Small Break LOCA Evaluation Model

Revision 1

Non-Proprietary

March 2017

REVISION HISTORY

Revision	Date	Page	Description
0	September 2014	All	First Issue
1	March 2017	8, 17~74, 76, 78	Revised to reflect RAI-8337

ABSTRACT

This report presents the small break loss of coolant accident (SBLOCA) analysis methodology that is used in Section 15.6.5 of the design certification document (DCD) Tier2 for the APR1400. The contents of this document include description of the computer code, analysis methodology, and results of the APR1400 SBLOCA analysis. The methodology for the analysis of radiological consequences is not described in this technical report.

This report provides an overview of the applicable methodology and a description of the specific models incorporated in the following codes used to analyze SBLOCA, as well as a discussion of the bases for applying these codes and methods to the APR1400. Validation of these codes by comparison with computer codes that have been approved by the NRC is presented.

The CEFLASH-4AS computer program is used to determine the primary system hydraulic parameters during the blowdown phase, and the COMPERC-II computer program is used to determine the system behavior during the reflood phase. Fuel rod temperatures and clad oxidation percentages are calculated using the STRIKIN-II and PARCH computer programs.

It was concluded that the applied codes and methodologies are appropriate for the APR1400 safety analysis.

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ACRONYMS AND ABBREVIATIONS

CFR	code of federal regulations
DCD	design certification document
DVI	direct vessel injection
ECC	emergency core cooling
ECCS	emergency core cooling system
EDG	emergency diesel generator
FSAR	final safety analysis report
IRWST	in-containment refueling water storage tank
LBLOCA	large break LOCA
LOCA	loss-of-coolant accident
LOOP	loss-of-offsite power
LPSIP	low pressure safety injection pump
NRC	nuclear regulatory commission
NSSS	nuclear steam supply system
PCT	peak clad temperature
PLHGR	peak linear heat generation rate
POSRV	pilot operated safety relief valve
PWR	pressurized water reactor
RCS	reactor coolant system
RCP	reactor coolant pump
SBLOCA	small break LOCA
SIAS	safety injection actuation signal
SIP	safety injection pump
SIS	safety injection system
SIT	safety injection tank
SIT-FD	safety injection tank equipped with a fluidic device
SRP	standard review plan

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1 INTRODUCTION

The APR1400 is an advanced pressurized water reactor (PWR) with a direct vessel injection (DVI) system; however, it retains the principal features of conventional 2-loop plants. Each loop consists of a hot leg, a steam generator (SG), two pump suction legs, two reactor coolant pumps (RCPs) and two cold legs. The major system parameters of the APR1400 design are given in Table 1-1. The core thermal power is 4,062.66 MWt (102% of nominal). The mass flow rates of the reactor coolant system (RCS) and the core are 20,991 kg/s and 20,361 kg/s, respectively. The active core is 3.81m long. In the core, 241, 16 by 16 type PLUS7 fuel assemblies are loaded. Detailed specifications of PLUS7 are presented in Table 1-2.

The features of the emergency core cooling system (ECCS) are as follows:

- DVI
- Four independent trains of high-pressure safety injection pumps (SIPs)
- Safety injection tank equipped with a fluidic device (SIT-FD)

All the emergency core cooling water of the four SIT-FDs and four SIPs are injected solely into the upper annulus of the pressure vessel. The DVI nozzle is 2.1 m above the center of the cold leg (Figure 1-1); nozzle azimuthal configuration is shown in Figure 1-2. The nozzles are 90^{o} apart from each other. The four trains of the SIPs have been designed to be mechanically and electrically independent. Each of the four emergency diesel generators (EDGs) independently provides a power source to each SIP. According to the worst single failure assumption, three SIPs are operable. However, for additional conservatism, only two SIPs are assumed to be operable in the present SBLOCA analysis. The configuration is shown in Figure 1-2. The SIT-FD controls the injection flow as a function of the water level.

The purpose of this technical report is to present the SBLOCA computer codes and methodologies for the SBLOCA analysis in Standard Review Plan (SRP) Section 15.6.5, except dose evaluation, for advanced PWRs such as APR1400. The SBLOCA methodology using the following codes is very similar to the conventional SBLOCA methodology used for currently operating US CE-fleet PWRs;

- CEFLASH-4AS Calculates thermal hydraulic behavior during the blowdown phase
- COMPERC-II Calculates thermal hydraulic behavior during the reflood phase
- STRIKIN-II Calculates the cladding and fuel temperature before the flow reversal time
- PARCH Calculates the cladding and fuel temperature after the flow reversal time

This report describes:

- Section 2 General Description and Characteristics of SBLOCA
- Section 3 Description of Model Components
- Section 4 SBLOCA Analysis
- Section 5 Requirements for SBLOCA
- Section 6 Conclusions

Table 1-1 Major System Parameters and Initial Conditions for SBLOCA Analysis

Parameters	Value
Power (102% of Nominal), MWt	4,062.66
Average linear heat generation rate, kW/m (kW/ft)	18.75 (5.715)
Peak linear heat generation rate (PLHGR), kW/m (kW/ft)	49.2 (15.0)
Gap conductance at PLHGR, kcal/hr-m^2-℃ (Btu/hr-ft^2-°F)	10,289 (2,107)
Fuel centerline temperature at PLHGR, ℃ (°F)	1,965 (3,568)
Fuel average temperature at PLHGR, ℃ (°F)	1,200 (2,192)
Hot rod gas pressure, kg/cm^{2}A (psia)	52.0 (740)
Initial reactor vessel inlet temperature, ℃ (°F)	290.6 (555)
Initial reactor vessel outlet temperature, ℃ (°F)	324.4 (615.9)
Moderator temperature coefficient, Δρ/℃ (Δρ/°F)	0.0×10^{-4} (0.0×10^{-4})
Initial RCS flow rate, kg/s (10^6 lbm/hr)	20,991 (166.6)
Initial core flow rate, kg/s (10^6 lbm/hr)	20,361 (161.6)
Initial RCS pressure, kg/cm^{2}A (psia)	158.2 (2,250)
Low pressurizer pressure reactor trip setpoint, kg/cm^{2}A (psia)	109.3 (1,555)
SIAS setpoint on low pressurizer pressure, kg/cm^{2}A (psia)	109.3 (1,555)
SIT gas pressure, kg/cm^{2}A (psia)	41.1 (584.7)

Table 1-2 Specifications of PLUS7 Fuel

Parameters	Value
Fuel assembly length, m (in)	4.5276 (178.25)
Active core length, m (in)	3.810 (150.0)
Fuel assembly pitch, m (in)	0.2078 (8.18)
Number of guide tubes per fuel assembly (outer + center)	5 (4+1)
Number of protective grids	1
Number of top Inconel grids	1
Number of bottom Inconel grids	1
Number of middle grids	9
Length of middle grid span, m (in)	0.3993 (15.719)
Number of fuel rods per fuel assembly	236
Fuel rod pitch, m (in)	0.01285 (0.506)
Clad material	ZIRLO
Fuel rod length, m (in)	4.0945 (161.20)
Fuel rod outer diameter, m (in)	0.0095 (0.374)
Clad thickness, m (in)	0.000572 (0.0225)
Pellet diameter, m (in)	0.008192 (0.3225)

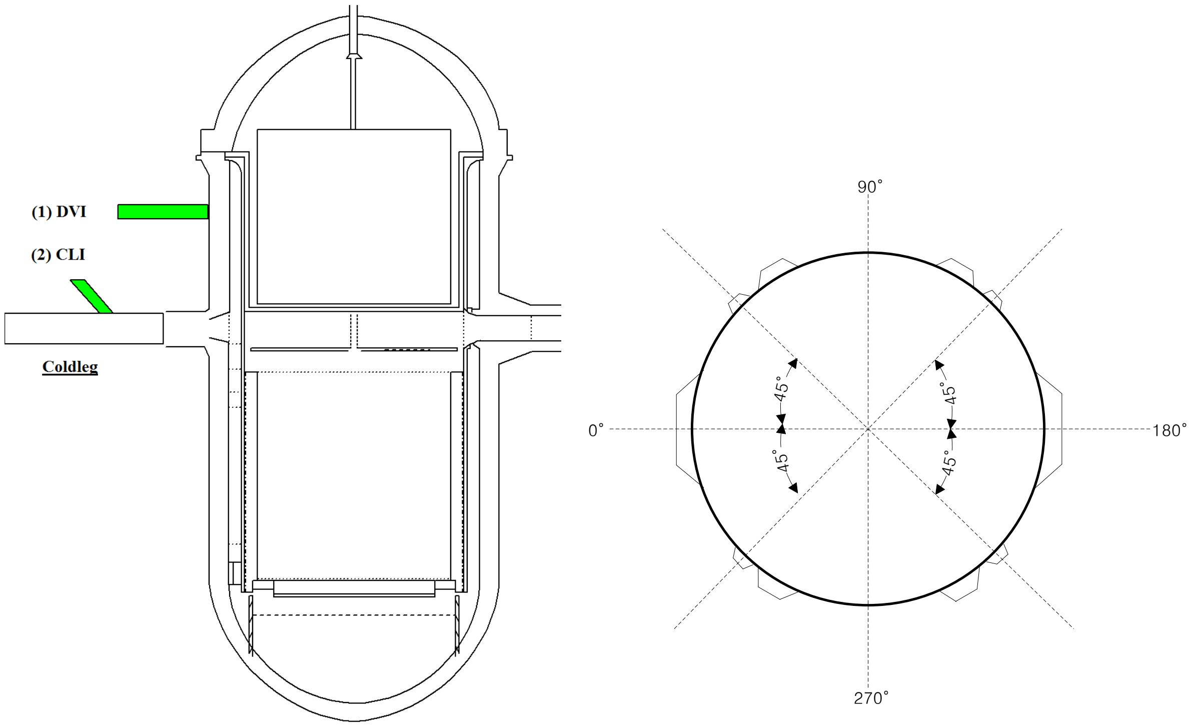

Figure 1-1 DVI Nozzle Location

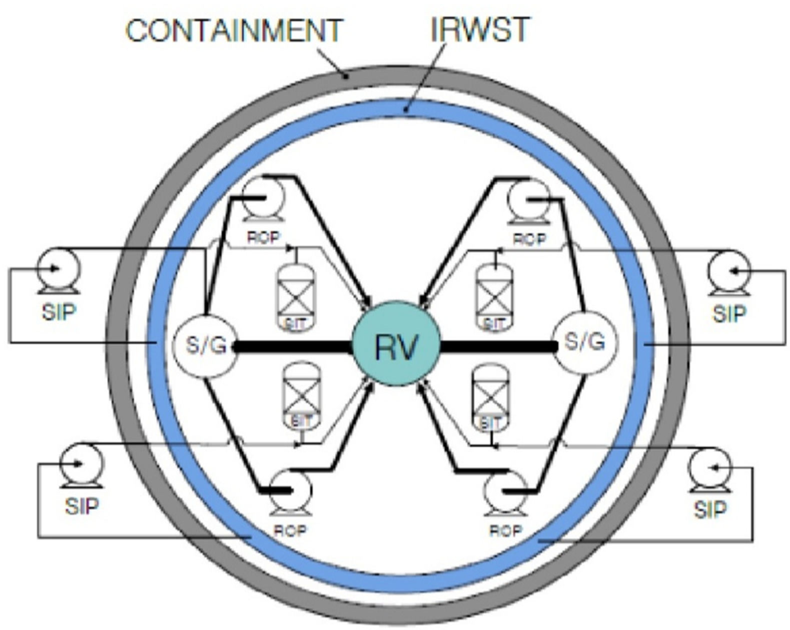

Figure 1-2 Configuration of the APR1400 Safety Injection System

2 GENERAL DESCRIPTION AND CHARACTERISTICS OF SBLOCA

2.1 Definition and Characteristics of SBLOCA

LOCAs are hypothetical accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system. The smallest SBLOCA is defined as a break that cannot be made up by charging pump. The largest SBLOCA is generally 1.0 ft^2. CENP SBLOCA methodology, however, through a sensitivity study for break size, has asserted that the largest SBLOCA is 0.5 ft^2.

SBLOCAs show fairly different behavior according to break size. When the break size is small, coolant inventory can be maintained by the SIP. In such a case, cooling and depressurization using the steam generator are performed by the operator. If the break size is large, release flow cannot be sufficiently compensated for by the SIP. Furthermore, SBLOCAs behave differently depending on break location. A cold-leg break SBLOCA, known as the most limited case, can be divided into five phases. The duration of each phase and its occurrence varies depending on the break size and the system characteristics.

2.2 Computer Codes

The calculations reported in this section were performed using the small break evaluation model, described in Reference 1, and were approved by the USNRC in Reference 2. The CEFLASH-4AS (Reference 3) computer program was used to determine the primary system hydraulic parameters during the blowdown phase; the COMPERC-II (Reference 4) computer program was used to determine system behavior during the reflood phase. Fuel rod temperatures and clad oxidation percentages were calculated using the STRIKIN-II (Reference 5) and PARCH (Reference 6) computer programs. The data transfer between these programs is discussed in detail in Reference 1.

