Enclosure 3
Radiological Source Term Methodology for the B&W mPower™ Reactor
MPWR-EPP-005010
(Redacted)

babcock & wilcox mPower

Position Paper on Radiological Source Term Methodology for B&W mPower™ Reactor
MPWR-EPP-005010
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a progressive energy solution

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This is a pre-application document and includes preliminary B&W mPower Reactor design or design supporting information and is subject to further internal review, revision, or verification.

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1. Introduction

As indicated initially in B&W letter BW-JAH-2010-235 to the U.S. Nuclear Regulatory Commission (NRC), dated November 18, 2010, a position paper describing the methodology for calculating accident source term for the mPower™ Reactor design would be provided to the NRC. The purpose of the following is to provide that paper as per our previous commitment.

The principal safety objective of nuclear power plants is to protect individuals, society and the environment by establishing and maintaining an effective defense against radiological hazard. The design of the B&W mPower Reactor incorporates features consistent with the goals of reducing system complexity and the potential for radiological consequences.

When crediting the features in the B&W mPower reactor, the [ ] accident source term is expected to be less than [ ] [CCI per Affidavit 4(a)-(d)] total effective dose equivalent (TEDE) at a plant's exclusion area boundary, [ ] [CCI per Affidavit 4(a)-(d)] expressed in 10 CFR 100 of the U.S. Code of Federal Regulations (Reference 1). U.S. regulatory expectations are further clarified in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," NUREG-1465, "Accident Source Terms for Light Water Nuclear Power Plants," and NUREG-0800, Standard Review Plan (SRP), Chapter 15 (References 2 – 4). Currently, the language in these regulatory statements does not endorse use of a [ ] [CCI per Affidavit 4(a)-(d)] evaluation in support of plant licensing.

While the B&W mPower reactor has been engineered to practically eliminate radiological source term from design-basis events and significantly reduce the likelihood of core damage events, B&W mPower is committed to an application of the existing regulatory framework, while investigating [ ] [CCI per Affidavit 4(a)-(d)]. It is expected that the resulting quantification of accident source term defines the exclusion area boundary (EAB) and emergency planning zone (EPZ) [ ] [CCI per Affidavit 4(a)-(d)]. As such, B&W mPower supports a reexamination by the U.S. Nuclear Regulatory Commission (NRC) of the existing regulatory framework to reflect design advances being made by the SMR community, thus resulting in a more appropriate designation of the EAB and EPZ for SMRs.

1.1 Defense-in-Depth Philosophy for the B&W mPower Reactor

Like similar nuclear power plant design and deployment projects, engineering for the B&W mPower reactor is structured around a defense-in-depth philosophy. The requirements established for advanced light water reactors (LWRs) are very stringent with regard to the consideration of defense-in-depth and the radiological impact on the public. To demonstrate the safety of the plant, the following basic objectives should be fulfilled per Reference 5:

i) prevention of abnormal operation and failures,

ii) control of abnormal operation and detection of failures,

iii) control of accidents within the design basis,

iv) control of severe plant conditions, including the prevention of accident progression and mitigation of the consequence of severe accidents, and

v) mitigation of radiological consequences of significant releases of radioactive materials.

The safety goal of the B&W mPower reactor is to enhance these safety objectives by:

- extensive review of initiating events used to confirm the adequacy of the safety provisions,
- minimizing or, if possible, eliminating the occurrence of complex phenomena and "cliff-edge" effects during normal operation, anticipated operational occurrences (AOOs), and accidents,
- providing for long periods when operator action other than monitoring is not necessary, and
- simplifying the whole operations architecture by:
  - the application of "As Low As Reasonably Achievable" (ALARA) principles for the protection of workers against the radiation exposure in particular when implementing the necessary corrective actions under accident conditions,
  - minimizing and mitigating of hazards other than radiological ones (e.g., chemical hazards),
  - minimizing the production of wastes and effluents and developing accommodation for their lifecycle, and
  - preventing, by design, possible types of malevolence and proliferation, and minimizing their potential consequences.

Specifically regarding radiological assessment, the B&W mPower reactor employs features that

- prevent accidents with harmful consequences resulting from a loss of control over the reactor core or other sources of radiation, and mitigate the consequences of any accidents that do occur,
- ensure that for all accidents taken into account in the design, any radiological consequences would be below the relevant limits, and
- ensure that the likelihood of occurrence of an accident with serious radiological consequences is extremely low and that the radiological consequences of such an accident would be mitigated to the fullest extent practicable.

