Non-LOCA Safety Analysis Methodology

for the APR1400

Revision 0

Non-Proprietary

September 2013

Revision History

Revision	Date	Page	Description
0	September 2013	All	Original Issue

ABSTRACT

This report presents the non-LOCA methodology that is used in the design certification document (DCD), Tier 2 Chapter 15 safety analysis for the APR1400. The contents of this document include description of the computer code, code validation, acceptance criteria and event specific methodology. The methodology for the analysis of radiological consequence is not described in this report.

The purpose of this report is to provide information to the NRC to facilitate efficient and timely review of the accident analysis to be provided in the DCD as part of the design certification license application.

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

- CESEC-III Simulate the NSSS
- TORC/CETOP Determine core thermal margin and minimum DNBR
- COAST Simulate RCP coastdown
- HRISE Calculate Minimum DNBR in case of return-to-power (RTP)
- STRIKIN-II Calculate the cladding and fuel temperature for an average or hot fuel rod
- HERMITE Determine short term dynamic response of reactor core

This report also provides a history of methodology changes for non-LOCA analysis codes. For CESEC-III, new models for POSRV and centrifugal charging pump are incorporated to implement the APR1400 advanced design features. For COAST, the initialization method used in COAST is changed from head-flowrate iteration with constant system friction to system friction iteration with constant flowrate to prevent diverging during the initialization step. For CETOP and STRIKIN-II, the critical heat flux (CHF) correlation is changed from CE-1 for C-E standard fuel and Guardian™ fuel to KCE-1 for PLUS7™ fuel with mixing vane grids. The correlation form is not changed but the coefficients are revised to predict the thermal characteristics of the improved fuel assembly. For STRIKIN-II, two types of cladding material are added. The properties of ZIRLO™ and M5™ are added to the properties of OPTIN™ which are already included in STRIKIN-II.

The event classification and associated acceptance criteria that will be used by KHNP for each non-LOCA event included in the DCD are presented, ordered by the broader event categories defined by the SRP Chapter 15 and Regulatory Guide 1.206.

On the basis of the information in this report, it is concluded that the applied codes and methodologies are appropriate for the APR1400 safety analysis. Also, it is concluded that the information provided in this report supports its purpose to provide key technical information related to the computer codes and methodology as well as the acceptance criteria of the APR1400 related with the representing non-LOCA safety analysis to the NRC to facilitate an efficient and timely review of the design certification license application.

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List of Acronyms

1-D	One-dimensional
3-D	Three-dimensional
AC	Alternate Current
ADV	Atmospheric Dump Valve
AOO	Anticipated Operational Occurrence
AFWP	Auxiliary Feed Water Pump
APR1400	Advanced Power Reactor 1400
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CCP	Centrifugal Charging Pump
CE	Combustion Engineering
CEA	Control Element Assembly
CHF	Critical Heat Flux (synonym of DNB)
CFR	Code of Federal Regulations
COLSS	Core Operating Limit Supervisory System
CPC	Core Protection Calculator
CVCS	Chemical and Volume Control System
DCD	Design Control Document
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
EOC	End of Cycle
GDC	General Design Criterion
gpm	Gallons per Minute
FLB	Feedwater Line Break
HEM	Homogeneous Equilibrium Model
HFP	Hot Full Power
HSGL	High Steam Generator Level
HZP	Hot Zero Power
IRWST	In-containment Refueling Water Storage Tank
KHNP	Korea Hydro and Nuclear Power Corporation
KEPCO E&C	Korea Electric Power Corporation Engineering & Construction Company
LHGR	Linear Heat Generation Rate
LOCV	Loss of Condenser Vacuum
LOOP	Loss of Offsite Power
MDNBR	Minimum DNBR
MFCS	Main Feedwater Control System
MFIV	Main Feedwater Isolation Valve
MSIV	Main Steam Isolation Valve
MSIS	Main Steam Isolation Signal
MSSV	Main Steam Safety Valve
MTC	Moderator Temperature Coefficient
lpm	Litters per Minute

LSGL	Low Steam Generator Level
Non-LOCA	Non-Loss of Coolant Accident
NRC	US Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PA	Postulated Accident
Pe	Pecklet Number
PLCS	Pressurizer Level Control System
POSRV	Pilot Operated Safety Relief Valve
PPS	Plant Protection System
PWR	Pressurized Water Reactor
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RIA	Reactivity Initiated Accident
RG	Regulatory Guide
RPS	Reactor Protection System
RTP	Return-to-Power
RV	Reactor Vessel
SAFDL	Specified Acceptable Fuel Design Limit
SBCS	Steam Bypass Control System
SER	Safety Evaluation Report
SIAS	Safety Injection Actuation Signal
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SIS	Safety Injection System
SLB	Steam Line Break
SRP	Standard Review Plan
TBV	Turbine Bypass Valve
TDC	Thermal Diffusion Coefficient
TEDE	Total Effective Dose Equivalent

1. INTRODUCTION

The purpose of this report is to present the non-LOCA computer codes and methodologies that adopted by KHNP for the analysis of all non-LOCA events in the standard review plan (SRP) Chapter 15 [1, 2], except LOCA and dose evaluation, for Advanced Pressurized Water Reactors such as the APR1400. The KHNP non-LOCA methodology using the following codes is very similar to the conventional non-LOCA methodology used for currently operating US CE- fleet PWRs;

- CESEC-III — Simulate the NSSS
- TORC/CETOP — Determine core thermal margin and minimum DNBR
- COAST — Simulate RCP coastdown
- HRISE — Calculate Minimum DNBR in case of return-to-power (RTP)
- STRIKIN-II — Calculate the cladding and fuel temperature for an average or hot fuel rod
- HERMITE — Determine short term dynamic response of reactor core

KHNP performs the non-LOCA safety analysis for the DCD, Tier 2 Chapter 15 events using these computer codes and methodologies. This report describes:

- Chapter 2- Transients and Accidents
- Chapter 3- Analysis Methodology
- Chapter 4- Requirements for DCD, Tier 2 Chapter 15 non-LOCA events
- Chapter 5- Results
- Chapter 6- Conclusions

2. TRANSIENTS AND ACCIDENTS

This chapter presents analytical evaluations of the APR1400 nuclear steam supply system (NSSS) response to postulated disturbances in process variables and to postulated malfunctions or failures of equipment. Such incidents (or events) are postulated and their consequences analyzed despite the many precautions which are taken in the design, construction, quality assurance, and plant operation to prevent their occurrence. The effects of these incidents are examined to determine their consequences and to evaluate the capability built into the plant to control or accommodate such failures and situations.

2.1 Event Frequency

Each postulated initiating event has been assigned to one of the following categories:

- Increase in Heat Removal by the Secondary System
- Decrease in Heat Removal by the Secondary System
- Decrease in Reactor Coolant Flow Rate
- Reactivity and Power Distribution Anomalies
- Increase in RCS Inventory
- Decrease in RCS Inventory
- Radioactive Material Release from a Subsystem or Component

The assignment of an initiating event to one of these seven categories is made according to Reference 2.

2.1.1 Event Frequencies

Reference 2 subjectively classifies initiating events in the following qualitative frequency groups in Table 2.1-1:

- Anticipated Operational Occurrences (AOOs): "... those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power" (10CFR50, Appendix A).
- Postulated Accidents (PAs): Those events which are not expected to occur during the life of the nuclear power unit, but are postulated because they pose the potential for the release of a significant amount of radioactivity

2.1.2 Events and Event Combinations

The events and event combinations in this chapter are those identified by References 1 and 2, and are presented with respect to the event specific acceptance criteria specified therein. For each applicable acceptance criterion in an event category/frequency group, only the limiting event or event combination is presented in analytical detail. Qualitative discussions are provided for all other events or event combinations explaining why they are not limiting.

All event analyses resulting in a turbine generator trip considered the loss of offsite power (LOOP) while applying the same acceptance criteria for the event with and without LOOP, i.e. without re-classifying the event. In the analyses for which the LOOP is assumed to result from a turbine trip, the time delay between the turbine trip and LOOP is assumed to be zero. However, a 3-second time delay can be assumed between reactor trip breakers opening and the turbine trip because of the turbine trip delay circuits. This time delay is assumed in the CEA misoperation and CEA ejection events while the other events do not use the time delay conservatively.

For event combinations which require consideration of a single failure, the limiting failure is selected from those listed in Table 2.4-1. Pre-existing failures are equipment failures existing prior to the event initiation which are not revealed until called upon during the event (e.g., a failure of an auxiliary feedwater pump). High probability dependent occurrences are always included in the event analysis, if they have an adverse impact. Interactive control system failures are not more limiting than the active failures listed.

According to the 10 CFR 50 Appendix A, a single failure means an occurrence which results in the loss of capacity of a component to perform its intended safety functions. However, failures are considered of not only the safety-related systems whose operation may be required, but also non-safety related systems whose failure could produce results more severe than failure of a safety system. If a non-safety related system would help to mitigate a transient, then the analysis of that transient assumes the system is in the manual mode of operation. Non-safety related systems are assumed to be in the automatic mode of operation if the system makes the consequences of a transient more adverse.

The purpose of the single failure list is to identify the single failures which could create the most adverse conditions during a given transient, regardless of the safety related status of that component or system. In most cases, the automatic action of safety-related systems overrides the operation of non-safety systems. The justification for choosing the most limiting single failure is explained in the individual analyses of this chapter.

Table 2.1-1 (sh. 1 of 2)

Initial Events and Category of Events

Chapter Number	Events	Category of Events
15.1	Increase in Heat Removal by the Secondary System	
15.1.1	Decrease in Feedwater Temperature	AOO
15.1.2	Increase in Feedwater Flow	AOO
15.1.3	Increased Main Steam Flow	AOO
15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve	AOO
15.1.5	Steam System Piping Failures Inside and Outside Containment	PA
15.2	Decrease in Heat Removal by the Secondary System	
15.2.1	Loss of External Load	AOO
15.2.2	Turbine Trip	AOO
15.2.3	Loss of Condenser Vacuum	AOO
15.2.4	Main Steam Isolation Valve Closure	AOO
15.2.5	Steam Pressure Regulator Failure	N/A
15.2.6	Loss of Non-Emergency AC Power to the Station Auxiliaries	AOO
15.2.7	Loss of Normal Feedwater Flow	AOO
15.2.8	Feedwater System Pipe Breaks	PA
15.3	Decrease in Reactor Coolant Flow Rate	
15.3.1	Total Loss of Reactor Coolant Flow	AOO
15.3.2	Flow Controller Malfunctions	N/A
15.3.3	Single Reactor Coolant Pump Rotor Seizure with Loss of Offsite Power	PA
15.3.4	Reactor Coolant Pump Shaft Break with Loss of Offsite Power	PA
15.4	Reactivity and Power Distribution Anomalies	
15.4.1	Uncontrolled Control Element Assembly Withdrawal from Subcritical or Low Power Conditions	AOO
15.4.2	Uncontrolled Control Element Assembly Withdrawal at Power	AOO

Table 2.1-1 (sh. 2 of 2)

15.4.3	CEA Misoperation	AOO
15.4.4	Startup of an Inactive Reactor Coolant Pump	AOO
15.4.5	Flow controller malfunction causing an increase in BWR core flow rate	N/A
15.4.6	Inadvertent Deboration	AOO
15.4.7	Inadvertent Loading of a Fuel Assembly into the Improper Position	AOO
15.4.8	Control Element Assembly (CEA) Ejection	PA
15.5 Increase in RCS Inventory		
15.5.1	Inadvertent Operation of the ECCS	AOO
15.5.2	CVCS Malfunction-Pressurizer Level Control System Malfunction with a Loss of Offsite Power	AOO
15.6 Decrease in RCS Inventory		
15.6.1	Inadvertent Opening of a Pressurizer Safety/Relief Valve	PA
15.6.2	Double-Ended Break of a Letdown Line Outside Containment	AOO
15.6.3	Steam Generator Tube Rupture	PA
15.6.4	Radiological consequences of main steam line failure outside the containment (BWR)	N/A
15.6.5	Loss-of-Coolant Accident	PA
15.7 Radioactive Material Release from a Subsystem or Component		
15.7.1	Radioactive Gas Waste System Failure	PA
15.7.2	Radioactive Liquid Waste System Leak or Failure	N/A
15.7.3	Postulated Radioactive Releases due to Liquid-Containing Tank Failures	PA
15.7.4	Fuel Handling Accident	PA
15.7.5	Spend Fuel Cask Drop Accidents	PA
15.8 Anticipated Transient without Scram (ATWS)		N/A

2.2 Initial Conditions

The events discussed in this chapter have been analyzed over a range of initial values for the principal process variables. The ranges have been chosen to encompass all steady state operational configurations (with the exception of part loop operation).

Analysis over a range of initial conditions is compatible with the monitoring function performed by the core operating limit supervisory system (COLSS) which is described in the APR1400 DCD, Tier 2 Section 7.7 and the flexibility of plant operation which the COLSS allows. This flexibility is produced by allowing parameter trade-offs by monitoring the principal process variables, synthesizing the margin to fuel thermal design limits, and displaying to the reactor operator the core power operating limit. The range of values of each of the principal process variables that was considered in analyses of events discussed is listed in Table 2.2-1.

2.2.1 Input Parameters and Characteristics

2.2.1.1 <u>Doppler Coefficient</u>

The effective fuel temperature coefficient of reactivity (Doppler Coefficient) as shown in the APR1400 DCD, TIER 2 Section 4.3 is multiplied by a weighing factor to conservatively account for higher feedback effects in the higher power density portions of the core and to account for uncertainties in determining the actual fuel temperature reactivity effects.

The effective fuel temperature correlation is discussed in the APR1400 DCD, TIER 2 Section 4.3. This correlation relates the effective fuel temperature, which is used to correlate Doppler reactivity, to the core power.

2.2.1.2 <u>Moderator Temperature Coefficient</u>

The events analyzed in this chapter model moderator reactivity as a function of moderator temperature instead of a moderator temperature coefficient. This method is used in order to more accurately calculate reactivity feedbacks due to the large moderator temperature variations which may occur during these events.

The moderator temperature coefficients corresponding to these moderator reactivity functions range from [ ]TS. These values include all uncertainties, and bound the expected moderator temperature coefficients for all power levels, control element assembly (CEA) configurations, and boron concentrations.

The most conservative, allowable value for the moderator temperature coefficient is assumed for each individual analysis.

2.2.1.3 <u>Shutdown CEA Reactivity</u>

The shutdown reactivity is dependent on the CEA worth available on reactor trip and the axial power distribution. For most transient analyses, conservative total CEA worth of -8 %$\Delta\rho$ and -5.5 %$\Delta\rho$

were used for hot full power (HFP) and hot zero power (HZP), respectively. For some events, more conservative values were used (i.e., less negative). However, in the case of steam line break events a CEA worth of -5.0 %$\Delta\rho$ was used for the full power cases. This worth is consistent with the moderator reactivity versus moderator temperature function used in these analyses (see the APR1400 DCD, Tier 2 Subsection 15.1.5). The foregoing values include uncertainties, the most reactive CEA stuck in the fully withdrawn position, and the effect of temperature on CEA worth for events initiated from HZP (the APR1400 DCD, Tier 2 Subsection 4.3.2.4.3).

The shutdown reactivity worth versus position curve which is employed in the majority of Chapter 15 analyses is shown in Figure 2.2-1 and is applicable for an axial shape with an axial shape index (ASI) of +0.3. This shutdown worth versus position curve yields a conservatively slower rate of negative reactivity insertion than is expected to occur during the majority of operations, including power maneuvering. Accordingly, it is a conservative representation of shutdown reactivity insertion rates for reactor trips which occur as a result of the events analyzed.

For some events, a dynamic axial power function is utilized based on the HERMITE code (see Subsection 2.5.5 of this report).

2.2.1.4 Effective Delayed Neutron Fraction

The effective neutron lifetime and delayed neutron fraction are functions of fuel burnup. For each analysis, the values of the neutron lifetime and the delayed neutron fraction are selected to be consistent with the time in life analyzed.

2.2.1.5 Decay Heat Generation Rate

Analyses assume decay heat generation based upon infinite reactor operation at the initial core power level identified for each event.

