Severe Accident Analysis

Technical Report

Non-Proprietary

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List of Acronyms

AC	Alternative Current
AFW	Auxiliary Feed Water
AICC	Adiabatic Isochoric Complete Combustion
ALWR	Advanced Light Water Reactor
AOO	Anticipated Operational Occurrence
ATWS	Anticipated Transient without Scram
BDBA	Beyond Design Basis Accident
CCFP	Conditional Containment Failure Probability
CCW	Component Cooling Water
CET	Core Exit Temperature
CFS	Cavity Flooding System
CSS	Containment Spray System
DBA	Design Basis Accident
DCH	Direct Containment Heating
DDT	Deflagration-to-Detonation Transition
DPS	Diverse Protection System
ECSBS	Emergency Containment Spray Backup System
EDG	Emergency Diesel Generator
ES	Equipment Survivability
ESW	Essential Service Water
EVSE	Ex-Vessel Steam Explosion
FA	Flame Acceleration
FCI	Fuel Coolant Interaction
FLC	Factored Load Category
HPME	High Pressure Melt Ejection
HVT	Holdup Volume Tank
ICI	In Core Instrumentation
IRWST	In-Containment Refueling Water Storage Tank
ISLOCA	Intersystem Loss of Coolant Accident
IVSE	In-Vessel Steam Explosion
LBLOCA	Large Break LOCA
LOCA	Loss of Coolant Accident
LOOP	Loss Of Offsite Power

List of Acronyms

MAAP Modular Accident Analysis Program
MBLOCA Medium Break LOCA
MCCI Molten Core Concrete Interaction
MWR Metal Water Reaction
PAR Passive Autocatalytic Recombiner
POSRV Pilot Operated Safety and Relief Valve
PRA Probabilistic Risk Assessment
RCGV Reactor Coolant Gas Vent
RCP Reactor Coolant Pump
RCS Reactor Coolant System
RD Rapid Depressurization
RV Reactor Vessel
RVLMS Reactor Vessel Level Monitoring System
SBLOCA Small Break LOCA
SBO Station Black Out
SCS Shutdown Cooling System
SIS Safety Injection System
TLOESW Total Loss of Essential Service Water
TLOFW Total Loss of Feed Water
VB Vessel Breach
MCCI Molten Core Concrete Interaction
MWR Metal Water Reaction
PAR Passive Autocatalytic Recombiner
POSRV Pilot Operated Safety and Relief Valve
PRA Probabilistic Risk Assessment
RCGV Reactor Coolant Gas Vent
RCP Reactor Coolant Pump
RCS Reactor Coolant System
RD Rapid Depressurization
RV Reactor Vessel
RVLMS Reactor Vessel Level Monitoring System
SBLOCA Small Break LOCA
SBO Station Black Out

List of Acronyms

SCS	Shutdown Cooling System
SIS	Safety Injection System
TLOESW	Total Loss of Essential Service Water
TLOFW	Total Loss of Feed Water
VB	Vessel Breach

1.0 INTRODUCTION

The likelihood of a severe accident, which postulates reactor core meltdown beyond the scope of design basis accidents and consequently can lead to releases of large amounts of radionuclides into the environment, is extremely low. However, in view of the postulated severe damage to the reactor core, the social and economical consequences of such an accident can be very significant.

The APR1400 is designed for the prevention and mitigation of severe accidents, based on in-depth phenomenological analyses that identify potential design and operational vulnerabilities and address them appropriately in a manner that minimizes the consequential risk to the public and to the environment. The design and construction of the severe accident prevention and mitigation features are in compliance with USNRC regulations. These design features comply with USNRC SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs."(Reference 1) In principle, a series of potential phenomena which can threaten the integrity of the containment needs to be screened, reviewed and assessed, since various individual phenomena can occur under severe accident conditions depending upon the design characteristics of the plant.

This report consists of the main body and supplemental appendices. The main body of this report has two parts. The first part describes the severe accident prevention and mitigation features. The second part covers the phenomenological analysis and assessment. The appendices contain the details for the following severe accident analysis.

- Hydrogen control
- Molten core-concrete interaction(MCCI) and Debris Coolability
- High pressure melt ejection(HPME) and direct containment heating(DCH)
- Fuel-coolant interaction(FCI)
- Containment performance capability
- Equipment survivability

2.0 SEVERE ACCIDENT PREVENTION AND MITIGATION FEATURES

This section describes the APR1400 features that are designed to prevent and mitigate severe accidents. The severe accident evaluation for the APR1400 design is consistent with the guidance in SECY-93-087 as well as the corresponding Staff Requirements Memorandum (SRM), dated July 21, 1993. The reactor and containment system designs are a vital portion of the defense-in-depth philosophy. Current reactors and containments are designed to withstand a loss-of-coolant accident (LOCA) and to comply with the siting criteria of 10 CFR 100 (Reference 2), General Design Criteria (GDC) of 10 CFR 50 Appendix A (Reference 3), and the Three Mile Island (TMI)-related requirements of 10 CFR 50.34(f) (Reference 4), 10 CFR 50.44 (Reference 5), NRC RG 1.216 (Reference 6), and SECY-90-016 (Reference 7).

2.1 Severe Accident Prevention

The APR1400 design includes features aimed at preventing the onset of a severe accident, including the severe accident precursors identified in SECY-90-016 and SECY-93-087. These precursors include anticipated transient without scram (ATWS), mid-loop operation, station blackout (SBO) event, fire, and intersystem loss-of-coolant accident (ISLOCA). Preventive features are described below for each of these events.

2.1.1 Anticipated Transient without Scram

An ATWS happens when an anticipated operational occurrence (AOO) occurs but is not followed by an automatic reactor trip. Reactor trip is necessary to terminate the transient and to shut down the plant. The APR1400 design includes a digital safety system and a diverse protection system (DPS) to minimize the possibility of an ATWS.

The plant protection system (PPS) is normally available to prevent and mitigate an ATWS. The PPS includes the electrical and mechanical devices and circuitry required to perform the functions of the reactor protection system (RPS) and the engineered safety features component control system (ESF-CCS). The RPS is the portion of the PPS that trips the reactor when required. A coincidence of two signals, due to the two-out-of-four trip logic, is required to generate a reactor trip signal. The ESF-CCS is the portion of the PPS that activates the engineered safety features (ESFs). Additionally, the reactor trip system includes the RPS portion of the PPS, reactor trip switchgear system (RTSS), and components that perform a reactor trip after receiving a signal from the RPS (either automatically or manually).

The DPS provides a diverse backup to the PPS when the PPS is not working. The DPS initiates a reactor trip signal on high pressurizer pressure to decrease the possibility of an ATWS and provides an auxiliary feedwater actuation signal (backup to the ESF-CCS of the PPS) to mitigate an ATWS.