In so doing, plant event sequences that could result in high radiation doses or radioactive releases are practically eliminated and plant event sequences with a significant frequency of occurrence have no or only minor potential radiological consequences. mPower is targeting a [ ] [CCI per Affidavit 4(a)-(d)] TEDE accident radiological consequence, an objective of which is that the necessity for off-site intervention measures to mitigate radiological

consequences is [ ] [CCI per Affidavit 4(a)-(d)].

1.2 Radiological Consequence Requirements

As described in RG 1.183, an accident source term assumes a major accident involving significant core damage and is typically postulated to occur in conjunction with a large loss-of-coolant-accident (LOCA). Although the LOCA is typically the maximum credible accident, experience has indicated the need to consider other accident sequences of lesser consequence but higher probability of occurrence. Accident source term is characterized by radionuclide composition and magnitude, chemical and physical form of the radionuclides, and the timing of the release of these radionuclides. Beyond the guidance for analyses addressing RG 1.183, several requirements are imposed on nuclear power plant (NPP) siting and operation to assure that the consequences of design-basis events remain well below that which would be expected to result in adverse impact on public health and safety. These requirements are summarized in the form of acceptance criteria in NUREG-0800, Section 15.0.3, reiterated below.

- Section 50.34(a)(1) of 10 CFR 50, "Contents of applications; technical information," as it relates to the evaluation and analysis of the offsite radiological consequences of accidents with fission product release. Of particular note:
  - "An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE), and
  - An individual located at any point on the outer boundary of the LPZ[1], who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 25 rem TEDE."
- GDC 19 of Appendix A to 10 CFR 50, "Control room," as it relates to maintaining the control room in a safe condition under accident conditions by providing adequate protection against radiation.
- Section 100.21 of 10 CFR Part 100, "Non-seismic siting criteria," as it relates to the evaluation and analysis of the radiological consequences of accidents for the type of facility to be located at the site in support of evaluating the site atmospheric dispersion characteristics.
- Paragraph IV.E.8 of Appendix E, to 10 CFR Part 50, "Emergency Planning and Preparedness for Production and Utilization Facilities," as it relates to adequate provisions for an onsite technical support center (TSC) from which effective direction can be given and effective control can be exercised during an emergency.

[1] i.e., "Low Population Zone"

2. General Design Overview

The B&W mPower reactor is an advanced pressurized LWR system featuring an integral reactor design in which the reactor core, control rod drive mechanisms (CRDMs), steam generator, and pressurizer are contained within a single reactor pressure vessel (RPV). This design overview section describes the principal systems employed during design basis events under both faulted conditions that rely on the safety-related structures, systems, and components (SSCs) credited in safety analysis and nominal conditions that employ non-safety systems providing broad operational flexibility and defense-in-depth. The system design information contained herein is considered preliminary by B&W mPower and is subject to change.

2.1 Accident Mitigating Structures, Systems, and Components

Safety-related SSCs support one or more of the critical safety functions of reactivity control, reactor coolant system (RCS) pressure and inventory control, RCS heat removal, containment environment and isolation, containment integrity, and radiation and radioactive effluent control. The front-line SSCs important to preventing and mitigating radiological consequences consists of the fuel rods, CRDMs, RPV, the containment structure, and all of the fluid and electrical systems needed to support reactor operation.

2.1.1 Reactor Coolant System

The RCS consists of the reactor core, CRDMs, reactor coolant pumps (RCPs), steam generator, and pressurizer. All of these components are contained within a bolted vessel assembly as illustrated on Figure 1. The reactor core, supporting core former, and incore instrumentation and guides are located in the lower vessel. The control rod guide frames and CRDMs are physically inside the lower vessel but are supported by a flange captured between the lower and upper vessel. The upper vessel contains a single once through steam generator (OTSG) with a central riser, and the pressurizer. Eight, canned motor, RCPs are bolted to a pump support plate at the base of the pressurizer.

The integral design of the RCS eliminates [

] [CCI per Affidavit 4(a)-(d)] RCS operating conditions are provided in Table 1.

