Small Break LOCA Evaluation Model

Technical Report

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Abstract

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

The purpose of this technical report is to provide information to the NRC, Nuclear Regulatory Commission, to facilitate the efficient and timely review of the accident analysis to be provided in the design certification document (DCD) as part of the Design Certification License Application.

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 APR1400. Validation of these codes by comparison with computer codes that have been approved by the NRC is presented.

- 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 applied codes and methodologies are appropriate for APR1400 safety analysis. Also, it was concluded that the information provided in this technical report supports its purpose to provide key technical information related to the computer codes and methodology to facilitate an efficient and timely review of the Design Certification License Application.

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List of Acronyms

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

1.0 Introduction

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 APR1400 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)
- Elimination of low-pressure SIP (LPSIP)

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); its 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, only two SIPs are assumed to be operable in the present SBLOCA analysis for additional conservatism. The configuration is shown in Figure 1-2. The SIT-FD is equipped with a fluidic device, as shown in Figure 1-3, which controls the injection flow as a function of the water level. The fluidic device is a combination of a stand pipe and a vortex chamber. The SIT-FD provides high injection flow, equivalent to that of a traditional safety injection tank (SIT) as far as the water level is higher than the top of the stand pipe; afterwards, the SIT-FD provides low injection flow equivalent to that of the LPSIP.

The purpose of this technical report is to present the SBLOCA computer codes and methodologies for the SBLOCA analysis in the Standard Review Plan (SRP) Chapter 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	
Average linear heat generation rate, kW/m (kW/ft)	
Peak linear heat generation rate (PLHGR), kW/m (kW/ft)	
Gap conductance at PLHGR, kcal/hr-m^2-℃ (Btu/hr-ft^2-°F)	
Fuel centerline temperature at PLHGR, ℃ (°F)	
Fuel average temperature at PLHGR, ℃ (°F)	
Hot rod gas pressure, kg/cm^{2}A (psia)	
Initial reactor vessel inlet temperature, ℃ (°F)	
Initial reactor vessel outlet temperature, ℃ (°F)	
Moderator temperature coefficient, Δρ/℃ (Δρ/°F)	
Initial RCS flow rate, kg/s (10^6 lbm/hr)	
Initial core flow rate, kg/s (10^6 lbm/hr)	
Initial RCS pressure, kg/cm^{2}A (psia)	
Low pressurizer pressure reactor trip setpoint, kg/cm^{2}A (psia)	
SIAS setpoint on low pressurizer pressure, kg/cm^{2}A (psia)	
SIT gas pressure, kg/cm^{2}A (psia)	

Table 1-2 Specifications of PLUS7 Fuel

Parameters	Value
Fuel assembly length, m (in)	
Active core length, m (in)	
Fuel assembly pitch, m (in)	
Number of guide tubes per fuel assembly (outer + center)	
Number of protective grids	
Number of top Inconel grids	
Number of bottom Inconel grids	
Number of middle grids	
Length of middle grid span, m (in)	
Number of fuel rods per fuel assembly	
Fuel rod pitch, m (in)	
Clad material	
Fuel rod length, m (in)	
Fuel rod outer diameter, m (in)	
Clad thickness, m (in)	
Pellet diameter, m (in)	

