ATWS Evaluation

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Non-Proprietary

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ABSTRACT

This document provides justification for the applicability of CENPD-158 to the Advanced Power Reactor 1400 (APR1400). APR1400 has various design features to mitigate the consequences of anticipated transients without scram (ATWS) events. The components or systems mitigating ATWS events are auxiliary feedwater, pilot-operated safety relief valves (POSRVs), main steam safety valves (MSSVs), steam bypass control system (SBCS) and turbine trip. Licensing requirements and regulations for ATWS are the integrities of the fuel and cladding, the reactor coolant pressure boundary (RCPB), and the containment. However, in this report, the ATWS events are evaluated only in view of reactor coolant system (RCS) over-pressurization.

The ATWS evaluation consists of two phases. First, qualitative evaluation has been performed to select the candidate ATWS events that are expected to have maximum RCS peak pressure among the anticipated operational occurrences (AOOs). After that, quantitative analysis has been conducted to simulate the thermal-hydraulic behaviors of the selected ATWS events in order to find the most limiting ATWS event in terms of RCS integrity. The maximum RCS peak pressures occur at the low fuel burnup. The limiting ATWS event has been found to be a loss of normal feedwater (LONF) without turbine trip event, which is the same result as was found in CENPD-158. Therefore, the applicability of CENPD-158 to the APR1400 for ATWS events has been justified.

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ACRONYMS AND ABBREVIATIONS

AC	alternating current
ADV	atmospheric dump valve
AFAS	auxiliary feedwater actuation signal
AFW	auxiliary feedwater
AFWS	auxiliary feedwater system
AFWST	auxiliary feedwater supply sank
AOO	anticipated operational occurrence
APR1400	Advanced Power Reactor 1400
ATWS	anticipated transients without scram
BOC	beginning of cycle
CCF	common cause failure
CE	Combustion Engineering
CEA	control element assembly
CEDM	control element drive mechanism
COL	combined license
CPC	core protection calculator
DCD	design control document
DPS	diverse protection system
EOC	end of cycle
ESFAS	engineered safety features actuation system
FTC	fuel temperature coefficient
FWCS	feedwater control system
GDC	general design criteria
HPP	high pressurizer pressure
I&C	instrumentation and controls
IRWST	in-containment refueling water storage tank
LOCV	loss of condenser vacuum
LONF	loss of normal feedwater
LOOP	loss of offsite power
LSGL	low steam generator level
MSIS	main steam isolation system
MSIV	main steam isolation valve
MSSV	main steam safety valve
MTC	moderate temperature coefficient

NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
PLCS	pressurizer level control system
POSRV	pilot operated safety relief valve
PPCS	pressurizer pressure control system
PRA	probabilistic risk analysis
PSA	probabilistic safety analysis
PWR	pressurized water reactor
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RG	regulatory guide
RPCS	reactor power cutback system
RRS	reactor regulating system
RTS	reactor trip system
SI	safety injection
SIAS	safety injection actuation signal
SAFDL	specified acceptable fuel design limit
SBCS	steam bypass control system
TBV	turbine bypass valve
TMI	Three Mile Island
UET	unfavorable exposure time
USI	unresolved safety issue

1 INTRODUCTION

An anticipated transient without scram (ATWS) is defined as an anticipated operational occurrence (AOO) followed by the failure of the reactor trip portion of the protection system in 10 CFR 50.62 (Reference 1), which are regulatory requirements regarding risk from ATWS events for light-water-cooled nuclear power plants. AOO is also defined in Appendix A of 10 CFR 50 as a condition of normal operation that is expected to occur one or more times during the life of the nuclear power unit.

The reliability of the reactor trip portion of the protection system has to be implemented according to the general design criteria (GDC) 20 through 29 (Reference 2), which are GDCs for protection and reactivity control systems. GDC 21 requires that the redundancy and independence designed into the protection system shall be sufficient to assure that no single failure results in loss of protection function. According to GDC 29, the protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs, which requires enhanced protection against ATWS. Since the protection system of the APR1400 is designed to satisfy the above regulatory requirements including single failure criterion, multiple failures or a common cause failure (CCF) must occur to cause the assumed failure of the reactor trip portion of the protection system during AOO events. The occurrence frequency of an AOO, in coincidence with multiple failures or a common cause failure of the reactor trip, is much lower than the occurrence frequency of any of the other events that are evaluated in APR1400 design control document (DCD) Chapter 15. Therefore, the ATWS event has historically been considered a beyond-design-basis event rather than either an AOO or a postulated accident.

The ATWS rule promulgated in 10 CFR 50.62 specifies the requirements for the pressurized water reactors (PWRs) of various vendor designs to reduce the probability of unacceptable consequences resulting from ATWS events. The U.S. Nuclear Regulatory Commission (NRC) based these requirements on the results of staff studies of operating PWRs under licensed conditions at the time of the ATWS rulemaking. Based on the study, NRC required that the PWRs manufactured by Combustion Engineering (CE) must have a diverse scram system from the sensor output to interruption of power to the control rods and this scram system must be designed to perform its function in a reliable manner and be independent from the existing reactor trip system.

