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APPLICABILITY OF NOTRUMP TO THE SAN ONOFRE NUCLEAR GENERATING STATION UNIT 1

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SAN ONOFRE NUCLEAR GENERATING STATION UNIT 1

(SONGS-1)

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EXECUTIVE SUMMARY

The information in the following provides a general discussion of the features of the Westinghouse NOTRUMP small break LOCA emergency core cooling system (ECCS) Evaluation Model and the assessments performed to validate the applicability of the calculational methodology to the San Onofre Nuclear Generating Station, Unit 1 (SONGS-1). The NOTRUMP small break LOCA ECCS Evaluation Model is concluded to be applicable to SONGS-1 when the stainless steel fuel rod cladding model option is activated and passive ECCS accumulator flow models are deactivated.

For ECCS design basis small break LOCA analysis calculations, the fuel rod model in the calculation of the reactor coolant system transient to a small break LOCA for SONGS-1 assumes the stainless steel cladding material properties, with the exception of the fuel rod rupture calculation. The fuel rod cladding rupture calculation is not performed for the core average fuel rod modeled in the NOTRUMP calculation. Confirmatory calculations are performed to assure that cladding rupture for the core average fuel rod does not occur. The hot assembly fuel rod thermal transient calculations performed with the small break version of LOCTA-IV assumes stainless steel cladding material properties including a calculation of the potential for the fuel rod rupture.

The existing small break LOCA ECCS Evaluation Model nodalization for the reactor coolant system is applicable to SONGS-1 when the fluid volumes and connections representing flow from the passive ECCS accumulators, which are not present in the SONGS-1 ECCS design, are deactivated.

In the following, information is provided regarding the historical background and capabilities of the NOTRUMP computer code and the current Westinghouse small break LOCA ECCS Evaluation Model. A brief overview of the important transient phenomena observed in typical design basis small break LOCA analysis calculations is provided.

The key design features of the SONGS-1 plant are compared to the key design features of other three loop PWRs which have previously been analyzed and licensed using the NOTRUMP small break LOCA ECCS Evaluation Model. The previous applications of the NOTRUMP small break LOCA ECCS Evaluation Model are discussed in relation to the SONGS-1 key design features.

The results of a scoping calculation performed to assess the transient response for some of the key design features in the SONGS-1 plant are provided, which indicate that the SONGS-1 small break transient response as calculated with NOTRUMP will be similar to the transient response calculated for other typical three-loop plants.

As an appendix, information is provided to supplement previous fuel rod model information when stainless steel cladding is represented in the NOTRUMP computer code.

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(SONGS-1)

1.0 INTRODUCTION

Small break loss-of-coolant accident (LOCA) analyses are performed for the San Onofre Nuclear Generating Station Unit 1 (SONGS-1) to demonstrate compliance of the emergency core cooling system (ECCS) with the requirements of the Interim Acceptance Criteria[1]. The results of the analyses are reported in the SONGS-1 updated final safety analysis report (UFSAR)[2]. Currently the small break LOCA analyses reported in the SONGS-1 UFSAR are based upon the Westinghouse October-1975 small break LOCA ECCS Evaluation Model[3], which used the WFLASH[4] computer code and the small break version of the LOCTA-IV[5] computer code. The WFLASH computer code was used to conservatively simulate the reactor coolant system thermal hydraulic response to a break in the primary pressure boundary of equivalent diameter less than 1.0-square foot, while the small break LOCTA-IV computer code was used to calculate the thermal transient response of fuel rods in the hot assembly of the core. To analyze the SONGS-1 plant, the fuel rod cladding models were specifically modified to represent the stainless steel fuel rod cladding material properties. The WFLASH computer code, however, is not currently available at Westinghouse to perform small break LOCA analyses.

The current Westinghouse small break LOCA ECCS Evaluation Model incorporates the NOTRUMP analysis technology[6,7]. The NOTRUMP small break LOCA ECCS Evaluation Model was approved by the NRC in May 1985. The NOTRUMP small break LOCA ECCS Evaluation Model has an option to allow the calculation to be performed assuming that the cladding material is composed of stainless steel rather than Zircaloy, although the code description[6] does not provide explicit details of the fuel rod model when stainless steel cladding is assumed in the calculations.

In the following, information is provided regarding the applicability of the NOTRUMP small break LOCA ECCS Evaluation Model to SONGS-1. Information regarding the small break LOCA regulatory requirements and a historical perspective for the small break LOCA analysis for SONGS-1 is provided. The capabilities of the NOTRUMP computer code and the current Westinghouse small break LOCA ECCS Evaluation Model are discussed. The discussion indicates that the NOTRUMP computer code is capable of calculating the complex two-phase fluid flow and thermal-hydraulic transient response to a small break LOCA for SONGS-1. A brief overview of the key transient phenomena which occur during a design basis small break LOCA is provided in this regard. A discussion of previous applications of the NOTRUMP computer code is also provided to further indicate the wide range of conditions over which NOTRUMP has been successfully applied.

The key design features of the SONGS-1 plant are compared to the key design features of other typical Westinghouse designed three-loop PWRs which have previously been analyzed and licensed using the NOTRUMP small break LOCA ECCS Evaluation Model. A scoping calculation was performed to assess the transient response for some of the key design features in the SONGS-1 plant which differ from the typical three-loop plant. The results of the scoping calculation indicate that the thermal hydraulic transient response for SONGS-1 will be similar to the transient response for other Westinghouse designed three-loop plants. As an appendix, information is provided to supplement the information regarding the fuel rod model in the NOTRUMP computer code when stainless steel cladding is employed.

Based upon the information in this report, it is concluded that the current Westinghouse small break LOCA ECCS Evaluation Model incorporating the NOTRUMP analysis technology is applicable to the SONGS-1 plant with only minor modifications. Therefore, NOTRUMP small break LOCA ECCS Evaluation Model analysis calculations may be used to demonstrate compliance with the requirements of the Interim Acceptance Criteria for SONGS-1.

1.1 Historical Background

In 1980, Westinghouse modified the WFLASH computer program and the small break version of the LOCTA-IV computer programs to represent the effects of stainless steel fuel rod cladding. The modifications included models to represent the stainless steel cladding material properties for thermal conductivity, heat capacity, linear expansion, etc. The modifications were made to perform small break LOCA analyses calculation for SONGS-1. The analysis results were compared to the small break LOCA behavior of the generic plant analyses performed with WFLASH[4]. A spectrum of breaks was analyzed and the results incorporated into the SONGS-1 updated safety analysis report (USAR)[2]. The analysis results demonstrated compliance with the Interim Acceptance Criteria[1].

1.1.1 Interim Acceptance Criteria

On June 29, 1971, the Atomic Energy Commission published its Interim Policy Statement[1], "Interim Acceptance Criteria (IAC) for Emergency Core Cooling Systems for Light-Water Power Reactors". This policy required analytic demonstration that the design of the ECCS was sufficient to meet the following criteria;

1. The calculated maximum fuel element cladding temperature does not exceed 2300°F,

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

3. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling, and

4. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long lived radioactivity remaining in the core.

In section IV.B., the IAC also required an evaluation of ECCS performance for each reactor using a suitable evaluation model. Appendix A to the IAC policy specified that suitable Evaluation Models were the AEC Evaluation Model for Pressurized Water Reactors, the General Electric Evaluation Model, or the Westinghouse Evaluation Model[8]. Use of one of these three models was acceptable to the Commission, but its use was not mandatory. ECCS analysis calculations using other models could be submitted to the Commission for review and approval.

The original analytical techniques found acceptable by the Commission for the Westinghouse IAC Evaluation Model[8] were documented in Reference [8] and additional restrictions were specified in Appendix A to the IAC policy. The SLAP computer program[8] was specified in the Westinghouse IAC Evaluation Models as the tool used to calculate the reactor coolant system (RCS) response to a small break loss-of-coolant accident (LOCA). By current standards, as with many of the models used to license nuclear power plant operation under the Interim Acceptance Criteria, relatively simple analytical models and techniques were employed in the SLAP computer program. SONGS-1 small break LOCA analysis calculations were originally performed using the SLAP code. The SLAP computer program was replaced by the WFLASH[4] computer program and the small break version of the LOCTA-IV[5] computer program.

1.1.2 Final Acceptance Criteria

In 1974, the Final Acceptance Criteria (FAC)[9], set forth in 10CFR50.46 as the Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors specified the ECCS analysis requirements for plants fueled with uranium oxide pellets within cylindrical Zircaloy cladding. Plants utilizing stainless steel clad continued to be licensed under the IAC policy.

With the advent of the Final Acceptance Criteria in 1974, the Westinghouse small break LOCA ECCS Evaluation Model incorporating the WFLASH computer program was used exclusively for analysis of the ECCS response to small break LOCAs in Westinghouse plants with Zircaloy clad fuel.

1.1.3 Post-TMI Requirements

Following the accident at Three Mile Island Unit 2, the Nuclear Regulatory Commission (NRC) staff focused additional attention on the small break loss-of-coolant accident (LOCA) and the analyses performed to demonstrate that the emergency core cooling system (ECCS) can meet the requirements of 10CFR50.46. Westinghouse submitted information[10] to the NRC providing details regarding the performance of the Westinghouse small break LOCA ECCS Evaluation Model which utilized the WFLASH computer program.

In NUREG-0611[11], the NRC staff outlined technical issues regarding the capability of certain models in the WFLASH computer program to simulate the reactor coolant system response to a small break LOCA. While specific models in WFLASH, such as the thermal equilibrium assumption relative to accumulator injection flow, were not able to predict the exact response of the physical phenomena, Westinghouse maintained that the overall ECCS Evaluation Model using the WFLASH computer program was suitably conservative.

Section II.K.3.30 of Enclosure 3 to NUREG-0737[12] clarified the Post-TMI requirements of the NRC regarding small break LOCA modeling. Section II.K.3.30 of NUREG-0737 required that the licensees revise the small break LOCA ECCS models along the guidelines specified in NUREG-0611 or justify the continued acceptance of the model. Furthermore, in section II.K.3.31 of Enclosure 3 to NUREG-0737, the NRC required that each licensee submit a new small break LOCA analysis using an NRC approved small break LOCA Evaluation Model which satisfied the requirements of NUREG-0737 section II.K.3.30.

1.1.4 Development of NOTRUMP

In response, the Westinghouse Owners Group directed Westinghouse to develop the NOTRUMP[6] computer program for reference in new small break LOCA ECCS Evaluation Model[7] calculations, based on the desire of the WOG to perform licensing evaluations with a computer program specifically designed to calculate small break LOCAs with greater phenomenological accuracy than capable with the WFLASH computer program.

The use of NOTRUMP for application to small break LOCA ECCS Evaluation Model analyses was approved by the NRC in May 1985. The staff concluded that the Westinghouse small break LOCA ECCS Evaluation Model incorporating the NOTRUMP computer program was acceptable for performing licensing calculations in compliance with Section II.K.3.30 of Enclosure 3 to NUREG-0737 for all Westinghouse designed nuclear steam supply systems. Subsequently, Westinghouse also received NRC approval for application of the NOTRUMP small break LOCA ECCS Evaluation Model to Combustion Engineering designed nuclear steam supply systems[13].

NRC Generic Letter 83-35[14] relaxed the requirements of Section II.K.3.31 of Enclosure 3 to NUREG-0737 by allowing a more generic response and providing a basis for retention of the existing small break LOCA analyses. Provided that the previously existing model results with the WFLASH computer program were demonstrated to be conservative with respect to the new small break LOCA model approved under the requirements of NUREG-0737 II.K.3.30 (NOTRUMP), plant specific analyses using the new small break LOCA Evaluation Model would not be required. To satisfy Section II.K.3.31 of Enclosure 3 to NUREG-0737, licensees had the option of performing plant specific calculations using a small break LOCA model approved under the requirements of NUREG-0737 II.K.3.30, or referencing generic calculations performed using a small break LOCA model approved under the requirements of NUREG-0737 II.K.3.30.

Westinghouse and the Westinghouse Owners Group demonstrated in generic studies that the results obtained from calculations with the WFLASH small break LOCA Evaluation Model were, in general, conservative relative to those obtained with the NOTRUMP small break LOCA Evaluation Model[15]. Licensees could then demonstrate compliance with Section II.K.3.31 of Enclosure 3 to NUREG-0737 by referencing the generic studies and providing some plant specific information.

1.1.5 <u>Reference to NOTRUMP for SONGS-1</u>

Following the NRC approval and issuance of the SER for the NOTRUMP small break LOCA ECCS Evaluation Model and NRC approval and issuance of the SER for the generic analyses[15], an assessment was performed in which it was judged that representation of stainless steel rather than Zircaloy cladding would have no appreciable effect on the calculated response to the small break LOCA for the NOTRUMP generic analyses[15]. The requirements of Section II.K.3.31 of Enclosure 3 to NUREG-0737 were then satisfied for SONGS-1 by referencing WCAP-11145 and indicating that the the WFLASH analyses would be conservative relative to NOTRUMP analyses performed on a comparable basis.

