Appendix-F

Severe Accident Analysis Report for Equipment Survivability Evaluation

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1.0 Introduction

Severe accidents, in which the reactor core is damaged, can lead to elevated temperatures and pressures in the containment, with significant concentrations of combustible hydrogen. In order to achieve a safe stable state, even in the event of vessel failure, it is necessary that certain equipment and instrumentation continue to function under these extreme conditions. The purpose of this Equipment Survivability (ES) assessment is to show that there is reasonable assurance that the equipment and instrumentation used to mitigate and monitor severe accident progression will perform their intended functions under the harsh environmental conditions of severe accidents.

The approach used in the APR1400 certification to demonstrate equipment survivability includes several steps:

- Identify the high level actions used to achieve a controlled, stable state,
- Define the accident time frames,
- Determine the equipment and instrumentation used to diagnose, perform and verify high level actions in each timeframe,
- Determine the bounding environment, and
- Demonstrate with reasonable assurance that the equipment will survive to perform its function in the severe environment.

Chapter 2 presents the applicable regulations and criteria for equipment survivability analysis. In Chapter 3, the equipment and instrumentation that require equipment survivability analysis are determined. In Chapter 4, the bounding thermal-hydraulic and radiation conditions during a severe accident are determined in each containment node. In Chapter 5, equipment survivability is evaluated for essential equipment and instrumentation.

2.0 Applicable Regulations and Criteria

Under design-basis accident conditions, without core damage, the ability of safety-related equipment to perform their required function is demonstrated through “equipment qualification.” Under severe accident conditions (beyond design basis) the environmental conditions are generally more challenging. The Nuclear Regulatory Commission (NRC) has developed criteria, described below, to provide reasonable assurance that necessary equipment will continue to function (i.e., survive) for the required time period during a severe accident. Thus these criteria are used to demonstrate “equipment survivability.”

2.1 SECY-90-016

On January 12, 1990, the NRC staff issued SECY-90-016 which requested Commission approval for the staff’s recommendations concerning proposed departures from current regulations for the evolutionary light-water reactors (LWRs). The issues in SECY-90-016 were significant to reactor safety and fundamental to the NRC decision on the acceptability of evolutionary LWR designs. The positions in SECY-90-016 were developed as a result of the following activities:

- NRC reviews of current-generation reactor design and evolutionary LWRs,
- Consideration of operating experience, including the TMI-2 accident,
- Results of PRAs of current-generation reactor designs and the evolutionary LWRs,
- Early efforts conducted in support of severe accident rulemaking, and
- Research to address previously identified safety issues.

The Commission approved some of the positions stated in SECY-90-016 and provided additional guidance regarding others in a Staff Requirements Memorandum (SRM) dated June 26, 1990. Section III (Mitigative Feature Issues) Part F (Equipment Survivability) of SECY-90-016 provides the following regulatory guidance.

F. Equipment Survivability

With regard to the Commission’s request concerning “The measures to ensure that systems and equipment required only to mitigate severe accidents are available to perform their intended function (e.g., environmental qualifications),” the staff believes that features provided for severe-accident protection (prevention and mitigation) only (not required for design basis accidents) need not be subject to (a) the 10 CFR 50.49 environmental qualification requirements (b) all aspects of 10 CFR Part 50, Appendix B quality assurance requirements, or (c) 10 CFR Part 50, Appendix A redundancy/ diversity requirements. The reason for this judgment is that the staff does not believe that severe core damage accidents should be design basis accident (DBA) in the transitional sense that DBAs have been treated in the past.

Notwithstanding that judgment, however, mitigation features must be designed so

there is reasonable assurance that they will operate in the severe-accident environment for which they are intended and over the time span for which they are needed. In instances where safety related equipment, (which is provided for design basis accidents) is relied upon to cope with severe accidents situations; there should also be a high confidence that this equipment will survive severe accident conditions for the period that is needed to perform its intended function. However, it is not necessary for redundant trains to be qualified to meet this goal.

During the review of the credible severe accident scenarios for specific ALWR designs, the equipment needed to perform mitigative functions, and the conditions under which the mitigative systems must function, will be identified. Equipment survivability expectations under severe accident conditions should include consideration of the circumstances of applicable initiating events (e.g., station blackout, earthquakes) and the environment (e.g., pressure, temperature, radiation) in which the equipment is relied upon to function. The required system performance criteria will be based on the results of these design-specific reviews. In addition, the staff concludes that severe-accident mitigation equipment for evolutionary ALWRs should be capable of being powered from an alternate power supply as well as from the normal Class 1E onsite systems. Appendices A and B to Regulatory Guide 1.155, "Station Blackout", provide guidance on the type of quality assurance activities and specifications which the staff concludes are appropriate for equipment utilized to prevent and mitigate the consequences of severe accidents.

2.2 SECY-93-087

On April 3, 1993, the NRC staff issued SECY-93-087 which sought Commission approval for the staff's positions pertaining to evolutionary and passive LWR design certification policy issues. This paper evolved from SECY-90-16. SECY-93-087 addresses various preventive features and issues, as well as mitigative features including Equipment Survivability.

The Commission approved some of the staff positions stated in SECY-93-087 and provided additional guidance regarding others in the SRM dated July 21, 1993. Section I (SECY-90-016 Issues), Part L (Equipment Survivability) of SECY-93-087 provides the following regulatory guidance.

In SECY-90-016, the staff recommended that the Commission approve the position that features provided only for severe-accident protection need not be subject to the environmental qualification requirements of 10 CFR 50.49; quality assurance requirements of 10 CFR Part 50, Appendix B; or redundancy/diversity requirements 10 CFR Part 50, Appendix A. The reason for this judgment is that the staff does not believe that severe core damage accidents should be treated in the same manner traditionally used for design-basis accident (DBAs) because of significant differences in their likelihood of occurrence. However, SECY-90-016 further stated that mitigation features must be designed to provide reasonable assurance that they will operate in the severe-accident environment for which they are intended and over the time span for which they are needed. In instances where safety-related equipment provided for DBAs is relied upon to cope with severe-accident situations, there should also be a high confidence that this equipment will survive severe-accident conditions for the period that is needed to perform its intended function.

During the review of the credible severe-accident scenarios for ALWR designs, the staff will evaluate the ALWR vendors identification of the equipment needed to perform mitigative functions and the conditions under which the mitigative systems must operate. Equipment survivability expectations under severe-accident conditions should consider the circumstances of applicable initiating events (such as station blackout or earth quakes) and the environment (including pressure, temperature, and radiation) in which the equipment is relied upon to function. The required system performance criteria will be based on the results of these design-specific reviews.

In its SRM of June 26, 1990, the Commission approved the staff's position. In its letter of May 6, 1991, the staff clarified its position that these criteria would be applied to features provided only for severe accident mitigation.

The EPRI requirements document and the evolutionary ALWR designers have indicated that their submittals are consistent with these criteria. The passive ALWR vendors have indicated that their designs will comply with the applicable EPRI requirements document. In its letter of August 17, 1992, ACRS agreed with the staff position discussed above.

2.3 Severe Accident Resolution

The basis for resolving the severe accident issues associated with the APR1400 design are the requirements of 10 CFR Part 52, as well as the NRC guidance in SECY-93-087, SECY-96-128, and SECY-97-044, as approved by the Commission.

As noted above, equipment that is classified as safety-related must perform its intended function under the environmental conditions associated with design-basis accidents; the level of assurance is demonstrated through "equipment qualification" and is governed by the requirements in 10 CFR 50.49. Severe accident environmental conditions are generally expected to be more extreme than conditions from design-basis events, so the NRC has established criteria to provide a reasonable level of assurance that necessary equipment will survive a severe accident for the time period it is required. This is referred to as "equipment survivability" and it is fundamentally different from equipment qualification.

The NRC requires the plant designer to perform analyses to demonstrate reasonable assurance of equipment survivability. The Commission approved the position that for the review of the credible severe-accident scenarios for ALWRs they will evaluate the design certification applicant's identification of equipment needed to perform mitigative functions as well as conditions under which the mitigative system must operate.