2.2.1.6 CEA Insertion Characteristics

The control element drive mechanism (CEDM) is designed to function during and after all normal plant transients. The CEA drop time for 90 % insertion shall be 4.0 seconds maximum. The drop time is defined as the interval between the time the power is removed from the CEDM coils and the time the CEA has reached 90 % of its fully inserted position.

2.2.1.7 Non Safety-Related Systems Assumed in the Analysis

Non safety-related systems are not required to mitigate the consequences of events. Only safety-related systems are credited in the APR1400 safety analyses. Nominal control system characteristics are modeled (best estimate) in the accident analyses only if they adversely impact the results.

2.2.1.8 Operator Action

Operator actions are required by plant emergency procedures following a DBE, and one or more actions are necessary to accomplish a safety-related function. Safety-related operator action is a manual

action required by plant emergency procedures that is necessary to cause a safety-related system to perform its safety-related function during the cause of any DBE. The successful performance of a safety-related operator action might require that discrete manipulations be performed in a specific order.

Operator action is credited in certain analyses to mitigate postulated events. In such cases, the action is not credited in the analysis before 30 minutes after event initiation even though the action can be performed from the main control room (MCR) within 30 minutes. In addition, operator errors are considered in developing event initiators and in considering limiting single failures (see Subsection 15.0.0.4.3 for a more detailed description).

Operator actions required to mitigate accidents are described in the individual event evaluation sections.

2.2.1.9 Loss of Offsite AC Power

All event analyses resulting in a turbine generator trip considered the loss of offsite power (LOOP) while applying the same acceptance criteria for the event with and without LOOP (i.e., without re-classifying the event). In the analyses for which the LOOP is assumed to result from a turbine trip, the time delay between the turbine trip and LOOP is assumed to be zero.

2.2.1.10 Methodology for Determining Uncertainties

Uncertainties that exist within an instrument signal are classified as either random or bias errors. Random errors are basic measurement uncertainties or variations that exist in any repeated measurement. The error is usually caused by the combination of numerous small effects that exist in any measurement. An exact value of a random error cannot be predicted for a specific measurement. Therefore, in order to account for the random errors, these unsystematic errors are enveloped by upper and lower limits, around the measured value, which bound the most probable value for the instrumentation output at any specific instance. The bias errors, on the other hand, do not exhibit the random normal distribution characteristics. Rather, they exhibit a correlated, predictable, fixed or systematic behavior. A bias exists where there is a known offset of measurement from the ideal value. Both the random and bias error effects of an instrument measurement loop are evaluated. Uncertainties inherent in the signal communication process are accommodated by the method of setpoint calculation recommended by ANSI/ISA-67.04-1994 (Setpoints for Nuclear Safety-Related Instrumentation).

To establish the total uncertainty in an instrument or measurement, the various random and bias error effects are appropriately combined. Those errors that are considered random are combined using statistical formulae such as Square-Root-Sum-of-the-Squares (SRSS). The bias errors, on the other hand, are algebraically combined. Finally, the resultant random and bias errors are algebraically combined to yield a total uncertainty.

Some events analyzed in the safety analysis result in a more severe environment for protection system equipment than other events. As a result, the expected total equipment uncertainties can be event-specific and a trip parameter can have an accident setpoint for each design basis event.

Table 2.2-1

Range of Initial Conditions

TS

Notes:

1) ASI = (A - B) / C
where: A = Core power in lower half of core
B = Core power in upper half of core
C = Total core power
For power less than 20%, TS is used.

2) Percent of distance between the wide range instrument taps.

3) Percent of distance between the narrow range instrument taps.

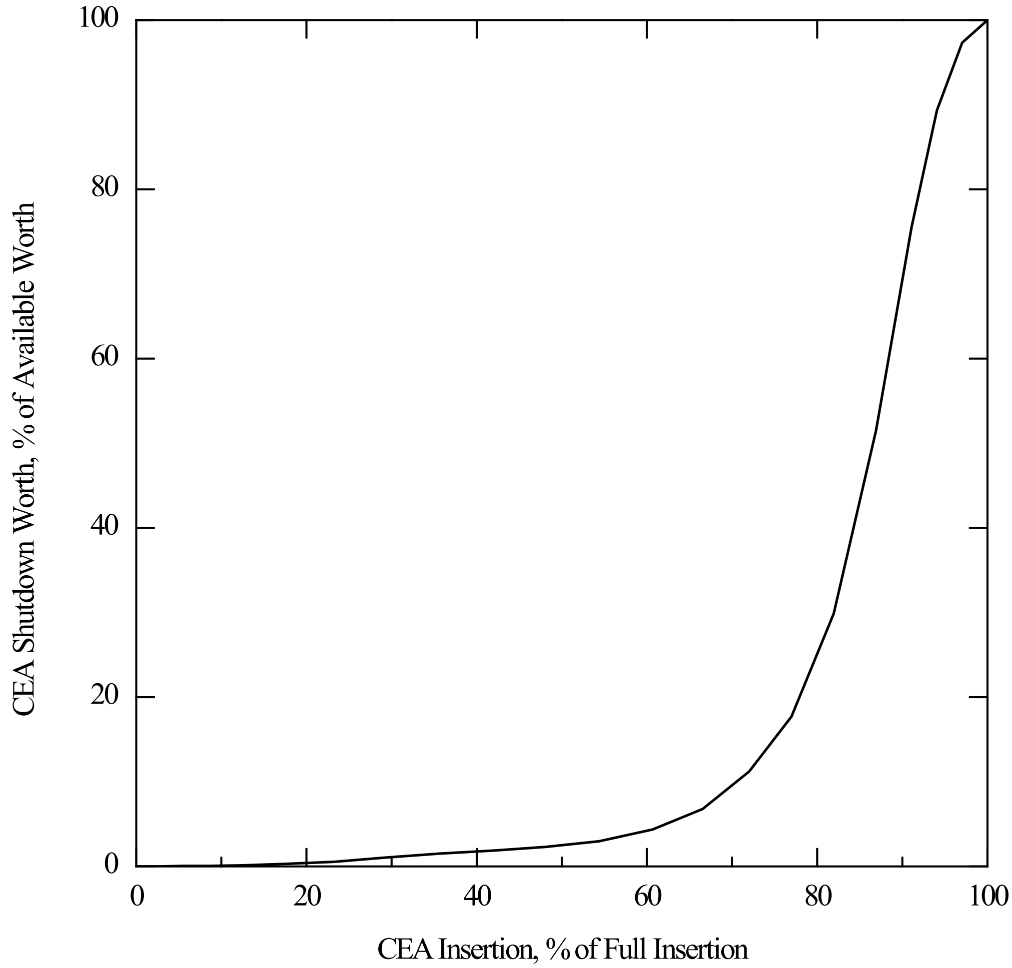

Figure 2.2-1
CEA Shutdown Worth vs. CEA Position

2.3 Major Assumptions

Major assumptions for non-LOCA analysis are different for each event. Specific assumptions for each AOO and PA are provided at relevant subsections of the APR1400 DCD, Tier 2 Chapter 15. The basic assumptions employed for all AOOs and PAs are based on the regulatory guidelines specific to each event, NSSS design features, computer code characteristics for thermal-hydraulic transient behavior, and experiences obtained from the former licensing processes, etc.

Major assumptions would be summarized as follows;

- No time delay is assumed between loss of off-site power (LOOP) and turbine trip caused by reactor trip. It means a loss of off-site power is assumed to occur concurrent with a turbine trip on reactor trip
- However, a 3-second time delay can be assumed between reactor trip breakers opening and the turbine trip because of the turbine trip delay circuits. This time delay is assumed in the CEA misoperation and CEA ejection events while the other events do not use the time delay conservatively.
- The control systems not required to perform the inherent safety functions are assumed to be in manual mode unless automatic operation of those systems adversely affects the consequences of the transient.
- Initial core power of 102% is assumed to maximize the post trip energy input to the RCS, if applicable. This will maximize the energy to be removed during the cooldown period and maximize the offsite radiological release.
- Operator action time is 30 minutes after the initiation of the event.
- For steam line break (SLB), in the calculation of moderator reactivity feedback, the moderator density is computed using the cold edge enthalpy of the affected side, which is defined as the enthalpy of the fluid from the cold legs of the loop with the ruptured steam generator without the effect of mixing with fluid from the intact loop.
- For steam generator tube rupture (SGTR), it is assumed that an immediate reactor trip after the initiation of the event occurs due to the high steam generator level trip condition. This will maximize the steam release through the MSSV with an assumed LOOP following turbine trip and hence the offsite radiological releases.

2.4 Single Failures

The purpose of the single failure list is to identify the single failures which could create the most adverse conditions during a given transient, regardless of the safety related status of that component or system. In most cases, the automatic action of safety-related systems overrides the operation of non-safety systems. The justification for choosing the most limiting single failure is explained in the individual analyses of this chapter.

The safety-related components in Table 2.4-1 are the:

TS

Table 2.4-1 (sh. 1 of 2)

Single Failures

a. Safety and Electrical System

TS

Table 2.4-1 (sh. 2 of 2)

b. Control System

TS

2.5 Computer Codes

The Nuclear Steam Supply System (NSSS) response to various events was simulated using digital computer programs and analytical methods.

Information about the computer codes used for analyzing specific events is provided in the following subsections. Any specialized modeling capabilities that are unique to a specific given event are provided in the relevant event analysis section.

2.5.1 CESEC-III Code

The CESEC-III is used to simulate the NSSS (unless specified otherwise for an event). CESEC-III is a version of CESEC which incorporates the ATWS model modifications documented in References 11 through 15 and includes additional improvements which extend the range of applicability of the models. CESEC-III explicitly models the steam void formation and collapse in the upper head region of the reactor vessel. It also includes a detailed thermal hydraulic model which explicitly simulates the mixing in the reactor vessel from asymmetric transients, an RCS flow model which calculates the time dependent reactor coolant mass flow rate in each loop, a wall heat model, 3-D reactivity feedback model, a safety injection tank model, and a primary-to-secondary heat transfer model which calculates the heat transfer for each steam generator node rather than for a steam generator as a whole. The CESEC-III is documented in References 4 and 5 and approved by NRC in Reference 3.

CESEC-III computes key system parameters during a transient including core heat flux, pressures, temperatures, and valve actions. A partial list of the dynamic functions included in this NSSS simulation includes: point kinetics neutron behavior, Doppler and moderator reactivity feedback, boron and CEA reactivity effects, multi-node average thermal hydraulics, reactor coolant pressurization and mass transport, reactor coolant system safety valve behavior, steam generation, steam generator water level, turbine bypass, main steam safety and turbine admission valve behavior, as well as alarm, control, protection, and engineered safety feature systems. The steam turbines, condensers and their associated controls are not included in the simulation. Steam generator feedwater enthalpy and flowrate are provided as input to CESEC-III.

During the course of execution, CESEC-III obtains steady-state and transient solutions to the set of equations that mathematically describe the physical models of the subsystems mentioned above. Simultaneous numerical integration of a set of first-order differential equations with time-varying coefficients is carried out by means of a simultaneous solution. As the time variable evolves, edits of the principal systems parameters are printed at pre-specified intervals. An extensive library of the thermodynamic properties of uranium dioxide, water, and zircaloy is incorporated into this program. Through the use of CESEC-III, symmetric and asymmetric plant response over a wide range of operating conditions can be determined.

2.5.2 COAST Code

The COAST is used to calculate the reactor coolant flow coastdown transient for any combination of active and inactive pumps and forward or reverse flow in the hot or cold legs. The program is described in Reference 7, and approved in Reference 17.

The equations of conservation of momentum are written for each of the flow paths of the COAST model assuming unsteady one dimensional flow of an incompressible fluid. The equation of conservation of mass is written for the appropriate nodal points. Pressure losses due to friction and geometric losses are assumed proportional to the flow velocity squared. Pump dynamics are modeled using a head-flow curve for a pump at full speed and using four-quadrant curves, which are parametric diagrams of pump head and torque on coordinates of speed versus flow, for a pump at other than full speed.

2.5.3 <u>STRIKIN-II Code</u>

The STRIKIN-II is used to simulate the heat conduction within reactor fuel rods and its associated surface heat transfer. The STRIKIN-II is described in Reference 8 and approved in References 16, 18 and 19.

The STRIKIN-II provides a single, or dual, closed channel model of a core flow channel to calculate the clad and fuel temperatures for an average or hot fuel rod, and the extent of the zirconium water reaction for a cylindrical geometry fuel rod. STRIKIN-II includes:

- Incorporation of all major reactivity feedback mechanisms,
- A maximum of six delayed neutron groups,
- Both axial (maximum of 20) and radial (maximum of 20) segmentation of the fuel element, and
- Control rod scram initiation on high neutron power.

2.5.4 <u>TORC and CETOP Code</u>

The TORC is used to simulate the three-dimensional fluid conditions within the reactor core. The TORC is described in References 20 and 21, and approved in Reference 22.

Results from the TORC include the core radial distribution of the relative channel axial flow rate that is used to calibrate CETOP, described in Reference 6 and approved in Reference 23. Transient core heat flux and thermal-hydraulic conditions from CESEC-III are input to CETOP which employs the KCE-1 critical heat flux correlation described in Reference 24.

2.5.5 <u>HERMITE Code</u>

The HERMITE solves the few-group space- and time-dependent neutron diffusion equation in order to consider integral effect of space and time in transient state. The HERMITE uses closed channel model or open channel model in TORC program as the thermal-hydraulic model to calculate the feedback effects of fuel temperature, coolant temperature, coolant density, xenon distributions and control rod motion. The 1D-, 2D- and 3D- neutron diffusion equation is solved with nodal expansion method. The fuel temperature model explicitly represents the pellet, gap and clad. The heat conduction equations are solved by a finite difference method. The HERMITE is described in Reference 10 and approved in

Reference 9.

2.5.6 HRISE Code

The HRISE is used to predict the transient DNBR for the thermal-hydraulic conditions beyond the range of applicability for the KCE-1 critical heat flux correlation employed in TORC and CETOP computer programs. The HRISE is described in Reference 25 and approved in Reference 26.

The HRISE performs thermal-hydraulic calculations using a closed channel model and calculates DNBR with various CHF correlations including MacBeth correlation which was approved by NRC for the post-trip steam line break analysis.

2.5.7 Reactor Physics Codes

Numerous computer codes are used to produce the input reactor physics parameters required by the NSSS simulation and reactor core programs previously described. These reactor physics computer programs are described in the APR1400 DCD, Tier 2 Chapter 4.

2.6 Analysis Process

Analysis processes for major events such as SLB, FLB, single RCP rotor seizure, control element assembly (CEA) ejection, and steam generator tube rupture are provided. Figures 2.6-1 through 2.6-5 show the analysis process for SLB, FLB, single RCP rotor seizure, CEA ejection, and steam generator tube rupture, respectively.

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Figure 2.6-1
Analysis Process for SLB

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Figure 2.6-2
Analysis Process for FLB

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Figure 2.6-3
Analysis Process for Single RCP Rotor Seizure

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Figure 2.6-4
Analysis Process for CEA Ejection

TS

Figure 2.6-5
Analysis Process for SGTR

3. ANALYSIS METHODOLOGY

As described in Chapter 1, the following computer codes are used for the non-LOCA safety analysis:

- CESEC-III — Simulate the NSSS
- TORC/CETOP — Determine core thermal margin and minimum DNBR
- COAST — Simulate RCP coastdown
- HERMITE — Determine short term dynamic response of reactor core
- STRIKIN-II — Calculate the cladding and fuel temperature for an average or hot fuel rod
- HRISE — Calculate the transient DNBR

Sections 3.1, 3.2, 3.3, 3.4, and 3.5 provide an overview of the plant system and mathematical models for each code, and Section 3.6 states the history of methodology changes.

3.1 CESEC-III Code [3, 4, 5]

3.1.1 Introduction

The CESEC-III is used to simulate the NSSS (unless specified otherwise for an event). CESEC-III explicitly models the steam void formation and collapse in the upper head region of the reactor vessel. It also includes a detailed thermal hydraulic model which explicitly simulates the mixing in the reactor vessel from asymmetric transients, an RCS flow model which calculates the time dependent reactor coolant mass flow rate in each loop, a wall heat model, 3-D reactivity feedback model, a safety injection tank model, and a primary-to-secondary heat transfer model which calculates the heat transfer for each steam generator node rather than for a steam generator as a whole.