2.2.1 CEFLASH-4AS

The CEFLASH-4AS computer code was used to simulate the blowdown hydraulics during small break LOCA. The CEFLASH-4AS code is a multi-node, multi-flow-path code with which the nuclear steam supply system (NSSS) is described as a series of volume nodes connected by flow paths that contain no volume. The equations of conservation of mass and energy are solved for the volume nodes at each time step. The static pressure in each node is determined at each time step using an equation of state and assuming that the fluid within each node is in thermodynamic equilibrium. The flow-paths connect the volume nodes at specified elevations. The conservation of momentum equation is solved for each flowpath assuming that the fluid within each flowpath is homogeneous and at thermodynamic equilibrium.

2.2.2 COMPERC-Ⅱ

The COMPERC-II computer code is designed specifically to describe the hydraulics in the reactor vessel during the reflooding of the reactor core by emergency core cooling (ECC) water. COMPERC-II describes the fluid within the reactor vessel as residing in five variable volume regions and one fixed volume region. The application of COMPERC-II analysis of an SBLOCA is distinguished by the high pressures occurring in the system during the reflood phase. The pressure varies down from that required for actuation of the ECC injection from the SITs.

2.2.3 STRIKIN-Ⅱ

The STRIKIN-II computer code was applied to the evaluation of fuel rod temperatures during the initial period of blowdown during a SBLOCA. The STRIKIN-II code solves the one-dimensional radial conduction equation at up to 20 axial locations along a fuel rod. It solves the equations of conservation of mass and energy and the equation of state in the fluid channel adjacent to the rod at each of the axial locations used for the conduction solution.

2.2.4 PARCH

The PARCH computer code was applied to the evaluation of fuel rod temperatures during the period following initial reversal of the coolant flow at the core inlet during an SBLOCA. The code describes the removal of heat from a fuel rod that is surrounded by a quasi-static fluid partially or totally covering its length. The code solves the one-dimensional radial conduction equation at up to 21 axial positions along the fuel rod. It solves the conservation of energy equation in both the two-phase and steam regions. The mass flow rate of steam in the steam region is determined from the boiloff and flashing rates computed for the two-phase region and is spatially uniform.

2.3 Data Transfer among Codes

The general method of data transfer among the four computer codes employed for SBLOCA analysis is schematically illustrated in Figure 2.3-1. In general, hydraulics data generated by either CEFLASH-4AS or COMPERC-II are passed to the temperature calculation codes, STRIKIN-II and PARCH. The pressure decay for the whole transient including reflood is calculated using CEFLASH-4AS. This pressure decay is then used in the other codes to determine the fluid properties as functions of time.

The sequence of calculation using the various codes is as follows: CEFLASH-4AS calculation of blowdown hydraulics and reflood pressure transient, STRIKIN-II temperature calculation, PARCH calculation of temperature up to the initiation of reflood, COMPERC-II calculation of reflood hydraulics and PARCH calculation of reflood temperature.

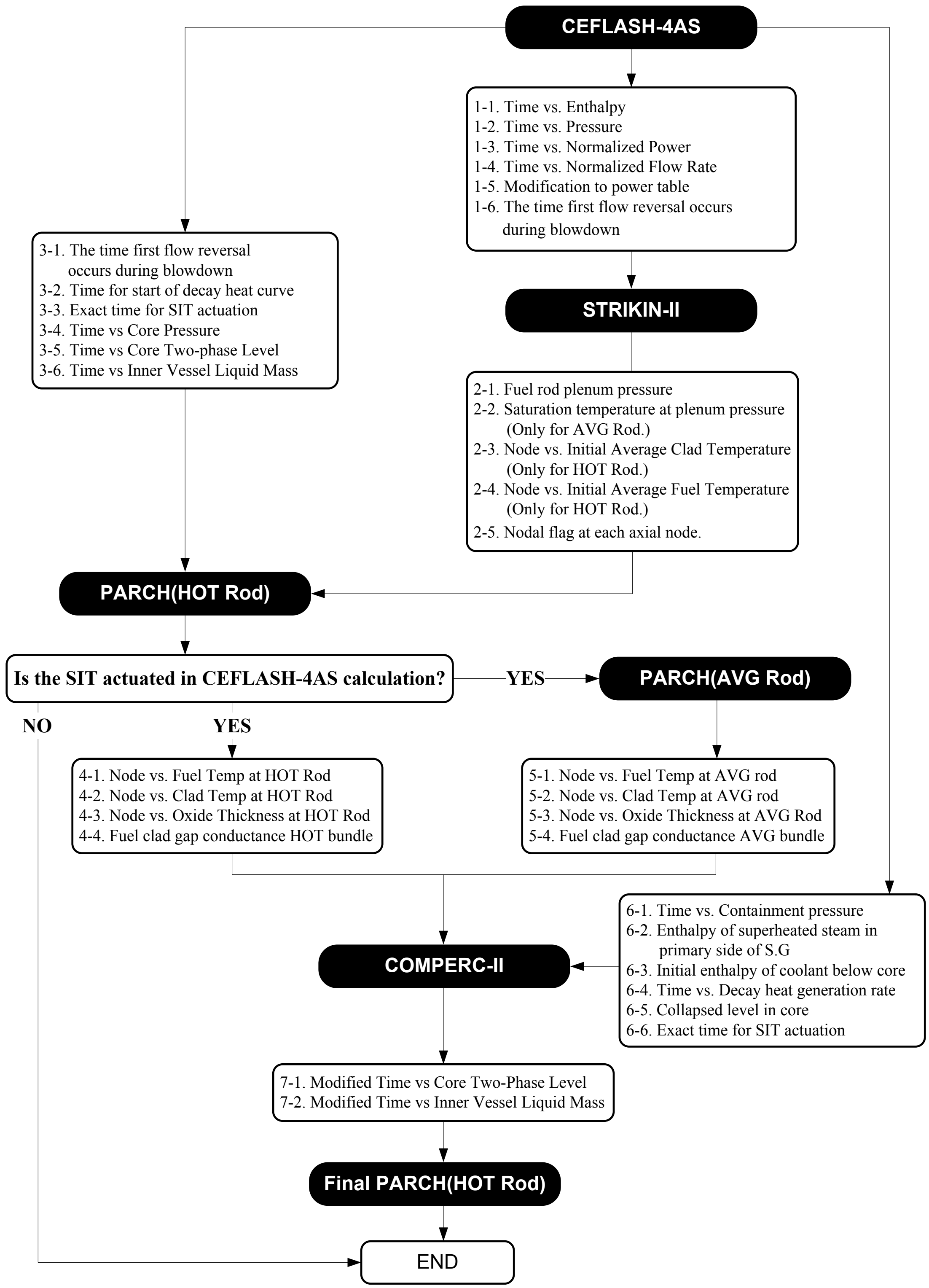

Figure 2.3-1 SBLOCA Analysis Code Flow Chart

3 DESCRIPTION OF MODEL COMPONENTS

Significant features of the SBLOCA analysis methods are described in this section. The use of computer codes to predict particular phenomenological events is discussed. Discussion of the calculation algorithms is limited to that necessary to show which physical processes are modeled. More detailed descriptions of the codes themselves are given in Reference 3 through Reference 6.

3.1 Blowdown Hydraulics

The blowdown period of an SBLOCA is defined as that period of time preceding initiation of ECC injection by the SITs. The hydraulics and depressurization characteristics throughout the primary coolant system during this period are calculated using the CEFLASH-4AS computer code. The principal outputs from the calculation are the transient core inlet flow rate, the two-phase fluid level within the core, the core pressure, and the time of SIT actuation.

3.1.1 CEFLASH-4AS features

The CEFLASH-4AS computer code is a version of the CEFLASH-4A code specifically modified to be applied to the analysis of the blowdown hydraulics during small break LOCAs. The important physical phenomena that are unique to SBLOCA are as follows.

- Phase separation model
  - The bubble rise model in CEFLASH-4AS has been modified to describe more realistically the effects of non-uniform bubble distribution with in a node.
- Heterogeneous flow paths
  - With the phase separation model employed for the transient computations, each control volume or node is described by a separate two-phase and steam region.
- Variable area volume node
  - To account for area discontinuities within the inner vessel, a variable area volume node has been incorporated into the coding. This modification permits proper representation of the coolant volume with the inner vessel when the node exit and inlet elevations are preserved.
- Core heat transfer
  - The blowdown transient can be divided into two distinct regimes for the purpose of heat transfer calculations. The first regime represents the positive flow coastdown when the core is completely covered with a two-phase mixture, while the second characterizes the remainder of the transient initiated by the first flow reversal.
- Loop seal entrainment
  - During the initial portion of the blowdown following SBLOCA in the pump discharge leg, the cold sides of the steam generators and cold legs drain since they are the nearest break. With the suction legs or loop seals modeled with two vertical control volumes to preserve height and volume, only the control volume adjacent to the steam generator drains loses its two-phase inventory. As the cold side of the system drains, two-phase is readily communicated through the loop seal flow path since the path is attached to the bottom of this region. Upon the loss of two-phase from the steam generator side of the loop seals, steam is vented from this section of the loop seal since the steam pressures in the upper

plenum, hot legs and steam generators are sufficient to maintain steam flow in the normal operating direction.

- Wall heat transfer
  - Since SBLOCA is characterized by relatively long blowdown times, heat deposition from the energy stored in the metal walls may have an influence on the transient responses. In particular, the time of actuation of the SITs is determined by the total energy deposited in the primary system fluid prior to the time of actuation.

- Steam generator heat transfer
  - Heat exchange between the primary and secondary sides of the steam generators is accounted for in the energy equation.

3.1.2 NSSS representation

A geometric rendering of the APR1400 NSSS design used for CEFLASH-4AS blowdown hydraulics calculations is shown in Figure 3.1-1. The node diagram represents a typical two-loop primary system with two hot legs and four cold legs. The two loops and the two cold legs of the broken loop are explicitly represented; the two cold legs of the intact loop are represented as one with twice the flow area and initial flow as those of a single cold leg. The leak path is located in the discharge leg, DVI line, or pilot operated safety relief valve (POSRV) line. Any break in the DVI line produces the most severe results for an SBLOCA.

Due to the importance of hydrostatic effects in the distribution of fluid within the primary system during an SBLOCA, a principle applied to NSSS modeling is that all volumes with significant vertical elevations must be explicitly represented as single volume nodes. This principle has been applied to modeling the inverted U-tube steam generators, the loop-seals, and the reactor vessel.

The node diagram shown in Figure 3.1-1 employs 19 volume nodes and 25 flow-paths to model the primary system. The secondary system is modeled by three volume nodes and two flow-paths. The leak is modeled by a flowpath connected to a volume node representing the containment building. The emergency core cooling system (ECCS) is modeled by flow paths connected to the lower annulus and containment nodes.

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Figure 3.1-1 CEFLASH-4AS Node Diagram

3.2 Reflood Hydraulics

The reflood period during an SBLOCA is defined as the period of time following initiation of ECC injection by the SITs. The fluidic devices of SITs, however, are not modeled in the reflood analysis because the water level of SITs does not drop to the top of the stand pipe in the reflood period during an SBLOCA. The hydraulics within the reactor core during this period were calculated using the COMPERC-II computer code. The principal output from the calculation was generally the transient level of two-phase fluid within the core. In cases that involve total core uncovery, the COMPERC-II code was used to calculate reflood convection heat transfer coefficients based on FLECHT data.

3.2.1 COMPERC-Ⅱ features

The COMPERC-II computer code is an assembly of several computer codes used to calculate various aspects of the reflood period. The principal subcode is the PERC code which is used to calculate the heat transfer and hydraulic characteristics within the reactor vessel. Peripheral subcodes are used to calculate the rate of discharge of the SITs, the initial core temperature distribution, the resistance to venting of the reflood-generated steam, the average reflood rates, the heat transfer coefficients based on FLECHT and the fluid properties.

- Fluid description
  - The fluid within the core is divided into three axial regions the sizes of which vary with time and are calculated as functions of their mass and energy contents. These regions are (1) subcooled water, (2) saturated water and steam (two-phase) and (3) saturated and super-heated steam.
- Core heat transfer
  - The transfer of heat from the core to the fluid is calculated using a simplified conduction/convection model. The core is represented by up to twenty axial and two radial zones. The initial fuel and clad axial temperature distributions for the two radial zones are obtained from PARCH calculations for average and hot fuel rods.
- FLECHT heat transfer correlations
  - Applying FLECHT data to SBLOCA analysis is somewhat unique compared to the use of such data for a LBLOCA because of the higher pressure that occurs during reflood following an SBLOCA. The FLECHT correlations are formulated for a pressure range from 1.05 to 6.33 kg/cm^{2}A(15 to 90 psia). SBLOCA cases in which FLECHT can be applied have pressures during reflood from about 2.81 to 16.17 kg/cm^{2}A(40 psia to about 230 psia). The FLECHT correlations are applied using the reference pressure at the time when the water level reaches the bottom of the core.

3.3 Fuel Rod Temperature

The SBLOCA transient is characterized by two regimes. During the initial blowdown period, while the main coolant pumps continue to operate, the primary system flow rates are high, nearly equal to the initial pre-rupture flow rates.

The later portion of the blowdown transient and the refill period are quiescent. Hydrostatic equilibrium is established between the hot and cold sides of the primary system, as fluid drains from, or resurges into, the inner vessel.