2.1.2 Mid-Loop Operation

During plant shutdowns, certain maintenance and testing activities require the controlled drain-down of the reactor coolant system (RCS) to a partially filled condition. When the reduced RCS level is within the hot leg, the risk of losing shutdown cooling is increased due to the possibility of vortex formation at the shutdown cooling suction line interface with the hot leg. If a vortex is formed in the shutdown cooling suction line, a substantial amount of air could be entrained into the shutdown cooling suction piping and degrade or interrupt the SC pump performance. If sufficient shutdown cooling is not reestablished, coolant heatup and vaporization/boiling can lead to uncovery of the reactor core.

The APR1400 design features can accommodate loss of residual heat removal during the operation with reduced reactor water inventory. These design features include:

2.1.2.1 Instrumentation for Shutdown Operations

Diverse, accurate, and redundant instrumentation provides the operator with continuous system status and precise information to monitor the operation with reduced reactor water inventory and respond to the loss of shutdown cooling events.

2.1.2.2 SCS Design

System design features that improve shutdown cooling system (SCS) performance include:

1. The shutdown cooling suction lines do not contain loop seals, thereby minimizing the potential to trap gas. The suction piping layout allows self-venting of accumulated gas (or air).

2. The two redundant shutdown cooling suction lines are completely independent.

3. There are no auto-closure interlocks on the shutdown cooling suction piping valves, minimizing the potential for shutdown cooling isolation events.

2.1.2.3 Steam Generator Nozzle Dam Integrity

The APR1400 design addresses the regulatory concern of preventing significant pressurization in the upper plenum of the reactor vessel during core boiling scenarios. The APR1400 procedural guidance recommends a nozzle dam installation and removal sequence, which consists of the following:

1. Installation: The nozzle dams are installed in the cold legs first and in the hot legs second.

2. Removal: The nozzle dams in the hot legs are removed first and in the cold legs second.

The installation procedure requires that the pressurizer manway be opened so that a hot side vent pathway exists prior to blocking both RCS hot legs with nozzle dams.

In the APR1400 design, the ability of the RCS to withstand abnormal pressurization during reduced-inventory operations with the nozzle dams installed is limited by the design pressure of the nozzle dams. Based on overpressure tests performed on nozzle dams, the design pressure is estimated to be 3.52 kg/cm^2 (50 psia). The design pressure is sufficient to withstand an abnormal pressurization transient.

In order to provide reasonable assurance that the nozzle dam design pressure is not exceeded during reduced-inventory operations with boiling conditions in the reactor vessel, the APR1400 design requires that a mid-loop vent pathway is opened via the pressurizer manway prior to reduced-inventory operation. When the pressurizer manway is opened to the containment atmosphere, the surge line provides sufficient venting capacity to prevent RCS pressurization and preclude subsequent nozzle dam failure. The pressurizer surge line vent pathway has sufficient capacity to prevent core uncovery due to pressurization of the hot side resulting from boiling coolant.

2.1.2.4 Alternate Inventory Additions and Decay Heat Removal Methods

If SCS is lost during Mode 5 reduced water inventory operations, containment spray (CS) pumps or the safety injection (SI) pumps are used to provide makeup. If all above methods of decay heat removal and inventory replenishment are unavailable, a charging pump or a boric acid makeup pump is used to provide makeup for Modes 5 and 6. If no method of pumped inventory addition is available, a source for gravity feed inventory addition can be used via the SI tanks.

2.1.3 Station Blackout

One alternate ac (AAC) source is provided to help mitigate the effects of an SBO. The AAC automatically starts and is manually aligned to provide power to a Class 1E 4.16 kV bus in case Class 1E emergency diesel generators (EDGs) fail to start and load during loss of offsite power (LOOP) events. This standby unit is independent and diverse from the Class 1E EDGs. Successful startup of the AAC together with turbine-driven auxiliary feedwater pumps is sufficient to prevent core damage in SBOs.

2.1.4 Fire Protection

The systems and components required for safe shutdown are physically separated from functionally similar or redundant systems or components to maintain the ability to perform safe shutdown functions in the event of a fire. Fire protection features such as fire detection, automatic and manual fire suppression, and fixed fire barriers provide reasonable assurance that the plant does not enter an unrecoverable state as a result of a fire incident.

2.1.5 Intersystem Loss of Coolant Accident

An ISLOCA is defined as a class of events in which a break occurs outside the containment in a system connected to the RCS, leading to a loss of primary system water inventory. This is considered as a beyond-design-basis event for systems connected with the RCS. Pressurization of an interfacing system could result from the inadvertent opening of a valve or valves, failure of containment isolation, or valves that are otherwise fully open (e.g., check valves that are stuck open). The APR1400 design addresses ISLOCA challenges by including the following design features:

1. The design pressure of equipment or systems has been increased to 64.3 kg/cm^2 (900 psig) for the low-pressure systems that are connected with the RCS.
2. Equipment and instrumentation has been added to alert the operator to an ISLOCA challenge or terminate and limit the scope of an ISLOCA event.
3. Parts of systems considered unnecessary are deleted because their functions can be replaced by other existing systems.
4. The refueling water tank is located inside containment.
5. Capability is provided for leak testing pressure isolation valves.
6. Pressure isolation valve position indication and control is provided in the main control room (MCR).
7. High-pressure alarms are added to warn the operator when increasing pressure approaches the design pressure of low-pressure systems.

In the APR1400 design, the safety injection system (SIS), SCS, and chemical and volume control system (CVCS) are directly connected to the RCS and are potentially susceptible to one or more ISLOCA events (i.e., they have one or more ISLOCA pressurization pathways).

The safety injection lines are connected to the reactor vessel directly and are primary interfaces through which an ISLOCA can begin. Pressurization is postulated to move from the direct vessel injection (DVI) nozzles and out of containment through the containment isolation valves to the low-pressure sections of the system. The SIS design satisfies the ISLOCA acceptance criteria because all sections of the system and interfaces are designed to withstand full RCS operating pressure or have a leak-test capability. In addition, the valve position indications in the control room function even when the isolation valve operators are de-energized, and high-pressure alarms sound to warn operators when pressure is approaching the design pressure. These design features protect the SIS lines and all interfacing systems from an ISLOCA challenge without adversely affecting performance or operations.

The shutdown cooling suction lines are connected to the RCS directly and are primary interfaces through which an ISLOCA event can begin. Pressurization is postulated from the hot leg and out of containment through the containment isolation valves to the low pressure sections of the system.

The shutdown cooling return lines are connected to the RCS directly and are primary interfaces through which an ISLOCA event can begin. Pressurization is postulated from the DVI nozzles and out of containment through the containment isolation valves to the low-pressure sections of the SCS.

This SCS line design satisfies the ISLOCA acceptance criteria because all sections of the system and interfaces are designed to withstand full RCS operating pressure, or they have leak-test capabilities, valve position indications in the control room that function even when isolation valve operators are de-energized, and high-pressure alarms to warn operators when pressure is approaching the design pressure. Deletion of the interfaces from the SCS lines eliminates the potential for an ISLOCA without adversely affecting the performance or operations of the SCS. These design features satisfy the ISLOCA acceptance criteria for the SCS line.