CESEC-III computes key system parameters during a transient including core heat flux, pressures, temperatures, and valve actions. A partial list of the dynamic functions included in this NSSS simulation includes: point kinetics neutron behavior, Doppler and moderator reactivity feedback, boron and control element assembly (CEA) reactivity effects, multi-node average thermal hydraulics, reactor coolant pressurization and mass transport, reactor coolant system safety valve behavior, steam generation, steam generator water level, turbine bypass, main steam safety and turbine admission valve behavior, as well as alarm, control, protection, and engineered safety feature systems. The steam turbines, condensers and their associated controls are not included in the simulation. Steam generator feedwater enthalpy and flowrate are provided as input to CESEC-III.

During the course of execution, CESEC-III obtains steady-state and transient solutions to the set of equations that mathematically describe the physical models of the subsystems mentioned above. Simultaneous numerical integration of a set of first-order differential equations with time-varying coefficients is carried out by means of a simultaneous solution. As the time variable evolves, edits of the principal systems parameters are printed at pre-specified intervals. An extensive library of the thermodynamic properties of uranium dioxide, water, and zircaloy is incorporated into this program. Through the use of CESEC-III, symmetric and asymmetric plant response over a wide range of operating conditions can be determined.

3.1.2 CESEC-III Application

The acceptability of CESEC-III to perform SRP Chapter 15 [2] licensing analyses for the following events;

1. Decrease in feedwater temperature (DCD, TIER 2 Subsection 15.1.1)
2. Increase in feedwater flow (DCD, TIER 2 Subsection 15.1.2)
3. Increase in main steam flow (DCD, TIER 2 Subsection 15.1.3)
4. Inadvertent opening of a steam generator atmospheric dump valve (DCD, TIER 2 Subsection 15.1.4)
5. Steam system piping failure (DCD, TIER 2 Subsection 15.1.5)
6. Loss of external load (DCD, TIER 2 Subsection 15.2.2)
7. Turbine trip (DCD, TIER 2 Subsection 15.2.3)
8. Loss of condenser vacuum (DCD, TIER 2 Subsection 15.2.5)
9. Loss of normal AC power (DCD, TIER 2 Subsection 15.2.6)

10. Loss of normal feedwater flow (DCD, TIER 2 Subsection 15.2.7)
11. Feedwater system pipe breaks (DCD, TIER 2 Subsection 15.2.8)
12. Total and partial loss of forced reactor coolant flow (DCD, TIER 2 Subsection 15.3.1)
13. Reactor coolant pump shaft seizure (DCD, TIER 2 Subsection 15.3.3)
14. Uncontrolled CEA withdrawal (DCD, TIER 2 Subsection 15.4.1 and 15.4.2)
15. CEA misoperation (DCD, TIER 2 Subsection 15.4.3)
16. Boron dilution (DCD, TIER 2 Subsection 15.4.6)
17. Spectrum of rod ejection accidents (DCD, TIER 2 Subsection 15.4.8)
18. Chemical and volume control system malfunction (DCD, TIER 2 Subsection 15.5.2)
19. Break of a letdown line (DCD, TIER 2 Subsection 15.6.2)
20. Steam generator tube rupture (DCD, TIER 2 Subsection 15.6.3)

3.1.3 CESEC-III Analytical Models

The fixed nodalization developed in CESEC-III is shown in Figure 3.1-1. The reactor vessel is symmetrically split with nodes (volume) representing the downcomer, inlet plenum, core, outlet plenum, and vessel head. The vessel head is common to both loops. The purpose of the symmetric division is to model asymmetric thermal-hydraulics in the reactor vessel. Each hot and cold leg of the primary piping is modeled as a single node. Each steam generator is modeled as two nodes representing the tube section and two nodes for the inlet and outlet plenum. The pressurizer is modeled as one node composed of two regions to simulate non-equilibrium conditions. The physical description of nodes is given in Figure 3.1-2.

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Figure 3.1-1
CESEC-III Nodal Scheme

NODE	PHYSICAL DESCRIPTION
1	Cold Leg A
2	Upstream Half of Inlet Plenum (before flow mixing)
3	Downstream of Inlet Plenum (after flow mixing)
4	Cold Leg B
5	Core
6	Upstream Half of Outlet Plenum
7	Downstream Half of Outlet Plenum
8	Hot Leg
9	Steam Generator Inlet Plenum
10	Upstream Half of Steam Generator Tubes
11	Downstream Half of Steam Generator Tubes
12	Steam Generator Outlet Plenum
13	Same as 1 in other Steam Generator Loop
14	Same as 2 in other Steam Generator Loop
15	Same as 3 in other Steam Generator Loop
16	Same as 4 in other Steam Generator Loop
17	Same as 5 in other Steam Generator Loop
18	Same as 6 in other Steam Generator Loop
19	Same as 7 in other Steam Generator Loop
20	Same as 8 in other Steam Generator Loop
21	Same as 9 in other Steam Generator Loop
22	Same as 10 in other Steam Generator Loop
23	Same as 11 in other Steam Generator Loop
24	Same as 12 in other Steam Generator Loop
25	Reactor Vessel Closure Head
26	Pressurizer

Figure 3.1-2
Physical Description of Nodes

3.1.3.1 Field Equations

Neutronics Model

CESEC-III calculates core neutronics using the point kinetics model. This model permits up to six groups of delayed neutron precursors. Gamma heating of the moderator is modeled as a constant fraction of the total power. The reactivity feedback from changes in the fuel temperature (Doppler) is calculated using a single-pin/single-axial-node model.

For steam line break events, the code calculates the moderator reactivity feedback using the cold edge temperature. The cold edge temperature is the coolant temperature at the core center plane along the core radial edge adjacent to the broken steam generator cold legs. This temperature subtracts out the inlet plenum mixing effects. The calculation of the cold edge temperature is based on several approximations. It assumes steady state symmetrically split core flow, mirror asymmetric planar enthalpy distribution, neglects core bypass flow and does not subtract density driven cross flow (separate from the asymmetric inlet plenum mixing flows) in the downcomer.

Reactivity contribution from control rod movement (input as a function of time and signal delay) and boron concentration (as calculated by the boron transport model) are modeled in the code. The kinetics equation is coupled to 1971 ANS Standard for decay heat. Upto 11 fission product groups can be specified. The 11 group decay constants used are from the ANS Standard for infinite reactor operation.

USNRC staff finds the model for neutronics and decay heat acceptable. The modeling of cold-edge temperature and reactivity control is acceptable.

Thermal-Hydraulic

The thermal-hydraulics for the reactor coolant system (RCS), excluding the pressurizer and its associated surge line, are calculated using homogeneous equilibrium model (HEM) assumptions. CESEC-III calculates a single RCS pressure, from which the thermodynamic states and fluid properties are derived.

This assumption decouples the momentum equations from the mass and energy equations. The code does not solve individual junction momentum equations per se. The RCS flows are driven by the calculated pump and gravity heads using separate loop “mechanical energy” equations (there are no vector effects). The equation for mechanical energy balance assumes the flow throughout the loop changes uniformly and is composed of a pump model, gravity terms, geometric and D’Arcy friction pressure losses. The core bypass flows are modeled as fixed fraction of the split core flow. The code imposes a fixed pressure drop across the divided core.

USNRC finds the simplified model of the system hydraulics acceptable. The applicability of the code is limited to transients which do not result in two-phase fluid conditions in the cold legs of the reactor coolant system.

Thermal Conduction Model

CESEC-III has the capability to model heat capacities in the primary system walls, the steam generator tubes, and the reactor vessel upper head intervals. Energy conduction from the component metal walls is calculated using Fourier equation. The wall energy distribution is solved using 13 radial nodes which include two nodes in the cladding.

Heat conduction in the steam generator tubes is modeled as a heat slab which assumes rapid equilibrium (tube time constant ~ 1 second) with the primary fluid. Conduction through the tubes is modeled quasi-statically with the tube resistance determined by the system initialization procedure.

The core heat generation model is a constant dimension single pin with a single axial node and a single coolant node. The pin is modeled as three equilibrium radial nodes with the cladding and gap homogenized into the third node. Equivalent conductivities are used in the lumped parameter heat conduction calculation with a fixed fraction of the neutronic power being deposited in each pin node as well as in the single coolant node. To couple this model with the thermal-hydraulic model of a split core, an average core enthalpy is used to calculate the temperature for the single coolant node.

USNRC staff has concluded that the simplified assumptions used in CESEC-III can be acceptably applied in licensing and scoping safety analyses.

Mixing Model

To simulate the amount of mixing in the reactor vessel downcomer/lower region and in the outlet plenum region, CESEC-III utilizes the mixing factors as described in page 2-2 and Fig. 2-1A of CESEC Report [5]. If $W_{i,j}$ is used to denote the flow from the i-th to the j-th node, the cross flows with regard to the density driven terms, $W_{14,2}$ and $W_{19,7}$, are determined by direct solution of the thermal hydraulic equations (Equations 2-1 and 2-2 of Reference 5). The coolant flows from the upper plenum nodes to the vessel head node ($W_{18,25}$ or $W_{6,25}$) are specified by user input fractions. The vessel head fluid returning into the outlet plenum nodes is assumed to be evenly distributed. The mixing flows, $W_{2,15}$, $W_{14,3}$, $W_{6,19}$, $W_{18,7}$, are calculated as described on page 2-2 of the CESEC Report [5] using experimentally determined mixing parameters, F_I and F_o defined such that:

$W_{2,15} = F_I W_{2,3}$

$W_{14,3} = F_I W_{14,15}$

$W_{6,19} = F_o W_{6,7}$ and

$W_{18,7} = F_o W_{18,19}$,

where $W_{2,3}$, $W_{14,15}$, $W_{6,7}$, $W_{18,19}$ are found from the solution of the time dependant conservation of mass and energy equations (Equations 2-1 and 2-2 of Reference 5).

CESEC-III doesn't model cross flow in the core. The effect of cross flow is included in the 3-D reactivity feedback only. APR1400 conservatively didn't consider the 3-D reactivity feedback on steam line break event which produces the greatest asymmetry in loop flows. Moderator reactivity feedback, other than the 3-D feedback, factors out the effect of mixing to use the density on the cold edge of the

core. For conservatism, the cold edge density determined from RCS pressure and the cold edge enthalpy. The cold edge enthalpy is defined as the enthalpy of the fluid from the cold legs of the loop with the ruptured steam generator, with the addition of cold heat up to the core axial midplane, and without the effect of mixing with fluid from the other loop. Figure 3.1-3 shows the sensitivity of maximum post-trip fission power to variations in the assumed mixing factors during the steam line break event. The change of post-trip fission power to mixing is due to the temperature variations during the event. The abscissa of the figures is a mixing factor multiplier which means increasing or decreasing both F_I and F_o by a fixed factor for an entire transient. A reduction in mixing of the order of 10% for this transient would yield an increase in maximum post-trip fission power of less than 10 MW out of a total of 100 MW.

The borated safety injection water can be supplied by the safety injection system. The borated injection flow rates vs. pressure are specified by input tables. Once the injection flow reaches the RCS node, it is assumed that the borated safety injection flow mixes instantaneously and homogeneously with the reactor coolant in that node. The boron is transported through the RCS by solving at each time step the continuity equation for each coolant node for the boron concentration:

$$M \frac{dC}{dt} = W_{in} C_{in} - W_{out} C$$

Where, C is the boron concentration

C_{in} is the inlet boron concentration

W_{in} is the inlet flow rate

W_{out} is the outlet flow rate

M is the mass inventory in the node

The boron concentration for the reactor core nodes is used to calculate the reactivity contribution due to boron via an input reciprocal boron worth. A time delay is input to CESEC-III to account for the time required to start the diesel generator and to bring the safety injection pumps to full speed. An additional time delay is accounted for the time required for the unborated water in the safety injection line to be swept out before borated water enters the reactor vessel. For steam line break event, minimum boron worth is conservatively selected by considering the cycle length and the core temperature variations to minimize negative reactivity insertion (to maximize the possible return to power occurrence).

3.1.3.2 Material Properties

The thermodynamic properties used in CESEC-III are based on the McSlinteck/Silverstri formulation. The saturation properties, temperature, enthalpy and specific volume agree with ASME steam tables to within tenths of a percent for a wide range of condition.

For subcooled and superheated regions, the agreement in specified heat is within two of percent. The thermodynamic derivatives are similar to those applied in CEFLASH-4AS for pressures above 38.67 kg/cm^2A (550 psia), and for pressures below 38.67 kg/cm^2A (550 psia), the thermodynamic properties are obtained from equations developed by the Brookhaven National Laboratory as incorporated into the THOR computer program (BNL-NUREG-50534, July 1976).

3.1.3.3 Heat Transfer Correlation

CESEC-III models the primary to secondary heat transfer using the Dittus-Boelter force convection model for subcooled primary system conditions and the Akets, Deans and Crosser Correlation for two-phase flow with condensation. The secondary system heat transfer is modeled using the CE modified Rohsenow correlation for pool cooling. During reverse (secondary to primary) heat transfer regimes, CESEC-III applies the Dittus-Boelter correlation on the primary side during subcooled forced convection conditions and Hoeld's modification to the Chen correlation modeled using McAdams single-phase free convection correlation.

The Dittus-Boelter correlation does not differentiate between nodes of heating versus cooling. Similarly, McAdams correlation neglects the low Grashof number region. The application of both the Dittus-Boelter and the McAdams correlations are standard and acceptable. The McAdams correlation has previously been approved for CEFLASH-4AS and is acceptable for CESEC-III.

USNRC staff concluded that the simplified assumptions applied in CESEC-III for modeling heat transfer is not a best-estimate of expected physics but an approximation to it. For mild transients, its application is acceptable. For severe transients, such as steam line breaks and feedwater line breaks, the code could have limitations. However, the methodology of application of the code has been shown to result in an acceptably conservative solution. This is verified by the staff's audit (described in Subsection 3.1.1).

3.1.3.4 Friction Correlation

CESEC-III applies the D'Arcy/Moody model for calculating single-phase fluid frictional flow resistance. During two-phase conditions, CE applies a combination of two-phase friction multiplier correlations, which include the Thom correlation for pressures above 17.58 kg/cm^{2}A (250 psia) and the Matinelli-Nelson correlation for pressures below 17.58 kg/cm^{2}A (250 psia). These models have been previously approved for application in CEFLASH-4AS and are appropriate for application in CESEC-III. The two-phase form losses are computed using homogeneous equilibrium assumptions. This is acceptable.

The non-recoverable frictional losses are based on an isothermal friction factor which is Reynolds number, Re, dependent. The D'Arcy friction factor, f, is determined by the following correlations:

$f = 64/Re$	$Re < 1250$
$f = (-0.000004)Re + 0.056$	$1250 \le Re < 6000$
$f = 0.184/Re^{0.2}$	$Re \ge 6000$

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Figure 3.1-3
Sensitivity of maximum Post-trip Fission Power to Mixing in RV

3.1.3.5 Critical Flow

The CESEC-III options for critical fluid flow of steam are the Murdock-Bauman and CRITICO correlations. Correlations available for modeling two-phase and subcooled fluid flow are: Henry-Fauske/Moody combination, Henry-Fauske, Moody and homogeneous equilibrium model (HEM). These models are applied as a function of the system stagnation enthalpy and pressure. The Murdock-Bauman and the Henry-Fauske/Moody options were previously approved for use in CEFLASH-4AS and are acceptable for use is CESEC-III. The HEM correlation is generally considered a best-estimate two-phase fluid flow model. This typically results in lower calculated flow rates than either the Henry-Fauske or the Moody models.

The CRITICO correlation was developed for two-phase flow. The CRITCO is valid for single phase steam flow within a couple of percent of D'Arcy formulation for the pressure range between 7.03 and 70.31 kg/cm^{2}A (100 and 1,000 psia).