1.2 NOTRUMP Modeling Capabilities

NOTRUMP is an advanced two-phase thermal-hydraulics computer code developed by Westinghouse. It includes detailed fluid flow and heat transfer models to accurately represent the thermal-hydraulic phenomena related to two-phase mass and energy convection. Special models are available to accurately represent the effects of two-phase flow, interfacial heat and mass transport, phase separation, and counter-current flow limitations for various configurations. Extensive heat transfer correlations represent regimes from liquid convection, through nucleate and transition boiling to stable film boiling, forced convection vaporization, and steam forced convection.

The spatial detail of the fluid system is modeled by elemental control volumes (nodes), interconnected by paths (links). The spatial and temporal solution is governed by the integral forms of the conservation equations in the nodes and links. The flexible noding capability in NOTRUMP permits a detailed full nodal treatment of both the primary and secondary sides of a nuclear power plant. NOTRUMP has a detailed momentum balance that permits an accurate calculation of inventory distribution among the fluid nodes and flow links. The drift flux and bubble rise models in NOTRUMP permit modeling of vertical slip flow including counter-current flow using flow regime maps. The treatment of phase separation (both natural and forced) permits an accurate calculation of the two-phase mixture level response within the primary and secondary reactor coolant systems in a pressurized water reactor.

The capability of realistically calculating the complex thermal-hydraulic response of single and two-phase fluid flow under various conditions has permitted the application of NOTRUMP to a broad spectrum of problems ranging from design basis small break LOCAs to natural circulation flow on the secondary side of steam generators. Examples of some of the applications of the NOTRUMP computer code are provided below;

1. Secondary side transients to address the consequences of postulated main feedline ruptures[16],

2. Severe primary side accident scenarios to examine various recovery actions to mitigate the consequences of inadequate core cooling scenarios when auxiliary feedwater is available in both UHI and non-UHI plants[17,18],

3. Transient response to small break LOCAs in support of studies of reactor vessel integrity issues for Westinghouse operating plants[19], and

4. Studies of two-phase natural circulation to address concerns related to the phenomena and recovery processes[20,21].

The NOTRUMP computer code and small break LOCA analysis methodology have been evaluated and approved by the NRC for use in calculating the performance of the ECCS to design basis small break LOCAs for both Westinghouse and Combustion Engineering NSSS designs in compliance with the requirements of Appendix K to 10CFR50. With the inherent two-phase thermal hydraulic capability, NOTRUMP can be used to evaluate the system response for a wide range of different reactor coolant system configurations under a wide range of accident analysis conditions.

1.3 Application of NOTRUMP to Small Break LOCA ECCS Analyses

In the current version of the Westinghouse small break LOCA ECCS Evaluation Model, the NOTRUMP computer code has replaced the WFLASH computer code in the calculation of the reactor coolant system transient response to the small break LOCA. The hot assembly fuel rod transient thermal performance is still calculated with the small break version of the LOCTA-IV computer code. The NOTRUMP computer code has the option of representing the stainless steel fuel rod cladding material properties. For application to SONGS-1, the NOTRUMP stainless steel cladding option is activated and modifications are made to represent the SONGS-1 specific configuration. The small break version of LOCTA-IV is modified, as was done in the previous small break LOCA analyses for SONGS-1, to represent the stainless steel cladding in the hot assembly fuel rod transient calculations.

In NOTRUMP small break LOCA ECCS Evaluation Model analyses, the primary and secondary reactor coolant system fluid volume spatial detail is represented by a network of fluid nodes interconnected by flow links. The structural metal mass is represented by metal nodes which are interconnected by heat links to represe * various heat transfer paths between metal structures and surrounding fluid. The NRC approved noding schemes for the NOTRUMP small break LOCA ECCS Evaluation Model analyses are shown in Figures 3-14-1 and 3-14-2 of Reference 7. The existing NOTRUMP small break nodalization of the reactor coolant system will be used for the analysis of SONGS-1, with the exception that the fluid volumes and connections representing the passive ECCS flow accumulators will be deactivated since the SONGS-1 ECCS design does not incorporate these features.

NOTRUMP utilizes state-of-the-art drift velocity bubble rise models to calculate phase separation within the fluid nodes. Various flow regime dependant drift velocity models are employed to allow NOTRUMP to accurately calculate the phase separation through out the RCS. NOTRUMP's state-of-the-art phase separation capabilities are enhanced by the node stacking and mixture level tracking capability. Multiple fluid nodes may be vertically stacked to accurately represent void fraction gradients wnile allowing a single mixture level to be tracked. This eliminates the unrealistic calculation of distinct mixture levels within each fluid volume in a set of vertically stacked non-homogenous fluid volumes. This feature allows NOTRUMP to utilize a more detailed nodalization of the RCS in order to accurately track the various density gradients and mixture levels within the reactor coolant system which are characteristic of a small break LOCA.

The convective transport of fluid mass around the reactor system is modeled in NOTRUMP with flow links which interconnect the fluid volumes. A single momentum equation is used to calculate flow through a single flow link. For two-phase flow situations NOTRUMP models the two-phase composition and relative slip between the phases in vertical flow paths with state-of-the-art flow regime dependent drift flux models. The drift flux models are consistent with, and complement the advanced fluid node phase separation capabilities described earlier.

The drift flux models are able to treat both co-current and counter-current flow, with counter-current flow limited by flooding correlations. For horizontal flow paths co-current and counter-current two-phase flow and slip may be modeled. The relative slip between the phases is modeled through the use of a horizontal interfacial shear correlation. This provides NOTRUMP with the capability to realistically represent phase separation and mass and energy transport throughout the reactor coolant system. In particular, liquid which accumulates in the hot legs and upflow side of the steam generators is accurately calculated to drain back into the reactor vessel upper plenum once two-phase co-current natural circulation has stopped. As a result, NOTRUMP calculates a more realistic mass inventory distribution than was calculated by the conservative WFLASH model, particularly in the core region. No changes to the drift flux and bubble rise models are necessary for application of NOTRUMP calculations to SONGS-1.

Thermal non-equilibrium is permitted within each fluid node between the upper, predominantly vapor, region and the lower, predominantly liquid region in the NOTRUMP small break LOCA ECCS Evaluation Model. This feature is essential in order to realistically model the non-equilibrium behavior that occurs in the reactor system during a small break LOCA.

Thermal non-equilibrium is particularily significant in the cold leg piping where subcooled liquid and saturated or superheated steam will co-exist as a result of the local injection of relatively cold ECCS water. The magnitude of the non-equilibrium interaction that occurs in the cold leg will influence the system transient response to a small break LOCA. The thermal non-equilibrium capability in NOTRUMP is complemented by mechanistic interfacial heat transfer models which calculate the heat transfer between the mixture and vapor regions within a node, should a thermal non-equilibrium condition exist. Of particular significance is the mechanistic interfacial heat transfer model that was developed specifically to represent the energy exchange between safety injection fluid and the saturated or superheated steam which surrounds the injection region. This model was developed to conservatively underpredict the interfacial energy exchange. For SONGS-1 the conservatism in the model has a larger effect due to the higher ECCS injection flow rates at the lower pressure. Since the existing model is conservative, however, it continues to be used in NOTRUMP analysis calculations for SONGS-1.

The thermal non-equilibrium capability also contributes to the more realistic modeling of energy transport between the primary and secondary sides of the reactor coolant system. Energy transport between the primary and secondary sides is important during a small break LOCA transient since it affects the pressure and mass distribution calculations. The NOTRUMP thermal non-equilibrium model results in a more realistic calculation of the heat transfer process.

Prior to the venting of steam through the pump suction leg loop seal, the steam generators act as heat sinks and are a significant mechanism for decay heat removal. Following the venting of steam through the pump suction leg loop seal, the steam generators tend to act as heat sources, as the primary reactor coolant system depressurizes below the steam generator secondary side pressure. During this period various modes of heat transfer are encountered on both the primary and secondary sides of the steam generator. These modes of heat transfer include subcooled forced convection, subcooled natural convection, two-phase mixture condensation, steam condensation, forced convection vaporization, nucleate boiling, steam forced convection, and steam natural convection. Each of these heat transfer regimes has its own heat transport characteristics which must be individually modeled in order to provide a realistic representation of the influence of the heat transfer process between the primary and secondary on the rest of the reactor coolant system. NOTRUMP utilizes mechanistic heat transfer models for each of these modes. These models are not changed for application of NOTRUMP small break LOCA calculations for SONGS-1.

The reactor coolant system mass inventory is to a large degree determined by the break flow models. For two-phase break flow, the Moody model is used as required by Appendix K to 10CFR50. For subcooled stagnation conditions, the NOTRUMP small break LOCA ECCS Evaluation Model uses the modified Zouledek model. These models are not changed for application of NOTRUMP small break LOCA calculations to SONGS-1.

The small break LOCA may be characterized by the relatively slow draining of the reactor coolant system and is accompanied by the formation of distinct mixture levels throughout the reactor coolant system. These mixture levels vary with time and are dependent upon the transient two-phase transport of mass and energy within the reactor coolant system during the course of the accident. Consequently, the degree of accuracy with which a system model is capable of simulating the transient response is dependent upon the capability of the model to accurately represent the transient mass and energy distribution. The NOTRUMP code can accurately calculate the transient mass and energy distribution and is therefore appropriate for application to small break LOCA ECCS Evaluation Model analysis calculations for the SONGS-1 plant.

1.4 Important Small Break LOCA Transient Phenomena

For any plant, the small break LOCA transient response is a function of the design of the facility, the size and location of the break, the assumptions regarding the availability of off-site power as it governs the operability of various auxiliary systems, the ECCS engineered safeguards characteristics, and the core power level. For design basis small break LOCAs in Westinghouse designed NSSS, the limiting break location is in the cold leg.

Following a small rupture of piping in the cold leg of the reactor coolant system, in which the primary fluid inventory loss exceeds the charging fluid makeup capability, depressurization will result in a reactor trip on low pressurizer pressure if not earlier from other engineered safeguards actions. Insertion of all but the most reactive rod control cluster assembly complements possible void formation in the core to result in a decrease of the heat generation rate to residual decay heat levels. The safety injection system is actuated when the appropriate setpoint, such as low low pressurizer pressure, is reached.

Depending upon the size of the break and the safety injection flow rate, the reactor coolant system may continue to depressurize and distinct mixture levels may develop. If the volumetric rate of the reactor coolant system fluid lost through the break plus the volumetric rate of fluid shrinkage due to condensation exceeds the volumetric rate of safety injection fluid into the reactor coolant system plus the vapor volumetric generation rate from decay heat induced fluid boiling and fluid flashing due to depressurization, the reactor coolant system will continue to depressurize until a balance is reached. The core decay heat level as influenced by the prior core operating conditions will affect the decay heat induced fluid boiling.

During the period when the break flow is all liquid, the reactor coolant system will depressurize to a pressure near the steam generator secondary side pressure. Consequently the secondary side conditions during the small break LOCA have an important influence on the transient response. If off-site power is lost, the secondary side may pressurize to the main steam safety valve setpoints. The secondary energy removal capability is then a function of the secondary side safety valve setpoint, the auxiliary feedwater flow rate, and the auxiliary feedwater temperature.

Operation of the reactor coolant pumps during the transient can have an important influence on the transient response since they can affect the mass distribution in the system and the quality of fluid at the location of the break. Higher mass flow rates due to the action of the reactor coolant pumps tends to preclude cladding heatup beyond the ambient fluid temperature.

After sufficient mass depletion, flow through the break will make a transition to two-phase or all-vapor flow. For breaks in the cold leg of the reactor coolant system, vapor generated by the core decay heat must vent through the pump suction leg loop seal section to exit through the break. During the loop seal steam venting process the inner vessel mixture level will be depressed as the inner vessel and downcomer responded to manometric pressure imbalance between the upper plenum and the top of the downcomer. This imbalance occurs in response to the hydrostatic head of water which remains in the upflow section of the pump suction leg loop seals as the liquid level in the downflow section of the loop seals is depressed below the cold leg elevation. As soon as steam is vented through the loop seal this pressure imbalance is relieved and the inner vessel mixture level quickly rises. The loop seal steam venting process affects the break flow response which affects the amount of liquid which remains in the reactor vessel after loop seal steam venting. Following loop seal steam venting the amount and distribution of mass in the reactor vessel will also be influenced by the design of the reactor vessel internals. Consequently, the primary reactor coolant system volume to core power ratio has an important influence on the transient response.

If there is sufficient mass inventory depletion from the primary reactor coolant system, the fuel rods will not remain covered by a two-phase mixture. When this occurs, decay heat removal from the fuel rods is less effective and fuel rod heatup will result. The decay heat power distribution, decay heat level, and fuel rod design then have an important influence on the transient response.