Beyond design basis events can generally be categorized into in-vessel and ex-vessel severe accidents, depending upon whether or not core damage leads to vessel failure. The environmental conditions resulting from these events are generally more limiting than those from design basis events. The applicable criteria for mechanical and electrical equipment and instrumentation required for recovery from in-vessel severe accidents are provided in 10 CFR 50.34(f) and are summarized below:

- Part 50.34(f)(2)(ix)(c) states that equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity

will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100 percent fuel-clad metal-water reaction including the environmental conditions created by activation of the hydrogen control system.

- Part 50.34(f)(3)(v) states that systems necessary to ensure containment integrity shall be demonstrated to perform their function under conditions associated with an accident that releases hydrogen generated from 100 percent fuel-clad metal-water reaction.

- Part 50.34(f)(2)(xvii) requires instrumentation to measure containment pressure, containment water level, containment hydrogen concentration, containment radiation intensity, and noble gas effluents at all potential accident release points.

- Part 50.34(f)(2)(xix) requires instrumentation adequate to monitoring plant conditions following an accident that includes core damage.

The applicable criteria for mechanical and electrical equipment and instrumentation required to mitigate the consequences of ex-vessel severe accidents are discussed in the "Equipment Survivability" section of SECY-90-016 (Section III Part F, see above) and its SRM and are summarized as follows:

- Features provided only for severe-accident protection (prevention and mitigation) need not be subject to the 10 CFR 50.49 environmental qualification requirements, 10 CFR Part 50, Appendix B quality assurance requirements, and 10 CFR Part 50, Appendix A redundancy/diversity requirements.

- Mitigation features must be designed to provide reasonable assurance that they will operate in the severe accident environment for which they are intended and over the time span for which they are needed. In cases where safety related equipment (provided for DBAs) is relied upon to cope with severe accident situations, there should be reasonable assurance that this equipment will survive accident conditions for the period that is needed to perform its intended function.

- Severe accident mitigation equipment should be capable of being powered from an alternate power supply as well as from the normal Class 1D onsite systems. Appendices A and B to Regulatory Guide 1.155, "Station Blackout," provide guidance on the type of quality assurance activities and specifications which are appropriate for equipment utilized to prevent and mitigate the consequences of severe accidents.

3.0 Identification of Required Equipment and Instrumentation for Equipment Survivability Assessment

In order to identify the required equipment and instrumentation for assessment of equipment survivability under severe accident conditions, the high level actions needed to achieve a controlled,

stable state in the plant are first determined. Then, the severe accident progression is divided into accident time frames. Lastly, the equipment and instrumentation used to diagnose, perform and verify high level actions in each time frame are identified.

2.4 High Level Actions to Achieve a Controlled, Stable State in Plant

During a core damage accident, operators are confronted with multiple failures of essential safety equipment and/or operator errors. For operators to effectively cope with this plant condition, they must make use of the available equipment and instrumentation with the ultimate goal of achieving safe shutdown of the plant and maintaining containment integrity. These equipment and instrumentation can be grouped according to their general function: RCS inventory control, RCS heat removal, reactivity control, and containment integrity.

RCS inventory control is primarily provided by the safety injection (SI) system. Should the SI system not be available, and if the RCS has depressurized below about 200 psia, then inventory control can also be provided via realignment of the containment spray or shutdown cooling (SC) system pumps to operate in injection mode.

RCS heat removal following a severe accident will be accomplished by establishing the auxiliary feedwater (AFW) system to at least one steam generator, or using “Feed and Bleed Operation”, where the operator feeds liquid inventory into the RCS via SI and bleeds off steam and/or water. Once a sufficiently low pressure is established in the RCS, long-term heat removal can be established via Shutdown Cooling (SC) functions using either the SC system or containment spray (CS) pumps, with associated heat exchangers.

Reactivity control is provided by insertion of control rods (typically done early in the transient) and by assuring the delivery of sufficiently borated water into the RCS.

Given the highly reliable containment isolation systems, containment integrity for the APR1400 depends on restoration of the containment heat removal function and the successful performance of seals in the Electric Penetration Assemblies (EPAs), Personnel Air Lock (PAL), and equipment hatch. If the reactor vessel fails and molten corium relocates into the containment, the cavity flooding system minimizes the Molten Core-Concrete Interaction (MCCI) and resulting non-condensable gas generation. Containment integrity also depends on hydrogen control and mitigation because hydrogen burns can create short but extreme temperature conditions in the containment.

2.5 Severe Accident Time Frames

Severe accident progression is divided into four time frames (identified here as Time Frames 0 though 3) to categorize the high level actions required and the corresponding equipment and instrumentation used. Also, the environmental conditions inside the reactor vessel and in the containment are related to these specific time frames.

2.5.1 Time Frame 0 – Pre-Core Uncovery

Time Frame 0 is the period from the initiation of an accident to the time of core uncovery. The high level action in this phase is to recover the reactor coolant system inventory and to remove heat so

as to maintain reactor vessel integrity. The equipment and instrumentation used in this phase are:

- Auxiliary Feedwater System (AFWS),
- Safety Injection System (SIS),
- Chemical and Volume Control System (CVCS) via charging pumps,
- Shutdown Cooling System (SCS),
- RCS/Pressurizer Pressure Indicators, and
- Reactor Vessel Level Measurement System (RVLMS).

Equipment survivability during Time Frame 0 is covered under the design basis equipment qualification (EQ).

2.5.2 Time Frame 1 – Core Heatup Phase

Time Frame 1 is the period from core uncovery to the onset of significant core damage. The onset of significant core damage is typically identified by a core-exit gas temperature measurement exceeding 1200 ^{o}F. Core overheating is accompanied by rapid fuel-clad metal-water reaction, producing hydrogen. The high level action in this phase is to recover the reactor coolant system inventory and to remove heat so as to maintain reactor vessel integrity.

The equipment and instrumentation used in this phase are:

- Auxiliary Feedwater System (AFWS),
- Safety Injection System (SIS),
- Chemical and Volume Control System (CVCS) via charging pumps,
- Shutdown Cooling System (SCS),
- RCS/Pressurizer Pressure Indicators,
- Reactor Vessel Level Measurement System (RVLMS), and
- Core Exit Thermocouples (CETs).

Equipment survivability during Time Frame 1 is covered under the design basis equipment qualification except for equipment and instrumentation inside the RCS.

2.5.3 Time Frame 2 – In-Vessel Severe Accident Phase

Time Frame 2 is the period from the onset of significant core damage to reactor vessel failure. A significant amount of hydrogen is generated in the reactor vessel during this phase. The core melts and relocates to the lower plenum.

The high level action in this time frame is to recover the reactor coolant system inventory and heat

removal, and to maintain containment integrity. To this end, the operator will open the POSRVs to depressurize the reactor vessel to allow SCS pumps to inject water into the RCS and to thereby minimize the possibility of Direct Containment Heating (DCH) if the reactor vessel fails. In addition, the operator will align the 3-way valves on POSRVs to direct the hydrogen-rich effluent to the steam generator compartment and containment dome, instead of the IRWST and annular compartment. The operator will turn on the igniters. The operator will also actuate the Cavity Flooding System (CFS) in anticipation of vessel failure and subsequent relocation of corium to the reactor cavity floor.

The equipment and instrumentation used to maintain reactor vessel integrity are:

- Auxiliary Feedwater System (AFWS),
- Safety Injection System (SIS),
- Chemical and Volume Control System (CVCS) via charging pumps,
- Shutdown Cooling System (SCS),
- RCS/Pressurizer Pressure Indicators,
- Reactor Vessel Level Measurement System (RVLMS),
- Core Exit Thermocouples (CETs), and
- Rapid Depressurization and Vent System (RDVS) and 3-way Valves.

The equipment and instrumentation needed to maintain containment integrity are:

- Hydrogen Mitigation System (HMS) – PARs and igniters,
- Cavity Flooding System (CFS),
- Containment penetrations – equipment hatch, personnel air lock, electrical penetration assemblies, and mechanical penetrations,
- Hydrogen monitors,
- Post Accident Sampling System (PASS),
- High level radiation monitors, and
- Containment temperature RTDs.