As a result of these markedly different blowdown regimes, two fuel rod temperature codes are needed in the SBLOCA analysis. STRIKIN-II, since it is a force convection heat transfer code, is used during the initial high-flow period immediately following a rupture. PARCH is a pool boiling fuel rod temperature code. It uses the time-varying two-phase level in the core to calculate fuel rod temperatures during the quiescent blowdown and refill portions of the SBLOCA transient.

3.3.1 Small break STRIKIN-Ⅱ model

- Fuel rod model
  - A single fuel rod representing the hottest rod in the core is modeled. The rod is divided into twenty equal-length axial segments. Temperatures are computed at the interfaces between segments, yielding temperatures at twenty one axial positions. Each axial interface is further subdivided into nine radial regions, one for the cladding, one for the gas gap and seven equal-thickness fuel pellet regions.
- Heat transfer regimes
  - The small break STRIKIN-II model considers the following force convection heat transfer regimes and respective correlation.
    - forced convection to subcooled water - Dittus-Boelter
    - nucleate boiling -Thom
    - transition boiling - McDonough, Milich and King
    - stable film boiling - Dougall-Rohsenow
    - heat transfer to steam - Dittus-Boelter
- Heat transfer regime selection logic
  - The heat transfer logic is the same as employed for large break analysis.
- Dynamic fuel rod model
  - The STRIKIN-II representations of variable gap conductance and fuel rod geometry changes, including clad swelling and rupture, are the same as employed for large break analysis.
- Coolant representation
  - The CEFLASH-4AS small break blowdown model does not contain an open core

representation. The axially variable mass flow rate and coolant enthalpy along the hot rod are therefore not obtained directly from the hydraulics calculation. For this reason a mass and energy balance must be performed in STRIKIN-II in order to determine local coolant conditions along the hot rod. The inlet flow and enthalpy are specified as boundary conditions. The momentum equation is solved in CEFLASH-4AS and its solution is reflected in the inlet flow used in STRIKIN-II.

– The coolant channel, like the fuel rod, is divided into twenty axial segments. Coolant properties are determined at the interfaces between segments, and are assumed to extend halfway into the adjacent segments.

3.3.2 Small break PARCH model

- Fuel rod model

  – After the transition time, temperature calculations are continued using the PARCH code. A single fuel rod representing the hottest rod in the reactor core is divided into twenty axial segments. An energy balance is performed at the ends of each segment giving temperatures at twenty-one axial positions along the rod.

  – PARCH uses a simplified three region radial conduction model with the regions being the fuel pellet, the cladding and the coolant. A quasi-steady state balance is performed for each of the three regions.

- Heat transfer regimes

  – The PARCH heat transfer regimes and correlations are:

    - pool nucleate boiling - Rohsnow
    - transition boiling - McDonough, Milich and King
    - stable film boiling in a pool - Modified Bromley
    - steam cooling - laminar flow - Sieder-Tate
    - steam cooling - turbulent flow - Dittus-Boelter
    - steam cooling - transition flow - interpolation
    - refill heat transfer - optional input table

- Dynamic fuel rod model

  – The PARCH small break model includes a dynamic gap conductance calculation similar to that used in the STRIKIN-II model. This model includes the contributions of conduction through contact points, conduction across the fill gas and radiation between the pellet and the cladding.

- Coolant representation

  – The transient two-phase level in the core is input to PARCH in tabular form. Pool boiling to saturated liquid is assumed below this surface. The saturation temperature corresponds to the transient inner vessel pressure computed in CEFLASH-4AS. This core pressure is also input to PARCH in tabular form. An energy balance in the coolant channel below the two-phase surface is performed in order to determine the rate of steam generation. The boiloff rate includes the integral of the surface heat flux below the two-phase surface, as well as the fraction of the decay heat directly generated in the coolant.

4 SBLOCA ANALYSIS

4.1 Input Parameters

Numerous parameters, which are input variables for the various computer codes have been given prescribed values as parts of the SBLOCA analysis methodology. Several of these parameters are options that are built into the various codes, themselves.

The safety injection system (SIS) consists of four DVI lines each supplying flow from one SIT and one safety injection (SI) pump. It is conservatively assumed that offsite power is lost upon reactor trip, and therefore the SIPs must await EDGs startup and load sequencing before they can start. The total time delay assumed to be 40 seconds from the time that the safety injection actuation signal (SIAS) setpoint is reached to the time that full SI flow is delivered to the reactor coolant system (RCS). For breaks in the DVI line, it is also assumed that all safety injection flow delivered to the broken line spills out of the break.

An analysis of the possible single failures that can occur within the SIS has shown that the worst single failure for the small break spectrum is the failure of one of the EDGs to start. According to the worst single failure assumption, three SIPs are operable. However, for additional conservatism, only two SIPs are assumed to be operable in the present SBLOCA analysis thereby minimizing the safety injection available to cool the core.

Therefore, based on the above assumptions, the following safety injection flows are credited for the small break analysis:

- For a break in the pump discharge leg, the SI flow credited is full flow from two SIPs and four SITs
- For a break in a DVI line, the SI flow credited is full flow from one SIPs and three SITs. The flow from the remaining active SIP and from one SIT is assumed to spill out of the break

The SIP flow rates assumed at each of the four injection points as a function of RCS pressure are described in DCD Tier2 Section 15.6.5.

The significant core and system parameters used in the small break calculations are presented in DCD Tier2 15.6.5. A peak linear heat generation rate (PLHGR) of 49.2kW/m is assumed to occur 15 % of the distance from the top of the active core. A conservative beginning-of-life moderator temperature coefficient of 0.0×10^{-4} Δρ/℃ was used in all small break calculations.

The initial steady state fuel rod conditions were obtained from the FATES3 (Reference 7) computer program. The small break analyses employed a hot rod average burnup that maximized the amount of stored energy in the fuel.

The small break analysis uses the containment parameters of the initial containment pressure and the maximum containment volume. However, containment parameters do not influence the small break analysis because the break flow maintains criticality.

4.2 Calculation Results

The 33 breaks analyzed at 4,062.66 MWt (102 % of nominal) include reactor coolant pump discharge leg breaks ranging in size from 506.69 cm^2 to 20.25 cm^2 and DVI line breaks ranging in size from 372 cm^2 to 11.43 cm^2. One break, equal in area to a fully open pressurizer safety valve, 27.9 cm^2, is postulated to occur at the top of the pressurizer. The 465 cm^2 discharge leg break is also analyzed for the large break spectrum and is defined as the transition break size (Reference 1).

The transient behavior of important NSSS parameters is shown in the figures listed in DCD Tier2 section 15.6.5. Table 4.2-1 summarizes the main results of this analysis. Times of interest for the various breaks analyzed are presented in Table 4.2-2. A plot of peak cladding temperature (PCT) versus break size is presented in Figure 4.2-49. The 126.7 cm^2 DVI break results in the highest cladding temperature (918℃) of the small breaks analyzed. Of the two break locations (pump discharge leg and DVI line), the DVI location is limiting due to the assumed loss of all safety injection flow to the broken line.

For the DVI line break location, as the break size becomes progressively smaller than 372 cm^2, the inner vessel two phase level follows a definite pattern:

- The time of initial core uncovery is delayed.
- The rate of level decrease and increase becomes slower.

This trend continues until the core does not uncover at all. These trends affect PCT.

The core uncover does not occur less than the 31.68 cm^2. Thus, by analyzing dense break spectrum over this range, the behavior of PCT versus break size is adequately determined.

The above behavior of core uncovery results from the design characteristics of the SIS. For DVI break sizes below 62 cm^2, the RCS pressure remains above SIT pressure and coolant flow injection to the reactor vessel is accomplished entirely by one SIP. For break sizes greater than 62 cm^2, the transient is terminated by the SITs and the SIP.

For cold leg breaks, the additional SIS flow resulting from being able to credit two SIPs precludes core uncovery for break sizes up to 153 cm^2. In addition, core uncovery for break sizes greater than 153 cm^2 is delayed and the depth and duration of uncovery decreases relative to DVI breaks that credit only one SIP. This more favorable behavior results in lower cladding temperatures relative to breaks in a DVI line.

In addition to the break discussed above, the rupture of an in-core instrument tube and very small break size of DVI and cold leg are considered as shown in Table 4.2-3. The typical phenomena of small break LOCA such as the core uncovery and loop seal clearing are not shown for the very small break size.

In the very small break LOCA, following rupture, the primary system depressurizes until a reactor scram signal and SIAS are generated due to low pressurizer pressure at 109.3 kg/ cm^2A. The assumed loss of offsite power causes the primary coolant pumps and the feedwater pumps to coast down. After the 40 second delay required to actuate the EDGs and the SIPs following SIAS, safety injection flow is initiated. Due to the assumed failure of one EDG, only two SIPs are available. (Four SITs are available but do not inject due to the high RCS pressure.)

The primary side depressurization continues accompanied by a rise in the secondary side pressure until the secondary side pressure reaches the lowest set point of the steam generator safety valves. The primary system pressure continues to fall until it is just slightly higher than the secondary side pressure. At this point, the flow from the two operating SIPs for the cold leg break and the rupture of an in-core instrument tube exceeds the leak flow. The flow from the one operating SIP for DVI break exceeds the leak flow. Comparison of leaking flow and charging flow by SIPs are summarized in the Table 4.2-3.

Therefore, the RCS will fill. The decay heat generated in the core is removed from the steam generators by steam flow through the secondary side safety valves. Thus, the core will remain covered and cooled in this condition.

Table 4.2-1 Results of SBLOCA Break Spectrum Analysis (1 of 3)

Case	Direct Vessel Injection Line Break size		Peak Cladding Temperature (℉)	Maximum Cladding Oxidation (%)	Maximum Core-Wide Oxidation (%)
	Diameter (in)	Area (ft^2)			
1	1.5	0.0123	1045.6	0.00096	<0.00033
2	2	0.0218	1042.6	0.00107	<0.00039
3	2.5	0.0341	1040.7	0.00129	<0.00035
4	3	0.0491	1053.6	0.00420	<0.00050
5	3.5	0.0668	1056.8	0.00111	<0.00019
6	4	0.0873	1060.3	0.01541	<0.00178
7	4.5	0.1104	1527.9	0.22653	<0.02017
8	5	0.1364	1683.9	0.55474	<0.05679
9	5.5	0.1650	1319.1	0.05786	<0.00558
10	6	0.1963	1229.9	0.02875	<0.00284
11	6.5	0.2304	1195.6	0.02019	<0.00206
12	7	0.2673	850.1	0.00081	<0.00010
13	7.5	0.3068	877.6	0.00099	<0.00017
14	8	0.3491	1015.2	0.00391	<0.00060
15	8.5	0.4006	1104.2	0.01106	<0.00133

Table 4.2-1 Results of SBLOCA Break Spectrum Analysis (2 of 3)

Case	Pump Discharge Leg Break size		Peak Cladding Temperature (℉)	Maximum Cladding Oxidation (%)	Maximum Core-Wide Oxidation (%)
	Diameter (in)	Area (ft^2)			
1	2	0.0218	1050.5	0.00114	<0.00030
2	2.5	0.0341	1040.9	0.00113	<0.00027
3	3	0.0491	1055.2	0.00111	<0.00024
4	3.5	0.0668	1061.1	0.00101	<0.00016
5	4	0.0873	1044.1	0.00098	<0.00015
6	4.5	0.1104	1050.6	0.00091	<0.00013
7	5	0.1364	1037.2	0.00087	<0.00012
8	5.5	0.1650	725.0	0.00008	<0.00005
9	6	0.1963	725.0	0.00017	<0.00006
10	6.5	0.2304	782.1	0.00029	<0.00008
11	7	0.2673	794.0	0.00033	<0.00008
12	7.5	0.3068	872.1	0.00095	<0.00015
13	8	0.3491	899.1	0.00121	<0.00021
14	8.5	0.4006	920.8	0.00245	<0.00039
15	9	0.4418	921.7	0.00192	<0.00032
16	9.5	0.4922	944.2	0.00196	<0.00029
17	10	0.5454	958.2	0.00225	<0.00032

Table 4.2-1 Results of SBLOCA Break Spectrum Analysis (3 of 3)

Case	Top of Pressurizer		Peak Cladding Temperature (℉)	Maximum Cladding Oxidation (%)	Maximum Core-Wide Oxidation (%)
	Diameter (in)	Area (ft2)			
1	2.3	0.03	1055.2	0.0004	<0.0002

Table 4.2-2 Time Sequence of SBLOCA Break Spectrum Analysis (1 of 3)

Case	Direct Vessel Injection Line Break size		SI Pump Flow Delivered to RCS (sec)	SI Tank Flow Delivered to RCS (sec)	Hot Spot Peak Clad Temperature Occurs (sec)
	Diameter (in)	Area (ft^2)			
1	1.5	0.0123	905.2	N/A[1]	786.0
2	2	0.0218	582.8	N/A	455.3
3	2.5	0.0341	396.8	N/A	297.7
4	3	0.0491	290.0	N/A	215.3
5	3.5	0.0668	195.0	2189.2	151.2
6	4	0.0873	155.0	1232.4	115.3
7	4.5	0.1104	129.0	872.2	936.0
8	5	0.1364	114.0	621.2	686.5
9	5.5	0.1650	93.0	524.8	562.1
10	6	0.1963	81.0	429.0	462.1
11	6.5	0.2304	75.0	356.8	378.1
12	7	0.2673	69.0	305.2	322.0
13	7.5	0.3068	66.0	261.2	275.9
14	8	0.3491	63.0	223.6	102.8
15	8.5	0.4006	60.0	191.9	219.0

