently applicable analyses.

   The probability of occurrence of these accidents is not increased, because an increased level of steam generator tube plugging as an initial condition for the accident has no bearing on the probability of occurrence of these accidents.

2. Does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The implementation of the increased steam generator tube plugging large break LOCA analysis into the North Anna Unit 1 design basis will not create the possibility of an accident of a different type than was previously evaluated in the UFSAR. No changes to plant configuration or modes of operation are implemented by the revised accident analysis. Therefore, no new mechanisms for the initiation of accidents are created by the implementation of the analysis.

3. Does not involve a significant reduction in a margin of safety.

   The North Anna Unit 1 operating characteristics, and accident analyses which support Unit 1 operation, have been fully assessed. The results of the revised large break LOCA analysis demonstrates that the consequences of this accident

are not increased as a result of the increased steam generator tube plugging up to 35% with a maximum reactor power of 95%. The results of the accident analysis remain below the limits established by the currently applicable analyses. Therefore, there is no significant reduction in the margin of safety.

Based on the above significant hazards consideration evaluation, Virginia Electric and Power Company concludes that the activities associated with this proposed license condition change satisfies the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.