The containment spray system (CSS) is not connected directly to the RCS during the modes of reactor operation for which an ISLOCA challenge can occur. However, there is an indirect interface through the SCS because the CS pumps, CS heat exchangers, SC pumps, and SC heat exchangers are interchangeable respectively. All connected CS sections are designed to 64.3 kg/cm^2 (900 psig). The only low-pressure system interface with the CSS is the spent fuel pool cooling and cleanup system (SFPCCS) connection to the refueling pool. This connection provides the ability to fill the refueling pool directly rather than through the reactor vessel. A spool piece connection is available to provide a method of physical separation of the low-pressure SFPCCS from any pressurization source in the CSS.

The CVCS letdown line is directly connected to the RCS and is a primary interface through which an ISLOCA event can begin. Pressurization is postulated from the letdown nozzle, through the regenerative and letdown heat exchangers, through the letdown orifices, and out of containment through the containment isolation and letdown control valves to the low-pressure sections of the system. The letdown line has a high-pressure alarm that is located downstream of the letdown control valves and warns the operator when the pressure is approaching the low-pressure system design pressure. When a warning is issued, the control room operator isolates the letdown line to terminate any further pressure communication downstream of the containment isolation valve.

The CVCS charging line is connected directly to the RCS and is a primary interface through which an ISLOCA event can begin. Pressurization is postulated from the charging nozzle, through the shell side of the regenerative heat exchanger, the charging control valve, and the charging pump to the low-pressure sections of the system. The charging pump suction line has a high-pressure alarm that warns the operator when the pressure is approaching the low-pressure system design pressure. When a warning is received, the control room operator isolates the charging line to terminate any further pressure communication downstream of the containment isolation valve. These design responses satisfy the ISLOCA acceptance criteria.

2.1.6 Other Severe Accident Preventative Features

The APR1400 design uses other features to prevent severe accidents including:

1. Feedwater can be supplied to a steam generator by a turbine-driven auxiliary feedwater

pump when the motor-driven auxiliary feedwater pumps are not available. Two independent turbine-driven auxiliary feedwater pumps are available in the APR1400 design.

2. If the CS pumps are inoperable during a LOCA event, then the SC pumps can be used as a backup.

3. The CS pumps and CS heat exchangers can be used as backups for the SC pumps and SC heat exchangers to provide cooling of the in-containment refueling water storage tank (IRWST) during post-accident feed-and-bleed operations when the steam generators are not available to cool the RCS.

4. Cooling during a loss of all feedwater can be accomplished via feed-and-bleed operation using the SIS and the pilot-operated safety and relief valves (POSRVs).

The component cooling water system (CCWS) is composed of two separate but interconnected two-division systems. The systems are designed to automatically isolate the cross connection in an accident. One or both of these systems operate independently after isolation because their designs provide a high level of performance reliability. If the CCWS is inoperable at any time during reactor coolant pump (RCP) operation, the RCP seal injection function is performed by the supply of seal injection via the auxiliary charging pump.

2.2 Severe Accident Mitigation

If a severe accident cannot be prevented by the above design features, other APR1400 features mitigate the effects of a severe accident. Of particular importance are the containment design and the ability of mitigating equipment to survive severe accident conditions. This section describes the mitigation features in the context of various severe accident phenomena that could be encountered during severe accident progression.

2.2.1 Overview of the Containment Design

2.2.1.1 Description of the Containment

The APR1400 containment is a pre-stressed concrete structure composed of a right circular cylinder with a hemispherical dome and is founded on safety-related common basemat. The APR1400 containment encloses the reactor vessel, steam generators, reactor coolant loops, and portions of the auxiliary and engineered safety features systems. The containment provides reasonable assurance that leakage of radioactive material to the environment does not exceed the acceptable dose limit as defined in 10 CFR 50.34 even if a LOCA occurred.

The cylindrical containment shell has a constant thickness of 1.37 m (4 ft 6 in) from the top of the foundation basemat to the spring line. The shell is thickened locally around the equipment hatch,

two personnel airlocks, feedwater, and main steam line penetrations. The containment reinforcing consists primarily of hoop and meridional steel. Prestressing tendons are also arranged in hoop and meridional directions. The roof of the containment is a hemispherical dome. The buttresses are extended up to 48 degrees into the dome to provide anchorage for the dome hoop tendons. The 6.0 mm (0.25 in) steel liner plate is attached to inside of the dome and the cylindrical wall to provide leak-tightness.

The containment provides a large free volume with its internal structures arranged in a manner to (1) protect the inner containment from missile threats, (2) promote mixing throughout the containment atmosphere, and (3) accommodate condensable and noncondensable gas releases from design basis and severe accidents. The internal structures, which are made of reinforced concrete, enclose the reactor vessel and other primary system components. The internal structures provide radiation shielding for the containment interior and missile protection for the reactor vessel and containment shell.

2.2.1.2 Containment Pressure Limits

In severe accident scenarios, the containment vessel is the last fission product barrier protecting the public from radiation release. Therefore, it is of paramount importance to provide a strong containment design to meet severe accident internal pressurization challenges.

The containment is designed in accordance with ASME Section III, Division 2 (Reference 8), and for the design pressure of 4.218 kg/cm^2 (60 psig) and design basis temperature of 416.5 K (290 °F). The containment is analyzed to determine all membrane, bending, and shear stresses resulting from the specified static and dynamic design loads.

As stated in SECY 93-087, the conditional containment failure probability (CCFP) must be less than 0.1 or meet a deterministic containment performance goal that provides comparable protection so the following general criterion is met: The containment maintains its role as a reliable, leak-tight barrier by providing reasonable assurance that the containment factored load category (FLC) requirements are met for a period of approximately 24 hours following the onset of core damage, and following this 24-hour period, the containment continues to provide a barrier against the uncontrolled release of fission products. The APR1400 containment meets the FLC requirement of ASME Section III, Division 2, Subarticle CC-3720.

2.2.1.3 Containment Penetrations

The containment pressure boundary is made up of the containment shell and several mechanical and electrical containment penetrations. The penetrations include one equipment hatch; two personnel airlocks; containment piping penetration assemblies to provide for the passage of process, service, sampling, and instrumentation pipelines into the containment; electrical penetrations for power, control, and instrumentation; and a fuel transfer tube. All large penetrations are explicitly considered in the containment shell ultimate pressure capacity analyses. Smaller penetrations are sufficiently strong that they do not prematurely compromise the integrity of

the containment shell.

2.2.2 Cavity Flooding System

The cavity flooding system (CFS) provides a means of flooding the reactor cavity during a severe accident to cool the core debris in the reactor cavity and to scrub fission product releases. The water delivery from the IRWST to the reactor cavity is accomplished by means of active components. The CFS is designed (in conjunction with the containment spray system) to provide an inexhaustible continuous supply of water to quench the core debris.

The components of the CFS include the IRWST, holdup volume tank (HVT), reactor cavity, connecting piping, valves, and associated power supplies. This system is used in conjunction with the containment spray system to form a closed recirculation water cooling system to provide a continuous cooling water supply to the core debris. The quenching of the corium produces steam, which is condensed by the containment spray flow. The CFS takes water from the IRWST and directs it to the reactor cavity. The water flows first into the HVT by way of the two HVT spillways and then into the reactor cavity by way of two reactor cavity spillways.