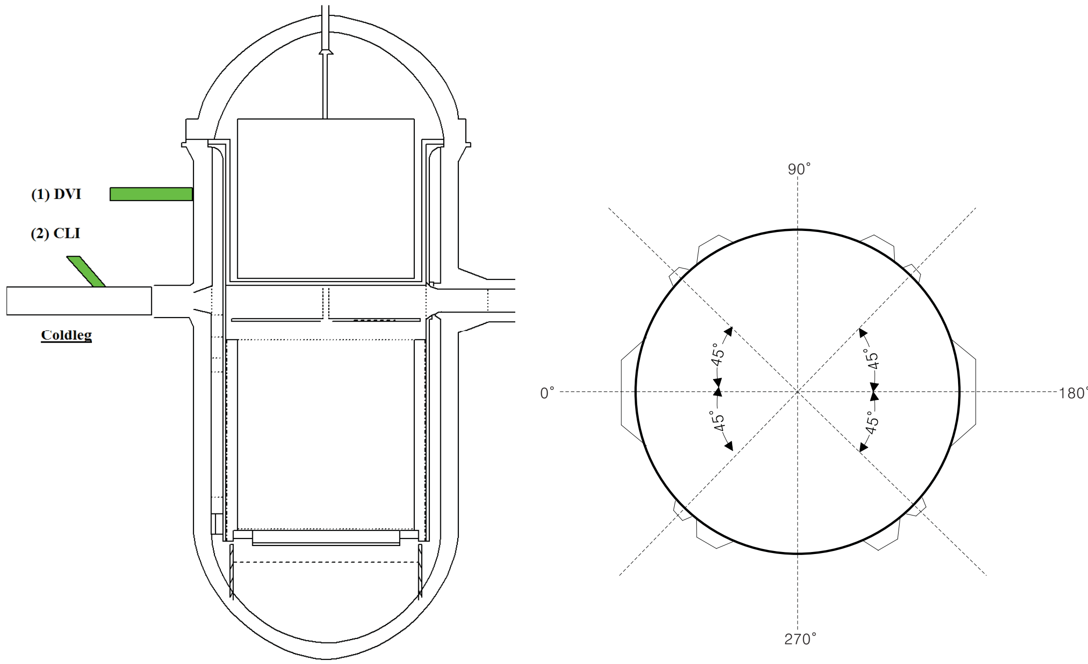

Figure 1-1 DVI Nozzle Location

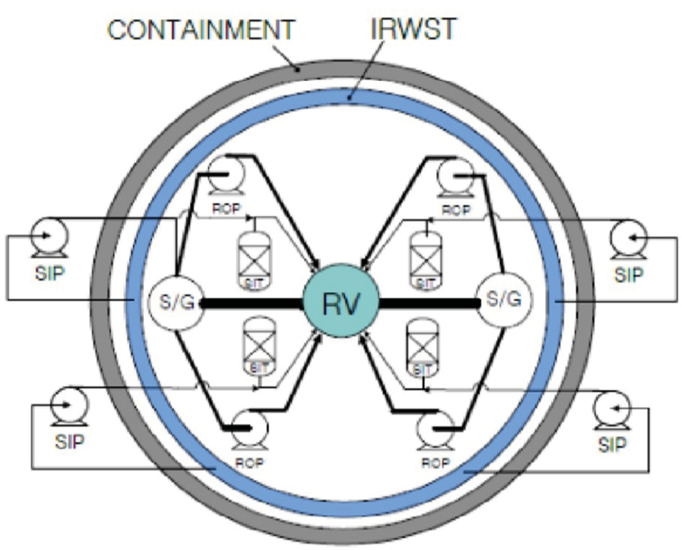

Figure 1-2 Configuration of the APR1400 Safety Injection System

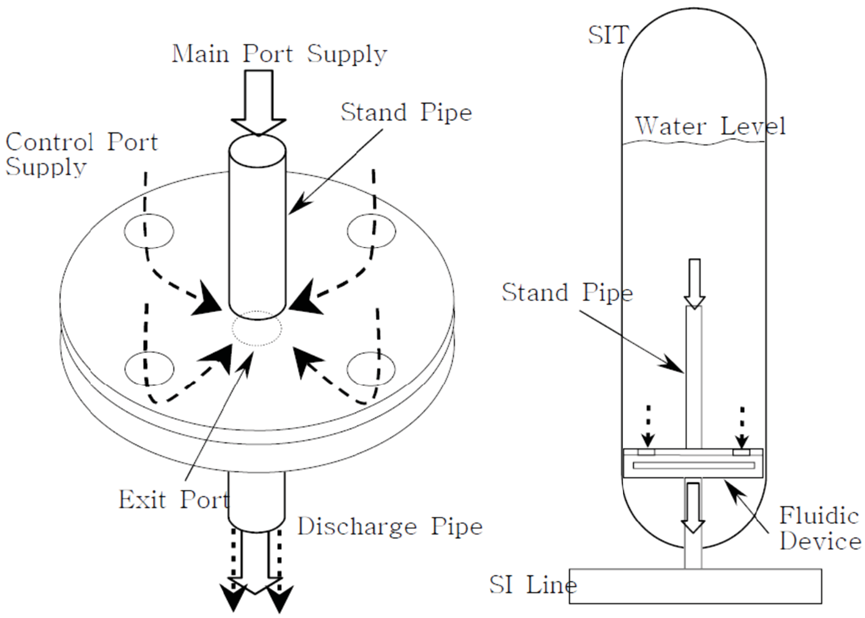

Figure 1-3 Schematic Diagram of Fluidic Device

2.0 General Description and Characteristics of SBLOCA

2.1 Definition and Characteristics of SBLOCA

LOCAs are hypothetical accidents that would 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 the break that cannot be made up by charging pump. The largest SBLOCA is generally 1.0 ft^2. CENP SBLOCA methodology, however, decided that the largest SBLOCA is 0.5 ft^2 through a sensitivity study for break size.

SBLOCA shows fairly different behavior according to break size. When the break size is small, coolant inventory can be kept by the SIP. In this case, cooling and depressurization using the steam generator are performed by the operator. If the break size is increased, release flow cannot be sufficiently compensated by the SIP. Furthermore, SBLOCA behaves differently depending on break location. The cold-leg break SBLOCA known as most limited one can be divided into five phases. The duration of each phase and occurrence varies depending on break size and 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 was 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, and 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 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 a 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 small break LOCA. 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 STRIKIN-II 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 a SBLOCA. It describes the removal of heat from a fuel rod, which is surrounded by a quasi-static fluid partially or totally covering the length of the fuel rod. 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 between Codes

The general method of data transfer between 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 is passed to the temperature calculations 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 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 temperatures.

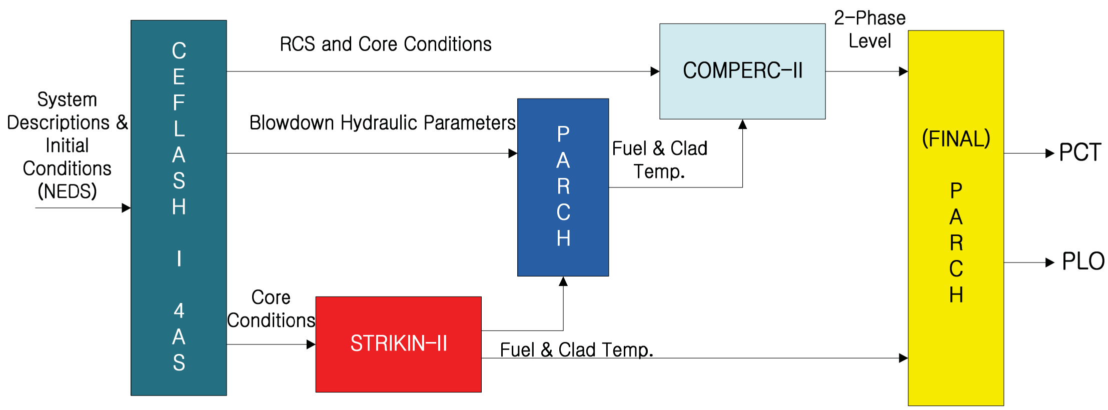

Figure 2.3-1 SBLOCA Analysis Code Flow Chart

3.0 Description of Model Components