2.5.4 Time Frame 3 - Ex-Vessel Severe Accident Phase

Time Frame 3 is the period from reactor vessel failure to the establishment of a controlled, stable state. The high level actions in this phase are to re-establish a coolable corium configuration on the containment floor, maintain containment integrity, and monitor the accident progression. To this end the operator will turn on the Emergency Containment Spray Backup System (ECSBS) if the controlled, stable state is not achieved 24 hours after the onset of core damage. The ECSBS is

designed to protect the containment integrity against challenge due to overpressure and prevent the uncontrollable release of radioactive material into the environment. It accomplishes this by removing heat for a period of approximately 48 hours in duration following the first 24-hour period after the onset of core damage.

The equipment and instrumentation needed to maintain containment integrity are:

- Hydrogen Mitigation System (HMS) – PARs and igniters,
- Containment Spray System (CSS),
- SCS pumps as a backup to containment spray pumps,
- Emergency Containment Spray Backup System (ECSBS),
- Containment penetrations – equipment hatch, personnel air lock, electrical penetration assembly, and mechanical penetrations,
- Hydrogen monitors, and
- Post Accident Sampling System (PASS).

2.6 Equipment and Instrumentation that Require Equipment Survivability Assessment

The list of equipment and instrumentation that require equipment survivability assessment are summarized in Table 2-1.

Table 2-1 Systems and Equipment/Instrumentation Requiring Equipment Survivability Assessments

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3.0 Severe Accident Environments

3.1 Method Discussion

In order to determine the environmental conditions expected during a severe accident, the MAAP 4.0.8 code is used to analyze a range of possible sequences. The equipment survivability (ES) profile in each containment node is constructed by selecting appropriate accident scenarios, determining the resulting environmental conditions in the node during each scenario, aggregating the results, and simplifying the curves in a conservative manner.

3.1.1 Description of the MAAP 4.0.8 Code

MAAP 4.0.8 (Modular Accident Analysis Program version 4.0.8) is a computer code that simulates the response of LWR power plants during accidents (Ref.1). Given a set of initiating events and operator actions, MAAP predicts the plant’s response as the accident progresses. MAAP4 has been benchmarked against major experiments and plant transient experiences.

MAAP is an integral code which treats the full spectrum of important phenomena that could occur during a severe accident, simultaneously modeling those that relate to the thermal- hydraulics and to fission products. It simultaneously models the primary system and the containment and reactor/auxiliary building. MAAP is designed to account for plant signals, signal delays and component strokes, and for equipment opening or actuation times.

The PWR primary system model calculates the thermal-hydraulic conditions in the reactor pressure vessel, the hot legs, the cold legs, the cross-over (intermediate) legs, and the primary side of the steam generators. (The pressurizer is treated in a separate model.) The primary system is modeled as two conceptual loops, referred to as the broken loop and the unbroken loop. The user specifies how many actual primary system loops are incorporated into each MAAP4 loop, and which loop contains the surge line to the pressurizer. (The terms broken and unbroken are misnomers, as breaks can be modeled in either or both of the loops. These terms are carryovers from earlier, more restricted versions of the code.)

There are 14 gas control volumes in the primary system model: core, upper plenum, broken and unbroken hot legs, broken and unbroken hot and cold leg tubes (hot and cold sides of the tubes) for U tube steam generators, broken and unbroken cross-over (intermediate) legs, broken and unbroken cold legs, downcomer, and reactor vessel dome. There are six water pools: the core (includes the upper plenum, the reactor vessel dome, the hot legs, and the hot side of U-tubes), broken and unbroken cold tubes (the cold side of U-tubes), broken and unbroken cross-over legs (includes the steam generator outlet plenum through to the outlet of the pumps), and downcomer (includes the horizontal portion of the cold legs). In addition, there are 19 primary system structural heat sinks, which are modeled as two-dimensional slabs. Because the number of gas volumes is larger than the number of water pools, a pool can occupy several gas volumes.

The pressurizer is modeled as a single control volume, with one water pool and one gas node. The water and gas can be at different temperatures (which are also distinct from the primary system fluid temperatures). Calculations of the TH (thermal-hydraulic) conditions in the pressurizer account for evaporation, condensation, steam stripping due to steam and non-condensable gases sparging through the water pool, and water and gas exchange (co-current and countercurrent), with the

primary system through the surge line, and with the containment through relief valves and safety valves via the quench tank. Mass and energy contributions from pressurizer sprays and heaters, as well as heat transfer to structures, are also included in the pressurizer model.

The core model predicts the TH behavior of the core, including the water and gas contained within the core boundary, and the response of core components during all phases of a sequence. The calculations are performed on a nodal basis. Users can specify up to 50 axial rows and 7 radial rings (channels). The code tracks the mass, energy, and temperature of the following constituents in each node:

- Fuel (UO_2)
- Cladding (Zr, ZrO_2, stainless steel, steel oxide, and U-Zr-O)
- Control rod or water rod (Ag-In-Cd or B_4C, stainless steel, steel oxide, Zr, and ZrO_2)
- Structural materials (Zr, ZrO_2, stainless steel, and steel oxide)

The MAAP steam generator model calculates the heat transfer, from the primary side water or condensing steam, to the secondary side water and/or uncovered portion of the tubes. The calculations start with the masses and energies of the secondary side water and gas. The pressure, the temperature of each phase, the water levels, and the rates of change of the mass and energy terms are then determined. The two-region model calculates individual water levels in each region. It also has a tube bundle heat transfer model that tracks the subcooled length, and includes a fully imposed fluid momentum model, a two-phase swell model, and a structural heat sink model. Both equilibrium and non-equilibrium thermodynamics between the gas and the water are included.

3.1.2 MAAP Model of APR1400

The environmental conditions in the containment during a severe accident, including hydrogen burns, are obtained by running the MAAP 4.0.8 code. The MAAP model for APR1400 is specified in the parameter file SKN34-408-X.par. The sequence definitions are specified in individual input files. The MAAP model for the APR1400 containment consists of 36 nodes and 88 junctions. The APR1400 nodalization diagrams can be found in Attachment A.

3.1.3 Description of MAAP4-DOSE Code

MAAP4-DOSE (Ref.2) is a radiation dose calculation code that reads input from MAAP4 output.

3.1.4 MAAP4-DOSE Model of APR1400

The MAAP4-DOSE geometry model for calculating radiation doses in the containment is constructed based on the assumption that the individual containment regions can be approximated by a rectangular box whose dimensions are shown in Figure 4-1. There are 13 containment regions represented by 13 boxes in the dose parameter file. Some regions, such as the steam generator compartments (nodes 6 to 9), are assumed to have a square base, while other regions are assumed to have a rectangular base. For example, the annulus region node 10 is modeled as a rectangular box with base dimensions of 6.7 m $\times$ 32.5 m and a height of 4.3 m. The width and

the height of the box are same as the actual width and height of the annulus. The 32.5 m length of the box is obtained from dividing the floor area by the width. For steam generator compartment nodes 6 to 9, the base of the region is assumed to be a square whose length is equal to the square root of the floor area. The height of the box is the same as the height of the node (already defined in the MAAP4 parameter file).

Dose parameter files DOSE_LLOCA-ES-0HF0P0-MCCI.PAR, DOSE_LOFW-ES-0HF0DP3-MCCI.PAR, and DOSE_SBO-ES-0HF0DP3-MCCI.PAR, were prepared and used for the analysis. They define the geometry of the containment regions, the dose points, and the control parameters needed for running the code.