Once actuated, movement of the water from the IRWST source to the cavity occurs passively due to the natural hydraulic driving heads of the system. Actuation of the CFS results in the opening of the HVT spillway valves, allowing water from the IRWST to flood the HVT. This flow is driven by the differences in the static heads of water between the IRWST and the HVT. Flooding of the HVT progresses until the water level in the HVT reaches the reactor cavity spillway, at which time reactor cavity flooding commences. Flooding ceases when water levels in the IRWST, HVT, and reactor cavity equalize at 6.4 m (21 ft) from the reactor cavity floor (EL. 69 ft 0 in).

The HVT and cavity spillways are located as low as possible to provide the greatest head and maximize usage of available water in the IRWST and HVT. Both spillways are equipped with remote manual motor-operated valves (MOVs). HVT flooding valves are normally closed and located in individual flow paths connecting the IRWST to the HVT. Reactor cavity flooding valves are normally closed and located in individual flow paths connecting the HVT to the reactor cavity. The valves are opened by the MCR operator to flood the reactor cavity in the event of a severe accident. Controls are provided to allow the valves to be opened either individually or simultaneously, to initiate reactor cavity flooding.

Flooding of the reactor cavity serves the following purposes in the strategy to mitigate the consequences of a severe accident:

1. Minimize or eliminate corium-concrete attack
2. Minimize the generation of combustible gases (hydrogen and carbon monoxide) and non-condensable gases
3. Scrub fission products released due to corium-concrete interaction
4. Remove heat from the core debris

The manual operation of the CFS provides a mechanism for the operator to most efficiently use plant resources and mitigate the consequences of a severe accident. It is envisioned that the CFS is actuated once a potential core melt condition is imminent or has been diagnosed as being in progress. Typical indications of core uncovery include (1) core exit temperature (CET) in excess of 922.04 K (1,200 °F), (2) reactor vessel level monitoring system (RVLMS) readings indicative of no liquid above the fuel alignment plate, and (3) significant changes in readings of self-powered neutron detectors (SPND).

It is understood that steam explosions may pose a non-negligible threat to the cavity and containment integrity. Thus, there may be an incentive to delay actuation of the CFS until vessel breach (VB) is imminent or when the reactor vessel lower head has failed. While actuation of the CFS before VB is presently deemed desirable, the consequences of delayed CFS actuation (prior to extensive concrete erosion) may also achieve similar results. Flooding of the HVT progresses until the water levels in IRWST, HVT, and reactor cavity equalize at 6.4 m (21 ft) from the reactor cavity floor (EL. 69 ft 0 in).

Thus, it is currently believed that an acceptable stable state can be achieved ex-vessel as long as the CFS has been actuated prior to VB. Although providing water to the reactor cavity may not immediately terminate the concrete erosion, having a water-filled reactor cavity initially reduces and ultimately terminates the erosion, while simultaneously providing scrubbing of fission products released in the molten core-concrete interaction process.

2.2.3 Hydrogen Mitigation System

During a degraded core accident, hydrogen is generated at a greater rate than that of the design basis LOCA. The containment hydrogen control system is designed to accommodate the hydrogen generation from a metal-water reaction of 100 percent of the active fuel cladding and limit the average hydrogen concentration in containment to 10 percent consistent with 10 CFR 50.34(f) and 10 CFR 50.44 for a degraded core accident. These limits are imposed to preclude detonations in containment that might jeopardize containment integrity or damage essential equipment.

The containment hydrogen control system (HG) consists of a system of passive autocatalytic recombiners (PARs) complemented by glow plug igniters installed within the containment. The PARs are capable of controlling hydrogen in all accident sequences with moderate hydrogen release rates, and are located throughout the containment. The igniters supplement PARs for accidents in which rapid hydrogen release rates are expected, and are placed near anticipated source locations to promote the combustion of hydrogen in a controlled manner. The HG design is composed of 30 PARs and 8 igniters. The HG PARs are strategically distributed so that the overall average concentration requirements are met. These locations are determined based on equipment and piping proximity as well as inspection and maintenance access. The PAR components and igniter assembly are designed to meet seismic Category I requirements.

The PARs are self-actuated and require no electric power. Therefore, no operator action is required. The igniters, which supplement PARs, are intended to control hydrogen concentration

within containment once the operator confirms that an extended core uncovery is in progress. The operators use specific accident management guidance that relies on RCS and containment instrumentation, such as in-vessel level monitoring instrumentation, core-exit thermocouples, containment and RCS pressure indications, and a direct measurement of containment hydrogen concentration.

Once activated, an igniter produces either periodic small local burns or a standing diffusion flame, either of which reduces the containment hydrogen concentration below the upward flammability limit. Thus, the HG system prevents hydrogen from accumulating to the point where a destructive hydrogen detonation might occur within the containment.

2.2.4 Rapid Depressurization Function

The rapid depressurization (RD) function is a multi-purpose dedicated system designed to serve important roles in severe accident prevention and mitigation.

In the APR1400 design, the POSRVs are designed to allow for depressurization of the RCS below the cutoff pressure for HPME to occur. For the APR1400 design, the rapid depressurization function is initiated by operator action as part of the severe accident management strategy. When CET exceeds 922.04K (1,200°F), the operator identifies entry into a severe accident condition and starts rapid depressurization by opening the required POSRVs.

The RD function design requirement related to severe accident mitigation is the capability to depressurize the RCS from approximately 175.8 kg/cm^2 (2,500 psia) to approximately 17.6 kg/cm^2 (250 psia) prior to reactor vessel breach. The target pressure of the RD function is determined on the basis of DOE/ID-10271 (Reference 9).

The power for each RD valve is supplied from a respective Class 1E direct current (dc) bus. The power is provided such that a bleed path can be established in case of a loss of offsite power, four EDGs, and the AAC source. Each train of dc loads is provided with a separate and independent battery charger and a standby charger. The battery chargers are powered from the 480 V ac Class 1E power distribution systems of the same trains. A load management strategy provides reasonable assurance of dc power availability for a minimum of 4 hours for Trains A and B and 16 hours for Trains C and D following an SBO.

The RD function provides a manual means of quickly depressurizing the RCS when normal and auxiliary feedwater are unavailable to remove core decay heat through the steam generators. This function is achieved via remote manual operator control. Whenever events, such as a total loss of feedwater (TLOFW), result in a high RCS pressure with a loss of RCS liquid inventory, the POSRVs may be opened by the operator, causing a controlled depressurization of the RCS. As the RCS pressure decreases, the SI pumps start, initiating feed flow to the RCS and restoring the RCS liquid inventory. The RD function allows for both short- and long-term decay heat removals.

The RD function also serves an important role in severe accident mitigation. In the event a high-pressure meltdown scenario develops and the feed portion of feed and bleed cannot be established due to unavailability of the SI pumps, the RD function can be used to depressurize the RCS and

prevent HPME following a VB.