Finally, the ECCS design and safety injection flow characteristics will have an important influence on the small break LOCA transient response. The ECCS design and safety injection flow characteristics governing the rate of mass inventory replacement which affects the primary side mass inventory balance. The pressure and rate of ECCS flow will then have a significant effect the fuel rod heatup transient. No changes to the NOTRUMP small break LOCA ECCS Evaluation Model are necessary to represent the SONGS-1 ECCS design, except for deactivation of the accumulator models which is accomplished through user inputs.

2.0 RANGE OF APPLICATIONS OF NOTRUMP SMALL BREAK LOCA ANALYSES

The NOTRUMP small break model has been applied to various transient simulations ranging from design basis ECCS analyses to support the licensing of nuclear power reactors to the simulation of loss of coolant tests in integral facilities such as Semi-Scale and LOFT, to the simulation of secondary side transients.

2.1 Comparison of Key Plant Parameters

The design of the SONGS-1 plant was examined to identify differences relative to those in typical three-loop PWRs which could affect the ability of the NOTRUMP small break LOCA ECCS Evaluation Model to accurately calculate the SONGS-1 transient response. Some of the key small break LOCA ECCS analysis parameters which have an important influence on the transient response were examined. The following parameters were compared for SONGS-1 and a typical Westinghouse designed three-loop plants to identify areas in the NOTRUMP small break LOCA ECCS Evaluation Model which require additional consideration:

KEY PLANT PARAMETERS	SONGS-1	TYPICAL 3-LOOP
Volume/Power (cu.ft./MWt)	5.15	3.22
Steam Generator	27	D-3
Reactor Coolant Pump	63	93A
MSSV Setpoint (psia)	985	1200
Fuel Rod Length (ft)	Approximately 10	Approximately 12
Fuel	UO_2	UO_2
Fuel Rod Cladding	SS	Zr-4
Core Power Level (MWth)	1347	2775
FQT	2.78	2.45
F-delta-H	1.57	1.62
Accumulator Pressure (psia)	N/A	Approximately 615
Pumped Safety Injection	MAIN FW	HHSI

The reactor coolant system volume to power ratio affects the small break LOCA transient response by affecting the rate of mass inventory depletion thereby affecting the timing of the small break LOCA transient response and the size of the limiting break. While a typical three loop plant has a volume to power ratio of approximately 3.4, NOTRUMP small break LOCA analyses have been performed for volume to power ratios ranging from approximately 3.15 to 5.38.

The steam generator type influences the small break LOCA transient response by affecting the primary to secondary heat transfer capability and the propensity for counter current reflux liquid flow to the reactor vessel upper plenum through the hot legs. Since the decay heat load is small relative to the available steam generator heat transfer surface area in all steam generator designs, the design differences would have an insignificant effect on the transient response. For SONGS-1 with a uniform steam generator tube plugging level of 20% as modeled in the large break LOCA analysis, the resulting primary pressure could show an increase of approximately 9 psi during the quasi-equilibrium pressure period when compared to a typical three-loop plant with a uniform steam generator tube plugging level 0%, assuming the same main steam safety valve setpoints.

The propensity to inhibit counter current reflux flow is slightly higher in the SONGS-1 steam generators assuming a 20% uniform steam generator tube plugging level than it is in a typical three-loop plant steam generator assuming 0% uniform steam generator tube plugging. However, levels of steam generator tube plugging higher than 20% have been represented in NOTRUMP small break LOCA ECCS analyses. The NOTRUMP counter current flow limitation models are applicable to equivalent levels of steam generator tube plugging of approximately 30%. Above that level the models tend to become overly conservative. Therefore the difference in steam generator type would not adversely affect the ability to apply NOTRUMP to the SONGS-1 plant.

The main steam safety valve (MSSV) setpoint affects primary side pressure by affecting the heat transfer between the primary and secondary sides. A lower MSSV setpoint pressure would have a beneficial effect on the small break LOCA transient response. MSSV pressures lower than the SONGS-1 values have been modeled in NOTRUMP small break LOCA analyses. Therefore this difference will not adversely affect the ability to apply NOTRUMP to the SONGS-1 plant.

While the SONGS-1 Model-63 reactor coolant pumps show some design differences when compared to Model-93A reactor coolant pumps in typical three loop plants, the homologous curves are very similar. Since the homologous pump curves are used as input in the NOTRUMP small break LOCA ECCS Evaluation Model calculations, the difference in reactor coolant pump design is considered to be insignificant to the analysis calculations and will not adversely affect the ability to apply NOTRUMP to the SONGS-1 plant.

The SONGS-1 fuel rod differs from the fuel rods used to perform previous NOTRUMP small break LOCA ECCS Evaluation Model analyses only in the length and cladding material type. NOTRUMP small break LOCA analyses have been performed for different fuel rod lengths in the analysis of Westinghouse designed plants with core lengths of 14-feet, the loss-of-fluid-test (LOFT) facility with a core length of 5.5-feet, and in the analysis of Combustion Engineering NSSS designs. Since the length of the fuel rod may be appropriately represented in the NOTRUMP small break LOCA ECCS Evaluation Model analyses, this difference will not adversely affect the ability to apply NOTRUMP to the SONGS-1 plant. The effect of the fuel rod cladding type will be discussed further in Section 2.2.

The ECCS design for SONGS-1 differs from typical three-loop plants in following ways;

1. The pumped safety injection flow is provided by switching the main feedwater pumps into the safety injection mode,

2. The safety injection flow shutoff head in the SONGS-1 ECCS design is lower than the shutoff head in typical ECCS designs, but provides substantially higher flow at lower pressures, and

3. The ECCS design does not incorporate passive ECCS flow accumulators.

The effect of the ECCS design differences are discussed in greater detail in Section 2.2.2.

2.2 SONGS-1 Design Differences Affecting Small Break LOCA Analyses

The only major design differences between SONGS-1 and a typical three-loop plant analysis which affect the ability to apply the NOTRUMP small break LOCA ECCS Evaluation Model to SONGS-1 are the fuel rod cladding material design and the ECCS safety injection design. The following provides a discussion of the fuel rod and ECCS design differences.

2.2.1 SONGS-1 Fuel Rod Cladding Design Effects

The SONGS-1 stainless steel fuel rod cladding affects the ability to demonstrate that the IAC criteria are met in the following manner;

1. To demonstrate that the calculated maximum fuel element cladding temperature does not exceed 2300°F, models for stainless steel cladding thermal conductivity, heat capacity, emissivity, oxidation, swelling, and rupture properties are necessary.

2. To demonstrate that the amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of cladding in the reactor, a model for the stainless steel metal water reaction is necessary.

3. To demonstrate that the clad temperature transient is terminated at a time when the core geometry is still amenable to cooling, a model for the stainless steel metal water reaction is necessary to demonstrate that the oxidation limits are not exceeded.

Models for the stainless steel material properties are available as an option in the NOTRUMP computer code. Special models were developed to conservatively represent the stainless steel metal and water reaction rate and to represent rupture of the stainless steel cladding. To perform small break LOCA analysis calculations for SONGS-1, the NOTRUMP computer code is modified to incorporate the stainless steel oxidation model and the small break version of LOCTA-IV is modified to represent the stainless steel material properties, the stainless steel oxidation model, and the stainless steel fuel rod rupture model. The stainless steel rupture model is not activated in the NOTRUMP computer code. Confirmatory calculations are performed to verify that the incidence of rupture of the core average fuel rod is not underestimated in the calculations.

2.2.1.1 Cladding Oxidation Rates

The importance of the cladding oxidation in the small break LOCA analysis response may be examined by comparing the stainless steel oxidation rate to the Zircaloy oxidation rate. The following stoichiometry represents the oxidation process for Zirconium and stainless steel:

Zirconium	$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$
Wustite	$Fe + H_2O \rightarrow FeO + H_2$
Hematite	$2Fe + 3H_2O \rightarrow Fe_2O_3 + 3H_2$
Magnetite	$3Fe + 4H_2O \rightarrow Fe_3O_4 + 4H_2$

The oxidation rate of materials may be compared by examining the rate of oxygen uptake in the material. Generally, the equation for the oxidation is given by:

$$w^2/t = K_o \exp [\Delta H / (R \times T)$$

where

w = weight gain, mg $(O_2)/cm^2$ of area

K_o = constant, mg^2/cm^4 - sec

ΔH = activation energy of reaction cal/mole

R = gas constant, cal/mole - ^oK

T = absolute temperature, ^oK

t = time, seconds

An empirical linear correlation of the oxidation reaction rate for Zirconium is presented by White[22].

$$W^2/t = 7.1 * 10^6 \exp [(-40,500 \pm 1,200)/(R*T)]$$

Values obtained from the correlation by White may be compared to the conservative Baker-Just[23] equation.

$$W^2/t = 4.1 * 10^6 \exp [(-45,000)/(R*T)]$$

Bittel[24] calculated an expression for the oxidation of stainless steel.

$$W^2/t = 2.4 * 10^{12} \exp [(-84,300 \pm 2,400)/(R*T)]$$

By equating these expressions, it is possible to determine the temperature at which the oxidation rate for the two materials is equal. Comparing White's experimental equations for Zirconium to stainless steel, the temperature of equal oxidation rate is approximately 1,960°F. Comparing the experimental data for stainless steel to the Baker-Just equation for Zirconium, the equal oxygen uptake point is approximately 2,015°F. The value obtained for White's experimental data provides a more conservative relationship for equal oxidation rate temperature than the conservative Baker-Just equation. As long as the stainless steel cladding remains below 1960°F, the oxidation rate for the Zircaloy cladding will bound that of stainless steel.

Similarly, the point at which the exothermic metal and water reaction for stainless steel cladding will approximately equal the metal and water reaction for Zircaloy may be calculated. Since the amount of heat generated by the stainless steel and water reaction is substantially lower than that for the Zircaloy and water reaction, the temperature at which the exothermic heat releases are equal is higher than the point at which the stainless steel and water reaction equals the Zirconium and water reaction. A conservatively low estimate of the temperature at which the exothermic heat release for stainless steel and water reaction equals the heat release for the Zircaloy and water reaction is approximately 2125°F.

A model was developed to represent the stainless steel and water reaction in NOTRUMP small break LOCA analysis calculations. The performance of this model does not become conservatve, relative to the Zircaloy and water reaction rate until the cladding temperature would exceed approximately 1960°F. If the cladding temperature is below 1960°F, the Zircaloy models provide a conservative estimate of the oxidation and heat generation rate. While either the Zircaloy or stainless steel models may be used to calculate the cladding oxidation rate below temperatures of 1960°F, only the stainless steel cladding oxidation model will be used for temperatures greater than 1960°F.

2.2.1.2 Cladding Swelling and Rupture

When the fuel rod internal pressure exceeds the reactor coolant system pressure, the fuel rod cladding may swell or rupture at the high temperature conditions characteristic of a loss-of-coolant accident. The incidence of rupture is a function of the cladding temperature and the difference between the fuel rod internal pressure and the reactor coolant system pressure. The cladding ductility affects the fuel rod internal pressure by governing the rate of cladding strain. Fuel rod rupture will occur when the cladding temperature exceeds the rupture temperature for the cladding hoop stress.

Typically, fuel rod rupture is not calculated to occur in Westinghouse small break LOCA ECCS Evaluation Model analyses with Zircaloy cladding models due to one or more of the following reasons:

1. The peak cladding temperature is relatively low, (typically less than approximately 1900°F),

2. The hoop stress is relatively low due to the high reactor coolant system pressures for the limiting size small break LOCAs,

3. The hoop stress is relatively low due to the relatively low rod internal pressure characteristic of the limiting time in life for small break LOCA analyses.

However, stainless steel has a lower rupture ductility than does Zircaloy cladding. Above 1500°F the stainless steel ductility remains below that of Zircaloy. Single rod burst tests indicate that swelling is only approximately two-thirds that observed in Zircaloy cladding. Thus stainless steel has a lower propensity for swelling than Zircaloy. The hoop stress for cladding rupture in stainless steel is approximately five times greater than that for Zircaloy. The strength considerations for stainless steel cladding make the degree of high temperature creep substantially lower than that observed in Zircaloy cladding.

A model for stainless steel cladding rupture as a function of the hoop stress was developed for use in the Westinghouse small break LOCA ECCS Evaluation Model analysis for SONGS-1. The stainless steel rupture model is not activated for the core average rod calculations in the NOTRUMP computer code analysis of the reactor coolant system response to a small break LOCA. Confirmatory calculations are performed to verify that the incidence of rupture, however, for the core average fuel rod is not underestimated. The stainless steel rupture model is activated for the hot assembly hot rod and average rod calculations in the small break version of the LOCTA-IV computer code.

Based upon the strength considerations and the lower propensity for stainless steel fuel rod cladding swelling, the cladding swelling model will be limited to the thermal expansion. This is conservative for calculations of the peak cladding temperature and fuel rod rupture. If rupture is calculated to occur, the existing flow blockage model in the NOTRUMP small break LOCA ECCS Evaluation Model is used.