3.1.5 Hydrogen Generation Equivalent to a 100% Fuel-Clad Metal Water Reaction for Equipment Survivability Assessment

According to the regulatory requirements (10 CFR 50.34(f)), it must be demonstrated that equipment necessary for achieving and maintaining safe shutdown of the plant, or maintaining containment integrity, will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of 100% fuel-clad metal-water reaction, including the environmental conditions created by activation of the hydrogen control system. The NRC position on the accident scenarios and methodologies to produce an amount of hydrogen equivalent to a 100% metal-water reaction is provided in the Safety Evaluation Report that was issued in support of the Hydrogen Control Owner Group (HCOG). The HCOG submittal noted that 100% fuel-clad oxidation could be achieved by an extended simulation that imposed a source of water/steam for the metal-water reaction. For the HCOG submittal, the Grand Gulf plant (BWR) was assessed and found to require an artificially extended steam generation rate of approximately 0.1 lb/sec to achieve 100% fuel-clad oxidation. The Grand Gulf reactor is rated at 3830 MW (thermal). This is the same order of magnitude as the APR1400 power rating. The method applied in the ALWR certifications followed this NRC approved general guidance of extending the steam/water inventory availability (if needed) until the hydrogen equivalent of 100% active fuel-clad oxidation was achieved.

To conservatively calculate the in-vessel hydrogen generation, the following model parameters related to oxidation are selected and changed from the default values:

Table 3-1 Parameters Selected for In-Vessel Hydrogen Generation

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FAOX is the multiplier for the cladding external surface area available for oxidation. This parameter is set greater than one to account for steam ingression inside the cladding after cladding rupture. The oxidation rate is increased when the oxidation surface area in increased. FGBYPA

is a parameter to divert gas flows in the core to the bypass channel. When FGBYPA is equal to 1, all channel flows are diverted to the bypass channel if the whole core is blocked for a given elevation, such that steam is unavailable for oxidation and heat transfer for nodes above these blocked nodes. When FGBYPA is equal to 0, all channel flow will “reappear” above the blocked nodes and thus be available for oxidation and heat transfer.

For reference, the MAAP4 TMI-2 benchmarking simulation uses the model parameter FAOX = 2 before the reflood of the damaged core. The predicted hydrogen mass history for TMI-2 is shown in Figure 3-2. The total hydrogen mass is consistent with the TMI-2 accident, for which the estimated hydrogen mass generated was approximately 1000 lb.

Figure 3-3 shows the comparison between the Phebus FPT0 test data and the MAAP result. Using FAOX = 2, MAAP predicts the hydrogen generation history reasonably well and somewhat over-predicts the total mass of hydrogen generated.

3.1.6 Ex-Vessel Hydrogen Generation Due to MCCI

All analyzed sequences consider ex-vessel hydrogen generation by MCCI. Following vessel failure, MCCI is controlled by two model parameters FCHF and ENT0C.

FCHF, a "Kutateladze number" multiplier to the flat plate critical heat flux, is the controlling input parameter for molten debris heat transfer to overlying water following vessel failure. ENT0C is a parameter controlling the corium-water interactions as the molten jet falls into a flooded reactor cavity. Appropriate values for FCHF and ENT0C have been studied in the technical report appendix B. In that report, one pair of values was chosen to match the results of more sophisticated MCCI codes, COREQUENCH 3.2 and MAAP5.01.1146, while ignoring hydrogen generation from molten jet-water interactions. A second pair of values was chosen to better represent hydrogen generation from molten jet-water interactions.

Each sequence in this analysis was analyzed with the two sets of parameters. Table 3-2 summarizes the set of values for FCHF and ENT0C that matches sophisticated MCCI code results. Those values are used in MAAP runs whose input file name has an "-MCCI" suffix. Table 3-3 summarizes the set of values for FCHF and ENT0C that have a more realistic hydrogen generation during molten jet-water interactions. Those values are used in MAAP runs whose input file name has an "-CP" suffix.

Table 3-2 Parameter Values that Match COREQUENCH MCCI Results: MAAP Input File Name with Suffix "MCCI" Uses These Values

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Table 3-3 Parameter Values that Match Steam/Hydrogen Generation from Molten Jet-Water Interaction: MAAP Input File Name with Suffix "CP" Uses These Values

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Figure 3-1 Geometric Model for Dose Analysis