Regulatory guide (RG) 1.206 (Reference 3) specifies that the physics and thermal-hydraulic phenomena of the plant response to ATWS events should be evaluated even though ATWS events are beyond-design basis events and are subject to probabilistic risk analysis (PRA). In this report, the results of evaluation of ATWS events regarding the reactor coolant system (RCS) integrity are described. The evaluation consists of two phases. The first phase is a qualitative evaluation to identify limiting ATWS events regarding RCS integrity; this phase requires quantitative analysis using computer code. The second phase is a thermal-hydraulic analysis for these limiting ATWS events.

1.1 Background

Safety issues associated with an ATWS have been evaluated since the early 1970s. During NRC's evaluations of various vendor models and analyses performed by vendors and NRC, the agency formally identified ATWS safety issue as an unresolved safety issue (USI) A-9 and presented the NRC staff's studies and findings regarding this issue in NUREG-0460 (Reference 4). ATWS analyses performed by CE have been depicted in references 5 and 6 used in ATWS rulemaking for the CE-fleet plants. Based on the extensive thermal-hydraulic studies summarized in reference 6, the complete loss of normal feedwater (LONF) without turbine trip turned out to be the most limiting ATWS event regarding RCS integrity. In 1986, the NRC resolved USI A-9 through the promulgation of 10 CFR 50.62, the ATWS rule. This rule required that PWRs have equipment from sensor output to a final actuation device, that is diverse from the existing safety-grade reactor trip system, and that will automatically initiate the auxiliary feedwater system and initiate turbine trip under conditions indicative of an ATWS. In addition to that, CE-fleet plants have been required to have a diverse scram system. According to these requirements,

APR1400, which has its origin in the CE fleet plant, has been equipped with the diverse protection system (DPS) described in detail in subsection 7.8.1 of APR1400 DCD.

Even though the APR1400 has many enhanced design features from System 80+, which is a CE-fleet design, the evaluation results in reference 6 were determined to be applicable to the APR1400 because the system configuration and design are basically similar between the two plants. In this report, an extensive study including quantitative analysis using computer code is described in order to verify the applicability of the analyses results in reference 6 regarding RCS integrity.

1.2 Description of ATWS

One of the major characteristics of ATWS events, which cause excursions in plant conditions, is a mismatch of power produced in the reactor core and power removed by the secondary system. For any AOO requiring reactor trip, failure of the reactor scram makes this mismatch larger than that of the AOO itself. Table 1-1 lists the AOOs described in DCD Chapter 15.

Table 1-1 AOO Events in DCD Chapter 15

DCD Section No.	AOO Event
15.1.1	Decrease in Feedwater Temperature
15.1.2	Increase in Feedwater Flow
15.1.3	Increased Main Steam Flow
15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve
15.2.1	Loss of External Load
15.2.2	Turbine Trip
15.2.3	Loss of Condenser Vacuum
15.2.4	Main Steam Isolation Valve Closure
15.2.6	Loss of Non-emergency AC Power to the Station Auxiliaries
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15.3.1	Total Loss of Reactor Coolant Flow
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15.5.1	Inadvertent Operation of ECCS
15.5.2	Chemical and Volume Control System Malfunction
15.6.2	Double Ended Break of a Letdown Line Outside Containment

Unexpected reactor power increases can be caused by positive reactivity insertion due to withdrawal of control element assembly (CEA) or increased steam flow in the secondary system. Unexpected decreases in RCS heat removal can come from various failures of systems and components that result in abrupt termination or reduction in steam flow, feedwater flow, or reactor coolant flow. Once any ATWS event resulting in excessive reactor power production beyond the heat removal capacity of the RCS occurs, the primary system pressure increases because of reactor coolant expansion and consequent in-surge to the pressurizer. One of the major objectives of this ATWS analysis is to evaluate the RCS pressure increase in terms of reactor coolant pressure boundary (RCPB) integrity.

Another consequence of the mismatch between reactor power and RCS heat removal is the increase of stored energy within the fuel and increased potential for fuel cladding degradation. Increased stored energy, and associated increased fuel temperature, can occur directly because of the inability of the fuel to rapidly conduct energy to its surface or the degradation of fuel surface heat transfer caused by decreases in the reactor coolant flow.

1.3 Design Features Used to Mitigate Consequences of ATWS

APR1400 has various design features to mitigate the consequences of ATWS events. These systems are credible in ATWS events because they are not influenced by ATWS events. In this subsection, typical design features used to mitigate ATWS events are described.

1.3.1 Auxiliary Feedwater

The auxiliary feedwater system (AFWS) provides an independent safety-related means of supplying auxiliary feedwater to the steam generators to mitigate ATWS events. The AFWS consists of two 100 percent capacity motor-driven pumps, two 100 percent capacity turbine-driven pumps, two 100 percent auxiliary feedwater storage tanks (AFWSTs), valves, two flow-limiting venturies, and instrumentation. One motor-driven pump and one turbine-driven pump are configured into one mechanical division.

An AFWS reliability analysis is performed in accordance with Three Mile Island (TMI) Action Item II.E.1.1 of NUREG-0737. The AFWS design meets the requirements in 10 CFR 50.62(c). The AFWS can be either manually actuated or automatically actuated by an auxiliary feedwater actuation signal (AFAS) from the engineered safety features actuation system (ESFAS) described in DCD subsection 7.3 or the DPS described in DCD subsection 7.8.1.1.