The user of CESEC-III can select any of the flow correlations listed above. In addition, the user can input a quality dependent discharge coefficient and a time dependent leak flow area. CESEC-III has an added option for simulating a leak, such as a 2.27 lpm (0.6 gpm) tube leakage used in Chapter 15 licensing evaluations and is not factored into the primary system mass and energy calculation.

3.1.3.6 Special Models

Enthalpy Transport

Enthalpy transport is a numerical means to transmit energy from one node to another. Rather than transmitting average nodal properties (i.e., temperature and enthalpy) from a volume containing a heat source (i.e., core), the enthalpy transport model calculates the exit fluid conditions and transfers those conditions into the downstream volume. This model is applied in the core and steam generators. This model provides more realistic outlet conditions and thereby a more realistic transient response. USNRC staff finds this model is acceptable.

Boron Transport

CESEC-III can model boron addition and boron dilution events. Boron addition is supplied by the safety injection systems. If selected, the charging/letdown contribution to these events can be neglected. The model can account for delays in startup of the SI pumps, diesel generators, and sweep-out of the ECC lines as the system aligns with the inside containment refueling water storage tank (IRWST). The boron concentration is solved using a transport model which assumes the instantaneous intra nodal mixing or can be input as a function of time.

USNRC staff finds the boron transport model acceptable when the reactivity changes slowly relative to the transport time of boron from the SI system.

3.1.4 Code Verification

3.1.4.1 USNRC Staff Audits

The USNRC staff performed audits of CESEC-III using the RELAP5 (an advanced thermal-hydraulic computer program developed by the Office of Nuclear Regulatory Research at NRC). These audit calculations were of the System 80 steam and feedwater line break events.

Figures 3.1-4 through 3.1-7 illustrate the good agreement between the two codes and the conservative trend predicted by CESEC-III. The audit calculation attempted to model the events with similar boundary and initial conditions as applied by CE. The feedwater line break audit confirmed the acceptability of CESEC-III to model severe pressurization events and the steam line break audit confirmed the acceptability of CESEC-III to model severe depressurization events.

The conclusions of these audits confirmed the acceptability of CESEC-III to calculate the thermal-hydraulic responses of the reactor coolant system with the boundary conditions employed. To predict conservative consequences, the applicant applies conservative constraints on the boundary conditions. The USNRC staff's audits confirmed the code's acceptability when applying these constraints.

The USNRC staff concluded that CESEC-III is an acceptable computer program for analyzing transients and accident events. The USNRC staff requires that all licensing analyses performed following issuance of this safety evaluation report (SER) be analyzed on CESEC-III.

3.1.4.2 Code Verification

The purpose of the assessment of CESEC-III is to examine the capability of the code to predict system response for PWR non-LOCA initiating events for a range of operating conditions. The activity includes verification of code/models through comparison to applicable experimental data and benchmarking of code/models through comparison to other code/models performing a similar calculation. The assessment of the code against plant test data satisfies an NRC request that experimental data be used in the verification of safety analysis computer codes.

Direct comparison of CESEC-III has been performed against the following CE codes:

1) COAST – four pump coastdown
2) CEFLASH-4AS – depressurization event

Qualification of the CESEC-III code against experimental results includes the following tests:

1) Hot zero power four pump coastdown
2) Turbine trip
3) Natural circulation cooldown

To show the accuracy of the pump model, a comparison has been made of the volumetric core flow fraction during a four pump coastdown transient. Figure 3.1-8 shows excellent agreement between

the CESEC-III and COAST codes. The COAST code is an NRC approved digital computer program used by CE to analyze coastdown transients for the reactor coolant pumps.

To show the adequacy of CESEC-III in predicting pressurization and depressurization transients, a comparison has been made of the code response during a full power turbine trip tests. The test data was compiled during the plant startup testing of a CE NSSS. The turbine trip is an event which results in a rapid increase in primary and secondary system pressures. For the test, the plant control systems were all in the automatic mode and operating normally except for the steam dump and bypass control system. One atmospheric dump valve (ADV) located downstream of the main stream isolation valves (MSIVs) was isolated and two ADVs located upstream of the MSIVs were in the manual mode.

Comparisons of the turbine trip test were performed. The steam generators 1 and 2 pressure responses are given in Figure 3.1-9. The pressure responses exhibit non-symmetric behavior caused by the non-symmetric steam flow. The calculated CESEC-III results agree well with the experimental results as seen from Figure 3.1-9. The peak pressures calculated by CESEC-III are higher than those recorded in the test.

Figure 3.1-10 shows the response of the pressurizer pressure. The pressurizer pressure calculated by CESEC-III agrees well with the test results over the entire transient time simulated. Figure 3.1-11 shows a comparison of the CESEC-III predicted water volume in the pressurizer against that calculated by related test data. As seen from Figures 3.1-10 and 3.1-11, the calculated CESEC-III results agree well with the experimental results.

To further substantiate CESEC-III's capability to simulate depressurization events, a comparison was made between CESEC-III and another CE thermal hydraulic code, CEFLASH-4AS. The event analyzed was a leak in the cold leg piping at the junction between the cold leg and the letdown line. The size of the leak was representative of twice the size of letdown line. The results shown in Figures 3.1-9 through 3.1-11 should only be reviewed in terms of the capability of both codes to simulate the system response resulting from such a leak (e.g., partial depletion of the reactor vessel water inventory) and not in terms of consequences resulting from a letdown line break as analyzed in Chapter 15.6 of a DCD (e.g., no depletion of reactor vessel water inventory). The assumptions for this analysis in terms of system operation (e.g., CVCS in manual), location of break (e.g., inside containment), etc., are not compatible with the assumptions made for Chapter 15.6 analysis (e.g., CVCS in automatic) for the limiting letdown line break which for dose consequences is located outside the containment.

The conclusion from this comparison is that both codes predict very compatible results. Calculation differences between both codes as shown in Figures 3.1-12 through 3.1-14 are attributed to modeling differences. For example, CEFLASH-4AS is basically a heterogeneous code with nodal fluid conditions evaluated at local pressures while CESEC-III is basically a homogeneous code with nodal fluid conditions evaluated at the average system pressure.

3.1.5 Limitations

The USNRC staff concludes that CESEC-III is an acceptable computer program for use in licensing applications for calculating DCD, Tier 2 Chapter 15 events. The approval is conditional upon an acceptable methodology of implementation. CESEC-III is composed of simplifying thermal-hydraulic assumptions which may not render best-estimate results for all analyzed events. An example is the modeling of the steam generator heat transfer response during a postulated feedwater line break. Imposing conservative boundary conditions on the model will provide a conservative and acceptable licensing analysis. For mild transient, CESEC-III with best estimate input assumptions calculates expected transient responses.

CESEC-III is not approved for events which lead to two-phase coolant conditions in the cold leg of the reactor coolant system, nor for events leading to two-phase stratified flow in the coolant loops. Should the applicant intend to apply this code for such events, further justification is required if the code is applied to resolve licensing concerns for recovery events where upper head inventory is replenished.

Conditional upon the limitations, KHNP concludes that CESEC-III is an acceptable computer program for analyzing transient and accident events for the APR1400.

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Figure 3.1-4
Primary System Peak Pressure as a Function of Time
For a Postulated Feedwater Line Break

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Figure 3.1-5
Peak RCS Pressure as a Function of Feedwater Line Break Size

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Figure 3.1-6
Pressurizer Pressure as a Function of Time for a Postulated Steam Line Break

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Figure 3.1-7
Total Reactivity as a Function of Time for a postulated Steam Line Break

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Figure 3.1-8
Four Pump Flow Coastdown Comparison of CESEC-III and COAST

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Figure 3.1-9
Secondary Pressure vs. Time
Turbine Trip CESEC Comparison (98% Power)

TS

Figure 3.1-10
Pressurizer Pressure vs. Time
Turbine Trip CESEC Comparison (98% Power)

TS

Figure 3.1-11
Pressurizer Water Volume vs. Time
Turbine Trip CESEC Comparison (98% Power)

TS

Figure 3.1-12
RCS Pressure vs. Time
(0.015 ft^2 Break)

TS

Figure 3.1-13
Inner Vessel Two-Phase Mixture Volume vs. Time
(0.015 ft^2 Break)

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Figure 3.1-14
Integrated Leak Flow vs. Time
(0.015 ft^2 Break)

3.2 TORC and CETOP Codes [6, 20]

3.2.1 Introduction

The minimum value for the departure from nucleate boiling ratio (MDNBR) which serves as a measure for the core thermal margin, is predicted for a CE-fleet reactor by the TORC (Thermal-Hydraulic of a Reactor Core).

The CETOP has been developed for determining core thermal margin for CE-fleet reactors. It uses the conservation equations for predicting the minimum DNBR (MDNBR) in its 4-channel core representation.

Details of TORC & CETOP codes including core modeling are described in References 6 and 20 (CETOP & TORC) and Thermal Design Methodology Technical Report (Reference 31).

3.2.2 TORC and CETOP Verification

USNRC staff had reviewed the TORC and CETOP topical report. The review included the conservation equations, constitutive equations, transport coefficients, method of solutions. For the CETOP, the benchmark results were compared to TORC.

Details of the codes verification are described in References 6 and 20 (CETOP & TORC) and Thermal Design Methodology Technical Report (Reference 31).

3.2.3 Limitations

Limitations of the TORC and CETOP codes are described in References 6 and 20 (CETOP & TORC) and Thermal Design Methodology Technical Report (Reference 31). The summary of the limitations are ;

- The code is acceptable for performing steady state calculations of the reactor core thermal hydraulic performance involving unblocked flow channels or subchannels (other than the minimal blockage offered by intact spacer grids.
- The application should be limited to conditions of single phase flow or homogeneous two-phase flow (such as bubbly flow regime).

TORC and CETOP are used to predict the minimum DNBR for core operating condition provided by the system transient analysis code, assuming steady state condition at each time step. KHNP concludes that TORC and CETOP are the acceptable computer programs for analyzing transient and accident events for the APR1400.

3.3 COAST CODE [7]

3.3.1 Introduction

The loss of coolant flow transient is computed by the COAST which calculates the individual loop flow rates and steam generator pressure drops as a function of time following coastdown of any number of reactor coolant pumps. The conservation equations of mass and momentum are solved for the fluid stream in each loop in conjunction with a torque balance on each reactor coolant pump.

The analysis utilizes fundamental pump data obtained by the manufacturer. The reactor coolant pumps are full-flow tested to provide the pump head-flow characteristic curve at operating speed. Model test data provides the pump head-flow-torque characteristics at off-design speed.

The mathematical model has been checked against field data from the Palisades' facility and is shown to be conservative.

3.3.2 COAST Model

The COAST code analyzes reactor coolant flow in a two loop-four pump plant. As shown in Figure 3.3-1, seven paths are considered; core region, two hot legs and four cold legs with reactor coolant pumps. The code can analyze the flow coastdown transient for any combination of active and inactive pumps including forward and reverse flow in any section.

As shown in Figure 3.3-1, the reactor coolant system is divided into seven flow sections and four nodal points. Under steady state four pump operation, all coolant flow passes through the first flow section consisting of the reactor vessel, core, and internals. One half the total flow enters each steam generator flow section which consists of the reactor vessel outlet nozzle, hot leg piping, and steam generator inlet plenum and tubes. One quarter of the total flow passes through each cold leg flow section which consists of a steam generator outlet nozzle, cold leg piping, reactor coolant pump, and reactor vessel inlet nozzle. The nodal points are located in the reactor vessel inlet and outlet plenums, and in the outlet plenum of each of the two steam generators. The nodal points are chosen at the junction of the loop flow sections where a common nodal pressure exists.

The equation for conservation of mass is written for each of the four nodal points. The equation for conservation of momentum is written for each of the seven flow segments assuming unsteady, one-dimensional flow of an incompressible fluid. Pressure losses due to friction, bends, and shock losses are assumed to be proportional to the flow velocity squared. Pump dynamics are modeled with the aid of the traditional head-flow characteristic curve for fully operable pumps and with the 4-Quadrant diagram (parametric curves of pump head and torque on coordinates of speed versus flow) for pumps at other than rated speed.

The pump dynamic equations are different in form depending on whether the pump is coasting down (inactive) or supplied with electrical power (active). For an active pump, it is assumed that the electrical torque provided by the pump motor is sufficient to maintain constant pump speed. In this case, the familiar head-flow characteristic curve is supplied as the pump dynamic equation. For an inactive pump, an equation expressing the conservation of angular momentum is written in terms of pump-motor

inertia and torque exerted on the pump impeller by the cool ant. The coolant momentum equation and pump momentum equation are coupled through the pump 4-Quadrant diagram. The governing conservation equations of fluid mass and loop momentum are solved simultaneously with the pump dynamics equations using conventional numerical integration techniques to obtain the transient flow coastdown.

Input to COAST includes the pressure loss coefficients for both forward and reverse flow and the fluid densities and momentum averaged fluid weights for each of the flow segments. The pump head-flow characteristic curve, pump inertia and the initial pump operating conditions are also code input. The code output consists of the time dependent speed of each pump and the transient flow rate in each flow section.

In the plant the low flow trip system sums the steam generator differential pressure measurements and trips the plant when the measurement is less than a pre-determined value. The COAST output contains the time dependent steam generator pressure data representative of the actual pressure measurements from nozzles in the hot leg piping and in the steam generator outlet plena. The values are computed in a separate subroutine which utilizes the transient loop flow rates and code input pressure loss coefficients (for both forward and reverse flow) for the flow path between the nozzle taps. The time dependent loop flow rates and the calculation of the steam generator differential pressure measurements are input to another digital code simulation for purposes of core thermal margin analysis.

3.3.3 Experimental Verification

Flow coastdown tests were conducted at the Palisades' plant in 1971 and 1972 [7]. Both the speed of each pump coasting down and the differential head of each pump as a function of time were obtained from Brush recorder traces. The data input to the COAST code was based on measured hydraulic parameters at Palisades for pump head, pump flow, steam generator and reactor vessel pressure drops.

The COAST code accepts one characteristic curve for the operating pumps and one 4-Quadrant diagram for the non-operating pumps. The characteristic curve (obtained by the manufacturer) for pump 5 was used for the former application and the characteristic curve for pump 4 was used for the latter application. This was intended to match as closely as possible the pump 4 coastdown (from four pump operation) test. The same code input data was used to predict all the coastdown cases since this is the way the code is used in design applications. However, in deriving pump flow from the measured pump parameters, each pump's own characteristic curve was utilized to determine as best as possible the actual transient flow rates in the plant tests. As a practical matter, this should not be an important point since the Palisades pumps' characteristic curves are quite similar particularly around the normal operating point and differ only in the run-out mode.

A four pump coastdown test was conducted with primary loop conditions of 532^{o}F and 2100 psia. The pump 6 data was taken as representative of all four pumps coasting down and is compared to the COAST code predictions in Table 3.3-1. The analytical model predicts a more rapid coastdown during the first five seconds of the transient and hence conservatively predicts the low flow trip setpoint.

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Figure 3.3-1
COAST Loop Geometry

Table 3.3-1
Four Pump Coastdown from Four Pump Operation

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A two pump (5, 6), opposite loop coastdown from a four pump initial condition was conducted with primary loop conditions of 277.8℃ and 147.6 kg/cm^{2}A (532^{o}F and 2,100 psia). The pump 6 data was taken to represent the coastdown behavior of both pumps 5 and 6. The pump 7 data was used to represent the run-out flow of both pumps 4 and 7 in calculating the transient core flow. The test data for pumps 6 and 7 are compared to the corresponding analytical predictions in Table 3.3-2. The COAST code predicts a lower core flow for the first 10 seconds of the transient and hence conservatively predicts the low flow trip setpoint.

It is interesting to note that the run-out flow of pump 7 (and 4) in the test was over-predicted by the COAST calculation. This is to be expected since in the COAST calculation the characteristic curve of pump 5 was used to represent the operating pumps. Pump 5 is the strongest pump, that is, it produces the largest flow for a given head in the run-out condition. If the characteristic curve of either pump 7 or 4 were used in the COAST code, a smaller run-out flow and a lower core flow would then be calculated throughout the transient. This would have indicated an even more conservative prediction of the low flow trip setpoint.