2.2.5 Reactor Cavity Design

The reactor cavity is configured to promote retention of, and heat removal from, the postulated core debris during a severe accident, thus serving several roles in accident mitigation. The large cavity floor area allows for spreading of the core debris, enhancing its coolability within the reactor cavity region.

The large free volume of the reactor cavity is a benefit when cavity pressurization issues are considered. Large, vented volumes are not prone to significant pressurization resulting from vessel breach or during corium quench processes. This design characteristic is illustrated in the cavity pressurization analysis results in that the possible peak pressure in the reactor cavity during severe accidents stays well below the allowable capacity.

The reactor cavity is designed to maximize the unobstructed floor area available for the spreading of core debris. The cavity floor is free from obstructions and comprises an area available for core debris spreading such that the floor area/reactor thermal power ratio is larger than 0.02 m^2/MWt. Uniform distribution of 100 percent of the corium debris within the reactor cavity results in a relatively shallow debris bed and consequently, effective debris cooling is expected in the reactor cavity. The containment liner plate in reactor cavity area is embedded 0.91 m (3 ft.) below from the cavity floor at the minimum.

Corium retention in the core debris chamber virtually eliminates the potential for significant DCH-induced containment loadings. When the vessel is breached under high pressure, the melt is ejected first followed by the high-speed steam and H_2 jet. The melt is entrained by the jet into small particles. The duration of the gas blowdown following melt ejection may be sufficiently long to cause complete sweep-out of the ejected melt. Therefore, it is reasonable, and conservative, to assume complete entrainment of ejected melt. Then, the mixture of steam, gas, and corium particles flow through available flow paths between the reactor cavity and the upper containment.

For flow entering the debris chamber, the lower-inertia steam/hydrogen/gas mixture negotiates right-angle turns and exit the reactor cavity while the corium particles carried by the flow impinge on walls and deposit in the subcompartment. For flow entering the in-core instrumentation (ICI) chase, the presence of the seal table prevents upward corium discharge through the instrument shaft. Even if the seal table fails due to overpressure in the reactor cavity, the flow first impinges on the wall at the end of the cavity and makes a 90-degree (upward) turn to the ICI chase where the seal table is located. It is shown that nearly all the entrained corium is captured by the impingement and only a small amount corium is released through the failed seal table. Therefore, the only flow path that leads directly to the upper containment without significant de-entrainment is the reactor pressure vessel (RPV) annulus. Because of the multiphase flow, the flow is choked in the reactor cavity, not in the reactor cavity access area. Thus, the fraction of the dispersed corium that enters the upper containment via the RPV annulus is given by the ratio of the area of RPV annulus, 1.96 m^2, to the total flow area, which is the sum of the area of PRV and the area of reactor cavity, 23.76 m^2, or 0.082.

2.2.6 Emergency Containment Spray Backup System

For a provision against a beyond-design-basis accident where either two SC pumps and two CS pumps or the IRWST is unavailable, the emergency containment spray backup system (ECSBS) is provided as an alternative to the CSS.

The ECSBS is designed to protect the containment integrity against overpressure and prevent the uncontrollable release of radioactive materials into the environment. The emergency containment spray flow path is from external water sources (the reactor makeup water tank, demineralized water storage tank, fresh water tank, or the raw water tank), through the fire protection system line via the diesel-driven fire pump, to the ECSBS line emergency connection located at ground level near the auxiliary building. The ECSBS flow rate provides sufficient heat removal to prevent containment pressure from exceeding 8.7 kg/cm^2 (123.7 psia).

3.0 SEVERE ACCIDENT PHENOMENOLOGICAL ANALYSIS AND ASSESSMENT

3.1 Hydrogen Control

3.1.1 Local Hydrogen Accumulation Evaluation and Summary

The evaluation of local hydrogen accumulation has been performed using MAAP 4.0.8 (Reference 10). This evaluation identifies local conditions within the containment with hydrogen exceeding 10%. The 10% hydrogen volume fraction is taken as a criterion for determining whether the accumulation should be further considered for deflagration-to-detonation transition (DDT) evaluation.

The analyzed accident sequences include five initiating events with a base case defined for each initiator type. For each initiator type, variations in the availability of accident mitigation systems are made such that their impact, if any, can be observed. The five initiating events are as follows:

- Large break LOCA (LBLOCA)
- Medium break LOCA (MBLOCA)
- Small break LOCA (SBLOCA)
- Station Blackout (SBO)
- Total loss of feedwater (TLOFW) that also represents total loss of essential service water (TLOESW) and loss of offsite power (LOOP)

These sequences represent the entire spectrum of severe accident conditions important to hydrogen accumulation and distribution in the containment. Other probabilistic risk assessment (PRA) accident sequences can be either represented or bounded by these analyzed sequences such that analysis of the specific sequence is not necessary.

All analyzed sequences have hydrogen generated in-vessel equivalent to 100% Metal Water Reaction (MWR) assuming minimum generation rate of 0.045 kg/s. The total amount of hydrogen generated during the analyzed sequences exceeded the equivalent of 100% MWR for sequences with MCCI.

The release of hydrogen was predicted at various containment compartments under severe accident scenarios. The possible hydrogen release points considered in the analysis include the hot-leg break (for LOCAs), IRWST spargers, failed reactor vessel lower head, and POSRV three-way valves. For LOCAs prior to vessel failure, hydrogen is released from the break in the hot leg into SG compartment. For non-LOCA sequences like SBO and TLOFW, hydrogen is first released to the IRWST through the pressure-lifted POSRVs. When three-way valve manual alignment is actuated, hydrogen is also released to SG compartment via the three-way valve. For high pressure sequences including SBO, TLOFW and SBLOCA, additional release point could come from the hot leg failure due to creep. After vessel failure, the failed lower head provides another hydrogen release point to the cavity area.

Figure 3-1 to Figure 3-4 show the hydrogen distribution in the dome region when applying all severe accident mitigation features. With all severe accident mitigation features available, the hydrogen

concentration is less than 10 percent.

In the screening of 10% hydrogen limit exceedance for various accident scenarios, it was found that

1. If the POSRV via the three-way valve is available, there is no limit exceedance anywhere in the containment except in the IRWST quarters and in the SG compartment for high pressure sequences (such as TLOFW, LOOP and TLOESW for delayed actuation timing of POSRV and three-way valve).

2. If the HG (i.e. igniters and PARs) is available and no containment sprays are actuated, there is no limit exceedance anywhere in the containment for all LOCA sequences.

3.1.2 DDT

The evaluation of the potential of DDT has been performed using MAAP 4.0.8. The DDT index condition refers to the gaseous mixture that has compositions that have the condition of flame acceleration (FA) and the detonation cell size that allow DDT to develop within the characteristic length of the compartment. When the DDT index condition is detected in the accident simulation, it means that all necessary conditions for DDT are present. Whether or not the condition is sufficient for DDT is beyond the capability of the criteria. However, as a conservative approach, one may assume that the presence of DDT index condition locally or globally in the containment means DDT will occur if ignited.

It was found that there is no DDT potential anywhere in the containment if the POSRV via the three-way valve is available.