1 N/A : calculation terminated before initiation of SIT discharge

Table 4.2-2 Time Sequence of SBLOCA Break Spectrum Analysis (2 of 3)

Case	Pump Discharge Leg Break size		SI Pump Flow Delivered to RCS (sec)	SI Tank Flow Delivered to RCS (sec)	Hot Spot Peak Clad Temperature Occurs (sec)
	Diameter (in)	Area (ft^2)			
1	2	0.0218	568.0	N/A	471.6
2	2.5	0.0341	396.0	N/A	301.0
3	3	0.0491	292.0	N/A	212.5
4	3.5	0.0668	192.0	2046.2	152.1
5	4	0.0873	156.0	1475.3	113.0
6	4.5	0.1104	129.0	1005.5	90.4
7	5	0.1364	114.0	634.0	73.4
8	5.5	0.1650	93.0	532.1	0.0
9	6	0.1963	81.0	419.0	0.0
10	6.5	0.2304	75.0	340.3	385.2
11	7	0.2673	69.0	286.1	321.1
12	7.5	0.3068	66.0	245.0	271.1
13	8	0.3491	62.0	212.8	102.7
14	8.5	0.4006	60.0	184.1	102.0
15	9	0.4418	58.0	166.7	89.9
16	9.5	0.4922	57.0	152.3	174.5
17	10	0.5454	55.5	138.7	160.1

Table 4.2-2 Time Sequence of SBLOCA Break Spectrum Analysis (3 of 3)

Case	Top of Pressurizer		SI Pump Flow Delivered to RCS (sec)	SI Tank Flow Delivered to RCS (sec)	Hot Spot Peak Clad Temperature Occurs (sec)
	Diameter (in)	Area (ft2)			
1	2.3	0.03	800.0	N/A	750.0

Table 4.2-3 The Evaluation Results for the Very Small Break LOCA Spectrum

Case	Break Location	Break Size		Break flowrate		ECCS flowrate	
		Diameter (In)	Area (ft^2)	lbm/sec	kg/sec	lbm/sec	kg/sec
1	DVI	0.5	0.0014	12.25	5.56	62.82	28.9
2		1	0.0055	48.13	21.80	62.82	28.9
3	PDL	0.5	0.0014	12.25	5.56	127.6	57.9
4		1	0.0055	48.13	21.80	127.6	57.9
5		1.5	0.0123	107.64	48.80	127.6	57.9
6	In-core Instrument tube	0.75	0.003	26.25	11.9	127.6	57.9

TS

Figure 4.2-1 102.5cm^2 (0.1104ft^2) Break in DVI Line: Nomalized Total Core Power

TS

Figure 4.2-2 102.5cm^2 (0.1104ft^2) Break in DVI Line: Inner Vessel Pressure

TS

Figure 4.2-3 102.5cm² (0.1104ft²) Break in DVI Line: Break Flow Rate

TS

Figure 4.2-4 102.5cm^2 (0.1104ft^2) Break in DVI Line: Inner Vessel Inlet Flow

TS

Figure 4.2-5 102.5cm^2 (0.1104ft^2) Break in DVI Line: Inner Vessel Two-Phase Mixture Level

TS

Figure 4.2-6 102.5cm^2 (0.1104ft^2) Break in DVI Line: Heat Transfer Coefficient at Hot Spot

TS

Figure 4.2-7 102.5cm^2 (0.1104ft^2) Break in DVI Line: Coolant Temperature at Hot Spot

TS

Figure 4.2-8 102.5cm^2 (0.1104ft^2) Break in DVI Line: Hot Spot Clad Surface Temperature

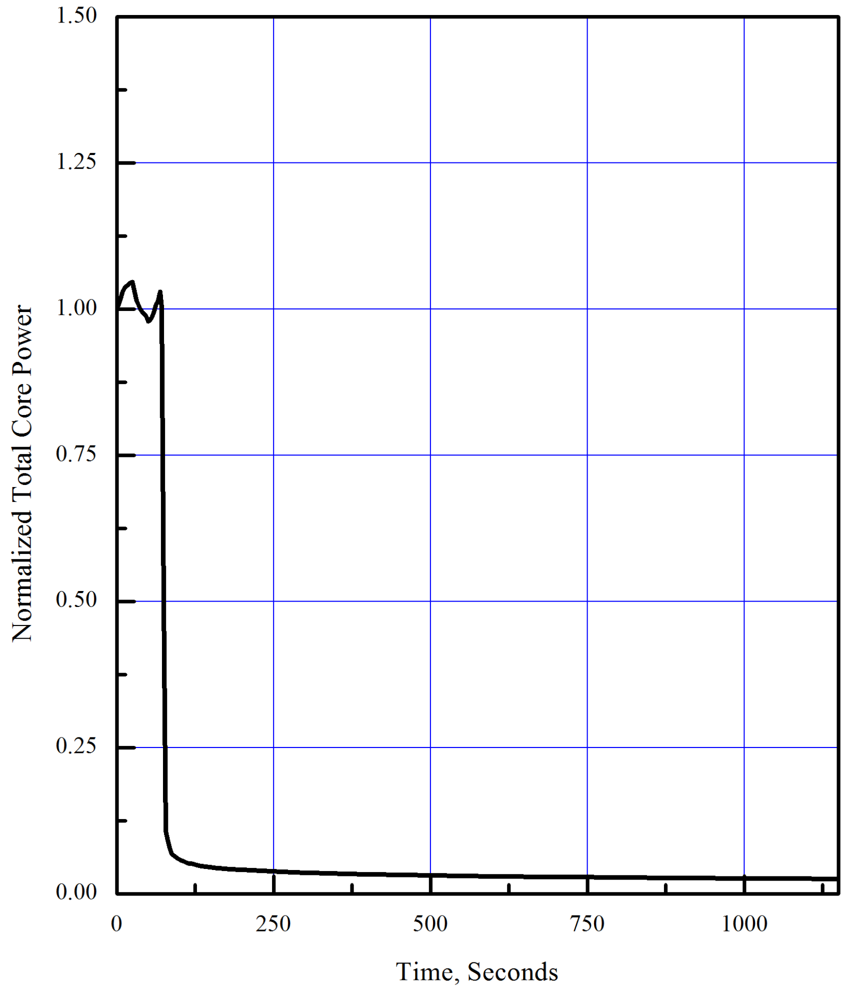

Figure 4.2-9 126.7cm^2 (0.1364ft^2) Break in DVI Line: Nomalized Total Core Power