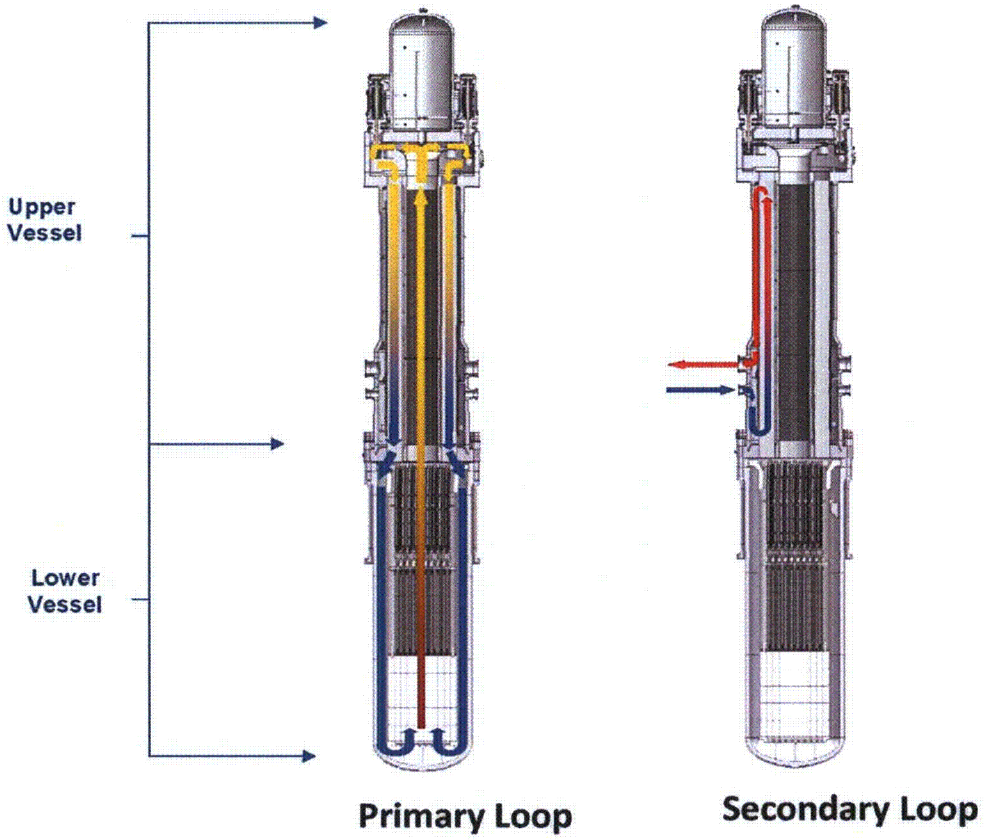

Figure 1 RCS General Arrangement

Table 1 General RCS Specifications

Operating power, MWt	530
Operating pressure, psia (MPa)	[ ] [CCI per Affidavit 4(a)-(d)]
Design pressure, psia (MPa)	2300 (15.9)
Cold leg temperature, °F (°C)	[
Hot leg temperature, °F (°C)	
Total coolant flow @ full power, millions of pounds per hour (kilograms per second)	]

[CCI per Affidavit 4(a)-(d)]

The reactor core consists of 69 fuel assemblies, each a shortened version of a conventional 17x17 square-lattice commercial pressurized water reactor (PWR) fuel assembly. The fuel assemblies include UO_2 fuel rods enriched up to 5wt% $_{235}U$, [

] [CCI per Affidavit 4(a)-(d)]

2.1.2 Emergency Core Cooling System

The emergency core cooling (ECC) system is the primary fluid safety system in the plant and ensures adequate water is available in the reactor vessel to provide core cooling. ECC is divided into [

] [CCI per Affidavit 4(a)-(d)]

2.1.3 Auxiliary Condenser System

The auxiliary condenser system (CNX) is a non-safety system consisting of a [

] [CCI per Affidavit 4(a)-(d)]

2.1.4 Reactor Coolant Inventory and Purification System

The reactor coolant inventory and purification (RCI) system is a multi-functioned non-safety system that is provided to perform the following:

- normal RCS cleanup and chemistry control,
- [ ] [CCI per Affidavit 4(a)-(d)]
- provide makeup and letdown to the RCS during startup and shutdown and if required during normal operation,
- [ ] [CCI per Affidavit 4(a)-(d)]

The RCI automatically provides makeup and letdown based on pressurizer water level and will initiate [ ] [CCI per Affidavit 4(a)-(d)]

2.1.5 Component Cooling Water System (CCW)

The non-safety component cooling water (CCW) system is designed to provide heat removal for the following major components:

- [

] [CCI per Affidavit 4(a)-(d)]

2.1.6 Reactor Containment

The reactor containment is an [ ] [CCI per Affidavit 4(a)-(d)]. The seismic category I structure is designed to withstand the maximum internal pressure from design basis accidents, including LOCA and steam line break. Wind, tornado and precipitation loads are not applicable because the structure is located completely below grade and is protected by the Reactor Services Building.