The significant features of the SBLOCA analysis methods are described in this section. The use of the 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 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 means of important physical phenomena which are unique to SBLOCA is as follows.

a. 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.

b. 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.

c. 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.

d. 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.

e. 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.

f. 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.

g. 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

The geometric description of APR1400's NSSS 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 a single cold leg. The leak path is located in the discharge leg, DVI line, or pilot operated safety relief valve (POSRV) line. The break in DVI line produces the most severe results from SBLOCA.

Due to the importance of hydrostatic effects on 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.

Figure 3.1-1 CEFLASH-4AS Node Diagram

3.2 Reflood Hydraulics

The reflood period during SBLOCA is defined as that 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 which 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 reflood-generated steam, the average reflood rates, heat transfer coefficients based on FLECHT and fluid properties.

a. 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.

b. 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.

c. FLECHT heat transfer correlations

Applying FLECHT data to SBLOCA analysis is somewhat unique compared to use for a LBLOCA because of the higher pressure which occurs during reflood following an SBLOCA. The FLECHT correlations are formulated for a pressure range from 15 to 90 psia. SBLOCA cases in which FLECHT would be applied have pressures during reflood from about 40 psia to about 230 psia. The FLECHT correlations are applied using a reference pressure at the time 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 the result of these markedly different blowdown regimes two fuel rod temperature codes are needed in the SBLOCA analysis. STRIKIN-II is used during the initial high-flow period immediately following the rupture, since it is a force convection heat transfer code.