1.3.2 Pilot-Operated Safety Relief Valve

Four pilot-operated safety relief valves (POSRVs) are mounted on the top of the pressurizer. The relieving capacity of the POSRVs is designed to provide sufficient overpressure protection function for the postulated transients presented in DCD Chapter 15. For ATWS events resulting in rapid RCS pressurization due to loss of heat balance between primary and secondary side, the operation of POSRVs plays a critical role in mitigating event consequences under the condition of loss of shutdown rod insertion.

1.3.3 Main Steam Safety Valve

Five spring-loaded main steam safety valves (MSSVs) are provided for each individual main steam line. Thus, a total of twenty MSSVs are installed on the four main steam lines from the two steam generators. Even though the primary purpose of the MSSVs is to provide overpressure protection for the secondary system, the MSSVs also provide a mitigating function against over-pressurizing of the RCPB for ATWS events. Particularly, when the steam bypass control system (SBCS) is not available due to loss of condenser, the MSSVs provide a heat sink for the removal of energy from RCS. Loss of condenser

vacuum without scram is an example of such a case.

1.3.4 Steam Bypass Control System

The turbine bypass control system consists of the turbine bypass valves and the SBCS. This system is designed mainly to increase plant availability by making full utilization of turbine bypass capacity to remove excess nuclear steam supply system (NSSS) thermal energy following events causing power mismatch between primary and secondary systems, such as turbine load rejection and turbine trip transients. If an ATWS event in which the condenser is available occurs, SBCS can provide RCS heat removal capability before the MSSVs open.

1.3.5 Turbine Trip

Turbine trip signal is generated whenever any reactor protection system (RPS) initiation signal is generated. Turbine trip is assumed not to occur followed any reactor trip signal in this ATWS evaluation because one of the reasons for the failure to trip the reactor after AOO is the failure of the reactor trip signal generation mechanism. Turbine trip is assumed to occur only in the AOOs when the turbine trip is a direct consequence of the transient.

1.4 Scope of Evaluation

CENPD-158 (Reference 6) is the ATWS evaluation report for CE-fleet plants; it was issued in 1976. According to the results of CENPD-158, the complete loss of main feedwater without turbine trip turned out to be the most limiting ATWS scenario in terms of RCS integrity. Even though the APR1400 has many enhanced design features from System 80+, which is a CE-fleet design, the evaluation results in CENPD-158 have been found to be applicable to the APR1400 because the system configuration and design are basically similar between the two plants.

This report provides the justification for the applicability of CENPD-158 to the APR1400. Quantitative analysis using computer code accompanied by qualitative evaluation has been performed for the justification.

2 LICENSING REQUIREMENTS AND REGULATIONS

The requirements for the protection system are described in GDC 20. GDC 20 states that the protection system functions shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded as a result of AOO and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

To reduce the risk from ATWS, 10 CFR 50.62 known as the ATWS rule requires licensees to install prescribed design features and to demonstrate their adequacy. 10 CFR 50.62 states that (1) each PWR must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system (RTS), to automatically initiate the auxiliary (or emergency) feedwater (AFW) system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the existing reactor trip system. (2) Each PWR manufactured by CE or by Babcock and Wilcox must have a diverse scram system from the sensor output to interruption of power to the control rods. This scram system must be designed to perform its function in a reliable manner and be independent from the existing reactor trip system (from sensor output to interruption of power to the control rods). RG 1.206 regulates that the combined license (COL) applicant should provide analyses to demonstrate conformance of instrumentation and controls (I&C) system with the requirements of 10 CFR 50.62. Thus, this report assumes that ATWS occurs with only the mechanical failure of scram rods because the reactor trip signal will be generated by DPS in spite of RPS failure.

10 CFR 50.46 (Reference 7) requires the integrity of cladding during ATWS. Under a situation in which the scram rods are not inserted, the cladding temperature may rise due to uncontrolled core power. Thus, 10 CFR 50.46 regulates the peak cladding temperature, maximum cladding oxidation, and coolable geometry needed to sustain the integrity of the cladding during an ATWS.

GDCs 12, 14, 16, 35, 38, and 50 are also related to ATWS. These are classified into three categories such as the integrity for fuel and cladding, RCPB, and containment integrity;

(1) Fuel and Cladding Integrity

Uncontrolled reactor without scram may cause a rapid core power rise that will threaten the integrity of the fuel and cladding during ATWS.

- GDC 12: Suppression of reactor power oscillations

  This regulates the reactor power oscillations which can result in conditions exceeding SAFDL.

- GDC 35: Emergency core cooling

  This requires that the licensee shall provide a system to provide abundant emergency core cooling. Fuel and cladding damage shall be prevented and cladding metal-water reaction shall be limited to negligible amounts.

(2) Reactor Coolant Pressure Boundary

ATWS may cause the primary system pressure to rise sharply.

- GDC 14: Reactor coolant pressure boundary

  This requires an extremely low probability of abnormal leakage, of rapidly propagating

failure, and of gross rupture for the primary system pressure boundary.