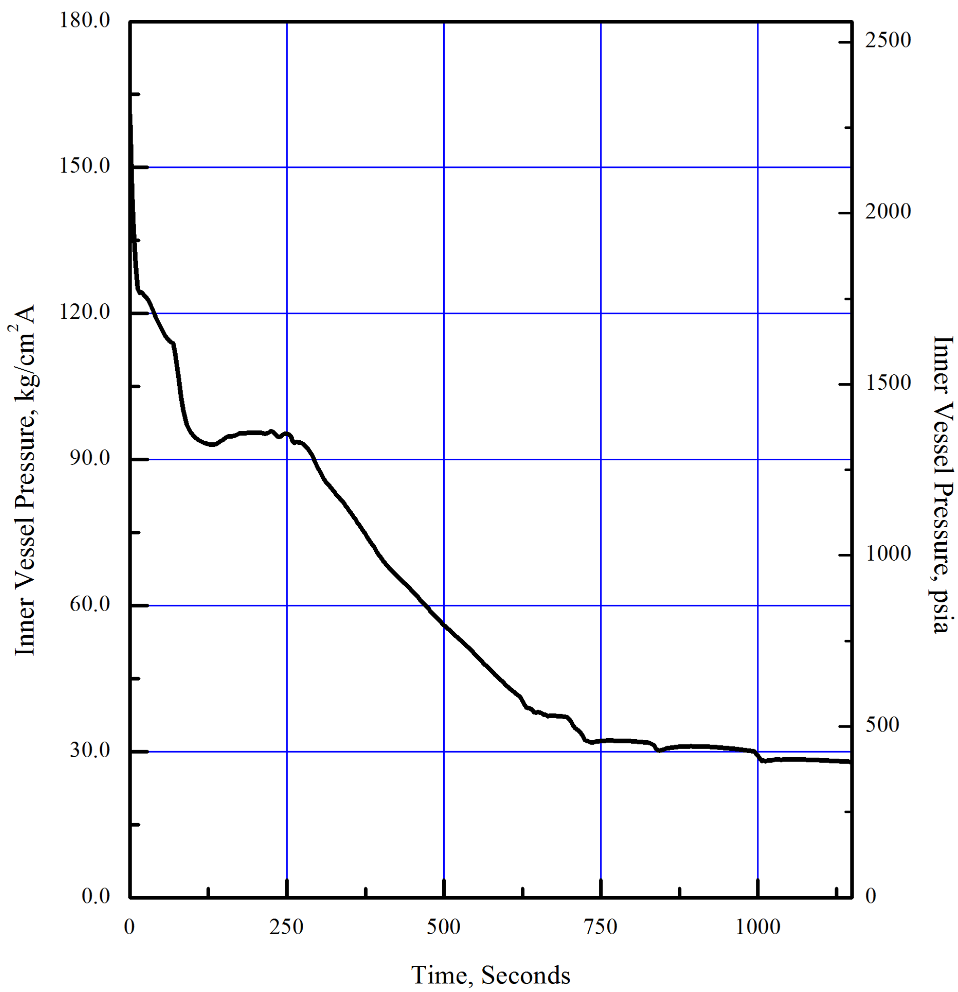

Figure 4.2-10 126.7cm² (0.1364ft²) Break in DVI Line: Inner Vessel Pressure

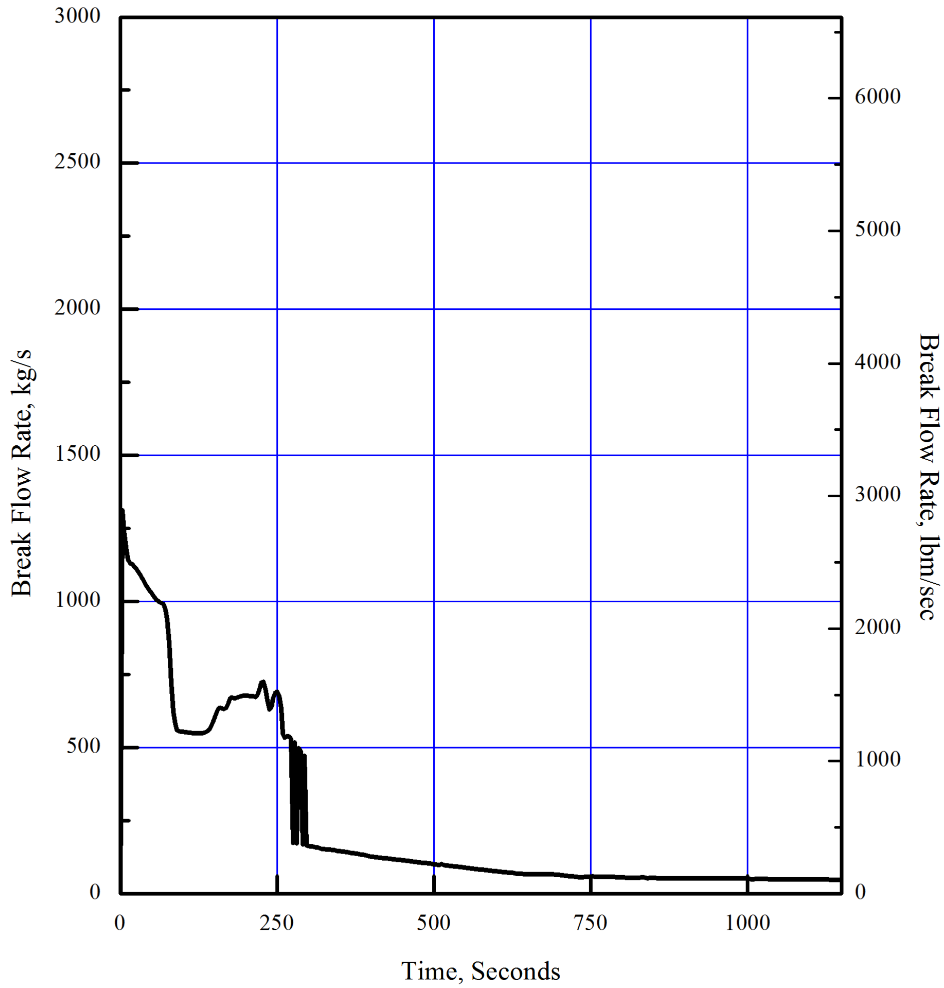

Figure 4.2-11 126.7cm^2 (0.1364ft^2) Break in DVI Line: Break Flow Rate

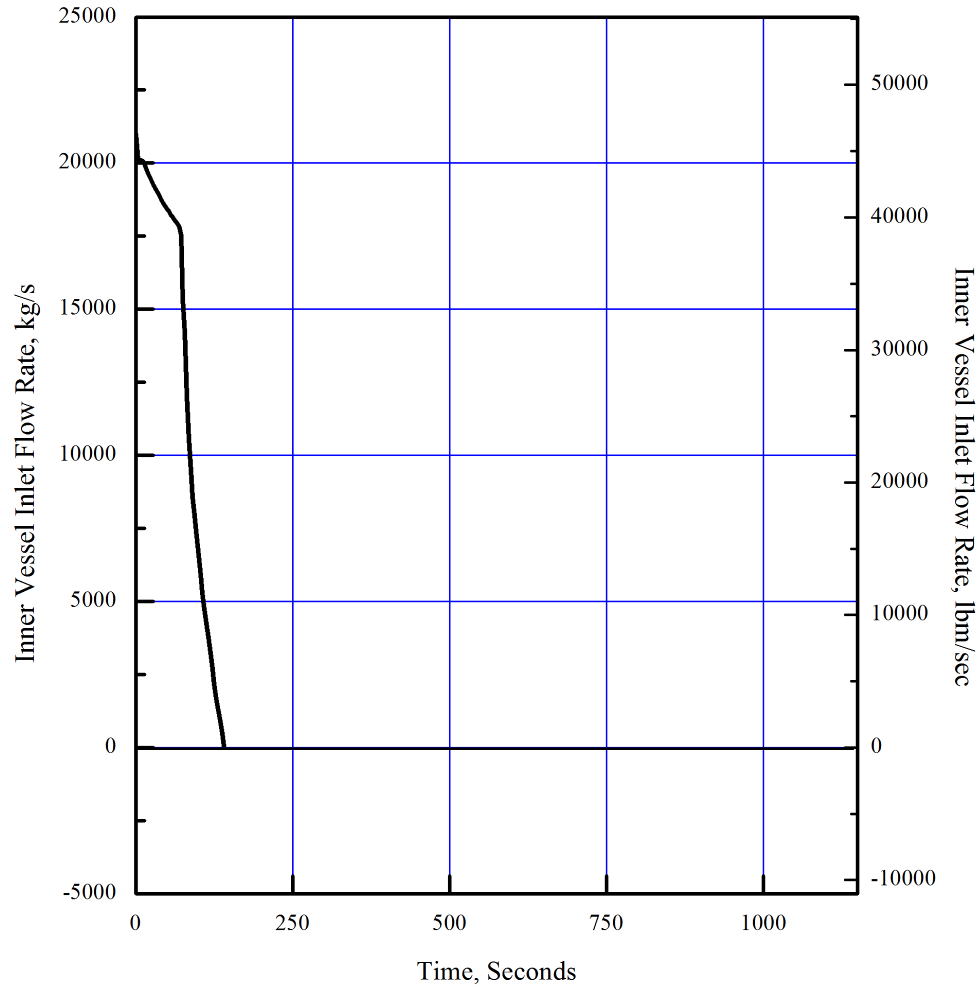

Figure 4.2-12 126.7cm^2 (0.1364ft^2) Break in DVI Line: Inner Vessel Inlet Flow

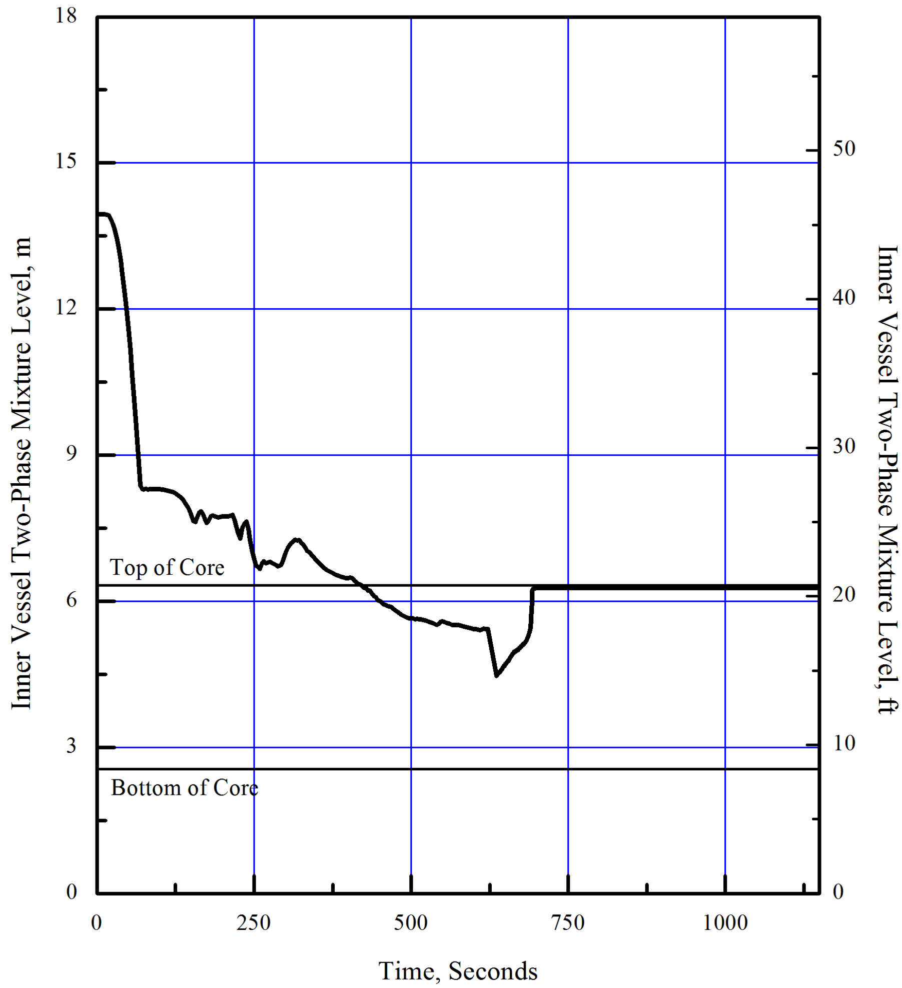

Figure 4.2-13 126.7cm^2 (0.1364ft^2) Break in DVI Line: Inner Vessel Two-Phase Mixture Level

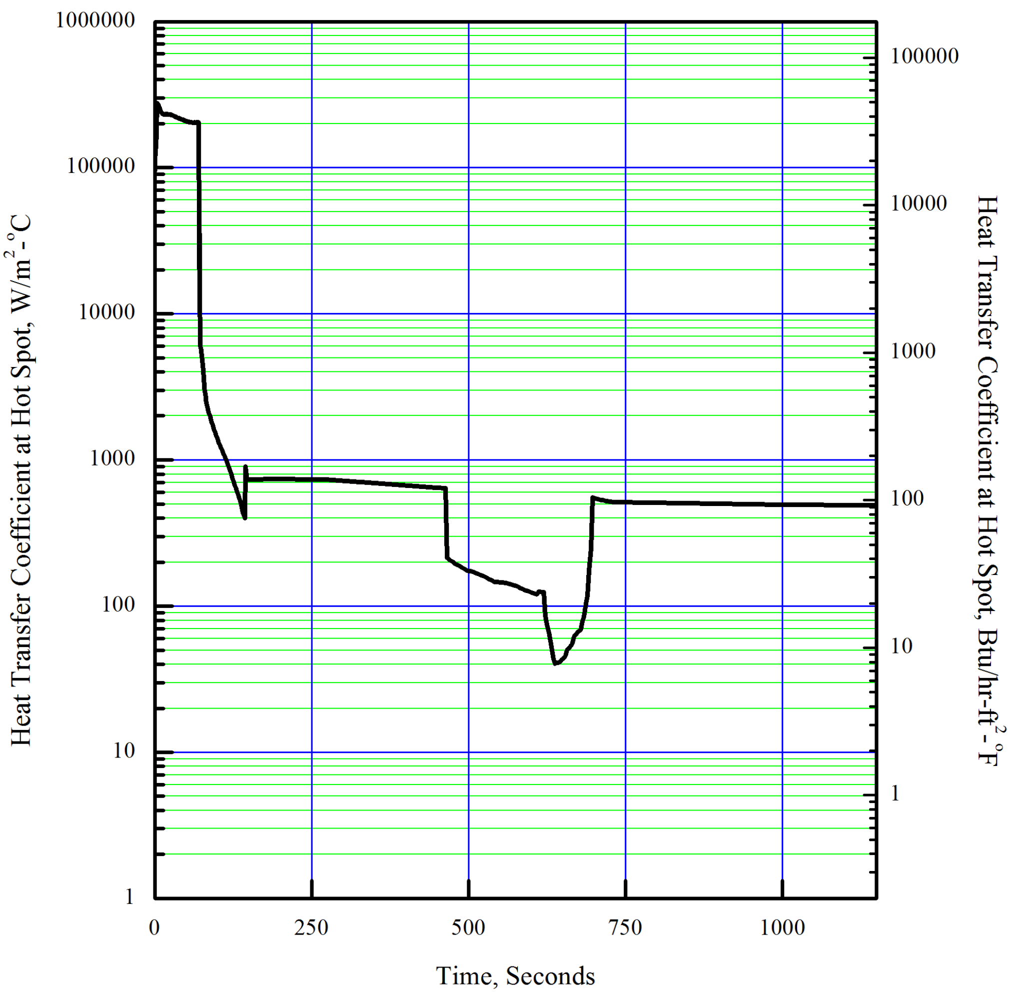

Figure 4.2-14 126.7cm^2 (0.1364ft^2) Break in DVI Line: Heat Transfer Coefficient at Hot Spot

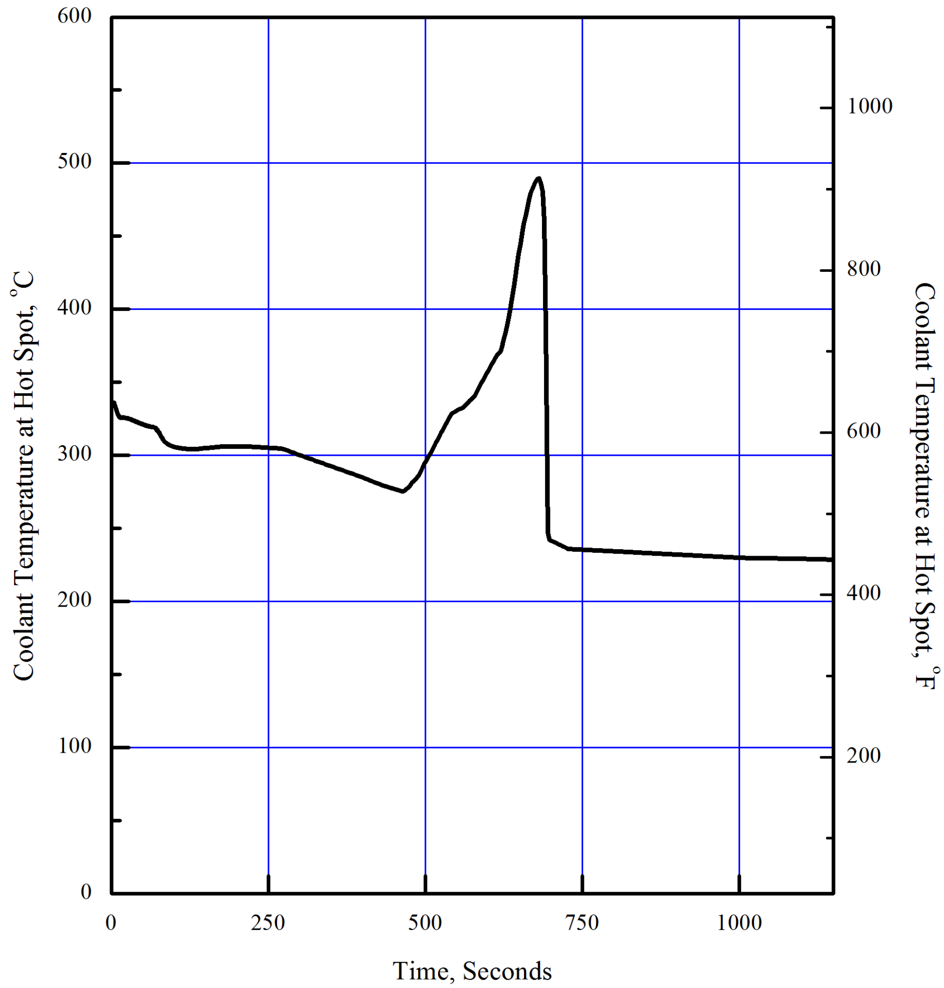

Figure 4.2-15 126.7cm^2 (0.1364ft^2) Break in DVI Line: Coolant Temperature at Hot Spot