A one pump (4) coastdown from a four pump initial condition was conducted with primary loop conditions of 277.8℃ and 147.6 kg/cm^{2}A (532^{o}F and 2,100 psia). The data is compared to the analytical predictions in Table 3.3-3. The predicted core flow is in slightly better agreement (over a much longer period of time) with the core flow derived from the pump test data when compared to the preceding two cases. This calculation is the most precise comparison to test data (as noted previously) in that the manufacturers characteristic curve for pump 5 was used to calculate the run-out of the operating pumps and the 4-Quadrant diagram for pump 4 was used for the pump coasting down. The COAST code predicts a slightly lower core flow and hence conservatively predicts the low flow trip setpoint.

Two coastdown tests from part-loop initial conditions were conducted with primary loop conditions of 200.1℃ and 59.8 kg/cm^{2}A (393^{o}F and 850 psia). The COAST code input data was identical to that used in the preceding three cases except for the change in fluid density due to the lower loop temperature and pressure. A one pump (4) coastdown from an initial three pump (4, 6, 7) condition is compared with the analytical predictions in Table 3.3-4. It is noted that pump 5 is initially in back-flow and hence a portion of the simulation not tested in the preceding three tests is verified. The COAST code predicts a slightly lower core flow and hence conservatively predicts the low flow trip setpoint.

Table 3.3-2
Pumps 5&6 Coastdown from Four Pump Operation

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Table 3.3-3
Pump 4 Coastdown from Four Pump Operation

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Table 3.3-4
Pump 4 Coastdown from 4-6-7 Operation

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The final test was a one pump (6) coastdown from an initial two pump (6,7) condition. The RPM signal pick-up from pump 6 did not function properly during this test. The measured pump d/p's are compared to predicted values in Table 3.3-5. A "core flow" based on pumps 4, 5, and 7 but excluding pump 6 was computed from the test data and is compared to a similarly derived COAST code prediction. The prediction is slightly non-conservative during the crucial first few seconds of the transient. This is due, at least in part, to over predicting the run-out flow of pump 7 with the COAST code calculation using the characteristic curve of the strongest pump (pump 5).

The predicted pump 6 head decreases more rapidly than the test data. This is similar to the predicted heads of the pumps coasting down in the four preceding tests. Most likely, the predicted flow of pump 6 also decreases more rapidly as shown in the four preceding tests for the pumps coasting down. It was assumed that the pump 6 flow test trace was higher than the predicted value by a constant 0.02 units; the smallest differential exhibited in the four preceding tests. A core flow based on the test data including the "guessed" pump 6 flow is compared to the COAST code prediction in the last column of Table 3.3-5. Although this is a somewhat artificial treatment, it is probable that the COAST code once again is slightly conservative in predicting the low flow trip setpoint for this coastdown mode. Certainly, the fundamental validity of the COAST model is borne out by the comparisons with the test data.

It is concluded that the fundamental mathematical model in COAST predicts the core flow conservatively during at least the first five to ten seconds of any mode of coastdown from all combinations of pumps initially operating. Since the plant is tripped within the first few seconds of any coastdown, the flow model is conservative for the prediction of core thermal margins since the core flow will exceed the amount indicated by analysis.

It is important to recognize that what is contained herein is a determination of the adequacy of the COAST mathematical model. As-built pump and loop parameters were used to generate the analytical predictions for comparison with test data. However, in the design stage, the moment-of-inertia of the rotating assembly is established based on coastdown and core thermal margin calculations using conservative assumptions. Maximum pressure loss coefficients and a minimum pump head-flow characteristic curve are selected to match the design coolant flow rate. The design coolant flow rate is the result of a series summation of all adverse tolerances in the reactor coolant system pressure drop calculations and the pump head-flow curve calculations. The resultant pump-motor moment-of-inertia is the minimum value that must be supplied by the manufacturer. This procedure ensures that the as-built loop hydraulics will not invalidate the DCD Loss-of-Coolant Flow analysis. Thus, conservatism is attained by determining the pump-motor moment-of-inertia utilizing conservative plant design data in a computer simulation which is based on a conservative mathematical model.

Table 3.3-5
Pump 6 Coastdown from 6-7 Operation

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3.3.4 Limitations

In the development of the mathematical model used in COAST, primary interest has centered on the early stages of coastdown, up to the point of low flow scram, in particular, because the program is used with a thermal margin analysis program to determine the low flow scram trip point. Results from COAST calculations are characterized as conservative relative to measured flow coastdowns, as they predict slightly lower flow rates over the first 5 seconds. Consequently, a slightly earlier low flow scram would be obtained from calculated than from measured parameters. From the results report thus far, the conservative characterization of the COAST code predictions on low flow trip point determination is considered marginal. As a result, DNBR predictions for loss of flow accident analysis in DCD's must be viewed as having a small margin of conservatism.

The loop segment momentum balance equations intentionally omit the momentum flux and gravity head terms. In the early stage of a coastdown, these terms would not contribute measurably in the transients. However, in the terminal stages the gravity head term would be a major contributor so that intentional neglect of this term will limit the applicability of COAST in coastdown calculation when used as a basis for clad temperature or core thermal margin calculations after low flow scram. In core thermal studies for loss of flow accidents, the initial temperature gradients are rapidly altered by reactor scram, turbine tripout, and the flow reduction so that neglect of the gravity force changes due to the loop temperature profile transient will result in an improper estimate of end-of-coastdown flow rate. As a result, use of the COAST program without gravity head effects, and without variable density parameters in the momentum equations will be of limited accuracy when used in a core thermal assessment near the end of coastdown.

Because the COAST calculations are made de-coupled from the thermal transient accompanying a flow coastdown, the general applicability of the COAST code for accident studies in DCD's is considered limited as to accuracy following the low flow scram, particularly in the case for the flow pump coastdown. For the partial flow loss from a single or two pump outage the flow coastdown predictions for isothermal conditions are shown to be reasonably accurate; however, no comparisons with measurements from typical operating transient temperature distributions have been made, so that the accuracy of COAST code predictions for these cases has not been fully evaluated.

Conditional upon the limitations, KHNP concludes that COAST is an acceptable computer program for analyzing four pumps coastdown for the APR1400.

3.4 HERMITE Code [9]

HERMITE 1-D model is used to determine the short-term response of the reactor core for some events. The benefit of using 1-D space-time kinetics comes from the larger scram reactivity compared to quasi-static calculated scram reactivity at a given CEA position. The additional scram reactivity will cause the core power to decrease at a faster rate following the initiation of the scram. This is true for all Design Bases Events where a trip is initiated.

3.4.1 HERMITE Application

HERMITE 1-D model is used to perform DCD Chapter 15 licensing analyses for the following events;

a. Total loss of reactor coolant flow (SRP Subsection 15.3.1)
b. Reactor coolant pump rotor seizure (SRP Subsection 15.3.3)
c. Reactor coolant pump shaft break (SRP Subsection 15.3.4)

3.4.2 HERMITE Analytical Models

The HERMITE program solves the few-group space- and time-dependent neutron diffusion equation including the feedback effects of fuel temperature, coolant temperature, coolant density and control rod motion. The neutronics equations are solved by a nodal expansion method or a finite difference method. The fuel temperature model explicitly represents the pellet, gap and clad. The heat conduction equations are solved by a finite difference method.

The HERMITE space-time kinetics and thermal-hydraulics code was first described in the HERMITE Topical Report, CENPD-188(A) [9], which was released in 1976. Since that time, more accurate methods and additional features have been added. Major additions and improvements have included the TORC open channel flow model, the nodal expansion method (NEM) neutronics solution and the FIESTA 1-D neutronics solution. The detailed descriptions are described in the HERMITE code manual [27].

3.4.2.1 Neutronics Model

In the generation of axial shapes in support of the process of preliminary safety analysis, the code employs a two-group steady-state finite difference diffusion theory model in which the fluxes are calculated at mesh boundaries and the cross sections are defined over mesh intervals. 1-D HERMITE uses the 1-D neutronics model from the approved code FIESTA [28].

The code employs macroscopic cross section models with microscopic cross sections for xenon and soluble boron, cross sections or coefficients for thermal-hydraulic and Doppler feedbacks, and macroscopic cross section changes to simulate control rod insertion. The fuel cross sections, including xenon and feedback terms, are derived from 3-D ROCS solutions. The rod cross sections which are modeled as changes in group 1 and 2 absorption, are derived by tuning to match ROCS rod worths.

With 1-D HERMITE, separate cross sections (all feedbacks) are derived from ROCS at each axial level. These axially variant cross sections more accurately reflect different radial flux distributions, nuclide distributions, and other conditions at each level of the 3-D ROCS solution.

In 1-D HERMITE the fuel temperature is first calculated as a function of radially averaged linear heat rate and burnup. Fuel cross section changes which include all cross section types for both groups are then calculated as an axial function of the fuel temperature.

The code integrates the power distribution axially from the bottom of the core to the mid-point of each mesh interval to obtain a local enthalpy. The local enthalpy is converted to moderator density and used for computing cross sections. The code models soluble boron as a moderator density dependent absorber.

The code models a pseudo-hot-pin in which the linear heat rate at each level is a multiple, determined from a radial peaking factor, of the core average linear heat rate at that level. In the code the radial peaking factor is adjusted for the presence of control rods.

3.4.2.2 Thermal-Hydraulic Model

The thermal-hydraulics calculation models a number of closed, parallel channels. The mesh used for the thermal calculation may be coarser than the mesh used for the neutronics solution. The code solves the one dimensional continuity and conservation of energy equations, along with the equation of state, for each channel. Axial expansion of the coolant and two phase slip flow are modeled. Water and steam properties are determined from tables based on the ASME steam tables. The core pressure, channel inlet temperatures, and channel inlet flow rates are user-specified. The surface heat flux is found using either the Dittus-Boelter correlation for the forced convection regime or the Jens-Lottes correlation for the nucleate boiling regime.

A single channel average fuel rod is modeled for each flow channel. The model defines three regions within a rod: the fuel pellet, the gap, and the cladding. The fuel and clad regions may be divided into any number of radial nodes.

No heat is generated in the gap or clad although a portion may be deposited directly into the coolant. The thermal properties of the fuel and clad are found as a function of temperature. The density of the fuel is assumed not to change. Gap conductance for steady state initialization is found as a function of linear heat rate. For transient calculations the gap conductance may be found as a function of gap temperature.

3.4.3 USNRC Approval

The HERMITE program was approved in Reference 9.

3.4.4 Code Verification

Extensive verification calculations have been performed and are summarized in HERMITE Topical Report [9].

3.4.5 History

a) CENPD-188-A, "HERMITE A Multi-Dimensional Space-Time Kinetics Code for PWR Transients," March 1976, Reprinted July 1976. (NRC Approval Letter dated June 10, 1976).

b) CE-CES-91 Rev. 2-P, "Users manual for HERMITE Space-Time Neutronics and Thermal Hydraulics Code," March 1992.

c) CEN-133(B), "FIESTA: A One Dimensional, Two Group Space-Time Kinetics Code for Calculating PWR Scram Reactivities," November 1979. Approval: Letter, R. A. Clark (NRC) to A. E. Lundvall, Jr. (BG&E), Docket Nos. 50-317 and 50-318, Approval of CEN-133(B), March 13, 1981.

d) The nuclear power plants used HERMITE for analyzing transients and accident events are as follows:

Arkansas Nuclear One Unit 2	Palo Verde Units 1, 2, and 3
Waterford Unit 3	St. Lucie Unit 2
San Onofre Units 1, 2, and 3	CESSAR-DC (System 80+)
Yonggwang Units 3, 4, 5, and 6	Ulchin Units 3, 4, 5, and 6
Shin Kori Units 1, 2, 3 and 4	Shin Wolsung Units 1 & 2
Shin Ulchin Units 1 & 2	

3.4.6 Limitations

The USNRC staff concluded that HERMITE was an acceptable neutron kinetics computer program for solving the few-group diffusion equations in one, two, and three dimensions. The USNRC staff also concluded that HERMITE was acceptable for reference in license applications (Reference 9).

KHNP concludes that HERMITE is an acceptable computer program for analyzing transient and accident events for the APR1400.

3.5 STRIKIN-II Code

The STRIKIN-II code is a FORTRAN computer program which is used for the hot channel heat-up calculations in the safety analyses. The code is used to calculate the transient DNBR, coolant enthalpy and fuel temperatures in the hot rod in the hot assembly.

This summary is applicable to the non-LOCA transient analysis version of the STRIKIN-II code. This version of STRIKIN-II was derived from the LOCA analysis version and has been maintained independently with various modifications since 1975.

3.5.1 STRIKIN-II Application

The acceptability of STRIKIN-II is to perform DCD Chapter 15 licensing analyses for the following events;

a. Spectrum of CEA ejection accidents (SRP Subsection 15.4.8)
b. Calculate fuel and cladding temperatures for SLB

3.5.2 STRIKIN-II Analytical Models

The STRIKIN-II computer program provides a single, or dual, closed channel model of a core flow channel to calculate the clad and fuel temperatures for an average or hot fuel rod. STRIKIN-II includes:

a. Incorporation of all major reactivity feedback mechanisms,
b. A maximum of six delayed neutron groups,
c. Both axial (maximum of 20) and radial (maximum of 20) segmentation of the fuel element, and
d. Control rod scram initiation on high neutron power.

The STRIKIN-II code solves conservation equations and the specified inlet flow to calculate mass flow rate. It also solves the one-dimensional (axial) conservation of energy equation and the equations of state for the fluid with provisions for local fluid expansion.

In a fuel rod the STRIKIN-II code solves (radially) the one-dimensional cylindrical heat conduction equation for each axial region along the rod. The conduction model explicitly represents the gas region and dynamically calculates the gap conductance in each axial region. A volume averaged temperature for each radial node is defined by assuming spatially constant material properties within each radial node for one time step. The material properties are assumed to be time and/or temperature dependent and are updated after each time step.

The solution of the volume averaged radial conduction equation for each radial node is obtained by analytically integrating the heat conduction equation by assuming that the material properties are constant and that the adjacent region temperatures and heat fluxes are linearly varying for one time step. The STRIKIN-II code uniquely determines a heat transfer regime at the clad/coolant boundary for the updated temperature distribution.

The fuel rod can be divided into a maximum of 20 radial nodes at each axial interface. The fuel rod conductivities and surface temperatures are determined at each axial interface at the appropriate radial position (see Figure 3.5-1). The fuel rod volumetric specific heats, heat generation rates, and fuel temperatures are determined at the volume averaged radius of each radial node.

Axially a multimode closed fluid channel model is assumed by the code (see Figure 3.5-1). A channel may be divided into a maximum of 20 axial nodes. The channel fluid properties are determined at the interfaces between the axial nodes resulting in one more set of fluid properties than axial nodes. The end point interfaces are treated as half-nodes since the fluid properties at each axial interface are assumed to extend halfway into each adjacent node. Thus, the 20 axial node options are equivalent to having 19 full-length nodes and 2 half-length nodes. All full-length axial nodes are of equal length.

The STRIKIN-II code has a dual channel model which is illustrated in Figure 3.5-2. The option, when exercised, can represent two parallel channels. In the first channel (i.e., average rod in hot assembly) the code performs a fluid energy balance to get the enthalpy distribution along the entire channel. The axially dependent dynamically calculated enthalpy in the first channel is transferred and used in the second channel (i.e., hot rod). The basic assumption in the calculation is that of perfect radial mixing of the fluid within the assembly.

3.5.2.1 Fuel Rod Cylindrical Heat Conduction Model

The temperature distribution in the cylindrical fuel element shown in Figure 3.5-3 is obtained from the solution of the energy equation in the fuel element.

$$\rho C(t,r)\,\frac{\partial T(r,t)}{\partial t} = \frac{1}{r}\frac{\partial}{\partial r}\left[r\,k(t,r)\,\frac{\partial T}{\partial r}(r,t)\right] + q(r,t)$$

3.5.2.2 Modes of Heat Transfer

The modes of heat transfer considered by STRIKIN-II are: forced convection, nucleate boiling, transition boiling, stable film boiling, film boiling in a pool, heat transfer to steam, adiabatic (refill), and reflood. Note that initially the code assumes that all axial regions are in forced convection or nucleate boiling depending on which regime yields the lower fuel rod surface temperature.