3.1.3 AICC Pressure Evaluation

The evaluation of adiabatic isochoric complete combustion (AICC) pressure has been performed. The upper bound value for the pressure load as a result of slow deflagrations of hydrogen produced from 100% metal-water reaction and uniformly distributed in the containment is determined to be 7.0 kg/cm^2 (99.84 psia). The upper bound AICC pressure load was determined based on the maximum-steam-concentration pre-burn condition allowed by the flammability limit curve. At this maximum-steam-concentration pre-burn condition, the steam volume fraction is 44.02%, and the volume fraction of hydrogen produced from 100% metal-water reaction is just 6.31%. The combustion would not be a complete burn if hydrogen concentration is below 8%. Hence, the calculated value of 7.0 kg/cm^2 (99.84 psia) for the upper bound AICC pressure load is also conservatively underlined by the complete combustion assumption. Considering the safety margin of the APR1400 containment, for the FLC, the pressure resulting from 100 percent metal water reaction of fuel cladding and uncontrolled hydrogen burning is determined as 8.7 kg/cm^2 (123.7 psia).

The details of hydrogen control analyses are provided in Appendix A.

3.2 MCCI and Debris Coolability

Potential threat to containment integrity due to MCCI was studied for the APR1400 plant. The postulated severe accident scenarios are the so-called “wet” cases, where the cavity flooding system (CFS) is assumed available to flood the reactor cavity following reactor core damage.

The computational tool used for this study is the severe accident code MAAP 4.0.8. Review of the modeling features of this code indicates that two modeling parameters, FCHF and ENT0C, must be calibrated against more sophisticated MCCI codes to achieve conservative predictions of key variables important to containment integrity, including concrete ablation depth and containment pressure. The MCCI code, CORQUENCH 3.03, was selected as the basis of the calibration process. This code has detailed modeling features for a corium-water interaction and a melt eruption. Based on the analysis results using this code, the ablation depth of a limestone common sand (LCS) concrete floor is within 0.3 m (1 ft) for the bounding large LOCA scenario. This result is considered conservative because it ignores initial molten corium jet breakup. If this effect was considered, the ablation depth would have been much smaller. The MAAP 4.0.8 input value of FCHF is calibrated to be 0.0235, which still leads to a conservative result for the ablation into the concrete floor, compared to the ablation depth predicted by CORQUENCH for the chosen bounding scenario. This ablation depth is much smaller than the depth of containment liner (about 0.9 m (3 ft)). Therefore, release of fission products from containment due to ablation damage is unlikely. The value of ENT0C, which is the entrainment coefficient in the initial jet breakup model, is set to a very small number, which is equivalent to ignoring the effect of jet breakup. It is conservative with respect to concrete ablation, because ignoring the jet breakup will generate a high temperature corium pool. The impact of this small value of ENT0C on containment pressurization has also been assessed and found consistent with modeling expectations.

The five severe accident scenarios were selected based on their core damage frequencies from Level I PRA analyses and potential bounding features, including such sequences as: loss of essential service water, loss of AC power with failure of auxiliary feedwater, medium break LOCA, and large break LOCA. Table 3-1 lists the identifiers of the scenarios along with brief scenario descriptions. For each scenario, up to 24 hours of the transient was simulated. The modeling parameter FCHF was set to 0.0235 and ENT0C was set to 1×10^{-5}.

Figure 3-5 and Figure 3-6 show the ablation depths and containment pressures of the scenarios for the first 24 hours. The large break LOCA is the most bounding scenario and it produces the maximum ablation depth and containment pressure, about 0.24 m (0.78 ft) and 7.55 kg/cm^2 (107.4 psia) respectively. According to SECY 93-087, the containment liner must be protected during MCCI, and the pressure resulting from MCCI should meet the FLC (Factored Load Category) requirement for 24 hours. The depth where the liner is located in the APR1400 cavity is about 0.9 m (1 ft). If the pressure does not exceed 8.7 kg/cm^2 (123.7 psia) in the APR1400, the FLC requirement is met. It can be seen that the ablation depth and containment pressure resulting from MCCI do not exceed the limits even for the most bounding scenario. For the MAAP analyses, it is assumed that the cavity floor is a flat surface without a sump. The CORQUENCH code is used to consider a cavity sump. Based on the CORQUENCH analyses, the corium in the sump is quenched before the ablation depth reaches the containment liner, with consideration of a cavity floor made of LCS concrete.

The details of MCCI and debris coolability analyses are provided in Appendix B.

3.3 High Pressure Melt Ejection and Direct Containment Heating

3.3.1 Direct Containment Heating

While DCH experiments have not been directed towards examining the consequences of DCH occurring in the APR1400 reactor, significant DCH pressurization of the APR1400 containment is judged to be most unlikely. It has been well established by the DCH laboratory research that containment structures and compartments have a first order mitigating influence on the pressurization potential of DCH. The number of compartments between the cavity and the containment dome of the APR1400 are not at all conducive to supporting a strong DCH event, as demonstrated in this report.

According to the NUREG/CR-6075 (Reference 12) methodology, the DCH issue is considered resolved if the containment failure probability due to DCH, obtained through a probabilistic evaluation of phenomenological analysis and its uncertainties, is found to be lower than a certain threshold value. The severe accident scenario that can lead to DCH in the APR1400 is identified as a small LOCA with RCS re-pressurization due to operator intervention and three representative scenarios were considered.

For each of the three scenarios, zero (0) containment failure case has resulted from 10,000 trials. Based on this outcome, the CCFP in the APR1400 due to DCH is estimated to be less than 0.01% (0.0001). This indicates that the APR1400 meets the success criterion established in NUREG/CR-6338 (Reference 13) for PWR large dry containment, where DCH problem is considered resolved if CCFP is less than 1% (0.01).

3.3.2 Rapid Depressurization Analysis

The RD function is diverse roles during design basis accidents (DBAs), beyond design basis accidents (BDBAs), and even severe accidents. Depressurization for a severe accident using the POSRVs enables operation of the low pressure systems, such as the SCS, thus enabling additional means of core cooling. The RCS is maintained at a low pressure for cases involving a loss of core heat removal functions. This prevents the occurrence of HPME phenomenon, and prolongs the reactor vessel integrity.

In this report, the main focus is on the following three conditions that are required for a successful de-pressurization: (i) RD function is performed after the CET exceeds 922.04K (1,200°F); (ii) the POSRVs can only be guaranteed to open when the temperature is lower than 644K (700°F); and, (iii) the RCS should be depressurized below 1.72 MPa (250 psia) before the RV fails. The depressurization evaluation was considered successful if these three conditions were met.

HPME can be prevented for the 10 analyzed sequences. For certain sequences, vessel failure occurs at a sufficiently low pressure, while for other sequences, operator intervention is necessary. Operation of only two POSRVs within a half hour of the plant entering a severe accident is sufficient for all the sequences that are being considered. The results are in compliance with SECY-93-087.

The details of the analyses for DCH and rapid depressurization capability are provided in Appendix C-1 and Appendix C-2, respectively.