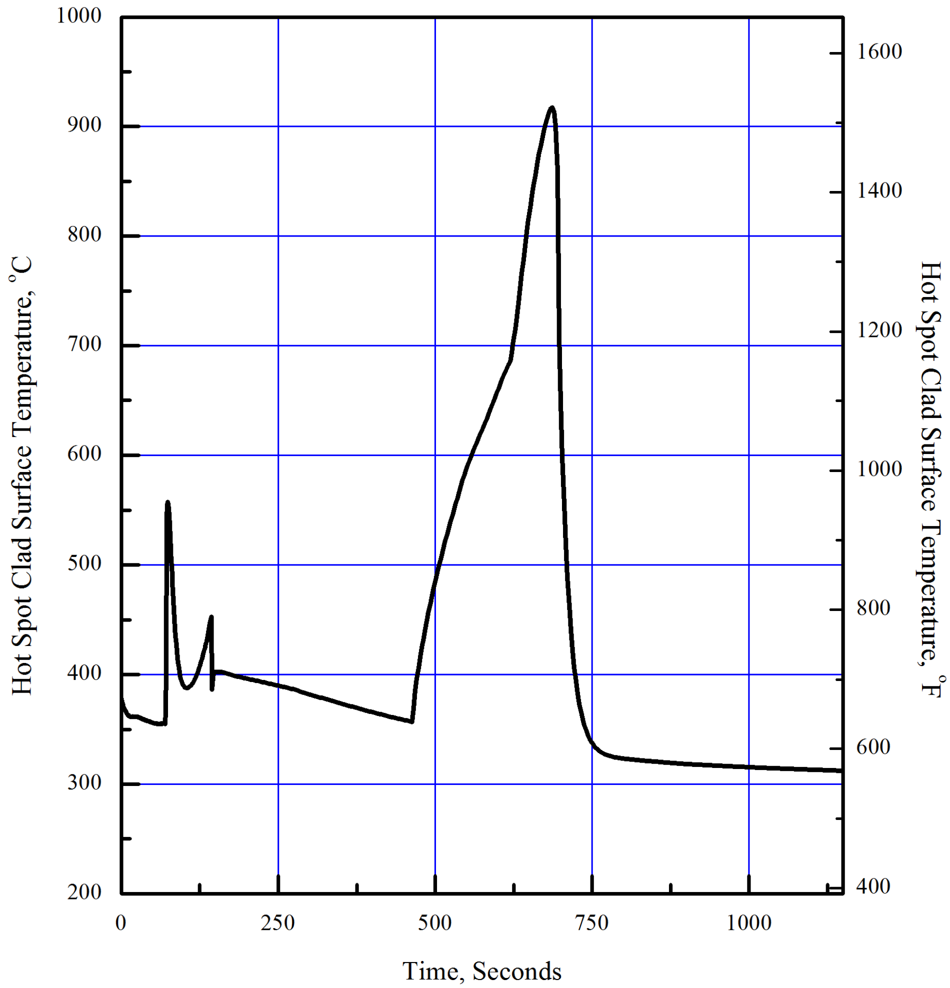

Figure 4.2-16 126.7cm^2 (0.1364ft^2) Break in DVI Line: Hot Spot Clad Surface Temperature

TS

Figure 4.2-17 153.3cm^2 (0.1650ft^2) Break in DVI Line: Nomalized Total Core Power

TS

Figure 4.2-18 153.3cm^2 (0.1650ft^2) Break in DVI Line: Inner Vessel Pressure

TS

Figure 4.2-19 153.3cm^2 (0.1650ft^2) Break in DVI Line: Break Flow Rate

TS

Figure 4.2-20 153.3cm^2 (0.1650ft^2) Break in DVI Line: Inner Vessel Inlet Flow

TS

Figure 4.2-21 153.3cm^2 (0.1650ft^2) Break in DVI Line: Inner Vessel Two-Phase Mixture Level

TS

Figure 4.2-22 153.3cm^2 (0.1650ft^2) Break in DVI Line: Heat Transfer Coefficient at Hot Spot

TS

Figure 4.2-23 153.3cm^2 (0.1650ft^2) Break in DVI Line: Coolant Temperature at Hot Spot

TS

Figure 4.2-24 153.3cm^2 (0.1650ft^2) Break in DVI Line: Hot Spot Clad Surface Temperature

TS

Figure 4.2-25 102.5cm^2 (0.1104ft^2) Break in Pump discharge Leg: Nomalized Total Core Power

TS

Figure 4.2-26 102.5cm^2 (0.1104ft^2) Break in Pump discharge Leg: Inner Vessel Pressure

Non-Proprietary

TS

Figure 4.2-27 102.5cm^2 (0.1104ft^2) Break in Pump discharge Leg: Break Flow Rate

TS

Figure 4.2-28 102.5cm^2 (0.1104ft^2) Break in Pump discharge Leg: Inner Vessel Inlet Flow

TS

Figure 4.2-29 102.5cm^2 (0.1104ft^2) Break in Pump discharge Leg: Two-Phase Mixture Level

TS

Figure 4.2-30 102.5cm^2 (0.1104ft^2) Break in Pump discharge Leg: Heat Transfer Coefficient at Hot Spot

TS

Figure 4.2-31 102.5cm^2 (0.1104ft^2) Break in Pump discharge Leg: Coolant Temperature at Hot Spot

TS

Figure 4.2-32 102.5cm^2 (0.1104ft^2) Break in Pump discharge Leg: Hot Spot Clad Surface Temperature

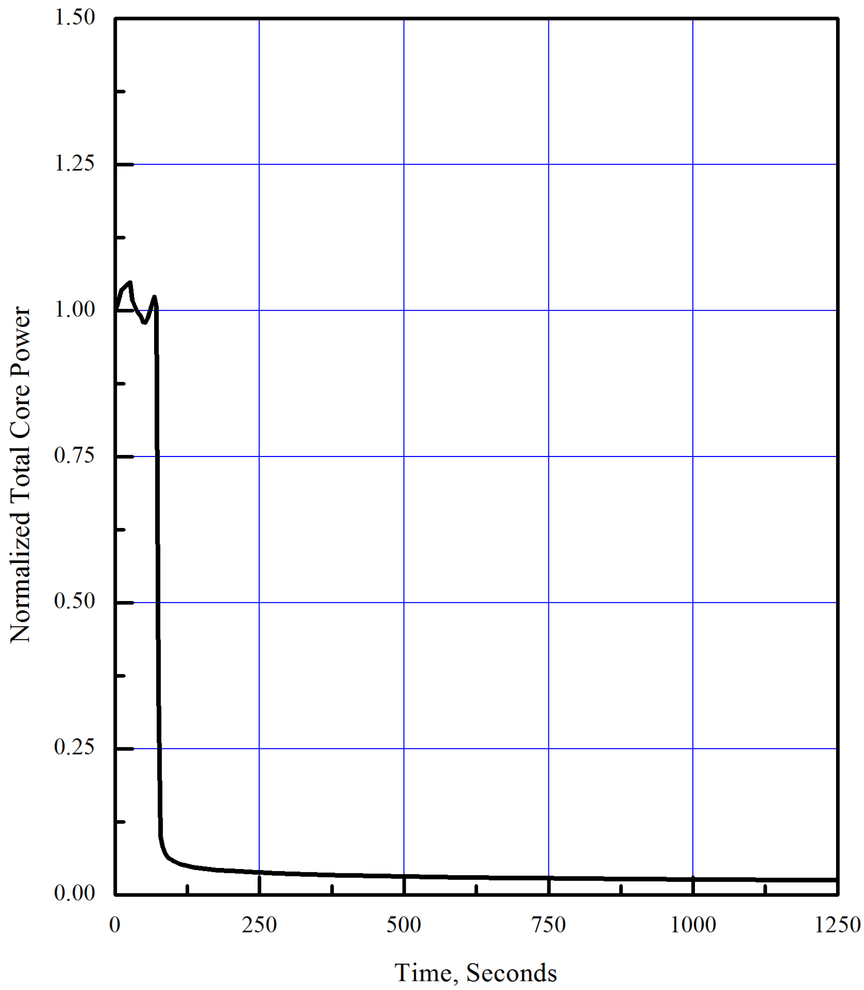

Figure 4.2-33 126.7cm^2 (0.1364ft^2) Break in Pump discharge Leg: Nomalized Total Core Power

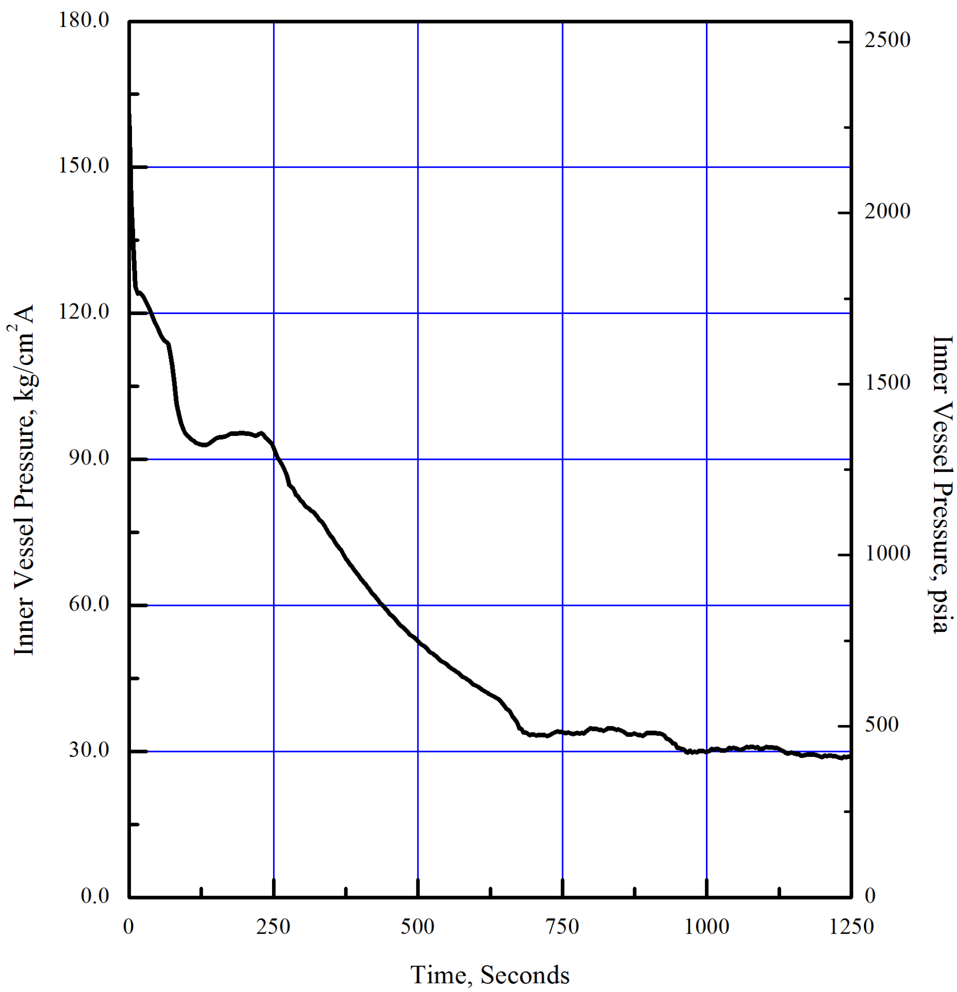

Figure 4.2-34 126.7cm^2 (0.1364ft^2) Break in Pump discharge Leg: Inner Vessel Pressure

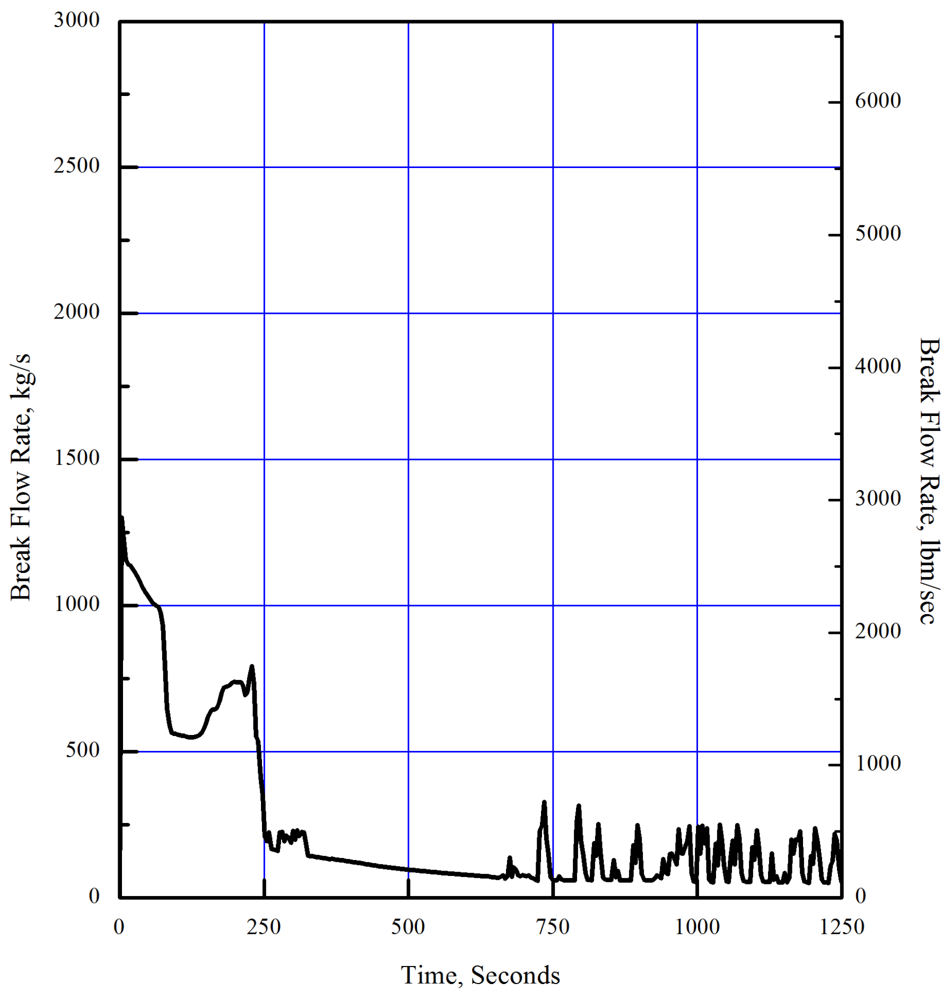

Figure 4.2-35 126.7cm^2 (0.1364ft^2) Break in Pump discharge Leg: Break Flow Rate