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

a. 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.

b. 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

c. Heat transfer regime selection logic

The heat transfer logic is the same as employed for large break analysis.

d. 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.

e. 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

a. 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.

b. 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

c. 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.

d. 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.

4.0 SBLOCA Analysis

4.1 Input Parameters

Numerous parameters, which are input variables for the various computer codes have been given prescribed values as part of the SBLOCA analysis methodology. Several of these parameters are options which 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 SI pumps must await diesel startup and load sequencing before they can start. The total time delay assumed is 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 emergency diesels to start. This failure causes a loss of two of the four SI pumps, 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:

a. For a break in the pump discharge leg, the SI flow credited is full flow from two SI pumps and four SITs

b. For a break in a DVI line, the SI flow credited is full flow from one SI pump and three SITs. The flow from the remaining active SI pump and from one SIT is assumed to spill out of the break

The SI pump flow rates assumed at each of the four injection points as a function of RCS pressure is described in DCD Tier2 15.6.5.

The significant core and system parameters used in the small break calculations are presented in DCD Tier2 15.6.5. The peak linear heat generation rate (PLHGR) of 49.2kW/m is assumed to occur 15 percent from the top of the active core. A conservative beginning-of-life moderator temperature coefficient of $0.0\times10^{-4}\ \Delta\rho/℃$ 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 which 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 nine breaks analyzed at 4,062.66 MWt (102 % of nominal) include reactor coolant pump discharge leg breaks ranging in size from 465 cm^2 to 46.5 cm^2 and DVI line breaks from 372 cm^2 to 18.6 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 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 4.2-1. The 372 cm^2 DVI break results in the highest cladding temperature (624℃) of the small breaks analyzed. Of the two break locations (pump discharge leg and a 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:

a. The time of initial core uncovery is delayed.
b. The depth of core uncovery is lower.
c. The rate of level decrease and increase becomes slower.

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

As the break size decreases, both the later time of initial core uncovery and the shallower depth of uncovery tend to mitigate the temperature transient. This trend continues until the core does not uncover as typified by the 18.6 cm^2 break. Thus, by analyzing several break sizes over this range, the behavior of PCT versus break size is adequately determined.

The above behavior of core uncovery with break size results from the design characteristics of the SIS. For DVI break sizes below 93 cm^2, the RCS pressure remains above SIT pressure and coolant flow injection to the reactor vessel is accomplished entirely by one SI pump. For break sizes greater than 93 cm^2, the transient is terminated by the SITs and SI pump.

For the cold leg breaks, the additional SIS flow resulting from being able to credit two SI pumps precludes core uncovery to break sizes up to 93 cm^2. In addition, the core uncovery for break sizes greater than 93 cm^2 is delayed and the depth and duration of uncovery decreased relative to DVI breaks that credit only one SI pump. This more favorable behavior results in lower cladding temperatures relative to breaks in a DVI line.

In addition to the break locations discussed above, the rupture of an in-core instrument tube is considered. A break, equal in size to a completely severed instrument tube (2.8 cm^2), is postulated to occur in the reactor vessel bottom head.

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 emergency diesel and the SI pumps following SIAS, safety injection flow is initiated. Due to the assumed failure of one diesel, only two SI pumps 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 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 SI pumps (62 kg/sec) exceeds the leak flow (35 kg/sec). 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

Break	Peak Cladding Temperature (℃)	Maximum Cladding Oxidation (%)	Maximum Core-Wide Oxidation (%)
465 cm^2/PD[1)]	498	0.0017	< 0.0003
325 cm^2/PD	492	0.0015	< 0.0002
93 cm^2/PD	565	0.0010	< 0.0001
46.5 cm^2/PD	568	0.0008	< 0.0002
372 cm^2/DVI[2)]	624	0.0195	< 0.0029
93 cm^2/DVI	569	0.0069	< 0.0009
46.5 cm^2/DVI	571	0.0018	< 0.0003
18.6 cm^2/DVI	616	0.0029	< 0.0006
27.9 cm^2/HL[3)]	568	0.0006	< 0.0002

