ATWS Evaluation

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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 with the fuel burnup of 2011 MWD/MTU regardless of ATWS events. 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
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
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
15.2.7	Loss of Normal Feedwater Flow
15.3.1	Total Loss of Reactor Coolant Flow
15.4.1	Uncontrolled CEA withdrawal from a Subcritical or Low Power Condition
15.4.2	Uncontrolled CEA withdrawal at Power
15.4.3	Single CEA Drop
15.4.4	Startup of an Inactive Reactor Coolant Pump
15.4.6	Inadvertent Deboration
15.4.7	Inadvertent Loading of a Fuel Assembly into the Improper Position
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.

(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

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3 EVALUATION APPROACH

3.1 Qualitative Evaluation

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3.2 Quantitative Analysis

3.2.1 Assumptions and Initial Conditions

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.

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

Table 3-2 Initial Conditions for ATWS Evaluation

Parameter	Value
Reactor Power, MWt	3,983
Pressurizer Pressure, kg/cm^{2}A (psia)	158.19 (2,250)
Core Inlet Temperature, ^{o}C (^{o}F)	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)

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4 ANALYSIS RESULTS

4.1 Loss of Normal Feedwater w/o Turbine Trip

4.1.1 Identification of Event

4.1.2 Sequence of Events

4.1.3 Results

4.2 Loss of Normal Feedwater w/ Turbine Trip

4.2.1 Identification of Event

4.2.2 Sequence of Events

4.2.3 Results

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4.4.2 Sequence of Events

4.4.3 Results

4.5 Turbine Trip

4.5.1 Identification of Event

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4.5.3 Results

4.6 Summary

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

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

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

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

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

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

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

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

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

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

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Figure 4-18 LONF w/ Turbine Trip: Reactivity

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Figure 4-19 LOCV: Reactor Power

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Figure 4-20 LOCV: Discharge Leg Pressure

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Figure 4-21 LOCV: Pressurizer Level

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Figure 4-22 LOCV: RCS Temperature

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Figure 4-23 LOCV: Steam Generator Pressure

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Figure 4-24 LOCV: Steam Generator Inventory

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Figure 4-30 LOOP: Pressurizer Level

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Figure 4-31 LOOP: RCS Temperature

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Figure 4-32 LOOP: Steam Generator Pressure

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Figure 4-33 LOOP: Steam Generator Inventory

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Figure 4-34 LOOP: POSRV Flow Rate

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Figure 4-35 LOOP: Core Inlet Flow Rate

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Figure 4-36 LOOP: Reactivity

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Figure 4-39 Turbine Trip: Pressurizer Level

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5 CONCLUSION

6 REFERENCES

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