3.4 Fuel-Coolant Interaction

Steam explosion is a remaining risk-significant issue in nuclear power plant due to the threatening of integrity of the defense-in-depth barriers by explosive dynamic loadings that could lead to release radioactive fission product to public.

Therefore, it is needed to evaluate steam explosion risk when a new design reactor like the APR1400 is considered. The steam explosion risk can be categorized into two groups in terms of the locations of steam explosion initiation, that is, in-vessel and ex-vessel.

Comprehensive analyses on both in-vessel steam explosion (IVSE) and ex-vessel steam explosion (EVSE) are conducted using the TEXAS-V computer code and the most updated technical information on the phenomena, risk evaluation, and analysis methodology are collected and utilized for the assessment of steam explosion risk in the APR1400 design. The analysis results can be used for the evaluation of structure integrity relevant to the steam explosion issue in the APR1400 design, especially in the case of ex-vessel severe accident progression.

The IVSE loads are evaluated using the TEXAS-V computer code. Based on this analysis, the integrity of the reactor vessel is assessed using the ABAQUS computer code. This analysis concludes that IVSE provides no threat to fail RPV. For the EVSE analysis, the risk due to EVSE is analyzed using the TEXAS-V computer code and the reactor cavity structure with EVSE pressure loading is evaluated using LS-DYNA. The maximum displacements, concrete cracks, liner plate stresses, reinforcement re-bar stresses, RPV column support anchor bolt stresses and strains due to the EVSE loading on the reactor cavity are summarized in Table 3-2. EVSE also provides no threat to fail the reactor cavity.

Details of the FCI analysis are provided in Appendix D.

3.5 Containment Performance

As stated in SECY 93-087, the containment stress should not exceed Factored Load Category (FLC) presented in ASME Code, Section III, Division 2, Subarticle CC-3720, approximately 24 hours following the onset of core damage. In addition, USNRC RG 1.216 requires the containment integrity analysis to consider the FLC requirement presented in requirement position 5 of USNRC RG 1.7.

Based on an AICC analysis with the assumption of the unavailability of hydrogen mitigation features, the pressure resulting from 100-percent metal water reaction of fuel cladding and uncontrolled hydrogen burning was determined as 8.7 kg/cm^2 (123.7 psia) including the safety margin of the APR1400 containment. For the pressure of 8.7 kg/cm^2 (123.7 psia), the consequential structural analysis for the APR1400 containment showed that the strain of containment liner plate did not reach the limit of ASME Code, Section III, Division 2, Subarticle CC-3720. Therefore, the criterion for the containment performance was determined as 8.7 kg/cm^2 (123.7 psia) with regard to the FLC requirement.

The response of the APR1400 containment to gradual pressurization under severe accident conditions was investigated by using the MAAP 4.0.8 code for a range of possible sequences. The containment pressurization sequences considered in the analysis included the overpressurization by steam and non-condensable gases.

The sequences analyzed covered the most likely severe accident initiators for the APR1400. In accordance with the design of the APR1400, Cavity Flooding System (CFS) and ECSBS were included in the sequences.

Among the sequences, a large break LOCA resulted in the highest pressure in containment at 24 hours following the onset of core damage. Due to the break, primary system pressure rapidly drops and causes the reactor to scram. The break flow from the RCS rapidly pressurizes containment and the core is quickly uncovered. Due to the low primary system pressure, the coolant in SITs is injected temporarily recovering and cooling the core. Once the inventory in SITs has been depleted, the core uncovers again and begins to heat up. Core exit temperature soon reaches 922.04K (1,200˚F), signaling the onset of core damage.

After 30 minutes following the onset of core damage, the actuation of the CFS, which allows the gravity-driven water flow from the IRWST into the reactor cavity, is assumed. The core begins to melt and relocate to the reactor vessel lower plenum. This produces a momentary pressure spike in the RCS due to the rapid steaming of the water pool that exists in the lower plenum.

The molten corium in the reactor vessel lower plenum eventually causes the vessel to fail at low pressure. The core debris slumps into the flooded reactor cavity. The corium causes the water pool in containment to heat up and boil. Due to the gradual steam generation in the cavity and non-condensable gas generation during MCCI, containment pressure steadily increases. Heat removal by the overlying water pool quenches the corium and eventually halts concrete ablation. At the time of complete quenching of the corium, the concrete ablation depth is much less than the depth of the steel containment liner in the reactor cavity.

Considering the further generation of steam after the concrete ablation stopped, the containment pressure does not reach the 8.7 kg/cm^2 (123.7 psia) within 24 hours after the onset of core damage. The ECSBS is actuated 24 hours after the onset of core damage at a flow rate of 2.84 m^3/min (750 gpm). It begins to cool and depressurize the containment atmosphere as shown in Figure 3-7. The result shows that the ECSBS is capable of controlling containment pressure for a period of 48 hours after 24 hours following the onset of core damage. The maximum pressure and temperature following the initial 24 hour period are enveloped by the maximum pressure and temperature during the initial 24 hour period.

In summary, the most likely severe accident initiators for the APR1400 are analyzed in the analysis. For initial 24 hours following the onset of core damage, the pressure of the APR1400 containment does not exceed 8.7 kg/cm^2 (123.7 psia), so that the strain of containment liner plate does not reach the limit of ASME Code, Section III, Division 2, Subarticle CC-3720.

Following the initial 24 hours period, the actuation of ECSBS enables to maintain the containment pressure lower than 8.7 kg/cm^2 (123.7 psia). It can be concluded that the APR1400 containment is prevented from the uncontrolled release of fission products into the environment.

The details of the analysis for containment performance are provided in Appendix E.

3.6 Equipment Survivability

The applicable regulations and criteria for equipment survivability assessment are reviewed. The required equipment and instrumentation for equipment survivability assessment are identified. Bounding radiation and thermal-hydraulic environmental conditions during a severe accident are determined. Each piece of essential equipment and instrumentation is assessed for equipment survivability.

Several potential design modifications may be considered if the equipment survivability of a given item cannot be demonstrated by analysis or use of existing test data. Typically localized high temperature conditions due to hydrogen burns within containment during postulated severe accident sequences constitute the greatest challenge to the component's survivability. Thus, detailed knowledge of the containment's response for a variety of bounding severe accident sequences can identify locations within the containment that have the least challenging thermal conditions. The relocation of some components to a position with less severe thermal conditions may better ensure their survivability.

Some devices, such as hydrogen igniters, must be appropriately located to ensure the successful operation of the system. Thus, relocating igniters may not be a useful approach given that they are intended to control the hydrogen in the regions of the containment where they are placed. These cases may require the use of redundancy or an understanding of the functional interval required for their operation.

Other design modifications include adding a thermal shield or wall to provide protection against the local severe accident environment, and modifying the component (replacing non-metallic material or changing its configuration).

The details of the methodologies for equipment survivability (ES) assessment are provided in Appendix F.