- SRP 15.8, NUREG-0800

  SRP is not a substitute for the NRC's regulations and compliance with it is not required. For PWRs, RCS pressure shall not exceed ASME Service Level C Limits (approximately 226 kg/cm^2A or 3200 psig) (Reference 8). Thus, this criteria has been just used to evaluate unfavorable exposure times (UETs) in this evaluation.

(3) Containment Integrity

Containment integrity is required to prevent the uncontrolled release of radioactivity to the environment. These GDCs are not directly related to ATWS, but are general regulations for safety.

- GDC 16: Containment design

  Licensee shall provide to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions required.

- GDC 38: Containment heat removal

  Licensee shall provide a system to remove heat from the reactor containment.

- GDC 50: Containment design basis

  This relates to ensuring that the containment does not exceed the design leakage rate when subjected to the calculated pressure and temperature conditions resulting from any accident that deposits reactor coolant in the containment

3 EVALUATION APPROACH

It will be shown that the results of evaluation on the ATWS events regarding RCS integrity. The evaluations consist of two phases. The first phase is a qualitative evaluation to identify limiting ATWS events regarding RCS integrity that require quantitative analysis using computer code. The second phase is a thermal-hydraulic analysis for these limiting ATWS events.

3.1 Qualitative Evaluation

Six anticipated transients are considered from the AOOs listed in Table 1-1 in order to evaluate ATWS effects regarding RCS integrity. These are representative events to evaluate the APR1400 NSSS design to cope with ATWS events with respect to RCS over-pressurization.

- Loss of external load
- Turbine trip
- Loss of condenser vacuum
- Main steam isolation valve closure
- Loss of nonemergency alternating current (AC) power to the station auxiliaries
- Loss of normal feedwater flow

The ATWS events listed above result in decrease in heat removal from the RCS. This leads to core power decrease as a result of heat up of the reactor coolant and consequent negative reactivity feedback. In each case, the expansion of heated reactor coolant produces an RCS pressure increase. Eventually, the pressurizer is filled with liquid which is then discharged through the primary system safety valves and the RCS pressure reaches a peak value.

In this report, to evaluate the capability of the APR1400 design against ATWS regarding RCS integrity, the following four events, but not the loss of external load event or the main steam isolation valve closure event, are quantitatively analyzed to assess their impact on the RCS overpressure protection.

- Turbine trip
- Loss of condenser vacuum
- Loss of nonemergency AC power to the station auxiliaries
- Loss of normal feedwater flow (w/ and w/o turbine trip)

TS

3.2 Quantitative Analysis

3.2.1 Assumptions and Initial Conditions

The ATWS events have been analyzed with best estimate analysis methods. The following best estimate assumptions and initial conditions are applied to the transient analyses.

a. It is assumed that an ATWS occurs due to only the mechanical failure of the reactor scram function. Even though the reactor trip signal is generated by RPS or DPS without failure, the scram rods are not to be inserted.

b. Non-safety grade NSSS control systems such as the pressurizer pressure control system (PPCS), the pressurizer level control system (PLCS), and SBCS are assumed to be in automatic mode of operation. For loss of condenser vacuum (LOCV) and loss of offsite power (LOOP) events, SBCS is assumed to be unavailable due to the loss of steam dumping capability. The reactor regulating system (RRS) and the reactor power cutback system (RPCS), which are the control systems related with the power control, are assumed to be unavailable in this analysis due to the mechanical failure of the reactor scram function.

c. Engineered safety features are assumed to actuate as designed.

d. For conservatism, the fuel cycle is assumed to be the first cycle. Moderator temperature coefficients (MTCs), which are the most influential factor in ATWS regarding RCS integrity, are the most adverse for the first cycle, not for the equilibrium cycle. The MTCs and fuel temperature coefficients (FTCs) used in this analysis are provided in Table 3-1.

e. No credit is taken for manual action by the operators during a period of 30 minutes.

f. AOOs are assumed to occur at nominal operating condition (Refer to Table 3-2).

g. Nominal design data such as POSRV capacity and ESFAS setpoints are applied to this analysis.

3.2.2 Analysis Tool

RELAP5/MOD3 computer program is used in the quantitative analysis for ATWS events. This program calculates NSSS thermal-hydraulic responses to the initiating events for a wide range of operating conditions. The RELAP5/MOD3 has been developed for best-estimate transient simulation of light water reactor coolant systems during postulated accidents. The code models the coupled behavior of the reactor coolant system and the core for loss-of-coolant accidents and operational transients, such as ATWS, LOOP, loss of feedwater, and loss of flow. A generic modeling approach is used that permits the simulation of variety of thermal hydraulic systems. Control system and secondary system components are included to permit modeling of plant controls, turbines, condensers, and secondary feedwater systems.