3.5.2.3 Critical Heat Flux Correlations

In the subcooled and two-phase flow regimes the critical heat flux is calculated by the W-3 and Macbeth correlations. The CE-1 and KCE-1 correlations are added for the critical heat flux calculation in the non-LOCA version of STRIKIN-II code.

3.5.2.4 Logic for Determining Heat Transfer Regimes

The rod surface film coefficient used in STRIKIN-II is dependent on the heat transfer regime. Table 3.5-1 relates the heat transfer regimes of a typical boiling curve (Figure 3.5-4) to the STRIKIN-II heat transfer regimes and describes the applicable equation for determining either the heat flux or film coefficient.

3.5.2.5 The HGAP Routine

The transient gap conductance HGAP routine programmed in STRIKIN-II is based on the model used in the steady-state fuel temperature FATES code. Applicable equations as programmed in STRIKIN-II are as follows:

a) Gap Conductance Equation

The HGAP routine accounts for heat passing through the fuel rod gap region in the following manner:

TS

b) ΔT Across Gap

The inside clad surface temperature (T_c) and outside fuel pellet temperature (T_f) are approximated by linearly extrapolating the temperature from the two adjacent known temperatures in the respective material. Once the value of T_f and T_c are known, they are used to obtain the temperature dependent values of k_c and k_f from the STRIKIN-II conductivity routines.

c) Thermal Conductivity and Jump Distance of a Gas Mixture

The conductivity of gas mixture in the gap (k_g) and the jump distance (d) are computed in the subroutine (TCON) as follows:

TS

where, A_{ij} is a dimensionless weighting factor, dependent on viscosity, molecular weight, and Surtherland constant of gases "i" and "j".
DJT is "distance" related to the transport and molecular properties of the gases.
k_i is the conductivity of gas "i".
X_i is the mole fraction of gas "i".
M_i is the molecular weight of gas "i".

The fill gas may be any combination of the following 10 gases:

1. Helium
2. Argon
3. Neon
4. Krypton
5. Xenon
6. Nitrogen
7. Oxygen
8. CO
9. CO_2
10. Water Vapor

d) <u>Mean Radius of Contact Spots and Arithmetic – Average Roughness</u>

The mean radius of the contact spots (a) and the sum of the arithmetic-average roughness (R_m) are related to the input relative roughness of the fuel (RRF) and clad (RRC) as follows:

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e) Emissivity of Clad and Fuel

The emissivity of the clad is found from the temperature dependent expression:

[ ]TS

The emissivity of the fuel is input constant and by default equal to 0.85.

f) Mechanical Interface Pressure

The mechanical interface pressure between the fuel and clad is defined as

$$
\begin{aligned}
P_m &= 0 && \text{If } \Delta R > 0 \\
&= (|\Delta R| * t * YM) / (R2H)^2 && \Delta R < 0 \\
&= PMAX && P_m > PMAX
\end{aligned}
$$

where: ΔR = (Radius of clad – Radius of fuel) $_{hot}$, inches
YM = Young's Modulus of Elasticity, psi
R2H = Hot clad inside radius, inches
t = Cold clad thickness, inches

The gap thickness R is determined dynamically throughout the transient from the fuel and clad dimension which responds to the system pressure and temperature as detailed below.

g) Clad Dimension

The inside clad dimension must be computed during the transient for the correct evaluation of the hot fuel-clad gap. The thermally expanded inside clad diameter is computed from the following equation:

[ ]TS

where: R2H = hot clad inside radius, inches
R2C = cold clad inside radius, inches
ν = Poisson's ratio, dimensionless
t = as manufactured cold clad initial thickness, inches
E = Modulus of elasticity, psi
PG = rod internal plenum pressure, psia
PW = instantaneous system pressure, psia
ε_θ^P = plastic strain at rupture

The first term on the right hand side of the equation is the cold clad radius (including the creep effect); the second term is the change in radius due to thermal expansion; and the third term is the change in radius due to the mechanical differential pressure across the clad. The fourth term is calculated only at time of rupture.

h) <u>The Transient Gap Pressure Determination</u>

The transient gap pressure is based on the ideal gas law

P V = n R T

where: P = Pressure, psia V = Volume, ft^3
T = Temperature, °R R = Constant
n = Total number of moles of gas

For the transient the total moles of gas in the rod are assumes to remain constant; thus $\overline{n} = \sum_{i=1}^{N} n_i$: where the sum is over all regions containing gas.

For each region

$$\frac{P_i V_i}{T_i} = R n_i$$

Summing both sides of above equation gives

$$\sum_{i=1}^{N} \frac{P_i V_i}{T_i} = R \sum_{i=1}^{N} n_i = R\overline{n} = \text{constant}$$

At each time step the gap in each i-th region is assumed to be at the same pressure $\overline{P}$; thus;

$$\overline{P} = \text{constant} \ / \sum_{i=1}^{N} V_i \ / \ T_i$$

In STRIKIN-II the sum in the above equation is over three different physical regions, 1) gas plenum, 2) gap between fuel and clad, and 3) the fuel dish and porosity volume. The gas plenum is assumed to have a transient volume and temperature. Each axial fuel region has transient temperature and a constant volume. The fuel dish volume and porosity may be specified as zero via input. It can also be computed at time zero by STRIKIN-II as the difference between the total volume and sum of the gas plenum volume and the gap region volume.

The plenum volume will change as the fuel rod cladding thermally expands or contracts in the axial direction (Note that the radial thermal expansion of the clad is accounted for in the volume of the gas gap). The axial expansion is an accumulative effect along the entire fuel rod and is restricted by the maximum physical rod length. The volume is a linear function of the fractional change in length which is computed as follows:

[ ]TS

where: $Vol_p(t)$ = The transient plenum volume, in^3.
Vol_{po} = Initial plenum volume, in^3.
l_o = Initial plenum length, in.
l_{max} = Maximum plenum length, in.
ΔZ = Length of all axial rod regions, in.
$\alpha_i(T_i)$ = Axial thermal expansion of clad, in/in.
α_{io} = Initial value of axial thermal expansion of clad, in/in.

i) Fuel Pellet Dimension

The cold ratcheted, densified, and swelled fuel pellet dimension is input to the STRIKIN-II code as a function of axial position. The fuel pellet is then thermally expanded using the instantaneous fuel pellet temperature distribution with a model which accounts for radial cracks forming in the fuel below 1400°C. For regions having temperatures above 1400°C, a restrained thermal expansion formulation is used; for regions with temperatures below 1400°C a free thermal expansion formulation is used. The restrained and free formulations are as follows:

[ ]TS

Where,

r_{cold} = cold ratcheted, densified, and swelled fuel pellet radius, inches.
$\Delta r = r_{cold}/50$, r_{cold} is divided into 50 regions to locate isotherm, inches.
r = average radius of radial region, Δr, inches.
T_r = average temperature of radial region Δr, °F.
$\alpha(T)$ = coefficient of thermal expansion as a function of temp., inches/inches.
r_{1400} = radius of first radial node having a temp. less than 1400°C, inches.

3.5.2.6 Time Step Calculation

The STRIKIN-II time step is selected between user input minimum and maximum values such that the time step used by the code is in the stable solution region (See Figure 3.5-5). In selecting the time step between Δt_{min} and Δt_{max} the following conditions must be met per time step:

a) The time constant in the system cannot change by more than the fractional change specified by

the user (default value = 0.005).

b) The fluid enthalpy in an axial node cannot change by more than the fractional change specified by the user (default value = 0.05).

c) The mass flow rate in an axial node cannot change by more than the fractional change specified by the user (default value = 0.05).

d) The mass flow rate below which c) does not apply is a user input value (default value = 0.5×10^6 lb/ft^2-hr).

3.5.2.7 Heat Generation

The volumetric heat generation, q, at any point in the fuel rod is assumed to be a function of space and time

$$q = \frac{Q}{(1+f_w)V_f} q_1(I)\, q_2(J)\, q_3(t)$$

where Q is the reactor initial power level, Btu/hr
f_W is the ratio of heat generated in the coolant to heat generated in the fuel rod
V_f is the volume of fuel, ft^3
q_1 (I) is the peaking factor for the I-th radial fuel rod node
q_2 (J) is the peaking factor for the J-th radial fuel rod node
q_3 (t) is the normalized time dependent heat generation rate

The initial power level, fuel volume, fraction of heat generated in the coolant, axial peaking factors, and radial peaking factors are obtained from design specifications. A distribution of rod radial peaking factors is input to represent the radial flux depression in the fuel pellet.

3.5.2.8 Coupling of Heat Conduction Equation and Fluid Energy Equation

In STRIKIN-II the surface temperature and heat flux are determined by a heat balance at the rod surface, which results in the following two coupled equations:

$$\phi = k\,(T_N - T_S)/\Delta r$$

$$\phi = h\,(T_S - T_C)$$

where k = conductivity of the clad
h = film coefficient
Δr = outer radius – average radius of outer clad region
ϕ = heat flux
T_S = clad surface temperature
T_C = bulk coolant temperature
T_N = average temperature of outer clad region

Note that the above 2 equations are equivalent to $\phi = h\,k\,(T_N - T_C) / (k + h\,\Delta r)$ and $T_S = (k\,T_N + h\,\Delta r\,T_C) / (k + h\,\Delta r)$ which show that the heat flux and surface temperature are uniquely determined from the film coefficient, the conductivity of the clad, the average temperature of the outer

clad region, and the bulk coolant temperature. The values of the average temperature of the outer clad region and conductivity of the clad are determined by the solution of the conductivity equation, while the film coefficient and bulk coolant temperature are dependent on the flow and the solution of the energy equation. Thus, coupling of the solution equations conserves the energy transfer at the film surface and yields a stable solution technique.

3.5.3 USNRC Approval

The STRIKIN-II is used to simulate the heat conduction within reactor fuel rods and its associated surface heat transfer. The STRIKIN-II was approved in Reference 29.

3.5.4 Code Verification

To evaluate the validity and conservatism of the synthesis method described in Section 2.0 of Reference 29 with STRKIN-II code for analyzing a CEA ejection transient, this synthesis method was compared with three-dimensional space-time calculations for both full and zero power initial conditions. The three-dimensional space-time analysis was made using the HERMITE computer code. More detail verification of CE method with STRIKIN-II code for CEA ejection transient are presented in Reference 29.

3.5.5 History

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

b) The nuclear power plants used STRIKIN-II for analyzing transients and accident events are as follows:

Arkansas Nuclear One Unit 2
Waterford Unit 3
San Onofre Units 1, 2, and 3
Yonggwang Units 3, 4, 5, and 6
Shin Kori Units 1, 2, 3 and 4
Shin Ulchin Units 1 & 2
Palo Verde Units 1, 2, and 3
St. Lucie Unit 2
CESSAR-DC (System 80+)
Ulchin Units 3, 4, 5, and 6
Shin Wolsung Units 1 & 2

3.5.6 Limitations

STRIKIN-II is used to simulate the heat conduction within reactor fuel rods and its associated surface heat transfer. STRIKIN-II provides a single, or dual, closed channel model of a core flow channel to calculate the clad and fuel temperatures for an average or hot fuel rod, and the extent of the zirconium water reaction for a cylindrical geometry fuel rod. The USNRC staff concluded that STRIKIN-II was an acceptable computer program for use in licensing applications for calculating CEA ejection event in the Reference 29. KHNP concludes that STRIKIN-II is an acceptable computer program for analyzing transient and accident events for the APR1400.

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Table 3.5-1
Heat Transfer Regimes in STRIKIN-II