Table 3-1 Selected Severe Accident Scenarios for MCCI Analyses

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TS

Table 3-2 Summary of Cavity Structural Integrity Analysis Result

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Figure 3-1 Mole Fraction of Hydrogen in the Dome Region for LBLOCA

TS

Figure 3-2 Mole Fraction of Hydrogen in the Dome Region for SBLOCA

TS

Figure 3-3 Mole Fraction of Hydrogen in the Dome Region for SBO-Three-way valve

TS

Figure 3-4 Mole Fraction of Hydrogen in the Dome Region for TLOFW-Three-way valve

TS

Figure 3-5 Ablation Depths within 24 Hours for Several Severe Accident Scenarios

TS

Figure 3-6 Containment Pressures within 24 Hours for Several Severe Accident Scenarios

TS

Figure 3-7 Containment Pressure for Large Break LOCA with ECSBS Actuated 24 hours after Onset of Core Damage

4.0 CONCLUSIONS

The preventive and mitigative design features implemented in the design of ARP1400 include the HG, CFS, ECSBS, and structural designs of the containment and reactor cavity for severe accident loads.

The structural design of the reactor cavity is intended to prevent failure from ex-Vessel severe accidents. In particular, the reactor cavity was designed to meet the requirements for prevention and mitigation of phenomena such as HPME/ DCH, EVSE and MCCI. The main design characteristics include a core debris chamber inside the reactor cavity, a convoluted gas vent path, large floor area, and the CFS.

The APR1400 are designed for safety functionality even during a very unlikely severe accident scenario. The design includes accident prevention and mitigation features that are based on the analyses of severe accidents. Sufficient robustness is provided in the design so that ample time exists for operator action to mitigate the consequences of a severe accident and minimize the radiological releases into the environment.

In the severe accident analysis for the APR1400, emphases have been laid on the following six topics: hydrogen control, MCCI and debris coolability, HPME and DCH, FCI, containment performance, and ES in order to ensure design adequacy and regulatory compliance. The regulatory compliance involves strict adherence to requirements and reasonable conformance with the guidelines set forth in USNRC, SECY-93-087 "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs" and other regulatory requirements such as 10 CFR 100, GDC of 10 CFR 50 Appendix A, and the TMI-related requirements of 10 CFR 50.34(f), 10 CFR 50.44, NRC RG 1.216, and SECY-90-016 (Reference 7). The six phenomenological issues considered here are listed and summarized below.

1) Hydrogen control
   In hydrogen control analysis, detailed containment hydrogen distribution was analyzed. The investigation of in-Vessel and ex-Vessel hydrogen release characteristics for different accident sequences considered the potential for DDT. It was concluded from the analysis that the APR1400 would have the capability to withstand the hydrogen generation and combustion risks following most severe core damage accident scenarios.

2) MCCI and debris coolability
   As a result of the MCCI analyses using the MAAP 4.0.8 and CORQUENCH computer codes, it was determined that the integrity of the containment liner would not be challenged by core concrete attack and it would maintain its integrity under severe accident conditions with the CFS operation.

3) HPME and DCH
   The assessment of containment integrity against the DCH risk evaluations concluded that the predicted CCFP values for scenarios V, Va, and VI were smaller than the criterion of 0.01 presented in NUREG/CR-6338. Accordingly, it was concluded that the likelihood of containment failure due to DCH loads was negligibly small. Based on the results of the

rapid depressurization analyses for preventing HPME under high RCS pressure conditions, it was confirmed that, if the operator could open two POSRVs at appropriate timings after core exit temperatures exceed 922.04K (1,200°F), the RCS pressure would decrease below the DCH cutoff pressure of 17.6 kg/cm^2 (250 psia).

4) FCI
The risk due to IVSE was analyzed by evaluating IVSE loads using the TEXAS-V computer code. Subsequently, the integrity of the reactor vessel was assessed using the ABAQUS computer code. It was confirmed that the integrity of the reactor vessel could be maintained even if an IVSE occurred. In the event there is a water pool in the reactor cavity due to the operation of the CFS, the resulting risk from EVSEs was analyzed using the TEXAS-V computer code. LS-DYNA computer code is employed in order to analyze the response of the cavity structures according to EVSE loads. All these evaluations confirmed that the integrity of the reactor cavity structure would be maintained even if ex-Vessel FCI occurred.

5) Containment performance capability
The criterion for the containment performance was determined as 8.7 kg/cm^2 (123.7 psia) with regard to the FLC requirement. For initial 24 hours following the onset of core damage, the pressure of the APR1400 containment didn't exceed 8.7 kg/cm^2 (123.7 psia) for which the strain of containment liner plate didn't reach the limit of ASME Code, Section III, Division 2, Subarticle CC-3720. After the initial 24 hours period, the actuation of ECSBS was able to maintain the containment pressure lower than 8.7 kg/cm^2 (123.7 psia). This prevents the uncontrolled release of fission products into the environment.

6) ES
The ES performance and operability of vital equipment and instrumentation for severe accident prevention and mitigation were verified. This was accomplished by calculating the severe accident environmental conditions inside the containment for in-vessel and ex-vessel events, and then assessing equipment and instrumentation survivability during designated time intervals.

5.0 REFERENCES

1. “Policy, Technical, and Licensing issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) designs,” SECY-93-087, USNRC, April 1993.
2. “Reactor Site Criteria,” 10CFR100, USNRC, July 2012.
3. “General Design Criteria for Nuclear Power Plants,” 10CFR50 Appendix A, USNRC, November 2012.
4. “Contents of Applications; Technical Information,” 10CFR50.34, USNRC, November 2012.
5. “Combustible Gas Control for Nuclear Power Reactors,” Title 10, Code of Federal Regulations, Part 50.44, U.S. Nuclear Regulatory Commission, Washington, DC, November 2012.
6. “Containment Structure Integrity Evaluation for International Pressure Loading above Design Basis Pressure” USNRC RG 1.216, Aug. 2010
7. “Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements,” SECY-90-016, USNRC, June 1990.
8. “Rules of Construction of Nuclear Facility Components - Code for Concrete Containments,” ASME Section III Division 2, ASME, July 2006.
9. DOE/ID-10271, “Prevention of Early Containment Failure due to High Pressure Melt Ejection and Direct Containment Heating for Advanced Light Water Reactors,” March 1, 1990.
10. Fauske & Associates, 2012(a), MAAP4 Modular Accident Analysis Program for LWR Power Plants, Transmittal Document for MAAP4 Code Revision MAAP 4.0.8, FAI/12-0005, February.
11. Breitung, W., et al. (2000). Flame Acceleration and Deflagration to Detonation Transition (DDT) in Nuclear Safety. State-of-the-Art Report, OECD Nuclear Energy Agency, Ref. NEA/CSNI/R/2000/7.
12. Pilch, M. M., Yan, H. and Theofanous, T. G., 1994, "The Probability of Containment Failure by Direct Containment Heating in Zion," NUREG/CR-6075, SAND93-1535, Sandia National Laboratory, Albuquerque, NM.
13. Pilch, M. M., Allen M. D., and Klamerus E. W., “Resolution of the Direct Containment Heating Issue for All Westinghouse Plants With Large Dry Containments or Subatmospheric containments,” NUREG/CR-6338, SAND95-2381, Sandia National Laboratories, Feb. 1996.