Table 3-1 Best Estimate MTC and FTC vs. Fuel Burnup (First Cycle)

Burnup (MWD/MTU)	MTC	FTC
		TS
0		
501		
1003		
2011		
3016		
4019		
5020		
6020		
7018		
8016		
9013		
10010		
12002		
13992		
14987		
15981		
17571		

Table 3-2 Initial Conditions for ATWS Evaluation

Parameter	Value
Reactor Power, MWt	3,983
Pressurizer Pressure, kg/cm^2A (psia)	158.19 (2,250)
Core Inlet Temperature, ^oC (^oF)	290.56 (555)
RCS Mass Flow Rate, 10^6 kg/hr (lbm/hr)	75.57 (166.6)
Pressurizer Volume, m^3 (ft^3)	33.98 (1,200)
Steam Generator Mass Inventory, kg/SG (lbm/SG)	96,039 (211,729)

4 ANALYSIS RESULTS

This section describes the analysis results for four ATWS events. Each ATWS event has been analyzed for seventeen burnups from the beginning of cycle (BOC) to the end of cycle (EOC) to find the most unfavorable burnup having maximum peak pressure. The peak pressure for each ATWS event occurred at the burnup of [ ]^{TS} regardless of ATWS events. Thus, all the analysis results described in this report are based on calculation results with burnup of [ ]^{TS} regarding RCS overpressure protection.

4.1 Loss of Normal Feedwater w/o Turbine Trip

4.1.1 Identification of Event

An LONF w/o turbine trip event may be initiated by losing two or more of the three operating main feedwater pumps or by a spurious signal being generated by the feedwater control system (FWCS) resulting in the closure of the feedwater control valve. If turbine trip does not occur during LONF, the decrease in secondary mass inventories accelerates rapidly. This phenomenon may cause RCS heat removal to be interrupted due to the reduced heat transfer from the primary system to the secondary system.

4.1.2 Sequence of Events

Table 4-1 shows the sequence of events for the LONF w/o turbine trip event. Figures 4-1 through 4-9 show the major parameters and the burnup sensitivity for LONF cases. The major phenomena are described as follows;

The LONF causes an increase in both the steam generator pressure and the temperature, with a reduction in the primary-to-secondary heat transfer. The RCS temperature increases due to the decrease in heat transfer, and the expansion of the primary coolant results in an increase in the pressurizer level and pressure.

A low steam generator level (LSGL) trip signal is generated at 35.6 seconds. However, the scram rods are not inserted due to the mechanical failure of the control element drive mechanism (CEDM). The primary system pressure rises rapidly and the steam generator inventories are depleted.

At 66.3 seconds, auxiliary feedwater is injected into the steam generators. However, the decrease in secondary mass inventories continues until 75.6 seconds. As shown in Figure 4-7, the POSRVs open at 68.1 seconds and the POSRV flow increases sharply due to increased RCS pressure. The increased RCS pressure and temperature reduce as the amount of the injected auxiliary feedwater increases.

At 70.6 seconds, the steam generator pressure reaches 60.112 kg/cm^{2}A generating the main steam isolation signal (MSIS); thus, the MSIVs close at 75.6 seconds. The closing of the MSIVs results in an increase in steam generator temperature and pressure.

At 99.1 seconds, the RCS peak pressure of [ ]^{TS} occurs. However, the large energy released through the POSRVs allows the pressurizer pressure to decrease rapidly to 151.08 kg/cm^{2}A, which is the closing setpoint of the POSRVs.

Figures 4-2 and 4-7 show the opening and closing of the POSRVs, until the reactor core power decreases sufficiently due to the moderator negative feedback resulting from the increased primary coolant temperature. The RCS pressure also begins to decrease at this time.

The MSSVs open at 182.0 seconds. Thereafter the auxiliary feedwater and the steam flow through MSSVs continue to cool down the RCS. After about 1000 seconds, reactor power remains below 9%. The decay heat is removed by auxiliary feedwater and MSSVs. After 1800 seconds, the operator will take actions to perform a controlled cooldown of the plant under an appropriate recovery procedure.

4.1.3 Results

The most limiting burnup which causes maximum RCS peak pressure is [ ]^{TS} in LONF w/o turbine trip event. The peak RCS pressure is [ ]^{TS} at 99.1 seconds.

4.2 Loss of Normal Feedwater w/ Turbine Trip

4.2.1 Identification of Event

The event initiation for an LONF w/ turbine trip is the same as that of the LONF w/o turbine trip event. The difference between these events is whether the turbine is tripped along with the LSGL trip signal.

4.2.2 Sequence of Events

Table 4-2 shows the sequence of events for the LONF w/ turbine trip event. Figures 4-10 through 4-18 show the variations of major parameters and burnup sensitivity for the LONF w/ turbine trip event. The major phenomena are described as follows;

The LONF causes an increase in both the steam generator pressure and the temperature, and subsequently a reduction in the primary-to-secondary heat transfer. The RCS temperature increases due to the reduction in heat transfer, and the expansion of the primary coolant results in an increase in the pressurizer level and pressure.

A low steam generator level trip signal is generated at 35.6 seconds, and a turbine trip occurs at 35.7 seconds. This is different from the case of the LONF w/o turbine trip event. Though a reactor trip signal has been generated by the low steam generator level, the scram rods are not inserted due to the mechanical failure of the CEDM. Thus, the primary system pressure rises rapidly and the secondary inventory is depleted continuously until the reactor power sufficiently deceases. Comparing this event to the LONF w/o turbine trip event, the decrease in the secondary mass inventories of this case is slower than that of the LONF w/o turbine trip.