TS

Figure 3.5-1
STRIKIN-II Geometric Model

TS

Figure 3.5-2
STRIKIN-II Dual Channel Data Transfer

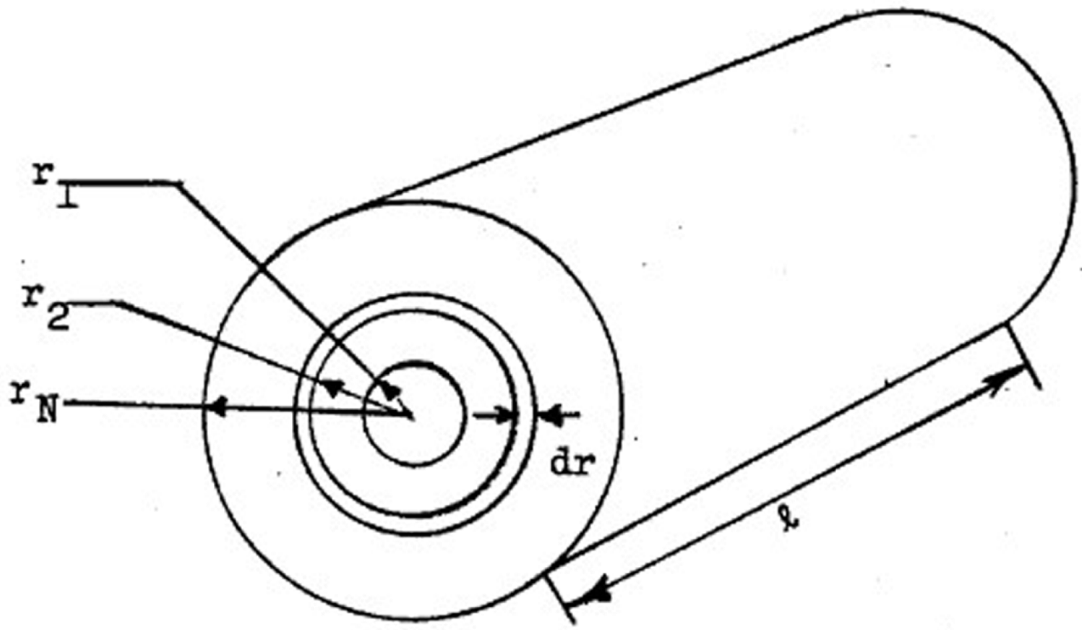

Figure 3.5-3
A Typical Cylindrical Fuel Element

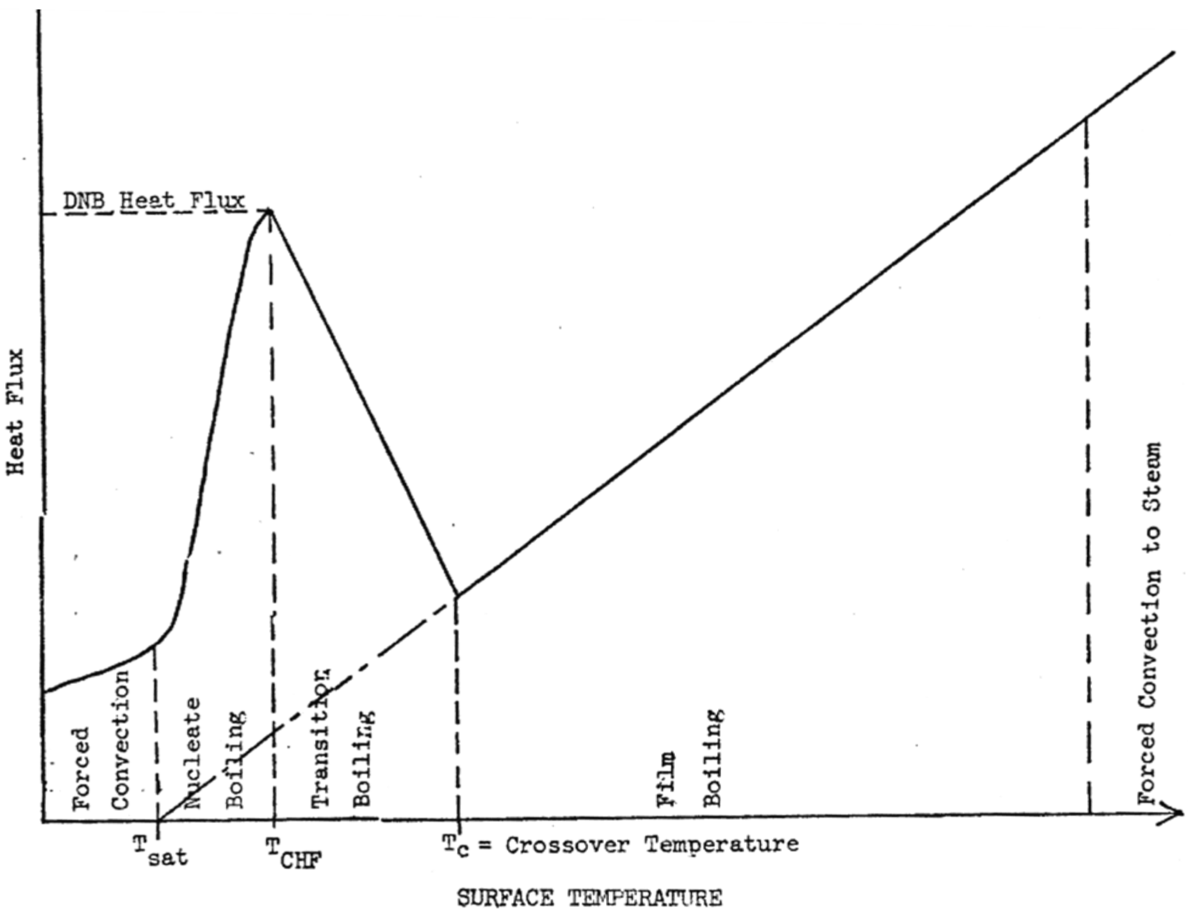

Figure 3.5-4
Variation of Heat Flux with Surface Temperature

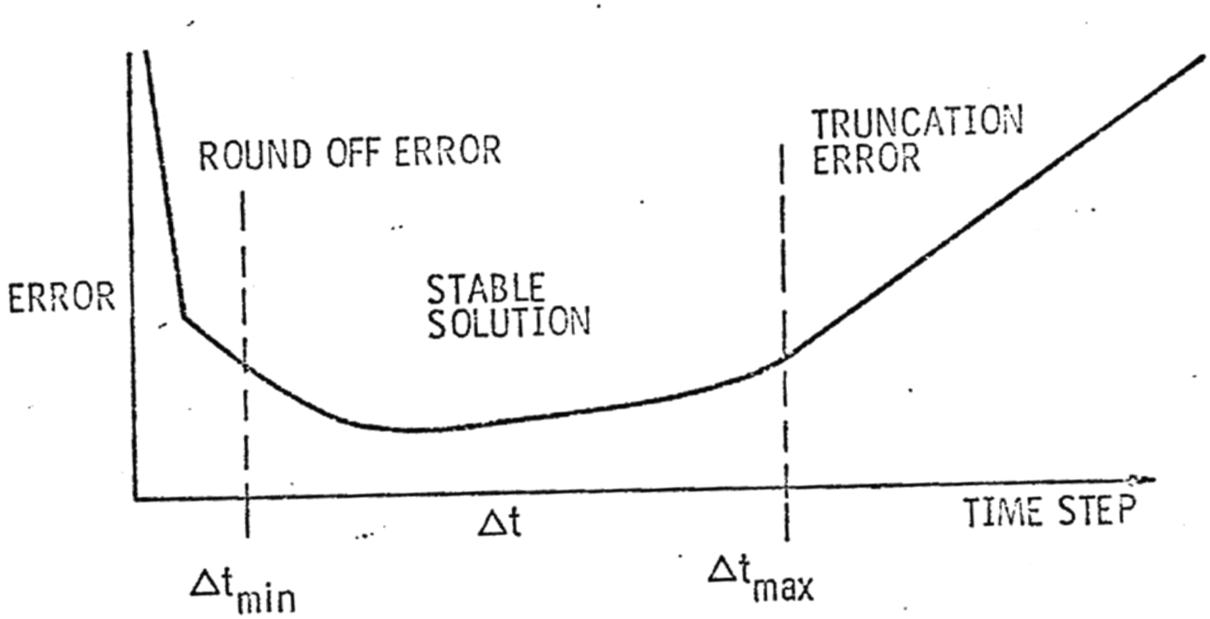

Figure 3.5-5
STRIKIN-II Time Step Selection

3.6 HISTORY OF METHOODOLOGY CHANGES

3.6.1 CESEC-III

3.6.1.1 Updated Contents for Pilot Operated Safety Relief Valve

For the APR1400, the type of RCS overpressure protection device is changed from pressurizer safety valve (PSV) to pilot operated safety relief valve (POSRV). Figure 3.6-1 shows the schematic diagram for POSRV.

Only the open setpoints of PSV are required to implement the open characteristics of PSV, however, the delay times are required to implement the open characteristics of POSRV. Therefore, the logic for the open characteristics of POSRV is modified to simulate the POSRV.

Due to the design change of POSRV, the PC version update was performed to generate the new executable file with the modified input.

3.6.1.2 Updated Contents for Centrifugal Charging Pump

The type of charging pump is changed from positive displacement charging pump (PDP) to centrifugal charging pump (CCP) to increase chemical and volume control system (CVCS) reliability. Two CCPs are utilized; one normally running and the other for backup. The pressurizer level control system should be changed due to the change of charging pump type. In addition, charging control valves, mini-flow heat exchanger and necessary instrumentation are added.

With PDP, the charging flow is determined as a number of operating charging pumps independent on the cold leg pressure. However, with CCP, the charging flow is supplied into the RCS depending on the cold leg pressure. Therefore, the input data for the charging flow vs. RCS pressure is used to simulate the CCP simulation.

Figure 3.6-2 shows the schematic diagrams for the charging system.

Due to the design change of CVCS, the PC version update was performed to generate the new executable file with the modified input.

3.6.1.3 Updated Contents for SIS Design Features (4 Train Lines and DVI)

SIS design features are changed from 2 low pressure safety injection (LPSI) and 2 high pressure safety injection (HPSI) pumps to 4 safety injection pumps (SIPs). The SIS consists of four (4) electrically independent divisions, and four (4) mechanically separated trains, each consisting of a SIP and a safety injection tank (SIT), and each taking suction its own suction source from IRWST. Figure 3.6-3 shows the schematic diagram for SIS 4-train and direct vessel injection (DVI).

For non-LOCA analysis, the safety injection flow and boron concentrations are critical parameters. Since there are no breaks on reactor inlet or outlet pipe for non-LOCA, the design change of SIS design features has no impact on non-LOCA safety whether the safety injection is injected to DVI or cold leg. Therefore, the code model changes are not required to implement the design features.

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Figure 3.6-1
Schematic Diagram for POSRV (Typical)

TS

Figure 3.6-2
Schematic Diagram for the Charging System

TS

Figure 3.6-3
Schematic Diagram for 4-Train SIS & DVI

3.6.2 TORC and CETOP

The TORC and the CETOP codes and its thermal margin model with the CE-1 CHF correlation were accepted by NRC (References 6, 23) for ANO-2 thermal margin calculations in 1981. The new CHF correlations, ABB-NV and ABB-TV developed by ABB-CE, were added in 2002~2003 by Westinghouse Electric Company. The TORC and the CETOP codes were changed due to the fuel assembly improvement (PLUS7) as follows:

1. The critical heat flux (CHF) correlation is changed. The CHF correlation is used to predict critical heat flux and the departure from nucleate boiling ratio (DNBR) from the coolant local condition of the core modeling. The CHF correlation is changed from CE-1 for C-E standard fuel and Guardian™ fuel to KCE-1 for PLUS7™ fuel with mixing vane grids. The correlation form was not changed but the coefficients were revised to predict the thermal characteristics of the improved fuel assembly.

   - <u>Statistics of KCE-1 CHF Correlation</u>

     TS

   - <u>Applicable Range of KCE-1 CHF Correlation</u>

     TS

2. The hardwired thermal diffusion coefficient (TDC) is changed for CETOP code. The TDC is represented as an inverse Pecklet number (1/Pe) in both TORC and CETOP code. For the TORC code, no corresponding modification is needed because 1/Pe is an input variable. Exact value of Pe is changed from 0.0035 for Guardian™ fuel to 0.0101 for PLUS7™ fuel with mixing vane grids.

3.6.3 COAST

In some initial conditions (specifically, in some head-flow tables used in COAST input files), COAST did not converge within specified iteration limit or diverged during the initialization step. This can be the failure of numerical process during the flow-head iteration in COAST.

To make this change of analysis method valid, initialization method used in COAST must be changed from head-flowrate iteration with constant system friction to system friction iteration with constant flowrate.

3.6.4 HERMITE

HERMITE code was initially approved by NRC in 1976 and the CE-Methodology with several improvements was approved in 1992 in the Amendment No. 61 of NPF-41 (Reference 30). HERMITE code version in 1992 was 1.5 while current version is 1.6 with added transient pressure option. There is no significant methodology change for this code after the last approval by NRC in 1992, thus this code is applicable to the APR1400 DCD preparation.

3.6.5 STRIKIN-II

3.6.5.1 Updated Contents for ZIRLO and M5 Cladding Material

Two types of cladding material are added to STRIKIN-II computer code. The properties of ZIRLO^{TM} and M5^{TM} are added to the properties of OPTIN^{TM} which are already included in STRIKIN-II. These properties used in STRIKIN-II are the temperature dependent thermal conductivity, specific heat, emissivity, Poisson's ratio and Young's modulus. STRIKIN-II has a capability of selecting one cladding material for analyzing transient and accident events for the APR1400.

3.6.5.2 Updated Contents for the CHF correlation of KCE-1

The critical heat flux (CHF) correlation is changed. The CHF correlation is used to predict critical heat flux and the departure from nucleate boiling ratio (DNBR) from the coolant local condition of the hot channel and local heat flux from the surface of the hot rod. The CHF correlation is changed from CE-1 for C-E standard fuel and Guardian^{TM} fuel to KCE-1 for PLUS7^{TM} fuel with mixing vane grids. The correlation form was not changed but the coefficients were revised to predict the thermal characteristics of PLUS7^{TM} . See Subchapter 3.6.2 "CETOP" for more details.

4. REQUIREMENTS FOR DCD, TIER 2 CHAPTER 15 NON-LOCA EVENTS

The methodology described in this report is applicable to the APR1400 plant design and modes of plant operation addressed in the non-LOCA accident analysis. In particular, the methodology described is related to the thermal-hydraulic aspects of the SRP Chapter 15 events for non-LOCA events for the APR1400 that challenge the cladding and reactor coolant system fission product barriers; the non-LOCA methodology does not include the dose consequence analysis of radiological releases, accidents that only apply to boiling water reactors (BWRs), or event that are beyond the design basis.

The accident analysis in the Design Certification and Combined License Applications will be organized consistent with the categories shown below in Table 4.0-1, as defined in the Standard Review Plan (SRP) NUREG-0800 and the most recent version of Regulatory Guide 1.206. Regulatory Guide 1.206 will serve the same purpose (format and content Guide) for new plants licensed under Part 52 as Regulatory Guide 1.70 serves for the current operating US plant Updated Final Safety Analysis Reports.

Table 4.0-1
Event Classification Categories

Category Number	Event Categorization By Effect on the Plant
15.1	Increase in Heat Removal by the Secondary System
15.2	Decrease in Heat Removal by the Secondary System
15.3	Decrease in Reactor Coolant System Flow Rate
15.4	Reactivity and Power Distribution Anomalies
15.5	Increase in Reactor Coolant Inventory
15.6	Decrease in Reactor Coolant Inventory

Due to the similarity between the APR1400 and the current generation of PWRs in the United States, KHNP has determined that no new categories of events (determined by effect on the plant) are required to bound the possible initiating events.

Each of the events categorized in Table 4.0-1 has different potential initiating events that can be further categorized according to their expected frequency of occurrence. Historically, the frequency of each event was categorized as a fault of moderate frequency (ANSI 18.2 Category II), limiting fault (Category III), or design basis fault (Category IV), and frequency-class-based acceptance criteria associated with each category applied to specific accidents. For new plants, the current SRPs no longer use the historical frequency categories by name or number, but instead, re-categorize each event as either an anticipated operational occurrences (AOOs) or postulated accidents (PAs). The following definitions of AOO and PA are derived from the SRPs:

- Anticipated operational occurrences (AOOs), as defined in Appendix A to 10CFR50, are those conditions of normal operation that are expected to occur one or more times during the life of plant. The SRP reiterates the 10CFR50 Appendix A definition of the term AOOs and adds that AOOs are also known as Condition II and Condition III events (referring to events that are categorized in Regulatory Guide 1.70 and Regulatory Guide 1.206 as incidents of moderate frequency and infrequent events). Incidents of moderate frequency and infrequent events have also been previously known as ANSI 18.2 Condition II and Condition III events, respectively.
- Postulated accidents (PAs) are unanticipated occurrence (i.e., they are postulated but are not expected to occur during the life of the plant). They are analyzed to confirm the adequacy of plant safety systems. These accidents have also been previously known as ANSI 18.2 Condition IV events or “Design Basis Accidents”.

Section 4.1 documents the acceptance criteria KHNP plans to use for AOOs and PAs based on the SRPs, modified as needed, to identify the key criteria and additional more restrictive criteria imposed by KHNP for each of the non-LOCA accidents to be provided in the DCD. The six event categories (15.1, 15.2, 15.3, 15.4, 15.5, and 15.6) in Table 4.0-1 are then expanded in Subchapters 4.2.1 through 4.4.2.6 to define all of the related initiating events, each which will be quantitatively analyzed in the APR1400 DCD, Tier 2.

4.1 Acceptance Criteria

Licensing analyses are performed to demonstrate that a plant can meet the applicable acceptance criteria for a limiting set of AOOs and PAs. The section provides the acceptance criteria used for the accident analyses of the APR1400.

The general design criteria are written such that the risk of an event, defined as the product of an event’s frequency of occurrence and its consequences, is approximately equal across the spectrum of AOOs and PAs. The first two subsections of Section 4.1 provide the general SRP acceptance criteria for the AOO and PA categorization of accidents. Additional event-specific SRP criteria are described in an appropriate event classification discussion.

4.1.1 AOO Acceptance Criteria

The followings are the generic criteria necessary to meet the requirements of GDC for AOOs:

i. Pressure in the reactor coolant (P_{RCS}) and main steam (P_{SG}) systems should be maintained below 110% of the design values in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.
ii. Fuel cladding integrity is maintained by ensuring the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit.
iii. An AOO should not generate a postulated accident without other faults occurring independently or result in a consequential loss of function of the reactor coolant system (RCS) or reactor containment barriers.

General Design Criterion (GDC) 10 within Appendix A to 10 CFR 50, establishes that specified acceptable fuel design limits (SAFDLs) should not be exceeded during any condition of normal operations, including the effects of AOOs. Further guidance for interpreting this regulation is provided in SRP 4.2.

4.1.2 PA Acceptance Criteria

A list of the basic criteria necessary to meet the requirements of GDC for postulated accidents appears below.

i. Pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits.
ii. Fuel cladding integrity is maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit. If the minimum DNBR does not meet this limit, then the fuel is assumed to have failed.
iii. The release of radioactive material shall not result in offsite doses in excess of the guidelines of 10 CFR Part 100. Any event-specific accident limits for allowable radiological releases are described in the appropriate section (i.e., for specific reactivity initiated accidents) below.
iv. The postulated accident shall not, by itself, cause a consequential loss of required functions of systems needed to cope with the fault, including those of the RCS and the reactor containment system.

For the reactivity initiated accidents (RIAs), SRP 4.2 Appendix B (Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents) provides the following additional acceptance criteria regarding core coolability (which are considered an extension of criteria iv. above):

1. Peak radial average fuel enthalpy must remain below 230 cal/g.

2. Peak fuel temperature must remain below incipient fuel melting conditions.

3. Mechanical energy generated as a result of (1) non-molten fuel-to-coolant interaction and (2) fuel rod burst must be addressed with respect to pressure boundary, reactor internals, and fuel assembly structural integrity.

4. No less of coolable geometry due to (1) fuel pellet and cladding fragmentation and dispersal and (2) fuel rod ballooning.