Internal containment pressure and temperature is controlled by an ultimate heat sink (UHS) [ ] [CCI per Affidavit 4(a)-(d)]

2.1.7 1E DC Power Supply

The 1E DC/UPS (DC) system provides reliable power for safety-related equipment including the [ ] [CCI per Affidavit 4(a)-(d)]

2.1.8 Auxiliary Power System

The auxiliary power (ACX) system includes the standby diesel generators and associated fuel oil storage and transfer systems. Each unit has [

] [CCI per Affidavit 4(a)-(d)] standby diesel [

] [CCI per Affidavit 4(a)-(d)]

2.2 System Level Defense-in-Depth for Design Basis Events

The B&W mPower reactor is designed so that in a plant upset condition, multiple non-safety systems can maintain the RCS within its safe operating envelope. Key defense-in-depth systems include the [

] [CCI per Affidavit 4(a)-(d)] Some of these scenarios are outlined briefly below.

[

]

Figure 2 System Defense-in-Depth Approach

[CCI per Affidavit 4(a)-(d)]

2.2.1 Turbine Trip

Normal plant response - A turbine trip will result in closure of the turbine throttle valve raising the steam pressure in the inlet line. This will automatically cause the turbine bypass valves to open and initiate a decrease in feedwater flow until 20% flow is reached. Reactor power will follow feedwater flow with core outlet temperature being held constant and pressurizer level rising to its programmed level at 20% power. The plant will remain in this condition until the turbine is brought on line, or the plant operators begin an orderly shutdown.

Faulted plant response – A turbine trip with faults [

] [CCI per Affidavit 4(a)-(d)]

2.2.2 Loss of Normal Feedwater Flow

Normal plant response – [

] [CCI per Affidavit 4(a)-(d)]

2.2.3 Loss of Off Site Power

Normal plant response – [

] [CCI per Affidavit 4(a)-(d)]

<u>Station blackout response</u> – [

] [CCI per Affidavit 4(a)-(d)]

2.2.4 LOCA

Even though the B&W mPower plant is limited to [

] [CCI per Affidavit 4(a)-(d)]

2.2.5 Anticipated Transient Without Scram (ATWS)

Any operating transient that generates RCS parameters outside of the acceptable operating envelope will result in the initiation of a [

] [CCI per Affidavit 4(a)-(d)]

2.3 Long-term Coping Period

The B&W mPower plant is designed to provide multiple ways of maintaining the RCS within its acceptable operating envelope. [

] [CCI per Affidavit 4(a)-(d)]

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3. Design Basis Events with Potential Radiological Consequences

10 CFR 50.34 requires presentation of a suite of radiological assessments associated with safety analyses applicable to the B&W mPower reactor. Table 2 provides a summary of the accidents with potential radiological impacts that are applicable to the B&W mPower reactor, as informed by NUREG-0800 SRP. This information is condensed from that appearing in Reference 5. Most transients and accidents relevant to the B&W mPower reactor are either identical or very similar to those applicable to commercial operating PWRs or to advanced PWRs with passive safety systems. In addition to those events appearing in NUREG-0800, the B&W mPower reactor RCI includes an interface leading outside containment. As such, failures of that interface must also be addressed; however, by virtue of liquid water barriers, it is expected to be bounded by the steam generator tube rupture.

Table 2 Applicability of NUREG-0800 Transients to B&W mPower Reactor

15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve	[
15.1.5.A	Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR	
15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	
15.6.3	Radiological Consequences of Steam Generator Tube Failure	
15.6.5.A	Radiological Consequences of a Design Basis Loss-of-Coolant Accident Including Containment Leakage Contribution	
15.7.3	Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	
15.7.4	Radiological Consequences of Fuel Handling Accidents	]

[CCI per Affidavit 4(a)-(d)]

Events described under Section 15.7.3 will be performed employing a separate evaluation methodology prepared for events addressing SRP Chapter 11, Radioactive Waste Management. Events described under Section 15.7.4 must assume *a priori* the amount of fuel failures occurring coincident with the accident. The details of the evaluation methodology related to these events are outside the scope of this report.

For all events, analyses shall consider the following source term scenarios:

- [

] [CCI per Affidavit 4(a)-(d)]

Regarding the latter two items, the enhanced safety features of the B&W mPower reactor have been designed to eliminate their occurrence in design-basis events.