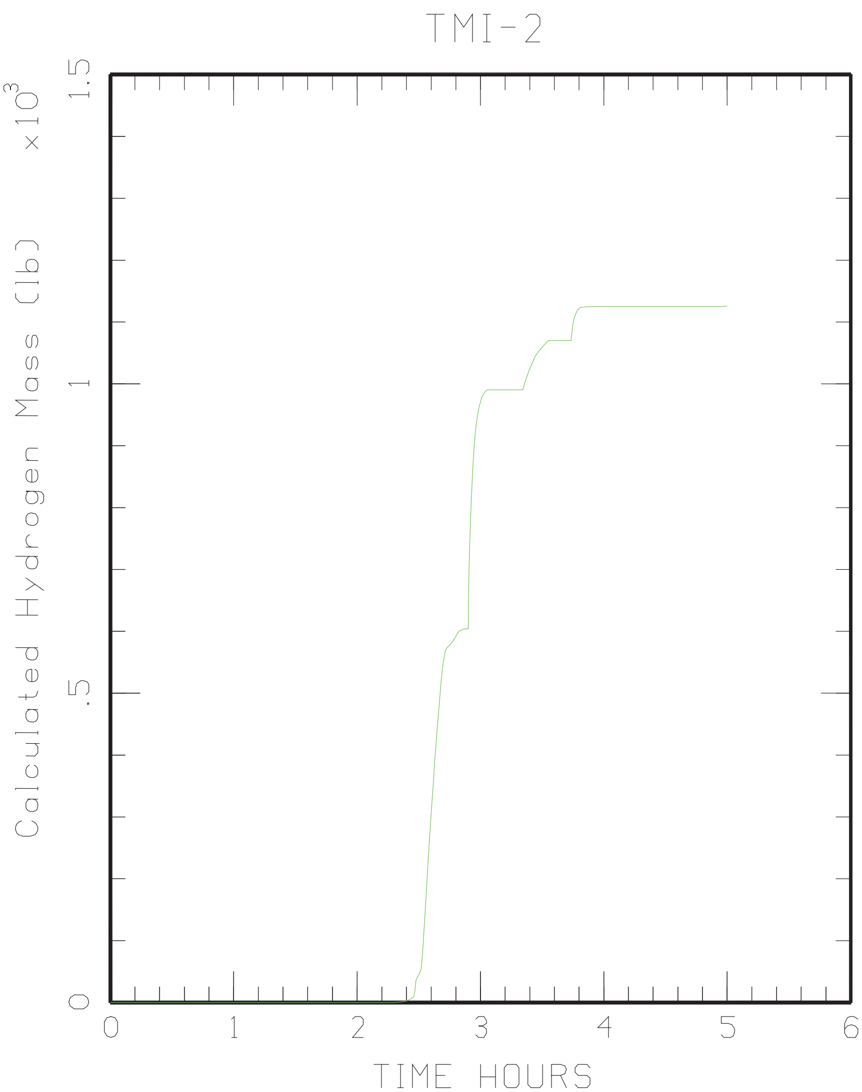

Figure 3-2 TMI-2 Generated Hydrogen Mass History

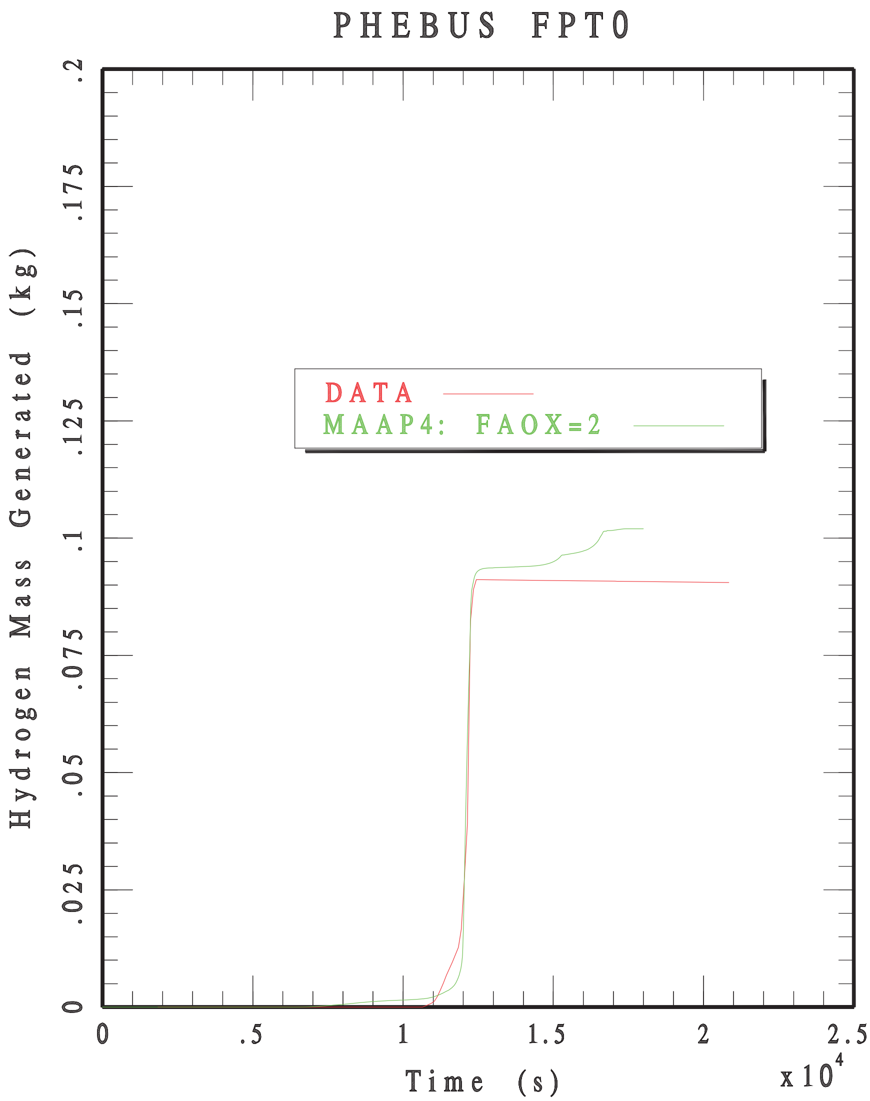

Figure 3-3 Comparison of PHEBUS FPT0 Results with MAAP

3.1.7 Construction of ES Profiles

Each accident sequence was analyzed with MAAP 4.0.8 to produce time-temperature histories for various locations in the containment (i.e. for the different MAAP containment nodes) over the 24 hours after onset of core damage. All severe accident mitigation features, including the Emergency Containment Spray Backup System (ECSBS), are assumed to be available for the purpose of evaluating equipment survivability. ECSBS is actuated 24 hours after the onset of core damage. The spray begins to cool and depressurize the containment atmosphere, removing the harsh environmental conditions that essential equipment and instrumentation were exposed to up to that point. Therefore, it is only necessary to evaluate equipment survivability during the first 24 hours after the onset of core damage.

Due to variability in the timing of severe accident phenomena, the option for bounding the entire set of results with a single curve is unnecessarily conservative. For example, a hydrogen burn subjects a compartment to extreme temperatures for a short duration. The time at which a hydrogen burn occurs varies with each scenario. Therefore, a simple curve which bounds all scenarios would model a compartment as experiencing extreme temperatures for an excessively long duration.

The temperature history of a compartment can be discretized and treated as a histogram. The bins in this histogram can then be reordered, not by time but by decreasing order of temperature magnitude, to create a monotonically decreasing characterization of compartment temperature which reproduces the same integrated value of temperature vs. time. This preserves the duration and the magnitude of high-temperature conditions, while minimizing the effect of uncertainty in phenomena timing. This reordering is conservative, since it maximizes the duration at extreme conditions. For example, two hydrogen burn events, each of 10 second duration, would be represented as a single high temperature event of 20 second duration. By reordering scenario results in this manner, the results of different sequences can be easily compared, and a simple bounding curve for all results may be constructed. This method of constructing ES curves accounts for the varied progression of accident sequences while maintaining conservatism.

A FORTRAN program was written to convert a temperature history to an equivalent reordered temperature histogram. The program also generates the time-temperature integrals before and after reordering, to verify that the appropriate integral value of temperature with time is preserved.

Except for the reactor cavity and the IRWST, where massive hydrogen burns cause extreme conditions, the temperatures in containment can be bounded by superposition of a long term elevated temperature of 460 K and short term temperature transients due to hydrogen burn, which can rise to as high as 900 K. Peak temperatures may last up to 10 seconds. Then, the temperatures decrease to 460 K over several hundred seconds. A bounding equipment survivability curve may consist of up to four regions as shown in Figure 3-4. Region I is a rapid initial temperature rise, indicated by the region from t_0 to t_1. This region corresponds to a rapid increase in compartment temperature, typically due to gas combustion or reactor coolant release. Equipment is initially at a normal operating temperature (T_{normal}) and experiences a rapid temperature increase to T_{max} over a short duration ($t_1 - t_0$). Note that the sorted histogram method described previously obscures the rate of temperature rise. Therefore an alternate method is used to create this region of an ES curve.

Regions II, III, and IV are conservatively constructed based on the sorted histogram method to

bound the aggregate MAAP results.

Region II represents a sustained application of high temperature, typically associated with gas combustion or reactor coolant release. This is shown in Figure 3-4 in the time period from t_1 to t_2, where equipment is subject to a constant temperature T_{max} over an interval on the order of 10-100 seconds.

Region III is a cooling period, representing a compartment temperature decrease from its high value to a steady-state temperature (T_{steady}) over an extended interval (from t_2 to t_3) on the order of 1000 seconds. The temperature decrease rate in this region may vary.

Region IV is an extended interval of elevated temperature, representing accident conditions which have approached a steady state (T_{steady}) above normal operating conditions. This is shown in the region after t_3; this period lasts on the order of 1000 to 100,000 seconds.

3.1.8 Bounding Radiation Environment

This analysis provides the cumulative radiation dose in selected containment regions during severe accidents for the purpose of equipment survivability assessment. The severe accident sequences selected for this analysis include large LOCA, loss of feedwater, and station blackout. The analysis was performed using the MAAP4-DOSE code. Necessary input information for the MAAP4-DOSE analysis can be obtained directly from the output files of the MAAP4 runs.

For each dose node, dose points are defined where cumulative radiation doses are calculated and reported. Note that in Figure 4-1 the MAAP containment node numbers (shown inside squares) are different from the dose node numbers (which are shown inside circles). For most nodes, the dose point is assigned to be the central point of the box. For nodes adjacent to the containment floor, such as the lower SG and lower annular regions (containment nodes 6, 8, 10 and 11), long-term radiation dose is expected to be highest near the floor, due to the deposition of airborne fission products. For these nodes, the dose point is located 0.5 m above the floor center to yield a higher dose reading than the node center.

Note that the radiation dose calculated by MAAP4-DOSE is a whole-body dose, and is based on the flux-to-dose rate conversion factors of ANSI/ANS 6.1.1-1991 standard (Ref.3). From the work of O'Brien and Sanna (Ref.4), which is discussed by Boerner and Chapman (Ref.5), the absorbed dose to various organs in the human body is 56.1% of the exposure, due to the body's self-shielding. Hence, the radiation exposure can be estimated by multiplying a factor 1/0.561 = 1.783 to the dose value calculated by MAAP4-DOSE. Dose absorbed by cables is calculated by

$$\text{Dose Absorbed by Cables} = 0.86 \times 1.783 \times \text{MAAP4-Dose Calculated Dose}$$

where it is assumed that cables absorb 86% of radiation exposure.

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Figure 3-4 Essential Aspects of an Equipment Survivability Curve

3.2 Selection of Severe Accident Sequences

Accident scenarios were selected both by deterministic and probabilistic means. The ten most likely core damage scenarios were selected from the PRA results as shown in Table 3-4, representing 87.6% of core damage frequency.

Deterministic sequences were selected for their severity and coverage. They include large, medium, and small loss of coolant accidents (LLOCA, MLOCA, and SLOCA), station blackout (SBO), and loss of feedwater (LOFW) scenarios. Table 3-5 shows the deterministic sequences analyzed by MAAP. For each base sequence, several sensitivity cases, with and without accumulators and containment spray, were analyzed to produce bounding environmental conditions.