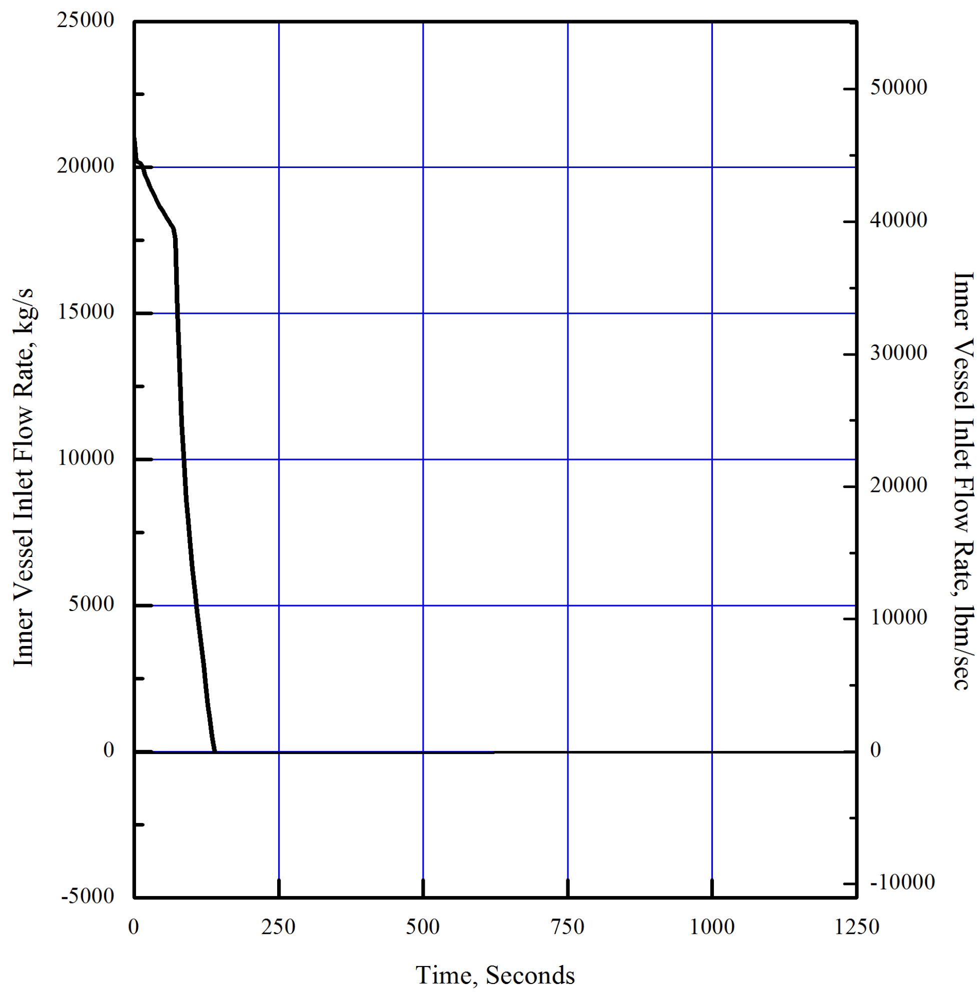

Figure 4.2-36 126.7cm^2 (0.1364ft^2) Break in Pump discharge Leg: Inner Vessel Inlet Flow

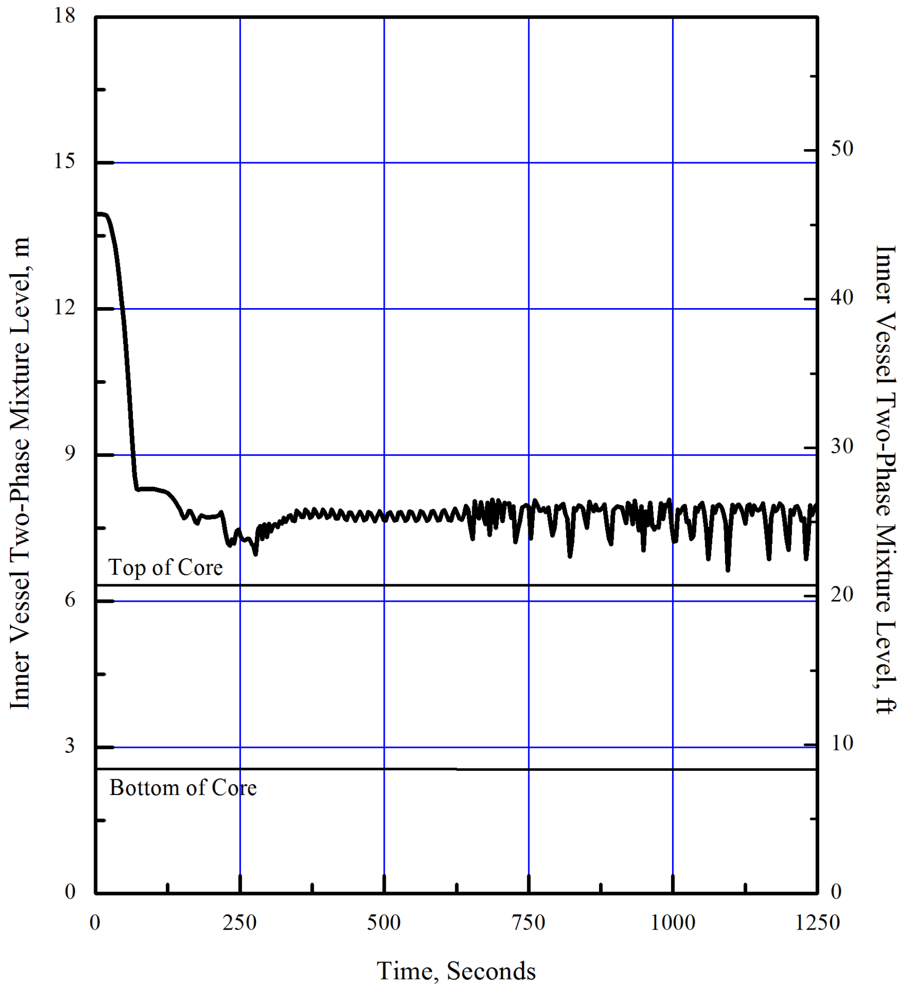

Figure 4.2-37 126.7cm^2 (0.1364ft^2) Break in Pump discharge Leg: Two-Phase Mixture Level

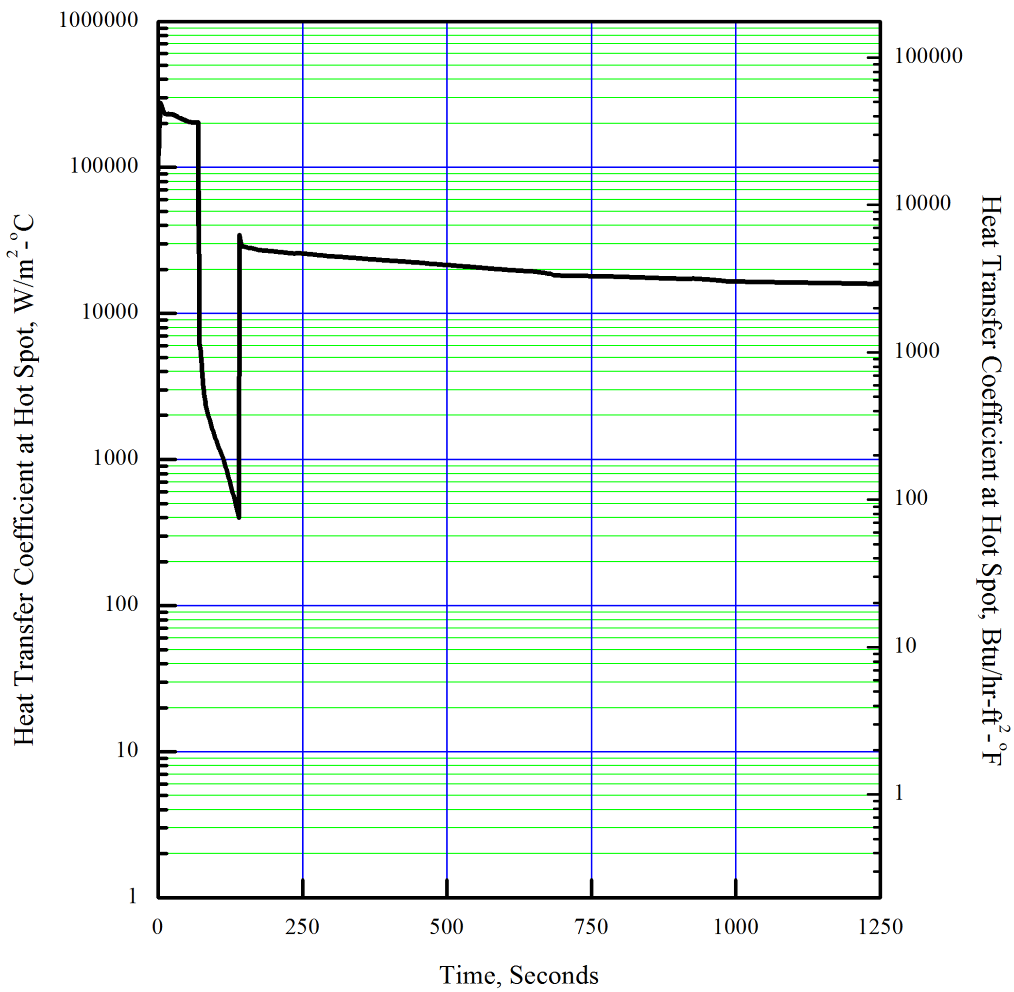

Figure 4.2-38 126.7cm² (0.1364ft²) Break in Pump discharge Leg: Heat Transfer Coefficient at Hot Spot

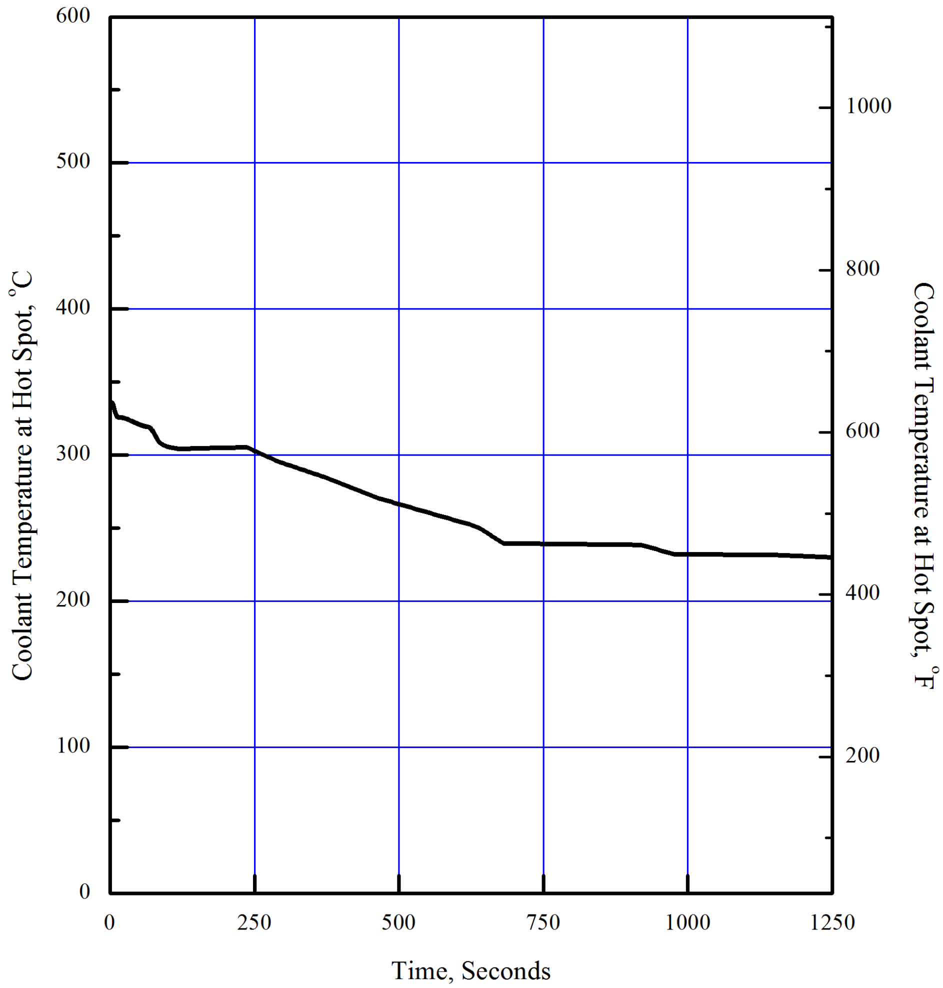

Figure 4.2-39 126.7cm^2 (0.1364ft^2) Break in Pump discharge Leg: Coolant Temperature at Hot Spot