1) PD : Pump Discharge Leg Break,
2) DVI : DVI Line Break,
3) HL : Hot Leg Break

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

Break	SI Pump Flow Delivered to RCS	SI Tank Flow Delivered to RCS	Hot Spot Peak Clad Temperature Occurs
465 cm^2/PD	57	152	167
325 cm^2/PD	62	218	105
93 cm^2/PD	138	1,128	100
46.5 cm^2/PD	248	2,984	208
372 cm^2/DVI	60	192	239
93 cm^2/DVI	138	1,096	100
46.5 cm^2/DVI	250	N/A[1)]	210
18.6 cm^2/DVI	624	N/A	1,184
27.9 cm^2/HL	795	N/A	750

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

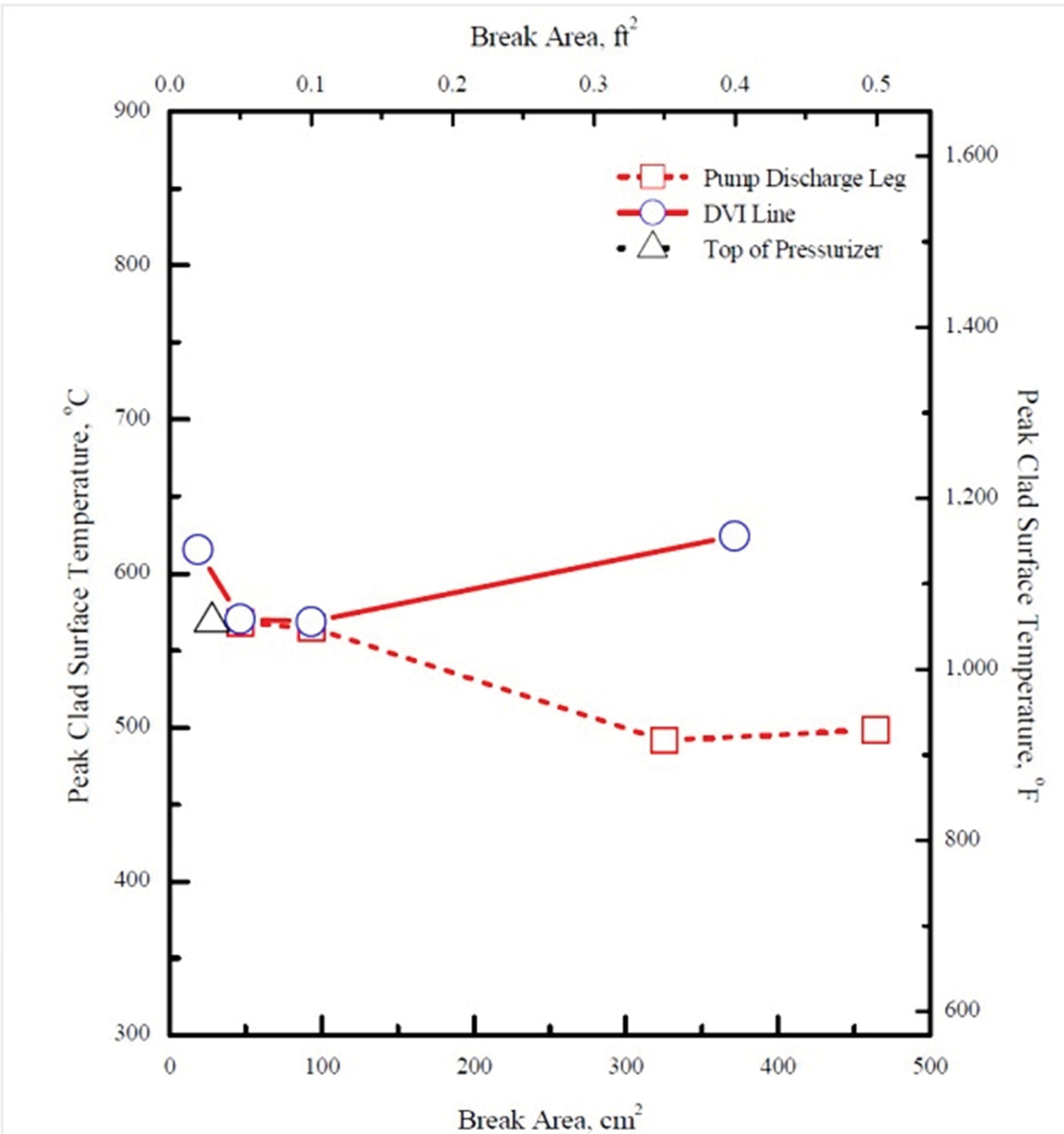

Figure 4.2-1 Peak Cladding Temperature vs. Break Area

4.3 Sensitivity Studies

During the course of development of the analysis methods, the influence of many parameters and models on the analysis results was evaluated. These evaluations were made in order to choose parameter values and reference models which would 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 described in Reference 1.

5.0 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 References 8. Conformances are as follows:

Criterion (1) Peak Cladding Temperature.
"The calculated maximum fuel element cladding temperature shall not exceed 1,204℃ (2,200°F)."

The ECCS performance analysis yielded a peak cladding temperature of 624℃ (1,156°F) for the 372 cm^2(0.4 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 0.0029% for the 372 cm^2(0.4 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 372 cm^2(0.4 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 small break 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 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 area associated with these integer diameters are too coarse to adequate identify the highest PCT.

Proposed additional break sizes analyzed on SBLOCA DVI break for NRC DC are 0.3 ft^2 and 0.2 ft^2. It is considered that these break sizes can be used to determine the worst break size sufficiently.

Also, SRP 15.6.5 (Rev.3) address 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 results in depression of the two-phase mixture level into the core and result in the worst case PCT.

The PCT of SBLOCA depends on the break location. Since the break size is small in SBLOCAs, it is possible that the break is located on the bottom, side or top of the pipe. Generally, the limiting location of the break is confirmed in the topical report.

But in CENPD-137P, there is no explicit description on the break location. It is because the CEFLASH-4AS code used for SBLOCA analysis does not have the model to treat the phase separation in the horizontal pipe.