Table 3-4 Dominant PRA Sequences

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Table 3-5 Deterministic Sequences for ES Curves

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3.3 Assumptions and Uncertainties

The thermal-hydraulic results produced by the selected sequences are expected to cover all credible environmental conditions that essential equipment and instruments could be exposed to during a severe accident. Additional sensitivity cases address uncertainties in operator actions (i.e. delay time for actuation of the safety depressurization and vent system) and in phenomena (i.e. burns with and without a diffusion flame). In all sequences, the severe accident mitigation features are assumed to be available as follows.

- Hydrogen Igniters are available – Igniters are placed near hydrogen release points to control rapid hydrogen releases during a severe accident. The igniters require AC power. Therefore, the igniters are not available during a station blackout.

- Passive Autocatalytic Recombiners (PARs) are available - PARs are a passive system effective in removing long-term hydrogen evolved from core-concrete interactions. A PAR efficiency of 50% is conservatively assumed.

- The Safety Depressurization and Vent System (SDVS) is available. Also, the Three-Way Valves are available. For loss of feedwater or station blackout sequences, the operator is supposed to open the POSRVs and align the Three-Way Valves to the steam generator compartment in order to depressurize the RPV after the onset of significant core damage. Without depressurization, massive release of hydrogen to the steam generator compartment will occur during a hot leg creep rupture event. Extreme temperatures of up to 1600 K are expected in the steam generator compartment due to a continuous hydrogen burn. Hence, with SDVS available, hot leg creep rupture is not considered. However, uncertainty in the delay time in the operator action is considered by running the sequences with both 5 minutes and 30 minutes delay time between core damage and actuation of the SDVS and Three-Way Valves. For station blackout sequences, the SDVS is actuated and Three-Way Valves are aligned to the SG compartment after the first opening of the POSRVs or one hour before battery depletion, whichever occurs first.

- In station blackout sequences, POSRV opening, due to high primary system pressure while significant water remains in the steam generators, are not used as a cue for the actuation of SDVS or the alignment of the Three-Way Valves to the SG compartment. "Significant water" is defined as a collapsed water level in the steam generator downcomers above the set point to generate an Auxiliary Feedwater Actuation Signal (3.317 m). This is done because primary system pressure is not expected to climb high enough to cause the POSRVs to lift while secondary side heat removal is still available, which delays the time at which SDVS initiation and Three-Way Valve alignment occur.

- The Cavity Flooding System (CFS) is available – The cavity flooding system is assumed to be actuated 30 minutes after the onset of significant core damage (i.e. core exit temperature is greater than 1200°F).

- The Emergency Containment Spray Backup System (ECSBS) is available – ECSBS is an alternate means of providing containment spray during a beyond design basis accident when both the Containment Spray (CS) and Shutdown Cooling (SC) pumps, and/or the IRWST, are not available. This system is actuated 24 hours after the onset of significant core damage (i.e. core-exit gas temperature increases above 1200°F). The harsh environmental condition in the containment is removed once the spray is turned on and cools the containment atmosphere. Therefore, equipment survivability assessment needs to be done only for the first 24 hours following the onset of significant core damage.

3.4 Thermal-Hydraulics and Radiation Envelopes

3.4.1 Containment Gas Temperatures

The temperature results for the thirty six compartments are compiled in Attachment B. For each compartment a figure with two temperature plots is provided. The temperature histories for all the various sequences are shown in the top plot. The corresponding sorted histograms are shown in the bottom plot. Time (x-axis variable) in the re-ordered temperature histories is plotted on a log scale so that the rapid temperature transients during hydrogen burns can be viewed.

Because of the sheer number of sequences analyzed, the analyzed sequences are divided into five groups:

- PRA sequences,
- Bounding sequences with model parameters FCHF and ENT0C tuned to match concrete erosion results of more sophisticated MCCI codes (case names have "-MCCI" suffix),
- Bounding sequences with model parameters FCHF and ENT0C tuned to give more realistic hydrogen generation rate during molten jet-water interactions (case names have "-CP" suffix),
- Bounding sequences with diffusion flame assumed on the SDVS 3-way valves (case names have "-DF" suffix),
- Bounding sequences with a 5 minutes delay time between core damage and actuation of the SDVS and 3-way valve (normally a 30 minutes delay time is assumed; case names have "-5M" suffix).

Severe accident temperature environments can be classified (in decreasing order of severity) as severely challenging, highly challenging, quite challenging, moderately challenging, or nominally challenging, depending on the magnitude and duration of extreme conditions. Severely challenging environments are characterized by highly-confined extreme conditions for a relatively long duration, such as in the reactor cavity and the IRWST. Highly challenging environments are areas close to a combustible gas source such as the steam generator compartments or the annular compartment

above the IRWST. Quite challenging and moderately challenging environments are areas where combustible gas may accumulate such as the containment dome. Nominally challenging environments are compartments where the containment atmosphere can be considered well-mixed and is inerted by a high steam concentration. The equipment survivability curves constructed for each of the five types of environments are shown in Figure 4-5 through Figure 4-9. They are also shown with the sorted histograms in Attachment B.

As described in Section 3.3, all severe accident mitigation features are assumed to be available for the purpose of constructing the bounding temperature environment for equipment survivability assessment. The cavity flooding system (CFS) insures the presence of water in the reactor cavity when the vessel fails and core debris is relocated to containment. The water pool formed in the reactor cavity following CFS actuation reaches the top of reactor cavity ceiling; the gas spaces in the corium chamber room, ICI chase, and reactor cavity access area are isolated from the gas space below the reactor vessel. Consequently, the environmental conditions in these areas are relatively benign whereas severe conditions are predicted in the reactor cavity and upper RPV annulus.

The bounding temperature profile expected in each containment node during a severe accident is summarized in Table 4-6.

3.4.2 Containment Pressure

The peak pressures in various locations in the containment for all ES sequences are listed in Table 3-7. The last row contains bounding pressures in each location. Clearly, there is no appreciable pressure difference between the different locations in the containment, and the bounding pressure can be conservatively set at 7.6 bar.

3.4.3 Radiation

Dose results presented here are the absorbed dose in units of rad, with an assumption that 86% of radiation exposure is absorbed by cables or equipment. Figure 3-10 shows the cumulative doses for selected containment regions during a 24-hour large break LOCA sequence. The region with the highest dose of 3.6E7 rad is found in the steam generator compartment node 7. Figure 3-11 shows the results for the LOFW sequence. The highest dose of 4.4E7 rad at 24 hours is found in the steam generator compartment node 7. The results for the station blackout are shown in Figure 3-12. The dose at 24 hours for this sequence is much lower than the other sequences because fission products are not released into the containment until after 4 hours. The maximum reading at 26 hours is 3.0E7 rad in the steam generator compartment node 7. Hence, the bounding sequence for radiation dose is the LOFW sequence, and the maximum dose is 4.4E7 rad.

Table 3-6 Summary of Temperature Envelopes for Equipment Survivability Assessment

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Table 3-7 Maximum Containment Pressure

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Figure 3-5 ES Curve for Nominally Challenging Environments

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Figure 3-6 ES Curve for Moderately Challenging Environments

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Figure 3-7 ES Curve for Quite Challenging Environments

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Figure 3-8 ES Curve for Highly Challenging Environments

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Figure 3-9 ES Curve for Severely Challenging Environments

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Figure 3-10 Cumulative Dose in Containment for Large Break LOCA Sequence

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Figure 3-11 Cumulative Dose in Containment for LOFW Sequence

Figure 3-12 Cumulative Dose in Containment for SBO Sequence

Equipment survivability is assessed by comparing reliable equipment qualification information such as equipment suppliers' documents, research results and experimental data with severe accident environmental conditions at the locations where the equipment is installed.

The principal cause for environments that are harsher than design basis events is hydrogen combustion. Severe accidents may produce significant quantities of hydrogen which can burn locally, globally, and either continuously or intermittently in different regions of the containment. Due to this potential for hydrogen burns it is necessary to assess the additional thermal loading on equipment that needs to survive for severe accident mitigation. This can be done by comparing the plant specific simulations of the thermal environments imposed by hydrogen burns against the available experimental basis.