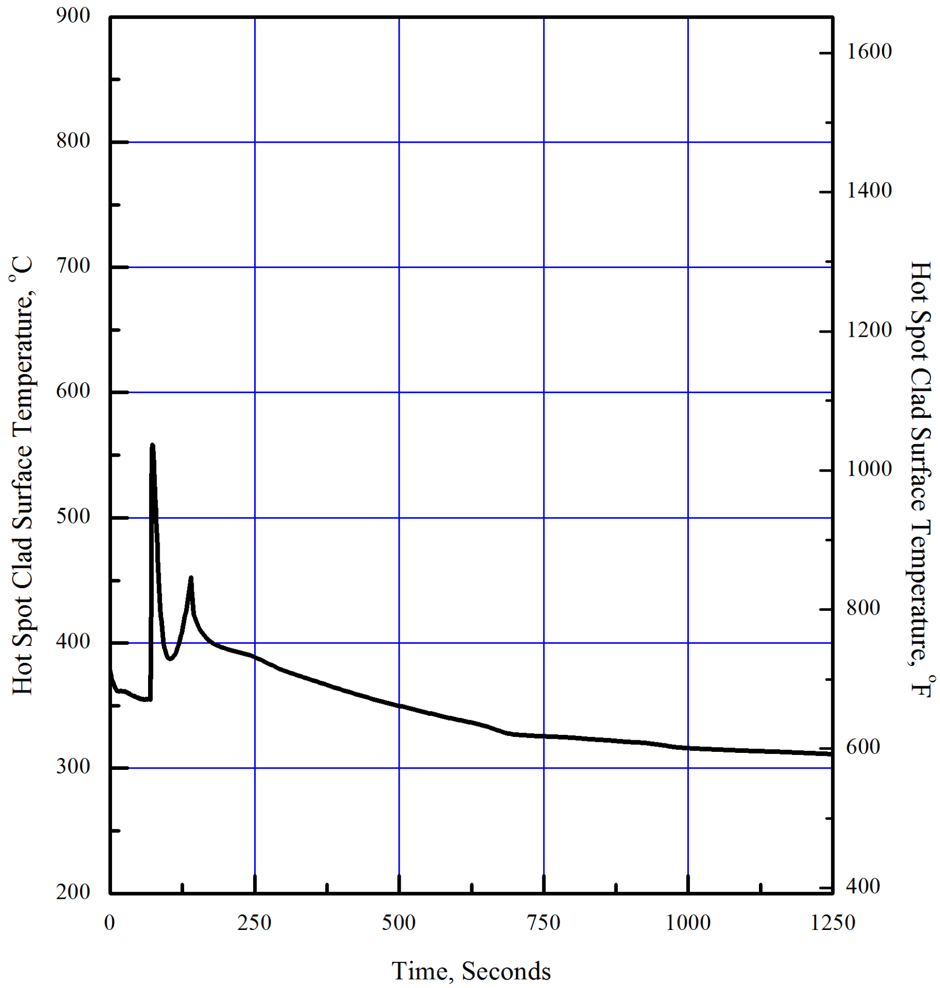

Figure 4.2-40 126.7cm^2 (0.1364ft^2) Break in Pump discharge Leg: Hot Spot Clad Surface Temperature

TS

Figure 4.2-41 153.3cm^2 (0.1650ft^2) Break in Pump discharge Leg: Nomalized Total Core Power

TS

Figure 4.2-42 153.3cm^2 (0.1650ft^2) Break in Pump discharge Leg: Inner Vessel Pressure

TS

Figure 4.2-43 153.3cm^2 (0.1650ft^2) Break in Pump discharge Leg: Break Flow Rate

TS

Figure 4.2-44 153.3cm^2 (0.1650ft^2) Break in Pump discharge Leg: Inner Vessel Inlet Flow

TS

Figure 4.2-45 153.3cm^2 (0.1650ft^2) Break in Pump discharge Leg: Two-Phase Mixture Level

TS

Figure 4.2-46 153.3cm^2 (0.1650ft^2) Break in Pump discharge Leg: Heat Transfer Coefficient at Hot Spot

TS

Figure 4.2-47 153.3cm^2 (0.1650ft^2) Break in Pump discharge Leg: Coolant Temperature at Hot Spot

TS

Figure 4.2-48 153.3cm^2 (0.1650ft^2) Break in Pump discharge Leg: Hot Spot Clad Surface Temperature

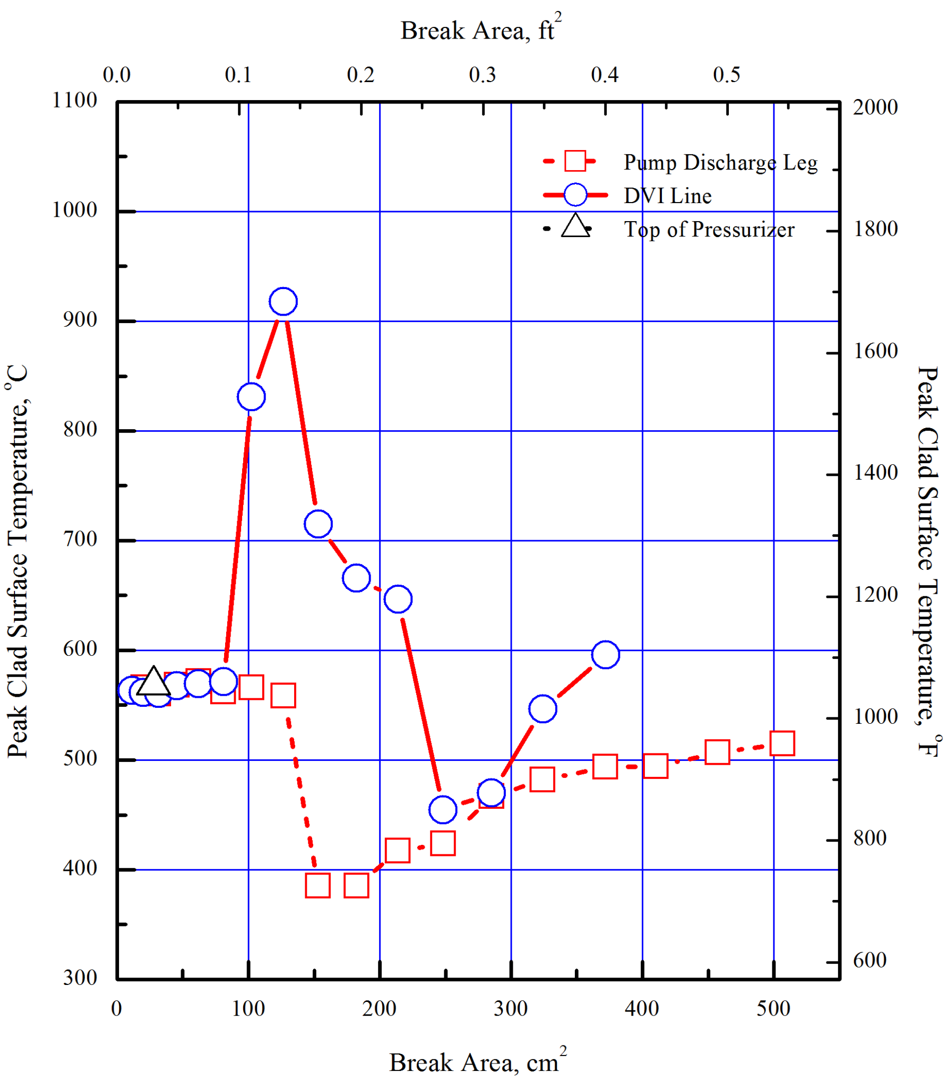

Figure 4.2-49 Peak Cladding Temperature vs. Break Area

4.3 Sensitivity Studies

During the course of development of the analysis methods, the influences of many parameters and models on the analysis results were evaluated. These evaluations were made in order to choose parameter values and reference models that could be used in the analysis methods. In general, the choices were made such that the analysis results would be more adverse than would be realistically expected.

The following sensitivity studies were performed.

- CEFLASH-4AS Phase Separation Model Sensitivity
- CEFLASH-4AS Heterogeneous Path Model Sensitivity
- CEFLASH-4AS Hot Leg Model Sensitivity
- CEFLASH-4AS Pump Degradation Sensitivity
- CEFLASH-4AS Wall Heat Transfer Sensitivity
- CEFLASH-4AS ECC Injection Location Sensitivity
- CEFLASH-4AS Time Step Sensitivity
- STRIKIN-II Coolant Enthalpy Sensitivity
- STRIKIN-II Coolant Flow Rate Sensitivity
- PARCH Critical Heat Flux Sensitivity
- PARCH Conduction Model Sensitivity
- PARCH Time Step Sensitivity
- Break Area Sensitivity

Detailed information about these sensitivity studies is provided in Reference 1.

5 REQUIREMENTS FOR SBLOCA

5.1 Acceptance Criteria for SBLOCA

10CFR50.46 (Reference 8) provides the acceptance criteria for the ECCS for light water-cooled reactors.

The results of the analyses demonstrate that the APR1400 SIS design meets the acceptance criteria of Reference 8. Conformances are as follows:

- Criterion (1) Peak Cladding Temperature.
  - "The calculated maximum fuel element cladding temperature shall not exceed 1,204℃ (2,200℉)."
  - The ECCS performance analysis yielded a peak cladding temperature of 918 ℃ (1,684 ℉) for the 126.7 cm^2 (0.1364 ft^2) DVI line break.
- Criterion (2) Maximum Cladding Oxidation.
  - "The calculated total oxidation of the cladding shall nowhere exceed 0.17 times [17%] of the total cladding thickness before oxidation."
  - The ECCS performance analysis yielded a maximum cladding oxidation percentage of less than 0.55474% for the 126.7 cm^2 (0.1364 ft^2) DVI line break.
- Criterion (3) Maximum Hydrogen Generation.
  - "The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 [1%] times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react."
  - The ECCS performance analysis yielded a maximum core-wide oxidation of less than 1% for the 126.7 cm^2 (0.1364 ft^2) DVI line break.
- Criterion (4) Coolable Geometry.
  - "Calculated changes in core geometry shall be such that the core remains amenable to cooling."
  - The cladding swelling and rupture model, which is part of the ECCS performance evaluation model, accounts for the effects of changes in core geometry if such changes are predicted to occur. With these calculated changes in core geometry, core cooling was enough to lower temperatures. No further cladding swelling and rupture can occur since the calculations were carried to the point at which the cladding temperatures were decreasing and the RCS was depressurized. Thus, a coolable geometry has been demonstrated.
- Criterion (5) Long Term Cooling
  - "After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in

the core."

- The SBLOCA analyses show that the rapid injection of borated water from the SIS limits the PCT and cools the core within a short period of time. The post LOCA long term cooling analysis shows that the continued injection of borated water by the SIPs from the in-containment refueling water storage tank removes the decay heat resulting from the long-lived radioactivity remaining in the core.

5.2 Conformance with SRP

The Standard Review Plan (SRP) is intended to provide guidance to the regulatory body staff for evaluating the acceptability of a design.

SRP 15.6.5 (Rev.3) addresses the idea that, in the analysis of small breaks, evaluating integer diameter break size (i.e., 1,2,3,4-inch, etc.) is considered insufficient to determine the worst break because the break areas associated with these integer diameters are too coarse to adequate identify the highest PCT.

According to the SRP 15.6.5 (Rev.3), the dense break spectrum analyses were performed with less than integer diameter break size. The results were shown in the Section 4.2 in this report.

Also, SRP 15.6.5 (Rev.3) address the idea that the analysis of small break location should include the side and top of the discharge leg to ensure that the suction leg piping that fails to clear off liquid does not result in depression of the two-phase mixture level into the core and result in the worst case PCT.

The PCT of an SBLOCA depends on the break location. Since the break size is small in SBLOCAs, it is possible that a break will be located on the bottom, side or top of the pipe. Generally, the limiting location of the break is confirmed in the topical report (Reference 1). There is, however, no explicit description of the break location in the topical report. This is because the CEFLASH-4AS code used for SBLOCA analysis does not have a model to treat the phase separation in a horizontal pipe.

6 CONCLUSIONS

From the perspective of SBLOCA accident analysis, APR1400 is an advanced PWR design that is functionally similar to existing plants and fuel designs. The codes and methodologies that were used for APR1400 SBLOCA analyses are similar to NRC-approved codes and methodologies used to evaluate existing plants and fuel. The codes previously approved by the NRC have been described, justified, and validated by this report again.

The codes and methodologies examined were:

- CEFLASH-4AS Calculates thermal hydraulic behavior during the blowdown phase
- COMPERC-II Calculates thermal hydraulic behavior during the reflood phase
- STRIKIN-II Calculates the cladding and fuel temperature before the flow reversal time
- PARCH Calculates the cladding and fuel temperature after the flow reversal time

It was concluded that the existing codes and methodologies are appropriate for APR1400 analyses. Also, it is concluded that the information provided in this technical report supports the report's purpose to provide the followings; key technical information related to computer codes; key methods, models, and their applicability; and event-specific acceptance criteria for the NRC to facilitate an efficient and timely review of Design Certification Applications.

7 REFERENCES

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