As shown in Figure 4-16, the POSRVs open at 52.0 seconds. The reduced steam generator heat transfer area results in a large decrease in the steam generator heat removal capacity. At 60.3 seconds, auxiliary feedwater is injected into the steam generators. However, the secondary mass inventories drop until 97.4 seconds.

At 106.1 seconds, the RCS pressure reaches its peak value of [ ]^{TS}. However, the large energy released through the POSRVs allows the pressurizer pressure to decrease rapidly to151.08 kg/cm^{2}A, which is the closing setpoint of the POSRVs at 250.0 seconds.

After 1800 seconds, the operator will take actions to perform a controlled cooldown of the plant under an appropriate recovery procedure.

4.2.3 Results

The most limiting burnup having the RCS peak pressure is [ ]^{TS} in the LONF w/ turbine

trip event. RCS peak pressure is [ ]^{TS} at 106.1 seconds.

The turbine trip reduces the steam blowdown of the secondary side. Thus, the steam generator depletion is delayed and faster core power depression occurs due to the moderator reactivity feedback. The RCS peak pressure of this event is lower than that of the LONF w/o turbine trip event.

4.3 Loss of Condenser Vacuum

4.3.1 Identification of Event

A LOCV event occurs due to failure of the circulating water system to supply cooling water, failure of the main condenser evacuation system to remove non-condensable gages, or excessive air leakage. For conservatism, it is assumed that the main feedwater flow is not supplied and the turbine is tripped at the beginning of LOCV.

4.3.2 Sequence of Events

An LOCV concurrent with turbine trip results in a main steam system pressure increase. Rapid pressure increase on the secondary side causes the MSSVs to open at 8.3 seconds. The mass inventories of both steam generators decrease due to the mass release through MSSVs until the auxiliary feedwater is injected into the steam generators.

The reduced steam flow on the secondary side and the termination of feedwater flow cause a reactor trip on HPP at 8.3 seconds owing to the RCS cooling reduction. However, the scram rods are not inserted due to mechanical failure. The primary system pressure rises and the secondary inventory is depleted continuously, before the reactor power sufficiently deceases due to moderator reactivity feedback.

At 11.4 seconds, the pressurizer POSRVs open to suppress the primary system pressure increase. The steam discharged through the pressurizer POSRVs is released to the in-containment refueling water storage tank (IRWST).

At 79.5 seconds, auxiliary feedwater begins to be injected into the steam generators to recover the decreased steam generator water level. However, the secondary mass inventories still decrease up to 93.6 seconds in spite of auxiliary feedwater injection. The RCS peak pressure of [ ]^{TS} is reached at 109.7 seconds. However, the energy released through the POSRVs allows the pressurizer pressure to decrease rapidly to the closing setpoint of the POSRVs.

After 1800 seconds, the operator will perform a controlled cooldown of the plant under an appropriate recovery procedure.

With respect to the RCS peak pressure, the dynamic behaviors of the NSSS parameters for the LOCV are presented in Figures 4-19 through 4-27. Table 4-3 presents a chronological sequence of events for the LOCV.

4.3.3 Results

The most limiting burnup regarding the RCS peak pressure is [ ]^{TS} for an LOCV event. Then, the RCS peak pressure is [ ]^{TS} at 109.7 seconds. The steam generator blowdown of the LOCV event is lower than that of the LONF event at the beginning. This makes core heat removal temporarily difficult, and therefore the primary system causes a heatup and pressurizes in 20 seconds. However the steam generator depletion is delayed and faster core power depression occurs due to the moderator reactivity feedback. Due to these factors, the RCS peak pressure of the LOCV is lower than

that of the LONF w/o turbine trip.

4.4 Loss of Offsite Power

4.4.1 Identification of Event

An LOOP is a complete loss of power to the electrical buses to the main unit. A LOOP results in a turbine generator trip and loss of normal electrical power to station equipment. It is expected that the control systems, such as SBCS and FWCS, will not be available and all the RCPs will start to coastdown during a LOOP.

4.4.2 Sequence of Events

The sequence of events and the major NSSS parameters for the LOOP event are provided in Table 4-4 and Figures 4-28 through 4-36, respectively. A description for the LOOP event is as follows:

The loss of offsite power causes the plant to experience a simultaneous turbine trip, loss of main feedwater, condenser inoperability, and coastdown of all RCPs. The loss of steam flow due to the closure of turbine stop valve results in a rapid increase in the steam generator pressure. A sharp reduction in heat removal of secondary side and the loss of forced reactor coolant flow causes a rapid heatup of the primary coolant.

A low RCP shaft speed trip signal is generated at 1.2 seconds by the core protection calculator (CPC). However, the scram rods are not inserted owing to the mechanical failure of the CEDM. The RCS pressure rises and the secondary inventory is depleted until the reactor power sufficiently deceases.

MSSVs and POSRVs open at 8.8 seconds and 8.9 seconds, respectively. At 77.2 seconds, auxiliary feedwater enters the steam generators. However, the decrease in secondary mass inventories is continued until 222.0 seconds. The opening of the POSRVs and entrance of the auxiliary feedwater results in a decrease in the RCS pressure and temperature.