3.1 Inadvertent Opening of a Steam Generator Relief or Safety Valve

The inadvertent opening of a main steam relief or safety valve provides a direct pathway for secondary coolant to bypass the containment. [

] [CCI per Affidavit 4(a)-(d)]

3.2 Steam System Piping Failures Inside and Outside of Containment

The steam system piping failure or main steam line break (MSLB) transient is a postulated accident not expected to occur during the life of the NPP. [

] [CCI per Affidavit 4(a)-(d)]

3.3 Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment

This event postulates the failure of small lines outside of containment that are connected to the primary coolant pressure boundary. Such lines include [

] [CCI per Affidavit 4(a)-(d)]

3.4 Radiological Consequences of Steam Generator Tube Failure

The steam generator tube rupture (SGTR) accident is defined as a breach of the barrier between the RCS and the secondary side, with a double-ended break of one tube being the limiting case. The radiological concern relates to a release to the environment via the failed

tube and the secondary side of the plant. If the size of the tube rupture is too small to result in a measurable impact on the plant thermal-hydraulic conditions, the event should still be detectable by a monitored increase in radiation activity on the secondary side. The atmospheric releases consist of the secondary-side activities, RCS leakage via the ruptured steam generator tube, and normal primary-to-secondary leakage.

The evaluation of the radiological consequences of a SGTR considers the [

] [CCI per Affidavit 4(a)-(d)]

3.5 Radiological Consequences of a Design Basis Loss-of-Coolant Accident

The LOCA is considered to be the maximum credible accident resulting in the highest realistic accident-related radiological source term. It is described in detail in Section 4.

4. Maximum Hypothetical Accident

Preliminary systems analyses have shown that the biggest challenge to maintaining core coolant, thus preserving fuel integrity, is a pipe break in a relief, safety, or automatic depressurization line [

] [CCI per Affidavit 4(a)-(d)]

4.1 Accident Scenario

The LOCA is a postulated accident that results from a pipe break on the RCS pressure boundary. The distinguishing characteristic of a LOCA is that it results in a loss of reactor coolant at a rate in excess of RCS makeup system capabilities. As the RCS inventory decreases, the reactor will trip [

[1] This non-safety system is not credited in safety analysis addressing SRP Chapter 15.

] [CCI per Affidavit 4(a)-(d)]

4.2 Evaluation Methodology

The release of radioactive substances from a nuclear power plant to the environment (the source term) depends on the following factors:

- the inventory of fission products and other radionuclides in the core,
- the retention of radionuclides in the primary cooling system,
- the progression of core damage,
- the fraction of radionuclides released from the fuel and the physical and chemical forms of released radioactive materials, and
- the performance of means of confinement (e.g. emergency ventilation rate, filter efficiency, leak rate, liquid effluent release rate, radioactive decay due to time delay of release, deposition on surfaces and resuspension).

In addition, the doses associated with the source term depend on the release mode (single puff, intermittent, continuous) and the release point (stack, ground level, confinement bypass).

It is expected that the radiological consequence evaluation for the transients discussed in this section will be calculated employing [

] [CCI per Affidavit 4(a)-(d)]

4.3 Core Inventory of Fission Products

Accurate and problem-dependent nuclear fuel depletion analyses to ascertain the core radiological inventory during the core operating cycle are typically performed using computational systems that couple reactor physics transport codes with burnup codes. More specifically, the burnup codes solve the neutron transmutation and decay equations that define the time-dependent nuclide concentrations, and the transport codes are used to calculate reaction rates in the system from which effective cross sections are derived. These cross sections are, in turn, passed to the burnup code to calculate the change in nuclide compositions in the material with time. At intervals throughout an irradiation simulation, the reactor physics code must recalculate cross sections for the burnup phase of the analysis to reflect the changes with time in nuclide concentrations and other reactor operating conditions. This coupled transport-depletion computational methodology is used to determine the mPower core isotopic inventory.

The B&W mPower core isotopic inventory for radiological source term application is calculated with the [

] [CCI per Affidavit 4(a)-(d)]

4.4 Release Fractions, Timing of Release, Radionuclide Composition, and Chemical Form

In accordance with RG 1.183, Table 4, releases from the core to the containment are assumed to occur in two phases. The gap release phase starts at 30 seconds and lasts 30 minutes. The early in-vessel release phase starts at the conclusion of the gap release phase and lasts 1.3 hours.