4.2 Acceptance Criteria for non-LOCA Events

4.2.1 Increase in Heat Removal by the Secondary System

This category covers events that lead to heat removal exceeding the heat generation in the core potentially leading to a decrease in moderator temperature resulting in an increased power level and reduced shutdown margin. The events which cause an increase in heat removal by the secondary system are:

① Decrease in Feedwater Temperature
② Increase in Feedwater Flow
③ Increased Main Steam Flow
④ Inadvertent Opening of a Steam Generator Relief or Safety Valve, and
⑤ Steam System Piping Failures Inside and Outside of Containment

The following table summarizes the initiating events considered for the APR1400, their associated event classification, the computer codes used to analyze the event for compliance with applicable codes and regulations, and a listing of the event-specific acceptance criteria.

Table 4.2-1
Events in Increase in Heat Removal by the Secondary System

Event	Class	Code	Acceptance Criteria
1. Decrease in feedwater temperature	AOO	CESEC-III CETOP	• max P_{RCS} & P_{SG}<110% design • min DNBR>95/95 DNBR limit[1)]
2. Increase in feedwater flow	AOO	CESEC-III CETOP	• max P_{RCS} & P_{SG}<110% design • min DNBR>95/95 DNBR limit[1)]
3. Increase main steam flow	AOO	CESEC-III CETOP	• max P_{RCS} & P_{SG}<110% design • min DNBR>95/95 DNBR limit[1)]
4. Inadvertent opening of a steam generator relief or safety valve	AOO	CESEC-III CETOP	• max P_{RCS} & P_{SG}<110% design • min DNBR>95/95 DNBR limit[1)]
5. Steam system piping failure inside and outside containment	PA	CESEC-III CETOP	• max P_{RCS} & P_{SG}<110% design • fuel failure: maintain coolable geometry • radiological consequences [2)]

Note: (1) Indicate the key parameter/acceptance limit of concern
(2) Pre-accident iodine spike - within 10CFR50.34 guidelines (25 rem TEDE*)
Accident initiated iodine spike – small fraction of 10CFR50.34 guidelines (2.5 rem TEDE *)

* TEDE: total effective dose equivalent

4.2.2 Decrease in Heat Removal by the Secondary System

This category covers events that lead to unplanned decreases in heat removal by the secondary system. The events which cause a decrease in heat removal by the secondary system are:

① Loss of External Load
② Turbine Trip
③ Loss of Condenser Vacuum
④ Main Steam Isolation Valve Closure
⑤ Loss of Non-emergency AC Power to the Station Auxiliaries
⑥ Loss of Normal Feedwater Flow, and
⑦ Feedwater System Pipe Breaks

Table 4.2-2 summarizes the initiating events considered for the APR1400, their associated event classification, the computer codes used to analyze the event for compliance with applicable codes and regulations, and a listing of the event-specific acceptance criteria.

Table 4.2-2
Events in Decrease in Heat Removal by the Secondary System

Event	Class	Code	Acceptance Criteria
1. Loss of external load	AOO	CESEC-III CETOP	• max P_{RCS} & P_{SG}<110% design • min DNBR>95/95 DNBR limit[1)]
2. Turbine trip	AOO	CESEC-III CETOP	• max P_{RCS} & P_{SG}<110% design • min DNBR>95/95 DNBR limit[1)]
3. Loss of condenser vacuum	AOO	CESEC-III CETOP	• max P_{RCS} & P_{SG}<110% design • min DNBR>95/95 DNBR limit[1)]
4. Main steam isolation valve closure	AOO	CESEC-III CETOP	• max P_{RCS} & P_{SG}<110% design • min DNBR>95/95 DNBR limit[1)]
5. Loss of non-emergency AC power to the station auxiliaries	AOO	CESEC-III CETOP	• max P_{RCS} & P_{SG}<110% design • min DNBR>95/95 DNBR limit[1)]
6. Loss of normal feedwater flow	AOO	CESEC-III CETOP	• max P_{RCS} & P_{SG}<110% design • min DNBR>95/95 DNBR limit[1)]
7. Feedwater system pipe breaks	PA	CESEC-III CETOP	• max P_{RCS} & P_{SG} [2)] • fuel failure: maintain coolable geometry • radiological consequences [3)]

Note: (1) Indicate the key parameter/acceptance limit of concern
(2) Maximum RCS Pressure (max P_{RCS}):
Low probability events - Less than 110% of design
Very low probability events - Less than 120% of design
Maximum Steam Generator Pressure (max P_{SG}):
Low probability events - Less than 110% of design
Very low probability events - Less than 120% of design
(3) Small fraction of 10CFR50.34 guidelines (2.5 rem TEDE)

4.2.3 Decrease in Reactor Coolant Flow Rate

This category covers events that lead to unplanned decreases in reactor coolant flow rate. The events which cause a decrease in reactor coolant flow rate are:

① Total Loss of Reactor Coolant Flow
② Single Reactor Coolant Pump Rotor Seizure with Loss of Offsite Power
③ Reactor Coolant Pump Shaft Break with Loss of Offsite Power

The following table summarizes the initiating events considered for the APR1400, their associated event classification, the computer codes used to analyze the event for compliance with applicable codes and regulations, and a listing of the event-specific acceptance criteria.

Table 4.2-3
Events in Decrease in Reactor Coolant Flow Rate

Event	Class	Code	Acceptance Criteria
1. Total Loss of Reactor Coolant Flow	AOO	CESEC-III CETOP HERMITE	• max P_{RCS} & P_{SG}<110% design • min DNBR>95/95 DNBR limit[1)]
2. Single Reactor Coolant Pump Rotor Seizure with Loss of Offsite Power	PA	CESEC-III TORC HERMITE	• max P_{RCS} & P_{SG}<110% design • fuel failure: maintain coolable geometry • radiological consequences [2)]
3. Reactor Coolant Pump Shaft Break with Loss of Offsite Power	PA	CESEC-III TORC HERMITE	• max P_{RCS} & P_{SG}<110% design • fuel failure: maintain coolable geometry • radiological consequences [2)]

Note: (1) Indicate the key parameter/acceptance limit of concern
(2) Small fraction of 10CFR50.34 guidelines (2.5 rem TEDE)

4.2.4 Reactivity and Power Distribution Anomalies

This category covers events that lead to anomalies in reactivity and power distribution. The events which cause reactivity or power distribution anomalies are:

① Uncontrolled CEA withdrawal from a subcritical or low power condition
② Uncontrolled CEA withdrawal at power
③ Single CEA drop
④ Single CEA withdrawal
⑤ Startup of an inactive reactor coolant pump
⑥ Inadvertent deboration
⑦ Inadvertent loading of a fuel assembly into the improper position
⑧ CEA ejection

The following table summarizes the initiating events considered for the APR1400, their associated event classification, the computer codes used to analyze the event for compliance with applicable codes and regulations, and a listing of the event-specific acceptance criteria.

Table 4.2-4
Events in Reactivity and Power Distribution Anomalies

Event	Class	Code	Acceptance Criteria
1. Uncontrolled CEA withdrawal from a subcritical or low power condition	AOO	CESEC-III CETOP	• max P_{RCS} & P_{SG}<110% design • min DNBR>95/95 DNBR limit[1)]
2. Uncontrolled CEA withdrawal at power	AOO	CESEC-III CETOP	• max P_{RCS} & P_{SG}<110% design • min DNBR>95/95 DNBR limit[1)]
3. Single CEA drop	AOO	CESEC-III CETOP	• max P_{RCS} & P_{SG}<110% design • min DNBR>95/95 DNBR limit[1)]
4. Single CEA withdrawal	PA	CESEC-III CETOP	• max P_{RCS} & P_{SG}<110% design • fuel failure: maintain coolable geometry • radiological consequences [2)]
5. Startup of an inactive reactor coolant pump	AOO	N/A	• max P_{RCS} & P_{SG}<110% design • min DNBR>95/95 DNBR limit[1)]
6. Inadvertent deboration	AOO	N/A	• max P_{RCS} & P_{SG}<110% design • min DNBR>95/95 DNBR limit[1)]
7. Inadvertent loading of a fuel assembly into the improper position	AOO	N/A	• max P_{RCS} & P_{SG}<110% design • fuel failure: maintain coolable geometry • radiological consequences [2)]
8. CEA ejection [4)]	PA	CESEC-III STRIKIN-II	• max P_{RCS} & P_{SG}<110% design • fuel failure: maintain coolable geometry • radiological consequences [3)]

Note: (1) Indicate the key parameter/acceptance limit of concern
(2) Small fraction of 10CFR50.34 guidelines (2.5 rem TEDE)
(3) Well within 10CFR50.34 guidelines (6.25 rem TEDE)
(4) The interim acceptance criteria presented in SRP 4.2 Appendix B are discussed in Reference 32.

4.2.5 Increase in Reactor Coolant Inventory

This category covers events that lead to fuel damage or over-pressurization of the RCS due to an expected increase in RCS inventory. The events which cause an increase in reactor coolant are:

① Inadvertent Operation of ECCS
② CVCS Malfunction-PLCS Malfunction

The following table summarizes the initiating events considered for the APR1400, their associated event classification, the computer codes used to analyze the event for compliance with applicable codes and regulations, and a listing of the event-specific acceptance criteria.

Table 4.2-5
Events in Increase in Reactor Coolant Inventory

Event	Class	Code	Acceptance Criteria
1. Inadvertent operation of ECCS	AOO	N/A	N/A 2)
2. CVCS malfunction-PLCS malfunction	AOO	CESEC-III CETOP	• max P_{RCS} & P_{SG}<110% design • min DNBR>95/95 DNBR limit1)

Note: (1) Indicate the key parameter/acceptance limit of concern
(2) Safety injection pump shutoff head is below normal operating pressure. Above that pressure there will be no injection water into the system. Below SI pump shutoff head pressure when the shutdown cooling system is isolated the SI flow will increase RCS inventory and pressure until the pressure reaches the pump shutoff head pressure. During shutdown cooling system operation, the increase in RCS inventory and pressure will be mitigated by the shutdown cooling system relief valves.

4.2.6 Decrease in Reactor Coolant Inventory

This category covers events that lead to accidental depressurization of the RCS. The events which cause a decrease in reactor coolant are:

① Inadvertent Opening of a Pressurizer Pilot Operated Safety Relief Valve (POSRV)
② Double Ended Break of a Letdown Line Outside Containment
③ Steam Generator Tube Rupture

Table 4.2.6-1 summarizes the initiating events considered for the APR1400, their associated event classification, the computer codes used to analyze the event for compliance with applicable codes and regulations, and a listing of the event-specific acceptance criteria.

Table 4.2-6
Events in Decrease in Reactor Coolant Inventory

Event	Class	Code	Acceptance Criteria
1. Inadvertent opening of a pressurizer POSRV	PA	N/A 3)	• AOO 2) • Accidents 2)
2. Double-ended break of a letdown line outside containment	AOO	CESEC-III CETOP	• max P_{RCS} & P_{SG}<110% design • min DNBR>95/95 DNBR limit 1) • radiological consequences 4)
3. Steam generator tube rupture	PA	CESEC-III CETOP	• max P_{RCS} & P_{SG}<110% design • fuel failure: maintain coolable geometry • radiological consequences 5)

Note: (1) Indicate the key parameter/acceptance limit of concern

(2) <u>For AOO</u>:

- max P_{RCS} & P_{SG}<110% design
- min DNBR>95/95 DNBR limit

<u>For Accidents</u>:

Relative to ECCS performance (as specified in 10CFR50.46):

1) The peak cladding temperature shall not exceed 1204 ℃ (2200 ℉).
2) The maximum cladding oxidation shall not exceed 0.17 times the total cladding thickness.
3) The maximum hydrogen generation shall not exceed 0.01 times the hypothetical amount if all the fuel cladding were to react.
4) The core shall be maintained in coolable geometry.
5) Long term decay heat removal shall be demonstrated.

Relative to Radiological Consequences: Small fraction of 10CFR50.34 guidelines

(3) The evaluation of this event is performed in section presenting SBLOCA.

(4) Small fraction of 10CFR50.34 guidelines (2.5 rem TEDE)

(5) Pre-accident iodine spike - within 10CFR50.34 guidelines (25 rem TEDE)
Accident initiated iodine spike – small fraction of 10CFR50.34 guidelines (2.5 rem TEDE)

4.3 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. As stated in 10CFR50.34 (h), conformance with the SRP is not required provided the applicant demonstrate that the alternative complies with the regulations of the Commission. Appendix A shows detail evaluations for conformance with the SRP.

5. Results

The object of this report is to provide methods of analysis, details, and acceptance criteria of the application of KHNP non-LOCA accident analysis to the APR1400 such that questions or issues can be identified as early as possible in the licensing process. As discussed in Chapter 2, the CESEC-III, TORC/CETOP, COAST, STRIKIN-II, HRISE, and HERMITE computer codes are the principle computer codes that are used by KHNP for the APR1400 non-LOCA analyses. Depending on the specific nature and computational capabilities needed for specific accidents, these programs are either used alone or in combination with another. Events utilizing computer codes for the non-LOCA accident analysis fall into one of the following categories based on the combination of codes used:

- Analyzed using CESEC-III and CETOP in sequence
- Analyzed using CESEC-III and HRISE in sequence
- Analyzed using HERMITE and TORC/CETOP in sequence
- Analyzed using STRIKIN-II

The first category that uses CESEC-III in combination with the CETOP includes most of the non-LOCA transients that challenge the design limits for the RCS and main steam system pressure limits, as well as loop-symmetric accidents at full-flow conditions that fall within the capabilities of the simplified CETOP DNBR model. These accidents do not require detailed calculation of localized fuel parameters and do not require spatially dependent transient calculations for accident-specific power levels or power distributions.

The second category that uses CESEC-III in combination with the HRISE is used for accident that challenges the DNB design limits such as an RTP during SLB. The loop-dependent and core total flow, core inlet conditions, pressure and power are calculated using the CESEC-III, and then the HRISE code is used to determine the few-group space- and time-dependent neutron diffusion equation in order to consider integral effect of space and time in transient state. It could be used to calculate the feedback effects of fuel temperature, coolant temperature, coolant density, xenon distributions and control rod motion.

The third category that uses HERMITE in combination with TORC/CETOP is used for accidents that challenge the DNB design limits under reduced flow conditions such as the partial loss of flow, complete loss of flow, locked RCP rotor, or RCP sheared shaft conditions. The loop-dependent and core total flow, core inlet temperature, RCS pressure are input to the HERMITE code to determine the state point for TORC/CETOP code which is used to calculate the minimum DNBR.

The fourth category that uses STRIKIN-II is used for the CEA ejection accident that challenges the DNB design limit and the enthalpy design limit for the reactivity induced accident. STRIKIN-II code is used to determine the hot channel or hot spot fuel response including minimum DNBR, fuel temperatures, and cladding temperature.

In summary, this report demonstrates the wide spectrum of key analytical methods (combinations of codes) and specialized models used by KHNP in the non-LOCA analysis for the APR1400. These events represent the SRP accident categories such as AOOs and PAs.

6. CONCLUSIONS

The APR1400 is an advanced PWR design that is functionally similar to existing plants and fuel designs from the perspective of non-LOCA accident analysis. The advanced features of the APR1400 have not created accidents of a different type that are not covered by Chapter 15 of the existing NRC Standard Review Plan, and event classification and acceptance criteria to be used in the design certification license application have been defined based on existing regulations and regulatory guidance. KHNP uses codes and methodologies for non-LOCA analyses of the APR1400 that 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 are:

- CESEC-III Simulate the NSSS
- TORC/CETOP Determine core thermal margin and minimum DNBR
- COAST Simulate RCP coastdown
- HRISE Calculate Minimum DNBR in case of return-to-power (RTP)
- STRIKIN-II Calculate the cladding and fuel temperature for an average or hot fuel rod
- HERMITE Determine short term response of reactor core

On the basis of the information in this report, it is concluded that the existing codes and methodologies are appropriate for the APR1400 analyses. Also, it is concluded that the information provided in this report supports its purpose to provide key technical information related to the computer codes, key methods and models and their applicability, and event-specific acceptance criteria to the NRC to facilitate an efficient and timely review of the design certification application.

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APPENDIX A

CONFORMANCE EVALUATIONS WITH SRP

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