Compared to the LONF and the LOCV events, in this LOOP event the primary-to-secondary heat transfer is low due to the reduced core flow. Therefore, the steam generator depletion is delayed and the POSRVs periodically open and close to remove decay heat. After 1800 seconds, the operator will perform a controlled cooldown of the plant under an appropriate recovery procedure using atmospheric dump valves (ADVs).

4.4.3 Results

The most limiting burnup influencing the RCS peak pressure is [ ]^{TS} in the LOOP event. Then the RCS peak pressure is [ ]^{TS} at 9.3 seconds.

The core heat removal is reduced due to the reduced RCS flow. This causes the core heatup and pressurization of the primary system. However, the depletion of the steam generator mass inventory is delayed and the core power depression is accelerated. For these reasons, the peak RCS pressure of the LOOP is lower than that of the LONF.

4.5 Turbine Trip

4.5.1 Identification of Event

A turbine trip may result from a number of conditions that cause the turbine generator control system to initiate a turbine trip signal. A turbine trip initiates closure of the turbine stop valves.

4.5.2 Sequence of Events

A turbine trip results in a reduction in steam flow from the steam generators to the turbine. The SBCS, RRS and RPCS are normally in automatic mode and are assumed to be available upon turbine trip to accommodate the load rejection. However, it is also assumed that RPCS and RRS cannot be operated due to the mechanical failure of the CEDM. Thus, the heat on the secondary side is removed by being discharged through the TBVs. The TBV capacity is insufficient to remove the RCS heat because the reactor power does not decrease. Therefore, the temperature and pressure of the RCS increase and a reactor trip signal will be generated by the high pressurizer pressure at 17.9 seconds. However, the scram rods are not inserted due to the mechanical failure of the CEDM, and the primary system pressure rises rapidly.

Due to the increased RCS temperature, the moderator reactivity feedback causes the reactor power to decrease. The peak RCS pressure is [ ]^{TS} at 17.9 seconds. At 29.8 seconds the MSSVs are opened to alleviate the increased secondary system pressure and act as a heat sink for the NSSS. After the MSSVs open, a new quasi steady-state condition is reached.

Table 4-5 presents a chronological sequence for this event until operator action is initiated; the main parameters related to this event are shown in Figure 4-37 through Figure 4-45.

4.5.3 Results

The most limiting burnup influencing the RCS peak pressure is [ ]^{TS} for the turbine trip case. The RCS peak pressure is [ ]^{TS} at 21.0 seconds.

4.6 UET Evaluation

UET has been also calculated to evaluate the design APR1400 design. As depicted in Figure 4-48, all the MTC of the first cycle are higher than those of the equilibrium cycle. Thus, UET calculation regarded the first cycle as a limiting cycle and was based on this. There are three events which had exceeded 226 kg/cm^{2}A (3200 psig) of ASME Service Level C Limits. UETs are [ ]^{TS} as shown in Figure 4-47. For CE type plant, the expected result for UET is that unfavorable MTC is less than 50% (Reference 9). However, the probability that the turbine trip does not occur during ATWS would be extremely low because the DPS has equipment which initiates a turbine trip under conditions indicative of an ATWS. Turbine trip signal by DPS is generated after the reactor trip signal.

4.7 Summary

The analysis is performed using the fuel burnup sensitivities for the four ATWS events selected with respect to RCS overpressure. The analysis results are provided in Table 4-6, and in Figures 4-46, and 4-47. This is the most adverse burnup, during which all events lead to the [ ]^{TS} burnup; the LONF without turbine trip is the most limiting event among all the ATWS events.

Table 4-1 LONF w/o Turbine Trip: Sequence of Events

Time (sec)	Event
0.0	Loss of main feedwater occurs
35.6	LSGL trip setpoint is reached, but scram rods are not inserted
66.3	Auxiliary feedwater is injected into steam generators
68.1	POSRVs open
75.6	MSIVs close by MSIS (LSGP)
80.6	Pressurizer solid begins
99.1	Peak RCS pressure occurs ([ ]TS)
182.0	MSSVs open
1800.0	Operators take manual actions to control plant

Table 4-2 LONF w/ Turbine Trip: Sequence of Events

Time (sec)	Event
0.0	Loss of main feedwater occurs
35.6	LSGL trip setpoint is reached, but scram rods are not inserted
35.7	Turbine trip occurs
52.0	POSRVs open
60.3	Auxiliary feedwater is injected into steam generators
83.0	Pressurizer solid begins
106.1	Peak RCS pressure occurs ([ ]TS)
1800.0	Operators take manual actions to control plant

Table 4-3 LOCV: Sequence of Events

Time (sec)	Event
0.0	Loss of condenser vacuum occurs
8.3	MSSVs open
8.3	HPP trip setpoint is reached, but scram rods are not inserted
11.4	POSRVs open
79.5	Auxiliary feedwater is injected into steam generators
84.5	Pressurizer solid begins
109.7	Peak RCS pressure occurs ([ ]TS)
1800.0	Operators take manual actions to control plant

Table 4-4 LOOP: Sequence of Events

Time (sec)	Event
0.0	Loss of offsite power occurs
0.0	RCPs coastdown
1.2	Low RCP speed trip setpoint is reached, but scram rods are not inserted
8.8	MSSVs open
8.9	POSRVs open
9.3	Peak RCS pressure occurs ([ ]TS)
77.2	Auxiliary feedwater is injected into steam generators
1800.0	Operators take manual actions to control plant