2.2.2 SONGS-1 ECCS Design Effects

Typical Westinghouse three-loop plants have passive safeguards ECCS flow available from the accumulators. The passive accumulator flow significantly helps to mitigate the consequences of larger size small break LOCAs. The SONGS-1 ECCS design does not incorporate accumulators. However, the pumped safety injection flow in SONGS-1, at typical accumulator initiation pressures, is substantially larger than the flow available from the pumped safety injection systems in typical three-loop plants. The differences in the ECCS design could affect the thermal hydraulic transient response which could affect the ability of the NOTRUMP small break LOCA ECCS Evaluation Model to calculate the appropriate response to a small break LOCA. To assess the potential effects, a scoping analysis calculation was performed in which the pumped safety injection flow was scaled on a core power to safety injection flow basis to that of SONGS-1 and the the passive ECCS flow from the accumulators was disabled. The results of the scoping calculation are presented in Section 3.0

2.2.3 Conclusions

NOTRUMP has been applied to small break LOCA analysis calculations over a range of parameters which bounds the SONGS-1 design, except for the stainless steel fuel rod cladding and the passive ECCS accumulator flow.

The stainless steel material properties are included as an option in the NOTRUMP small break LOCA ECCS Evaluation Model. The material properties have only a small effect on the the transient calculation results. Special models for stainless steel cladding oxidation, swelling, and rupture were developed. Reactor coolant system transient analysis calculations using the NOTRUMP computer code utilize fuel rod cladding models for the stainless steel material properties and cladding oxidation. The thermal transient response of the hot rod and average rod in the hot assembly as calculated by the small break version of the LOCTA-IV computer program would utilize models for the stainless steel cladding material properties, oxidation, and rupture.

To assess the potential effects of the ECCS design differences, a scoping analysis was performed with the important features of the SONGS-1 ECCS design represented. The scoping analysis is presented in Section 3.0.

Appendix A provides information regarding the fuel rod models incorporated in the NOTRUMP computer code when the option for stainless steel cladding is specified. An evaluation of the performance Validation of the models is is discussed in Section 4.0.

3.0 SCOPING CALCULATION FOR APPLICABILITY OF NOTRUMP TO SONGS-1

As noted in Section 2.2, the SONGS-1 ECCS design differs from typical three-loop plants which have been previously analyzed using NOTRUMP. The ECCS design does not incorporate passive ECCS accumulator flow, while the pumped safety injection flow rate is substantially higher than that found in typical three-loop plants at the typical accumulator minimum setpoint pressure. A scoping calculation was performed to determine if fluid flow conditions during a small break LOCA transient in SONGS-1 would look different from the results obtained for previously analyzed three-loop plants. It was concluded that the thermal-hydraulic response and ECCS performance is approximately the same in SONGS-1 as it is in typical three loop plants.

3.1 Introduction

A NOTRUMP input deck for a typical three-loop plant licensed to a core power level of 2775 MWth was obtained. The NOTRUMP input deck for the typical three-loop plant assumed values appropriate for Model D3 steam generators with a uniform tube plugging level of 20%, main steam safety valve setpoints beginning at approximately 1200 psia, 17x17 Vantage-5 fuel, and a downflow barrel baffle. Only those input changes necessary to represent the unique SONGS-1 ECCS design features were made. This included the following changes;

1. Deactivating the fluid volumes representing the accumulators,

2. Deactivating the flow links connecting the accumulators to the reactor coolant system,

3. Scaling the pumped safety injection flow rate equivalent to that in SONGS-1 based on the ratio of the pumped safety injection flow to core power, and

4. Incorporating the safety injection flow delay time appropriate to SONGS-1.

Based upon these changes a break transient analysis was performed, and the preliminary results reviewed.

3.2 Differences in Key Plant Parameters

The design of the SONGS-1 plant was examined and compared to the design of the plant to be used for the scoping analysis to identify design differences which could affect the NOTRUMP simulation of the break transient. The following SONGS-1 key plant parameters were compared to the plant parameters for the scoping analysis:

KEY PLANT PARAMETERS	SONGS-1	TYPICAL 3-LOOP
Volume/Power (cu.ft./MWt)	5.15	3.22
Steam Generator	27	D-3
Reactor Coolant Pump	63	93A
MSSV Setpoint (psia)	985	1200
Fuel Rod Length (ft)	Approximately 10	Approximately 12
Fuel	UO_2	UO_2
Fuel Rod Cladding	SS	Zr-4
Core Power Level (MWth)	1347	2775
FQT	2.78	2.45
F-delta-H	1.57	1.62
Accumulators	N/A	Deactivated
Pumped Safety Injection	MAIN FW	SCALED

Since an existing NOTRUMP analysis input data set was used to perform the scoping analysis for SONGS-1, all of the key analysis parameters were not able to be appropriately represented. The following key analysis parameters differences were considered:

1. Reactor Coolant System Primary Volume to Core Power Ratio:
The reactor coolant system primary volume to core power ratio in the scoping analysis would be much lower than that in SONGS-1. However, the volume to core power ratio in the scoping analysis will result in a conservative calculation of the primary mixture mass transient and time of core uncovery.

2. Core Length:
The scoping analysis used the existing fuel length of 12.0-ft rather than the SONGS-1 value of 10.0-ft. This difference is not significant when the reactor coolant system primary volume to power ratio is considered.

3. Cladding Material:
The scoping analysis would represent the fuel rod cladding material with the existing Zircaloy cladding in the NOTRUMP input rather than changing the fuel rod model to stainless steel cladding. Based upon the transient results, this difference is not significant.

4. Main Steam Safety Valve Setpoint:
The scoping analysis used the existing main steam safety valve (MSSV) setpoint of 1200 psia which is higher than in SONGS-1. The higher MSSV setpoint will result in a conservative calculation.

5. Core Power Level and Peaking Factor Differences:
The core power level and peaking factor differences could affect calculations of the peak cladding temperature. The core average linear heat rate in the scoping study was 5.554 Kw/ft and the peak linear heat rate was 13.61 Kw/ft, while in SONGS-1 the core average linear heat rate is approximately 4.73 Kw/ft at 102% of licensed core power and the peak linear heat rate is approximately 13.2 Kw/ft. The average hot channel enthalpy rise per unit fuel rod length is slightly lower in the scoping study than in SONGS-1. However, since the power distribution is skewed toward the top of the core in the analysis, the hot channel enthalpy rise above the froth level is higher in the scoping calculation. The scoping analysis values are conservative.

6. ECCS Design Differences:
The primary focus of the scoping study was to determine the system response to the differences in the ECCS design. The accumulator models were disabled and the ECCS flow used in the scoping analysis was scaled based upon the total core power to mass flow rate to match the SONGS-1 head to flow performance for the core power. Therefore, the ECCS design differences were represented.

3.3 ECCS Analysis Results

To assess the capability of NOTRUMP to analyze the small break transient response for SONGS-1, a scoping analysis was performed simulating a 6-inch equivalent diameter break in the cold leg of the primary reactor coolant system. The 6-inch break was chosen for the following reasons;

1. Larger break sizes subjects the calculational model to greater fluid flow and heat transfer regimes thereby posing a greater calculational challenge to the NOTRUMP fluid flow calculations.

2. Typical three-loop plants rely upon accumulators to mitigate the consequences of breaks of this size in the reactor coolant system.

The scoping analysis results are shown in Figures 3-1 through 3-8. Upon initiation of the 6-inch break, there was a rapid depressurization of the RCS (Figure 3-1). At 1.6 seconds into the transient the pressure decreased to the low pressurizer pressure setpoint of 1860 psia for the scoping analysis plant, which generated a reactor trip signal. The loss of off-site power was assumed to occur coincident with reactor trip.

The loss of off-site power assumption generated reactor coolant pump trip, turbine trip and isolation, and main feedwater pump trip and isolation. Following main feedwater pump trip and isolation the auxiliary feedwater system was automatically actuated and began delivering flow to the steam generator secondaries approximately 60 seconds later. Although this differs from the SONGS-1 design, the effect of the earlier auxiliary feedwater flow is insignificant for a break of this size. The loss of off-site power assumption additionally resulted in the loss of condenser steam dump capability. Consequently, the steam generator secondary side was effectively isolated and pressurized to the steam generator safety valve setpoint of approximately 1200 psia (Figure 3-2) and released steam through the safety valves.

Following reactor trip, the RCS continued to depressurize and reached the pressurizer low-low pressure setpoint of 1715 psia at approximately 2.4 seconds into the transient. Consequently, a safety injection S-signal was generated. Throughout this initial depressurization period, there was only liquid flow out of the break (Figures 3-7 and 3-8).

By approximately 25 seconds into the transient, the rate of depressurization of the reactor coolant system was altered. Vapor production due to flashing and decay heat boiling changed the rate of depressurization. Primary to secondary heat transfer continued to influence the volumetric balance and the steam generators were active heat sinks during this period. However, the volumetric flow thorough the break continued to exceed the primary volume swell and the system depressurization continued. Also during this period, the loop seals remained plugged with liquid, but the break flow indicated some vapor flow as depressurization and flashing in the cold legs resulted in some two phase flow through the break. However, hot side vapor, generated by flashing and decay heat boiling, was unable to vent out the break.

At approximately 130 seconds, the loop seal in the broken loop began to clear and vent steam to the break. This occurred as a result of the liquid level in the downflow section of the pump suction leg being depressed low enough to allow steam to begin to vent through the loop seal. Steam previously trapped on the hot side of the reactor coolant system was now able to vent to the break. The quality of the break flow fluid increased. The increased vapor flow out the break changed the volumetric balance and the rate of depressurization in the reactor coolant system increased.

The steam generator secondary side pressure performance is a function of the heat transfer between the primary and secondary side, the heat removal capability of the auxiliary feedwater flow, and the MSSV flow capability when steam dump capability is lost. In the scoping analyses, the secondary pressure between the broken loop and intact loop steam generators is isolated. The broken loop steam generator pressure increases to the MSSV setpoint, then decreases as the heat transfer process reverses and the steam generator becomes a heat source. Then during the core uncovery period, superheated vapor on the primary side results in pressurization of the broken loop secondary. As the core is cooled and the primary continues to depressurize, the secondary supplies energy to the primary and depressurizes. The intact loop steam generator secondary pressure increases to the MSSV setpoint and remains near the MSSV setpoint throughout the transient. This is a result of the conservative modeling of the condensation effects of auxiliary feedwater flow in the NOTRUMP analysis and the relatively stagnant primary side flow in the intact loop steam generator tubes.

During the loop seal steam venting process, the inner vessel mixture level (Figure 3-3) was briefly depressed as the manometric pressure imbalance between the upper plenum and the top of the downcomer affected the inner vessel mixture level response. This imbalance occurred in response to the hydrostatic head of water which was formed in the upflow section of the pump suction leg as the liquid level in the downflow section of the pump suction leg was depressed below the cold leg elevation. As soon as steam was vented through loop seal (horizontal flow section of the pump suction leg), this pressure imbalance was relieved, and the inner vessel mixture level returned to an elevation above the top of the core.

Following the loop seal clearing, the inner vessel mixture level decreased as the core boil-off rate exceeded the rate of safety injection flow (Figure 3-6) plus condensation in the system. The boiloff core uncovery began at approximately 175 seconds. The core mixture level continued to decrease until the reactor coolant system depressurized to a point at which the pumped safety injection flow rate exceeded the break flow rate. In typical three-loop plants for breaks of this size, this occurs when the reactor coolant system depressurizes below the accumulator cover gas pressure, which results in passive ECCS accumulator injection flow. In the scoping calculation, the scaled pumped safety injection flow in combination with the conservatively low amount of condensation in the system resulted in the recovery of the core mixture level by approximately 310 seconds.

As the transient continues, the break flow rate is exceeded by the safety injection flow rate which recovers the core mixture level to approximately the hot leg elevation. At approximately 500 seconds, the safety injection flow roughly equals the break flow rate, and relatively stable recovery conditions are established. Due to the large amount of safety injection flow, the core uncovering transient is terminated relatively early when compared to the same size break in a typical three-loop plant.

A calculation for the hot assembly fuel rod transient performance was performed using the small break version of the LOCTA-IV computer code assuming Zircaloy cladding. The results (Figure 3-9) indicate that the peak cladding temperature on the hot rod is approximately 1507°F. The maximum oxidation rate at the hot spot was less than 0.24%. Rupture of the hot rod was not calculated to occur. As noted in Section 2.2.1.1, the oxidation rate and exothermic heat of reaction for Zircaloy cladding at this temperature bounds that of stainless steel. The maximum hoop stress at approximately 290 seconds into the transient would have required a cladding temperature of more than 1900°F for rupture to be calculated for stainless steel cladding. Therefore, the scoping analysis peak cladding temperature results are conservative relative to the hot assembly fuel rod temperature performance that would be calculated assuming the cladding were stainless steel.