The primary source of performance expectations of similar equipment in severe accident environments is an EPRI report (Achenbach, 1985) and supplemental information in NUREG/CR-5344 (SAND89-0327) (Clauss, 1989) and NUREG/CR-6530 (Blanchat, 1997). These reports describe programs which tested equipment types that have previously been qualified for design basis event environmental conditions. The temperature range in the test chamber for the EPRI program was 700 to 800°F for 10-20 minutes during a continuous hydrogen injection test. Additionally, the equipment in the EPRI program was exposed to significant hydrogen burns. The same equipment was exposed to and survived several events, both pre-mixed and continuous hydrogen injection, which provides confidence in the equipment's ability to survive a severe accident. The second program tested containment penetrations to high temperatures for long durations. Thus, reasonable assurance is achieved by applying available experimental databases to the expected containment conditions induced by hydrogen burns for the selected instrumentation and required functional intervals. Additionally, plant design features such as redundancy, separation of redundant capabilities and containment internal structures that shield equipment from burns can be used to construct arguments that provide reasonable assurance of the survivability of the designated equipment.

For the comparison with equipment supplier's documents, related documents are reviewed and the location of equipment is identified. The assessment of survivability is performed by comparing the equipment data to the ES profile, which defines the accident conditions the equipment will potentially be exposed to. If the equipment data cannot support survivability under the specified severe accident environmental conditions, survivability will be confirmed by consulting the equipment vendors.

A key input for the assessment of each individual piece of equipment or instrumentation is identification of its location within the containment. Furthermore, for each component on the equipment and instrumentation ES list, one may need to identify the thermally limiting (non-metallic) subcomponent. This identification is typically also performed as part of the design basis equipment qualification (EQ) activity. Usually equipment placed on the ES list is also included in the plant as part of design basis instrumentation such that the thermally limiting component for those items may already have been determined

Examples of non-metallic components and materials include compression seals, gaskets, cables (electrical power, control and instrumentation), and motor winding insulation. Non-metallic

materials often encountered in these components include ethylene propylene (EPDM), silicone rubber, and neoprene.

Thermal lag analysis may be used for equipment survivability assessments. For example, even though the local atmospheric condition could momentarily rise to an extreme temperature during hydrogen combustion, the temperature at the surface of the equipment may remain much lower than that in the gas space. Hence, equipment survivability can be assessed using simplified thermal lag analysis based on the temperature difference between the containment atmosphere and the equipment surface and/or internal components.

4.1 Hydrogen Igniters

Eight hydrogen igniters are distributed near expected hydrogen release points throughout the containment: reactor cavity access area, generative heat exchanger room, steam generator compartments, and pressurizer compartment. Of these, the harshest environment is expected to be the steam generator compartment. The ES profile in this area is characterized by a constant 900 K for 10 seconds, decreasing to 460 K at 600 seconds, and then staying constant.

The hydrogen igniters are protected with E-5A-3 fire wrap from 3M Corporation. The igniters can thus survive an accident environment temperature of 1073 K for four hours and still perform their intended function.

4.2 Passive Autocatalytic Recombiners (PARs)

The PARs are made of Pt or Pd catalyst in a stainless steel casing. They do not contain any organic material components which could be susceptible to thermal degradation. Therefore, they are expected to survive the harsh environment of a severe accident and continue to perform their intended function of hydrogen removal.

4.3 Cavity Flooding System (CFS) Motor Operated Valves (MOVs)

The CFS consists of two spillways between the IRWST and the Hold-up Volume Tank (HVT), two spillways between the HVT and the reactor cavity, and related valves. Four MOVs are installed in the Hold-up Volume Tank. The goal of the CFS is to pre-flood the reactor cavity prior to vessel failure so as to possibly prevent vessel failure or, in the event of vessel failure, to cool core debris which would accumulate on the reactor cavity floor. The MOVs need to operate only prior to vessel failure, under a relatively mild environment. In that case the cavity conditions should be bounded by the DBA EQ peak temperature of 455 K. Therefore, the CFS MOVs are expected to perform their intended function of opening the valves for cavity flooding.

4.4 Post-Accident Sampling System (PASS)

PASS is designed to collect and deliver representative samples of liquids and gases in various process systems to sample stations for chemical and radiological analysis. The RCS hot-leg sample isolation MOVs and their position transmitters are located in the steam generator compartment. The containment air sample isolation MOVs and their position transmitters are located in the annular compartment at the 121’6” elevation. The bounding ES profile in the steam generator compartments is characterized by a constant 700 K for 100 seconds, decreasing to 500 K

at 1000 seconds, and staying constant thereafter. The ES profile in the annular compartment is characterized by a constant 700 K for 10 seconds, decreasing to 460 K at 600 seconds, and staying constant thereafter. MOVs were included in the EPRI hydrogen burn experiments (EPRI NP-4354) and survived many transients.

4.5 Containment Hydrogen Monitoring System

The containment and IRWST hydrogen monitoring system contains hydrogen monitor inlet valves, hydrogen analyzers and piping. This system samples the containment atmosphere and measures the hydrogen concentration, employing sensing devices outside containment. The hydrogen analyzers and discharge valve are not subject to the harsh environment of a severe accident because they are located outside containment. However, the inlet valves are potentially subjected to a harsh environment because they are located inside containment. They are located in the dome, in the annular compartment, and in the IRWST.

The environmental condition in the containment dome is quite challenging. The environmental condition in the annular compartment is highly challenging. The environmental condition in the IRWST is severely challenging, with a peak temperature of 1200 K for 10 seconds, decreasing to 700 K at 2000 seconds, and staying at a constant 600 K thereafter. If the inlet valves fail during a severe accident, the Post-Accident Sampling System can be used to determine representative hydrogen concentrations since a relatively uniform hydrogen concentration is expected throughout the containment.

4.6 Containment Atmospheric Temperature Sensors

Thirteen temperature sensors are distributed throughout the containment: dome, steam generator compartments, pressurizer room, annular compartment above the operating deck, annular compartment below the operating deck, and reactor cavity. The sensor located in the reactor cavity will be exposed to a severely challenging environmental condition, with a peak temperature of 1,200 K for 10 seconds, decreasing to 700 K at 2,000 seconds, and staying at constant 600 K thereafter.

The sensor located in the steam generator compartments will be exposed to a highly challenging environmental condition, with a peak temperature of 900 K for 10 seconds, decreasing to 460 K at 600 seconds, and staying constant thereafter. In addition, sensors in the path of a diffusion flame on the IRWST vent stack and on the SDVS 3-way valves will likely not survive.

Although the EQ data conditions provided by the equipment vendor are less than the harsh environmental conditions expected during a severe accident in some locations, there are enough temperature sensors distributed throughout the containment to provide redundancy. Also, installation of fire wrap around these sensors would minimize the effect of high temperatures and shield against diffusion flames.

4.7 Containment Radiation Monitor System (RMS)

Two radiation monitoring systems are located in the south side of the upper operating area. The ES curve in this area is characterized by a constant 900 K for 10 seconds, decreasing to 460K at 600 seconds and staying constant thereafter.

The thermally limiting components in the radiation monitor system are PEEK insulators in the chamber and cable connectors. Based on the “Percentage retention of elongation at break after aging” test data for PEEK material, it is determined to have qualified life of 50 hours at 583 K. The test temperature is higher than the long term severe accident environmental temperature of 460 K. Due to thermal lag, short temperature transients due to hydrogen burns should not affect the insulators. Therefore, the PEEK insulators in the RMS are expected to maintain their integrity during a severe accident.

4.8 Equipment Hatch and Personnel Air Lock

The equipment hatch and personnel air locks are located in the annular compartment, two at the operating deck and one above the IRWST. The environmental conditions at the operating deck during a severe accident will be highly challenging, while those above the IRWST will be less harsh. The highly challenging ES curve is characterized by a constant 900K for 10 seconds, decreasing to 460 K at 600 seconds, and staying constant thereafter.