6.0 Conclusions

APR1400 is an advanced PWR design that is functionally similar to existing plants and fuel designs from the perspective of SBLOCA accident analysis. 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 its purpose to provide key technical information related to computer codes; key methods, models, and their applicability; and event-specific acceptance criteria to the NRC to facilitate an efficient and timely review of the Design Certification Application.

7.0 References

1. "Calculative Methods for the C-E Small Break LOCA Evaluation Model," CENPD-137P, August 1974 (Proprietary).

   "Calculative Methods for the C-E Small Break LOCA Evaluation Model," CENPD-137, Supplement 1-P, January 1977 (Proprietary).

2. Letter, O. D. Parr (NRC) to F. M. Stern (C-E), June 13, 1975.

   Letter, O. D. Parr (NRC) to A. E. Scherer (C-E), December 9, 1975.

   Letter, Karl Kniel (NRC) to A. E. Scherer (C-E), September 27, 1977.

   Letter, D. M. Crutchfield (NRC) to A. E. Scherer (C-E), July 31, 1986.

3. "CEFLASH-4AS, A Computer Program for Reactor Blowdown Analysis of the Small Break Loss-of-Coolant Accident," CENPD-133P, Supplement 1, August 1974 (Proprietary).

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4. "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," CENPD-134P, August 1974(Proprietary).

   "COMPERC-II, A Program for Emergency Refill-Reflood of the Core (Modifications)," CENPD-134P, Supplement 1, February 1975 (Proprietary).

   "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," CENPD-134, Supplement 2, June 1985.

5. "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," CENPD-135P, August 1974 (Proprietary).

"STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program (Modifications)," CENPD-135P, Supplement 2, February 1975 (Proprietary).

"STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," CENPD-135, Supplement 4-P, August 1976 (Proprietary).

"STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," CENPD-135, Supplement 5-P, April 1977 (Proprietary).

6. "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," CENPD-138P, August 1974 (Proprietary).

   "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup (Modifications)," CENPD-138P, Supplement 1, February 1975 (Proprietary).

7. "C-E Fuel Evaluation Model," CENPD-139-P-A, July 1974 (Proprietary).

   "Improvements to Fuel Evaluation Model," CEN-161 (B)-P-A, August 1989 (Proprietary).

8. 10CFR50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water-Cooled Nuclear Power Reactors," October 1988.

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