Overall the phenomena observed in the scoping calculation is similar to that observed in the NOTRUMP generic studies[15]. The reactor coolant system pressure, mixture level performance, and fluid flow response was similar to the phenomena observed for other typical three-loop plants with a similar break. The expected effects of the large amount of safety injection flow appropriate to the SONGS-1 plant were observed. There was no indication that the ECCS design difference would adversely affect the ability of the NOTRUMP small break LOCA ECCS Evaluation Model to calculation the appropriate transient response. In fact, Figure 3-5 shows that the reactor coolant system total mixture mass responds exactly as expected and essentially the same as in typical three-loop plants, when the ECCS design differences are considered.

FIGURE 3-1

SCE SCOPING ANALYSIS

REACTOR COOLANT SYSTEM PRESSURE

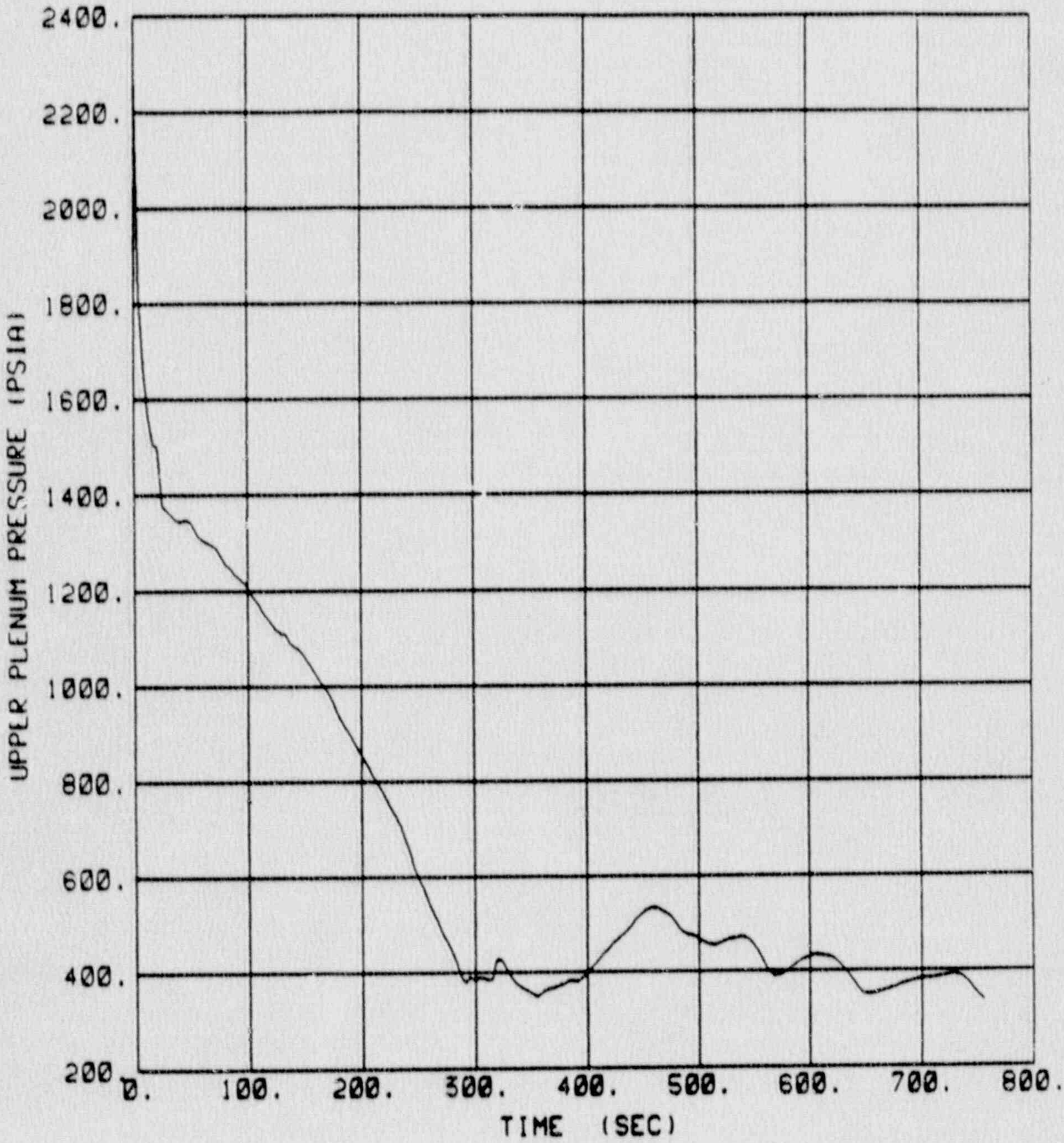


FIGURE 3-2

SCE SCOPING ANALYSIS

STEAM GENERATOR SECONDARY SIDE PRESSURE

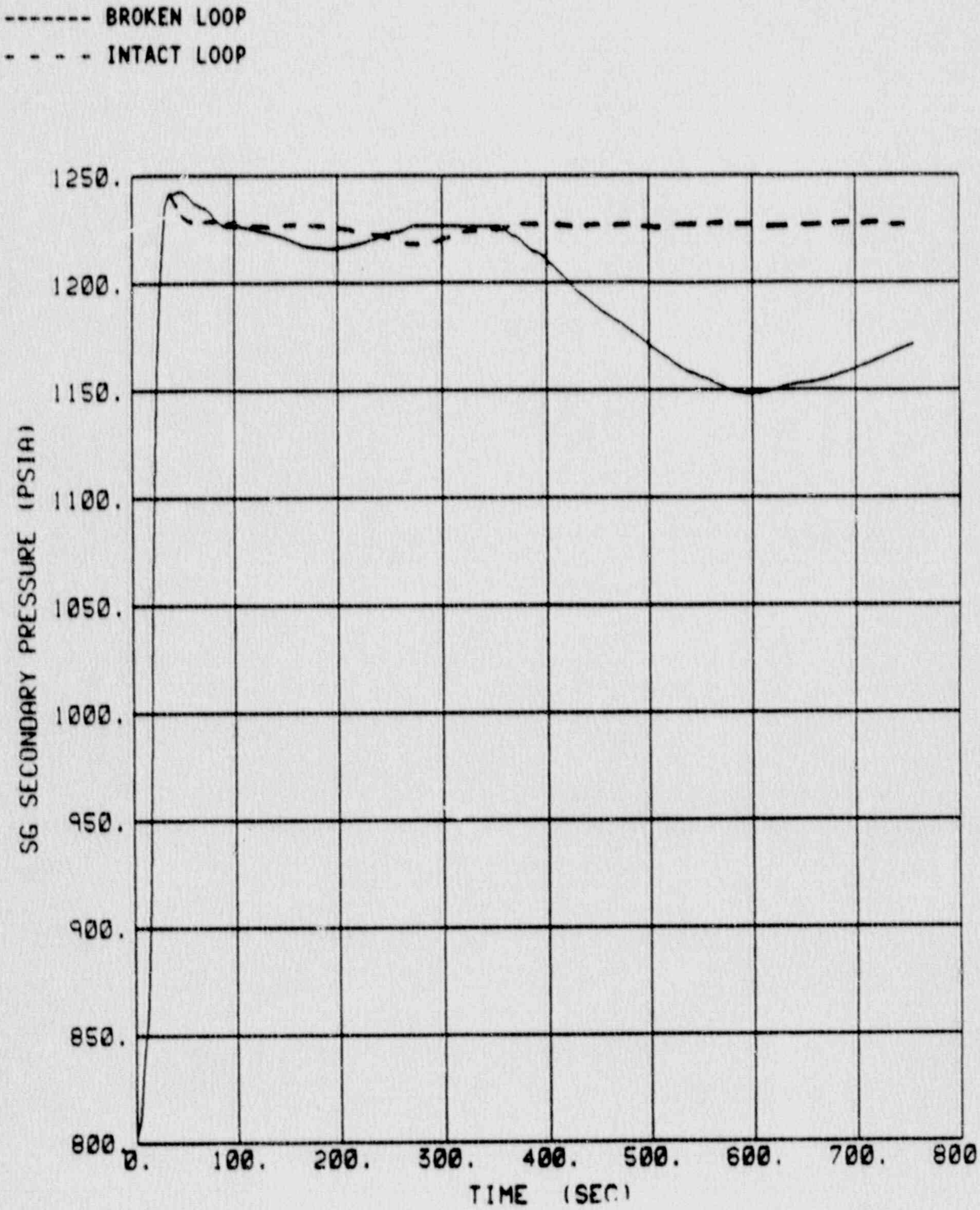
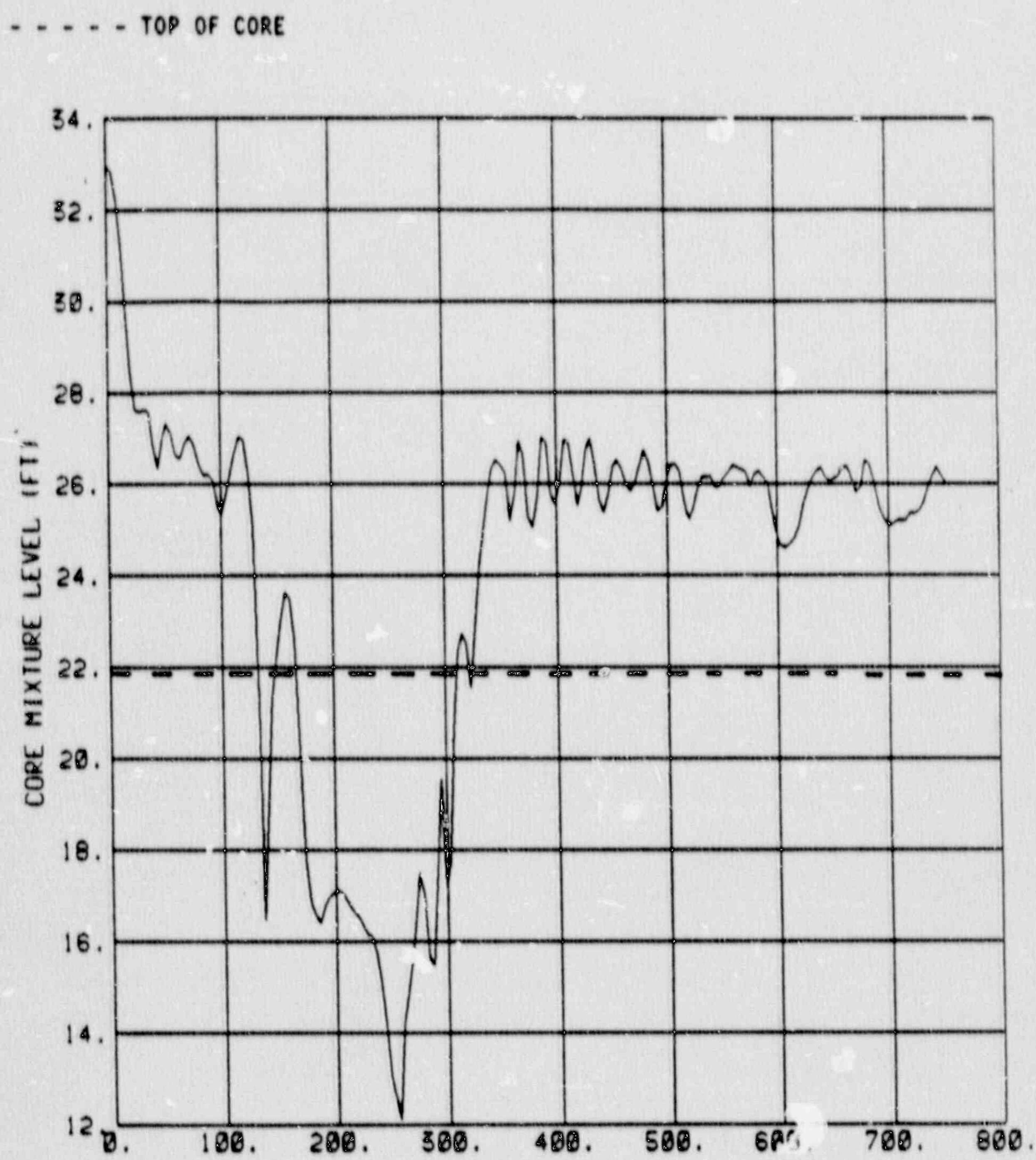