Thermally limiting components in the equipment hatch and personnel air lock are EPDM O-rings, compression seals, and gaskets. In the Sandia/CBI Personnel Airlock Testing, an actual full-scale airlock assembly was subjected to environmental conditions corresponding to severe accident events. In particular, Test 2C consisted of three thermal and pressure cycles. In the second cycle, the air temperature was raised to 700 K. Then, the pressure was increased to 300 psig. There was no measurable leakage of the inner door seal. In the tests, it was determined that the temperature at which the material deteriorates is approximately 600 K. Indeed, the peak temperature recorded on the door surface when the seal failed during the third cycle was 633 K. Results of Test 2C demonstrated that the EPDM seal material will survive the ambient temperature of 485 K over 24 hours, the long term bounding temperature of the highly challenging ES curve. When a thermal-lag calculation was done on the door seal, the short 900 K temperature spike in the atmosphere was not transmitted to the door seal. Hence, the seal and gaskets in the Equipment Hatch and Personnel Air Locks are expected to maintain their integrity during a severe accident.

4.9 Electrical Penetration Assembly (EPA)

The Electrical Penetration Assemblies (EPAs) are installed on the containment pressure boundary and are sealed with double O-rings. The EPAs are located in the annular compartment at various elevations above the operating deck. The environmental conditions in the annular compartment where EPAs are installed are bounded by the highly challenging ES curve. The highly challenging ES curve is characterized by a constant 900 K for 10 seconds, decreasing to 460 K at 600 seconds and staying constant thereafter.

The thermally limiting components in EPAs are Viton O-rings, polysulfone module conductor sealant, and polymide film conductor insulation. A Conax EPA was tested under severe accident conditions by Sandia National Laboratories. The EPA was a lower voltage penetration assembly with a typical cable mix for power, control, and instrumentation functions. The EPA was first irradiated and then thermally aged. Then, the EPA was exposed to steam at 135 psia and 644 K for 8 days. The temperature in the test chamber reached the maximum value, 644 K, about 45 minutes into the test. Temperature in the junction box reached a steady-state temperature of about 561 K at about 4 hours into the test. Temperature on the header plate reached the steady-state temperature of about 444 K at about 4 hours into the test. The leak integrity of the Conax EPA was maintained

during the entire 10 day period of the severe accident test. Clearly the test condition of 644 K exceeds the long term ES curve. Thermal-lag calculations show that the short temperature spike of 900 K in the ambient is not transmitted to the air inside the termination box, and certainly not to the sealant. Hence, the seal in the EPA is expected to maintain its integrity during a severe accident.

4.10 Mechanical Penetrations

Mechanical penetrations include the Main Steam (MS) and Feedwater (FW) flow penetrations and hot/cold process piping. These penetrations have no organic material and are designed to remain sealed under severe accident environmental conditions.

4.11 Safety Depressurization and Vent System (SDVS)

The POSRVs in the pressurizer compartment, and 3-way valves located in the steam generator compartment, are manually operated to rapidly depressurize the RCS following onset of core damage. The goal of the system is to depressurize the RCS from 2500 to 250 psia prior to vessel failure, thereby preventing high pressure melt ejection. The essential components of the system that need to be assessed for a harsh environment include:

- POSRVs and actuation circuitry
- POSRV position transmitters and indicators
- POSRV discharge branch line isolation valves and position transmitters, and indicators
- MOVs for 3-Way Valves

The system is not needed during a LOCA because the break will depressurize the RCS. Therefore, the essential components are not subject to the harsh environment of design basis accidents. Also, prior to actuation of SDVS the essential components will not be subjected to the harsh environment of a hydrogen burn or to the long term elevated temperatures following vessel breach. Hence, the SDVS equipment is expected to survive and perform its intended function during a severe accident.

4.12 Reactor Vessel Level Monitoring System (RVLMS)

The RVLMS consists of two probes with heated and unheated junction thermocouples. The heated junction thermocouple (HJTC) probes monitor the liquid inventory above the fuel alignment plate. The temperature difference between the heated and unheated junction thermocouples is a direct indication of the presence of liquid inventory. RVLMS provides useful information as the core uncovers and it provides confirmation of core recovery. Individual unheated junction thermocouples may also trend the progression of core degradation by monitoring the reactor vessel upper plenum gas temperature.

The HJTC probes utilize heated and unheated junction Type K thermocouples. Unlike the core exit thermocouple, the RVLMS thermocouple string is top mounted and it does not pass through the core. These thermocouples are calibrated to operate up to 1255 K, in accordance with the RVLMS

design requirements. Hence, these instruments are expected to function well after the start of core degradation

4.13 Core Exit Thermocouples (CETs) and Resistance Temperature Detectors (RTDs)

The CETs and RTDs used to monitor RCS inventory are relatively robust. They can survive well past design basis conditions and provide useful information until their temperature limits are exceeded. These temperature limits are about 1533 K for Type K thermocouples procured for “in-vessel” application and about 673 K for RTDs.

4.14 Pressurizer Pressure Sensors and SG Level Monitors

RCS pressure monitoring is necessary to trend RCS depressurization following operator actions taken to either establish “feed and bleed” operation or to confirm sufficiently low pressure to enter shutdown cooling (SDC) operation. In the event the operator has to depressurize the RCS via the steam generator (SG), the water level in the SG must be monitored to assure sufficient SG secondary-side inventory.

All pressure transmitting devices are located outside the secondary shield wall. A long, small diameter tube connects the RCS to the high pressure side of the pressure transmitter. The sensor tap of these pressure transmitting devices is typically filled with low viscosity fluid. The long length of the tube provides sufficient heat loss and thermal capacity to maintain the fluid temperature closer to the ambient. Therefore, the “in-vessel” environment will not influence the operation of these pressure transmitting devices.

All transducers, cables, and associated signal conditioning equipment inside the containment have been tested by Duke Power Company, and have shown to withstand a combined LOCA and hydrogen burn environmental condition.

4.15 Class 1E Power, Control, and Instrumentation Cables

Power, control, and instrumentation cables are distributed throughout the containment including the annular compartment, steam generator compartments, pressurizer compartment, holdup volume tank, and reactor cavity. They are exposed to a harsh environment of long- term elevated temperature as well as short-term temperature transients due to hydrogen burn. Except for the reactor cavity, the cables can be protected from the harsh environment using the E-54 series Fire Wrap made by 3M Company.

4.16 Equipment Located Outside Containment

The active components of the following equipment, required for severe accident mitigation and monitoring, are located outside containment. They are not subjected to the harsh environment of a severe accident.

- Safety Injection System (SIS)
- Auxiliary Feedwater System (AFWS)

- Containment Spray System (CSS)
- Emergency Containment Spray Backup System (ECSBS)
- Shutdown Cooling System (SCS)
- Containment hydrogen monitors
- Containment pressure sensor
- IRWST water level sensors

The check valve on the ECSBS spray headers is located inside containment, but it contains no organic material susceptible to thermal degradation.

4.17 Radiation Dose ES Results

The environment qualification report for safety related equipment usually contains radiation test data. Table 4-1 summarizes the radiation dose test data for essential equipment and instrumentation. All equipment was tested under at least five times the bounding radiation dose in the containment during a severe accident.

For most equipment and instrumentation, it is concluded there is reasonable assurance that instrumentation and equipment required to mitigate a severe accident and achieve a safe stable state perform their function as intended under severe accident environmental conditions.

Table 4-1 Tested Radiation Dose

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5.0 Summary and Conclusion

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).

6.0 References

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4. O’Brien et.al., 1976, “The Distribution of Absorbed Dose-Rates in Humans from Exposure to Environmental Gamma Rays,” Health Physics Journal, Vol. 30, pp. 71-80, January.

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ATTACHMENT A: Containment Nodalization

Table A-1 APR1400 Containment Node Description

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Figure A-1 Containment Nodalization Diagram for APR1400

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Figure A-2 Containment Nodes in Section A-A

Security Related Information- Withhold Under 10CFR2.390