Table 4-5 Turbine Trip: Sequence of Events

Time (sec)	Event
0.0	Turbine trip occurs
17.9	HPP trip setpoint is reached, but scram rods are not inserted
21.0	Peak RCS pressure occurs ([ ]TS)
29.8	MSSVs open
1800.0	Operators take manual actions to control plant

Table 4-6 RCS Peak Pressure vs. Burnup

Burnup, MWD/MTU	Peak Pressure, kg/cm^2A (Time, seconds)				
	LONF		LOCV	LOOP	Turbine Trip
	w/o Turbine Trip	w/ Turbine Trip			
0					
501					
1003					
2011					
3016					
4019					
5020					
6020					
7018					
8016					
9013					
10010					
12002					
13992					
14987					
15981					
17571					

TS

TS

Figure 4-1 LONF w/o Turbine Trip: Reactor Power

TS

Figure 4-2 LONF w/o Turbine Trip: Discharge Leg Pressure

TS

Figure 4-3 LONF w/o Turbine Trip: Pressurizer Level

TS

Figure 4-4 LONF w/o Turbine Trip: RCS Temperature

TS

Figure 4-5 LONF w/o Turbine Trip: Steam Generator Pressure

TS

Figure 4-6 LONF w/o Turbine Trip: Steam Generator Inventory

TS

Figure 4-7 LONF w/o Turbine Trip: POSRV Flow Rate

TS

Figure 4-8 LONF w/o Turbine Trip: Core Inlet Flow Rate

TS

Figure 4-9 LONF w/o Turbine Trip: Reactivity

TS

Figure 4-10 LONF w/ Turbine Trip: Reactor Power

TS

Figure 4-11 LONF w/ Turbine Trip: Discharge Leg Pressure

TS

Figure 4-12 LONF w/ Turbine Trip: Pressurizer Level

TS

Figure 4-13 LONF w/ Turbine Trip: RCS Temperature

TS

Figure 4-14 LONF w/ Turbine Trip: Steam Generator Pressure

TS

Figure 4-15 LONF w/ Turbine Trip: Steam Generator Inventory

TS

Figure 4-16 LONF w/ Turbine Trip: POSRV Flow Rate

TS

Figure 4-17 LONF w/ Turbine Trip: Core Inlet Flow Rate

TS

Figure 4-18 LONF w/ Turbine Trip: Reactivity

TS

Figure 4-19 LOCV: Reactor Power

TS

Figure 4-20 LOCV: Discharge Leg Pressure

TS

Figure 4-21 LOCV: Pressurizer Level

TS

Figure 4-22 LOCV: RCS Temperature

TS

Figure 4-23 LOCV: Steam Generator Pressure

TS

Figure 4-24 LOCV: Steam Generator Inventory

TS

Figure 4-25 LOCV: POSRV Flow Rate

TS

Figure 4-26 LOCV: Core Inlet Flow Rate

TS

Figure 4-27 LOCV: Reactivity

TS

Figure 4-28 LOOP: Reactor Power

TS

Figure 4-29 LOOP: Discharge Leg Pressure

TS

Figure 4-30 LOOP: Pressurizer Level

TS

Figure 4-31 LOOP: RCS Temperature

TS

Figure 4-32 LOOP: Steam Generator Pressure

TS

Figure 4-33 LOOP: Steam Generator Inventory

TS

Figure 4-34 LOOP: POSRV Flow Rate

TS

Figure 4-35 LOOP: Core Inlet Flow Rate

TS

Figure 4-36 LOOP: Reactivity

TS

Figure 4-37 Turbine Trip: Reactor Power

TS

Figure 4-38 Turbine Trip: Discharge Leg Pressure

TS

Figure 4-39 Turbine Trip: Pressurizer Level

TS

Figure 4-40 Turbine Trip: RCS Temperature

TS

Figure 4-41 Turbine Trip: Steam Generator Pressure

TS

Figure 4-42 Turbine Trip: Steam Generator Inventory

TS

Figure 4-43 Turbine Trip: POSRV Flow Rate

TS

Figure 4-44 Turbine Trip: Core Inlet Flow Rate

TS

Figure 4-45 Turbine Trip: Reactivity

TS

Figure 4-46 Event Sensitivity at Limiting Burnup

TS

Figure 4-47 RCS Peak Pressure vs. Burnup

TS

Figure 4-48 Comparison of MTC for First and Equilibrium Cycles

5 CONCLUSION

ATWS analysis has been performed regarding RCS over-pressurization. Four ATWS events are selected through the quantitative evaluation to evaluate the capability of the APR1400 design against ATWS regarding RCS integrity.

Each ATWS event is qualitatively evaluated by a burnup sensitivity study to find the maximum RCS peak pressure. The analysis results show that the most limiting burnup is [ ]TS in view of RCS integrity, regardless of ATWS events. This means that the most important parameter for ATWS is the moderator reactivity feedback. Among the selected four ATWS events, the most limiting event regarding RCS over-pressurization is found to be an LONF w/o turbine trip event. This ensures that the analysis results of CENPD-158 can be applied to the APR1400 design.

6 REFERENCES

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