FIGURE 3-3

SCE SCOPING ANALYSIS

CORE MIXTURE LEVEL



SCE-SS-2-901123

FIGURE 3-4

SCE SCOPING ANALYSIS

DOWNCOMER MIXTURE LEVEL

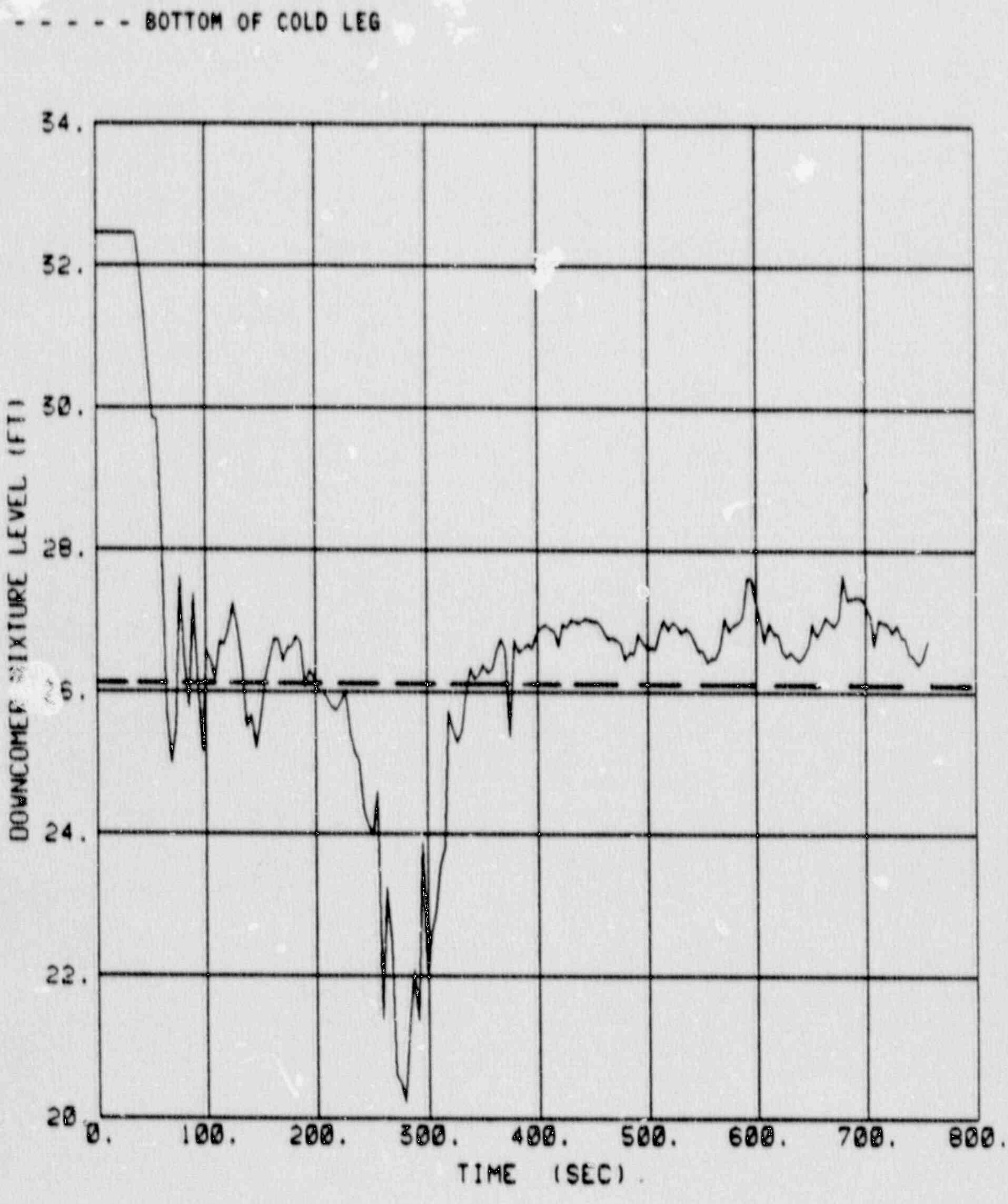


FIGURE 3-5

SCE SCOPING ANALYSIS

TOTAL MIXTURE MASS OF RCS

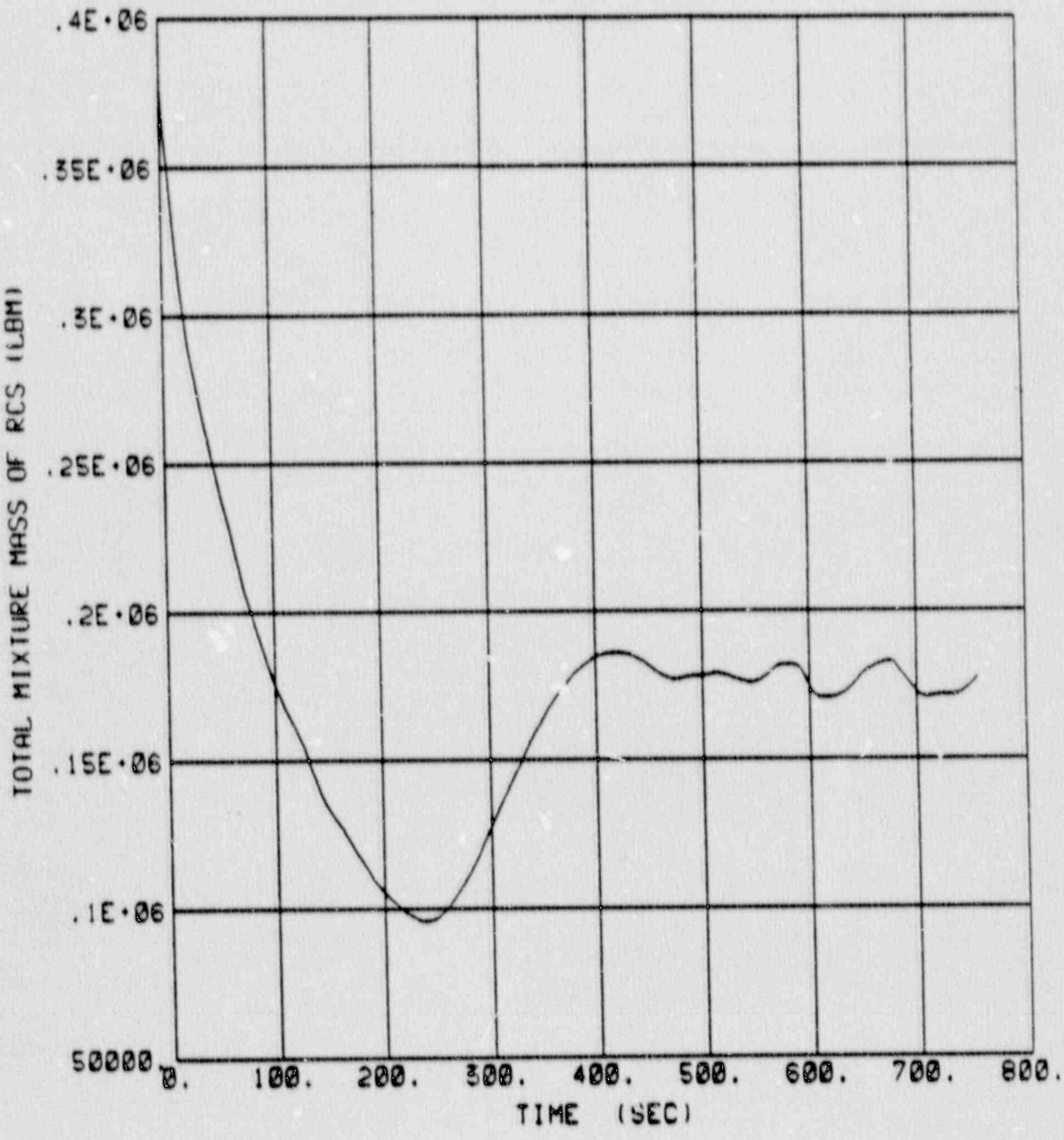


FIGURE 3-6

SCE SCOPING ANALYSIS

SAFETY INJECTION FLOW RATE

TO THE INTACT LOOPS

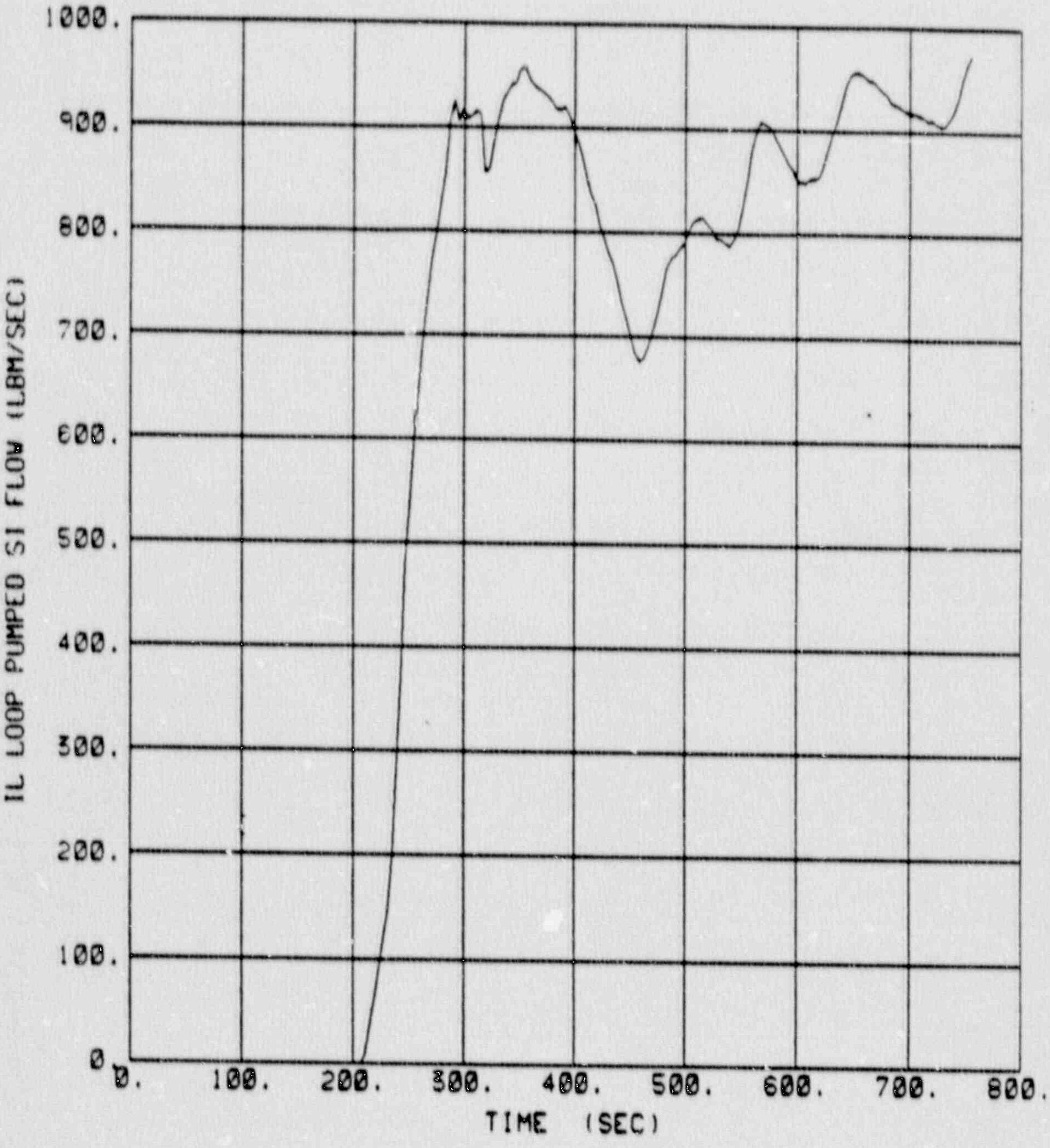


FIGURE 3-7

SCE SCOPING ANALYSIS

TOTAL BREAK FLOW RATE

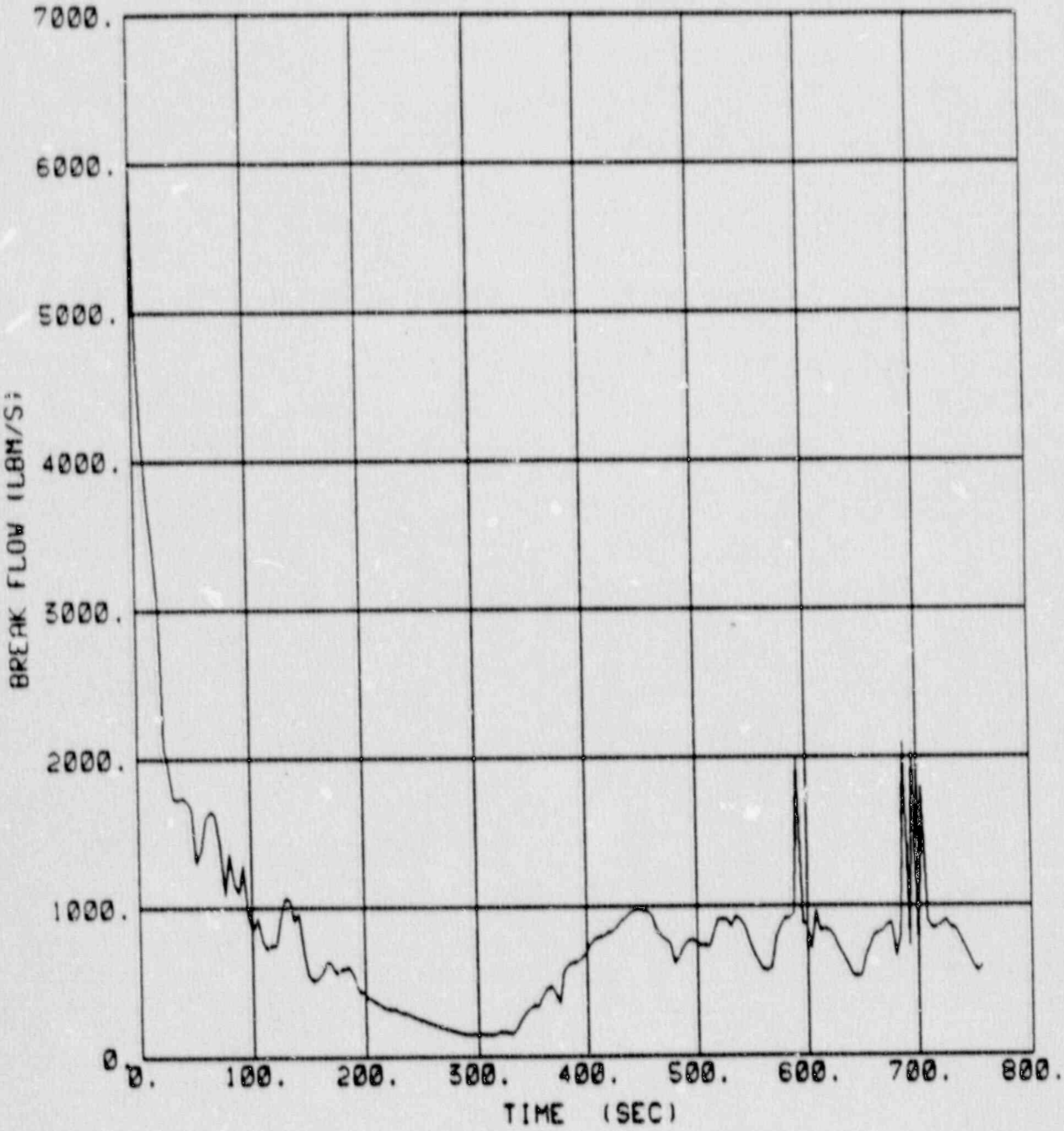


FIGURE 3-8

SCE SCOPING ANALYSIS

STEAM BREAK FLOW RATE

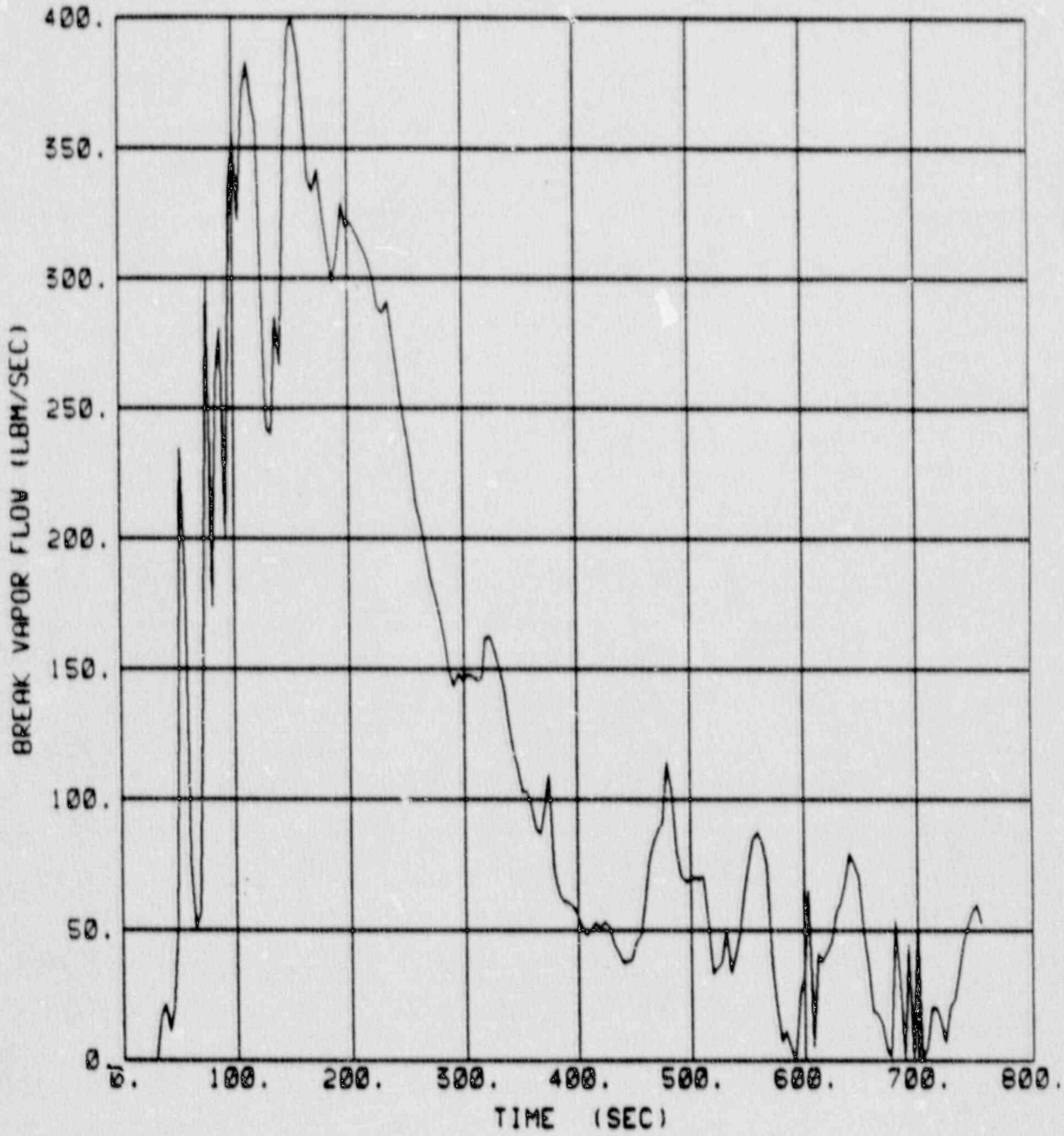


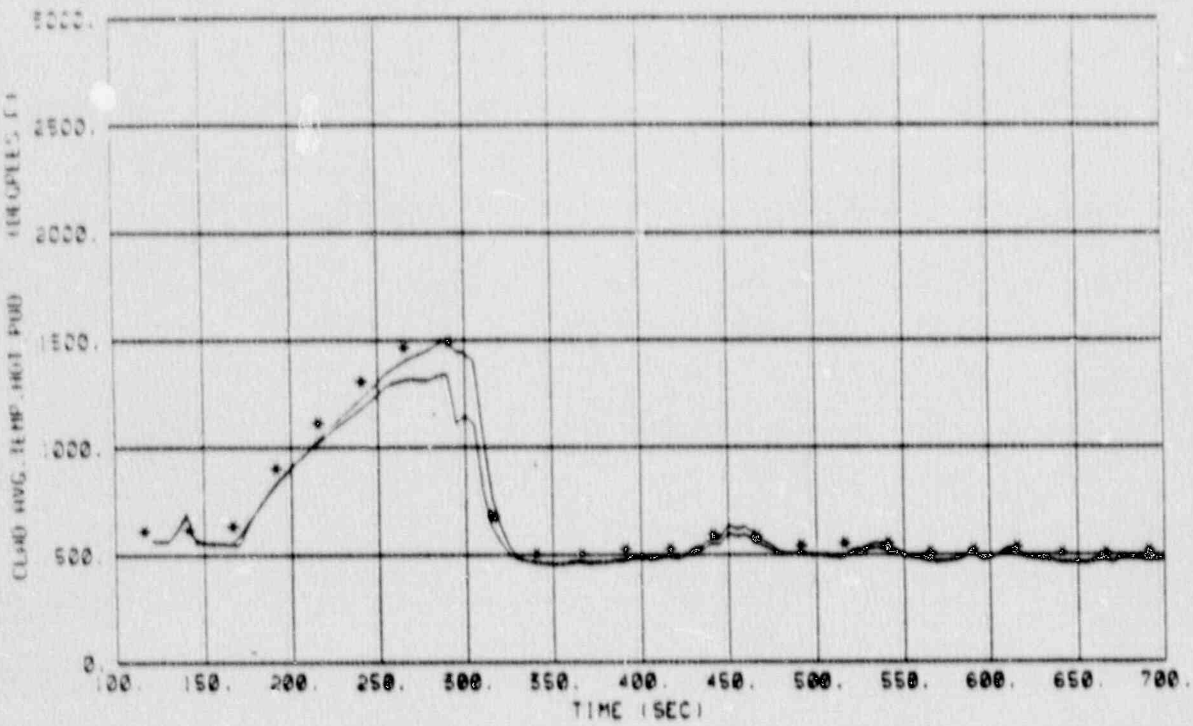
FIGURE 3-9

SCE SCOPING ANALYSIS

HOT ROD CLADDING AVERAGE TEMPERATURE

———— 9.50-ft peak power elevation

—*—*—*— 11.25-ft peak clad temp. elevation



4.0 EVALUATION OF THE NOTRUMP STAINLESS STEEL CLADDING MODEL

Appendix A provides information regarding the NOTRUMP fuel rod model with stainless steel cladding. This model was evaluated by activating the stainless steel cladding option and performing steady state and transient analyses for a typical three-loop plant. Hot assembly fuel rod heatup calculations were not performed, since the thrust of the evaluation was to assess the fuel rod performance in the NOTRUMP computer code.

A NOTRUMP input data set for a typical three-loop plant licensed to a core power level of 2775 MWth was obtained. The analysis assumed values appropriate for Model D3 steam generators with a uniform tube plugging level of 20%, main steam safety valve setpoints beginning at approximately 1200 psia, 17x17 Vantage-5 fuel, and a downflow barrel baffle.

The only change to the input was the selection of the stainless steel fuel rod cladding model option rather than the Zircaloy model. Although the fuel rod dimensions with stainless steel clad fuel would be different, the fuel rod dimensions were not changed for the evaluation analysis. This also allowed a direct comparison with previous steady state results which assumed Zircaloy clad fuel and provided a more detailed test of the fuel rod model solution convergence. In this evaluation, the oxidation and rupture models were not activated.

4.1 Steady State Simulation

Simulation of plant steady state operation at 102% of licensed core power was performed using NOTRUMP for 100 seconds. A review of the steady state simulation output indicated that the material properties were being calculated properly during the NOTRUMP steady state simulation. Comparisons to the previous steady state simulation assuming the Zircaloy cladding model indicated that there were small differences in the temperature profile in the fuel rod. Calculations were performed which confirmed that the small differences were due to the representation of the stainless steel cladding material properties. The steady state simulation assuming the stainless steel cladding model resulted in slightly more stored energy in the fuel and cladding. Considering that the fuel dimensions were not altered for the stainless steel cladding case, this result was as expected. Overall, the deference in the fuel rod cladding material did not affect the solution convergence and acceptable steady state simulation results, similar to those obtained for Zircaloy clad fuel rods, were obtained.

4.2 Transient Analysis Simulation

To assess the stainless steel fuel rod cladding model over the expected range of a small break LOCA transient, a 3-inch equivalent diameter break in the cold leg was simulated. The 3-inch break resulted in the highest calculated peak cladding temperature in the previous break transient simulation for Zircaloy clad fuel. Reviews of the output at various points in the transient indicated that the stainless steel cladding model was performing as expected and calculating the appropriate values for the cladding. It should be noted that the difference in the cladding material had an insignificant effect on the transient response.

Based upon the review of the steady state and break transient simulations and comparison with the previous calculations which assumed Zircaloy fuel rod cladding model, it was concluded that the stainless steel fuel rod model in NOTRUMP was performing correctly.

5.0 CONCLUSIONS

An assessment of the applicability of the NOTRUMP small break LOCA ECCS Evaluation Model to the San Onofre Nuclear Generating Station Unit 1 (SONGS-1) was performed. Previous applications of the NOTRUMP computer code were reviewed and it was found that NOTRUMP had been successfully applied to a wide range of engineering analysis problems for a broad spectrum of two-phase fluid flow configurations. The design differences between SONGS-1 and typical three-loop plants for which NOTRUMP analysis calculations serve as the small break ECCS licensing bases were reviewed. It was observed that only two plant design differences are significant to the ability to apply the NOTRUMP small break LOCA ECCS Evaluation Model to SONGS-1:

1. The fuel rod cladding material in SONGS-1 is stainless steel instead of Zircaloy as in typical three-loop plant applications, and

2. The ECCS design differences could result in thermal-hydraulic behavior which differs from that previously calculated using the NOTRUMP small break LOCA ECCS Evaluation Model.

Since the NOTRUMP computer code included the option to represent fuel rods with stainless steel cladding, an assessment was performed to validate that the appropriate calculations would result during the simulation of a small break LOCA transient for SONGS-1. The assessment concluded that the appropriate stainless steel cladding calculations would be performed for the small break LOCA transient conditions in SONGS-1. Additional models to represent the rupture and the metal and water reaction for stainless steel cladding were developed. The rupture and oxidation models will be used in the fuel rod heatup calculations, but only the oxidation model will be used in the reactor coolant system thermal-hydraulic transient calculation. Confirmatory analyses will be performed to assure that the incidence of cladding rupture is not underpredicted for the core average fuel rods in the reactor coolant system thermal-hydraulic transient calculation.

Since the ECCS design differs between SONGS-1 and typical three-loop plants, an assessment of the capability of the NOTRUMP small break LOCA ECCS Evaluation Model to calculate the transient response for the SONGS-1 ECCS design was made by performing a scoping calculation. The scoping calculation indicated the the SONGS-1 ECCS design differences would not result in a reactor coolant system transient response which is substantially different from that observed for other plants in terms of the pressure, mixture levels, and fluid flows. Conservative calculations of the peak cladding temperature and the amount of oxidation were performed assuming Zircaloy cladding. The results indicated that the peak cladding temperature was approximately 1507°F and the maximum amount of oxidation was less than 0.24% in the scoping calculation.

Based upon these observations, it is concluded that the modified NOTRUMP small break LOCA ECCS Evaluation Model is appropriate for application to SONGS-1.

6.0 REFERENCES

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APPENDIX A

SUPPLEMENTAL INFORMATION REGARDING THE NOTRUMP FUEL ROD MODEL FOR STAINLESS STEEL CLADDING

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SUPPLEMENTAL INFORMATION REGARDING

THE NOTRUMP FUEL ROD MODEL

FOR STAINLESS STEEL CLADDING

SUPPLEMENTAL INFORMATION REGARDING

THE NOTRUMP FUEL ROD MODEL

FOR STAINLESS STEEL CLADDING

I) INTRODUCTION

The fuel rod model as used in the NOTRUMP computer code is described in Appendix T, "Fuel Rod Model," of Reference 1. The fuel rod model in NOTRUMP determines the radial temperature profile in a fuel rod and the heat flux at the rod surface at the end of a time interval given the initial temperature profile, thermal properties, heat transfer coefficients, and energy generation rate. The model is essentially the same as the fuel rod model that is described in Reference 2.

In NOTRUMP, the fuel rod is modeled as three regions: the fuel pellet , the cladding, and a gap between the pellet and cladding. The radial heat conduction equations are solved in the pellet and cladding. Heat transfer coefficients are used to calculate temperature differences across the gap and between the cladding outer surface and the ambient fluid.

Appendix T in Reference 1 provides a description of the NOTRUMP fuel rod model, including the numerical solution techniques, a volumetric heat generation model for the Zircaloy clad metal water reaction, a model for heat transfer across the gap between the fuel pellet and the cladding, a model for the expansion of the fuel and cladding for Zircaloy clad fuel, and models for the fuel rod to coolant heat transfer. Appendix T in Reference 1 also provides the material properties used in the code for Uranium-Dioxide, Zircaloy cladding, and Zirconium-Dioxide.

SUPPLEMENTAL INFORMATION REGARDING

THE NOTRUMP FUEL ROD MODEL

FOR STAINLESS STEEL CLADDING

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The fuel rod model as used in the NOTRUMP computer code is described in Appendix T, "Fuel Rod Model," of Reference 1. The fuel rod model in NOTRUMP determines the radial temperature profile in a fuel rod and the heat flux at the rod surface at the end of a time interval given the initial temperature profile, thermal properties, heat transfer coefficients, and energy generation rate. The model is essentially the same as the fuel rod model that is described in Reference 2.

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The effect of different cladding materials is accommodated through representation of specific cladding material properties and special models for cladding oxidation, cladding swelling, and cladding rupture.

The following discussion provides a description of the models used when stainless steel clad fuel rods are modeled in NOTRUMP computer analysis calculations. Reference 1 indicates that there is an option for representing stainless steel clad fuel rods. However, specific details regarding the stainless steel clad fuel rod models are omitted from Appendix T to Reference 1. In the following discussion, only the additional information necessary to supplement the information in Appendix-T to Reference 1 in order to represent stainless steel clad fuel rods is provided.

II) FUEL ROD MODEL SOLUTION TECHNIQUE FOR STAINLESS STEEL CLAD FUEL

Heat Conduction Equations and Method of Solution

When stainless steel cladding is represented, the same fully implicit finite difference method as used for Zircaloy clad fuel rods is used for the numerical solution of the problem. The finite difference equations T-24, T-29, and T-34 remain unchanged except for the values of some of the coefficients. The effect of the stainless steel cladding enters only by affecting some of the coefficients through the volumetric heat generation rate in the cladding and through the material properties for thermal conductivity, density, and heat capacity. For stainless steel cladding, the volumetric heat generation rate array is affected by the metal water reaction rate. The solution of the tridiagonal matrix equation to implicitly provide the temperature distribution remains unchanged.

Heat Generation in Stainless Steel Cladding

The following stoichiometry represents the oxidation process for stainless steel:

Wustite	$Fe + H_2O$	-->	FeO	+	H_2
Hematite	$2Fe + 3H_2O$	-->	Fe_2O_3	+	$3H_2$
Magnetite	$3Fe + 4H_2O$	-->	Fe_3O_4	+	$4H_2$

The oxidation rate of materials may be compared by examining the rate of oxidation in the material. In general, the parabolic equation for this oxidation is expressed in the form:

$$W^2/t = K_o \exp [\Delta H / (R \times T)] \qquad \text{(A-1)}$$

where

W	= weight gain,	mg $(O_2)/cm^2$ of area
K_o	= constant,	mg^2/cm^4 - sec
ΔH	= activation energy of reaction,	cal/mole
R	= gas constant,	cal/mole - ^oK
T	= absolute temperature,	^oK
t	= time,	seconds

A linear depiction of the initial 304L stainless steel and water reaction rate is provided by the following[3]

$$W = 1.1 \times 10^5 \; t \exp [(-44,350)/(R*T)] \qquad \text{(A-2)}$$

A conservative representation for the oxidation rate for 304L stainless steel is provided by the parabolic kinetics region as follows[3]

$$W^2 = 2.4 \times 10^{12} \; t \; \exp [(-84,300)/(R*T)] \qquad \text{(A-3)}$$

Equation (A-3) may be differentiated and recast into a form similar to the derivation of the rate equation for Zircaloy cladding as found in Appendix T of Reference 1. For the conservative representation for the oxidation energy release for 304L stainless steel this would become

$$\frac{dq}{dt} = \frac{(x-x_0)}{\Delta t} * 7.24 \times 10^4 \qquad \text{(A-4)}$$

where $(x-x_0)$ represents the change in the per unit length oxidation thickness over the time step Δt.

The heat generated by the metal water reaction is applied as a volumetric heat source in the cladding node (of thickness Δr_c) in which the reaction is taking place. The heat generation rate, dq/dt, for stainless steel cladding is then converted to a volumetric heat generation rate as follows:

$$q''' = \frac{[dq/dt]}{[\ \Delta r_c\ ]} \qquad \text{(A-5)}$$

III) THE FUEL ROD GAP IN STAINLESS STEEL CLAD FUEL RODS

Fuel Gap Gas Pressure Calculation

For stainless steel clad fuel rods as for Zircaloy clad fuel rods, the fuel rod gap pressure, P_g, is calculated using equation T-55 in Appendix T of Reference 1. The fuel rod gap pressure is affected by the temperature distribution in the fuel rod and the gap volume. For an axial node j, the largest effect of stainless steel cladding will be on the gap temperature, T_{cj} and T_{gj}, and the gap volume, V_{gj}. The gap temperature is taken as the average of the fuel surface and inner cladding diameter temperatures. The gap volume, V_{gj}, is calculated by taking the difference between the expanded cladding and the envelope volume of the fuel column. The numerical solution process for the fuel rod gap pressure is not affected by different cladding materials, except through the temperature distribution in the fuel rod and the gap volume.

Expansion of Fuel and Cladding

The width of the gap used in determining the gap conductance is adjusted each time step to reflect the thermal expansion of the fuel and the thermal and elastic expansion of the cladding. For stainless steel clad fuel rods, the radial thermal expansion of the cladding is calculated by the equation

$$r^{th}_{c} = r_{c_0} (1 + \alpha_c * T'_c) \qquad \text{(A-6)}$$

where α_c, the coefficient of linear thermal expansion of the cladding, is given by:

$$\alpha_c(T'_c) = [\qquad]^{a,c} \qquad \text{(A-7)}$$

where T'_c is the volume weighted average clad temperature in °F, and $\alpha_c(T'_c)$ is in in/in/°F.

Elastic expansion of the stainless steel cladding due to differential pressure between the gas in the gap and the core coolant volume is also calculated by equation T-63 in Appendix T of Reference 1.

Poisson's ratio for the stainless steel cladding[4] is given by:

$$\mu = [\qquad]^{a,c} \qquad \text{(A-8)}$$

and Young's modulus for the stainless steel cladding[4] is given by:

$$E_{ss} = [\qquad]^{a,c} \qquad \text{(A-9)}$$

Where T_c is in °F.

Rupture of Stainless Steel Cladding

During a LOCA, the cladding is assumed to strain uniformly and plastically in the radial direction when both the temperature and the differential pressure across the cladding are sufficiently high. If the cladding temperature exceeds the rupture temperature, which is determined as a function of the instantaneous stress, the cladding is assumed to rupture. A rupture model is not activated in the NOTRUMP computer code, which is used to calculate the reactor coolant system response to a small break LOCA. Confirmatory calculations are performed to ensure that the incidence of rupture of the core average fuel rod is not under predicted assuming the model used to perform hot assembly fuel rod heatup calculations in the small break version of LOCTA-IV. The following model is used to calculate the rupture temperature:

$$T_{rupture} = [\qquad] \qquad \text{(A-10)}$$

where σ is the stainless steel cladding hoop stress.

Gap Conductance

The conductance of the interface between the fuel pellet and cladding at each time step is calculated from the conditions existing at the end of the previous calculational time step. If an open gap is calculated to exist between fuel and cladding the conductance is calculated using equation T-66 in Appendix T of Reference 1. If cladding and fuel contact is calculated to exist, the conductance is calculated based upon the following equations:

$$U_{gap} = [\qquad]^{a,c} \qquad \text{(A-10)}$$

or

$$U_{gap} = [\qquad]^{a,c} \qquad \text{(A-11)}$$

The conductivity of the gap gas mixture is calculated using equation T-68 in Appendix T of Reference 1 and is unaffected by the cladding material, except through the gap temperature effect in the calculation of the thermal conductivity of the individual gases.

The additional term added to the gap conductance to account for radiation between the fuel pellet and the cladding in Equation T-79 in Appendix T of Reference 1 is also used for fuel rods with stainless steel cladding, except that cladding emissivity, ϵ_c used in the equation is for stainless steel rather than oxidized zirconium.

IV) ROD TO COOLANT HEAT TRANSFER MODELS

The fuel rod to coolant heat transfer correlations in the NOTRUMP are provided in Appendix T of Reference 1 and are largely unaffected by the type of fuel rod cladding. The fuel rod to coolant heat transfer correlations for the subcooled fluid, nucleate boiling, transition boiling, saturated film boiling, and stable film boiling heat regimes are not affected by the cladding type.

Steam Cooling

The fuel rod to steam cooling heat transfer correlation is given by Equation (T-98) in Reference 1 is also used for fuel rods clad with stainless steel. The steam cooling convection coefficient, U_{SCON}, is not affected by the type of fuel rod cladding, but the coefficient used for radiation from the fuel rod to the coolant is affected through the radiosity. For stainless steel clad fuel rods, Equation (T-97) is used to calculate the radiosity and the fuel rod emissivity is based upon stainless steel.

V) PROPERTIES OF STAINLESS STEEL

Density

Equation (A-12) defines the density for stainless steel as a function of temperature[4]

$$\rho = [\qquad\qquad\qquad]^{a,c} \qquad \text{(A-12)}$$

where ρ is in lbm/in^3 and T is in °F.

Thermal Conductivity

Equation (A-13) defines the thermal conductivity for stainless steel as a function of temperature[4]

$$k_c = [\qquad\qquad\qquad]^{a,c} \qquad (A-13)$$

where T is in °F and k_c is in BTU/hr-ft-°F

Heat Capacity

Equations (A-14) and (A-15) defines the heat capacity for stainless steel as a function of temperature[4]

$$C_p = [\qquad\qquad\qquad]^{a,c} \qquad (A-14)$$

and

$$C_p = [\qquad\qquad\qquad]^{a,c} \qquad (A-15)$$

where T is in °F and C_p is in BTU/lbm-°F

Emissivity

Equation (A-16) defines the emissivity for stainless steel

$$\epsilon = [\qquad]^{a,c} \qquad (A-16)$$

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FNCLOSURE 3