Core nuclide release fractions to the containment are assumed to be in accordance with RG 1.183, Table 2. During the gap release phase, 5 percent of the noble gases, halogens, and alkali metals are released. During the early in-vessel release phase, 95 percent of the noble gases, 35 percent of the halogens, and 25 percent of the alkali metals are released, along with smaller fractions of other nuclides.

As specified in RG 1.183, Section 3.5, the chemical form of iodine released into the containment is assumed to be 95 percent cesium iodide, 4.85 percent elemental iodine, and 0.15 percent organic iodide. With the exception of noble gases, all other fission products are assumed to be in particulate form.

4.5 Aerosol Scrubbing in Reactor Vessel

The transport of aerosol-laden radionuclide gases are affected by water. Aerosols within gas bubbles can diffuse, sediment, or inertially impact the gas-water interface. Surface tension and van der Waals forces assure that when an aerosol particle reaches the gas-water interface, it will be trapped in the aqueous phase. This trapping is somewhat similar to the aerosol trapping that occurs when aerosol-laden gases sparge through steam suppression pools in boiling water reactor accidents (References 9-11).

4.5.1 Design Features that Preclude Core Uncovery

The B&W mPower reactor incorporates design features for the specific purpose of preventing core uncovery, thus preserving an aqueous medium around the reactor core for the trapping of aerosol-laden radionuclides. These are used throughout the RCS design starting with the core and include the RCS pressure boundary, isolation capability of the RCS, and fluid injection capability of the ECC. Some of these enhancements include:

- [

] [CCI per Affidavit 4(a)-(d)]

These design features make the B&W mPower more reliable in preventing fuel clad damage and assures long-term cooling of the core in normal operation and during design basis events.

4.5.2 Evaluation Methodology Crediting Scrubbing

As indicated above, core uncovery is prevented by design features available during a design-basis event. As such, there will be no fuel damage and the releases described in RG 1.183 would not occur. It is, however, conservative to assume that fuel damage does occur, resulting in the RG 1.183 releases, but credit is taken for retention of particulates in the vessel water during the transport from the fuel to the break location.

Powers and Sprung have evaluated the phenomenon of aerosol scrubbing from bubbles rising through the core, as described in NUREG/CR-5901 (Reference 12). The model described in this report considers aerosol capture by diffusion, sedimentation, and inertial impaction. The decontamination of aerosols produced during core debris interactions with concrete by a water pool of specified depth and subcooling is analyzed using a Monte Carlo method while considering uncertainties in a number of variables, including properties of the aerosols, the bubbles, the water, and the ambient pressure. The resulting decontamination factors at confidence levels of 50, 90, and 95 percent for pool depths ranging from 30 to 500 cm and subcooling levels of zero to 70°C are correlated by polynomial regression. The polynomial equations can be used to estimate decontamination factors at prescribed confidence levels.

The Powers and Sprung correlations will be used to calculate conservative aerosol decontamination factors suitable for the conditions in the mPower core.

4.6 Evaluation Methodology Crediting Containment Deposition

RG 1.183 allows credit for reduction in airborne activity in the containment by natural deposition on containment surfaces. For mPower, the MELCOR (Version 1.8.6) computer program (Reference 13) is used to calculate the deposition rates. MELCOR is a fully integrated, computer simulation code intended to model the progression of accidents in light water reactors.

Aerosol behavior in MELCOR is calculated using the sectional method developed by Gelbard, as described in the MELCOR Reference Manual. Gelbard discretizes the airborne mass into aerosol size distribution bins between 0.1 μm and 50 μm, typically using ten (or user-defined

number) equal logarithmic size sections. Agglomeration of smaller aerosol particles to form larger particles is considered, and deposition velocities are calculated for each section, including effects of thermophoresis, diffusiophoresis, and gravitational settling.

The MELCOR calculation of changes in aerosol distribution and location considers the following general processes:

1. Aerosol phenomenological sources such as release from fuel rods or during core-concrete interactions, and/or arbitrary user-specified sources.
2. Condensation and evaporation of water and fission products to and from aerosol particles.
3. Particle agglomeration (or coagulation), whereby two particles collide and form one larger particle.
4. Particle deposition onto surfaces or settling through flow paths into lower control volumes.
5. Advection of aerosols between control volumes by bulk fluid flows.
6. Removal of aerosol particles by engineered safety features (ESFs), such as filter trapping, pool scrubbing, and spray washout.

The MELCOR aerosol mechanics deposition model for each type of surface is made up of four contributions: gravitational deposition, Brownian diffusion to surfaces, thermophoresis, and diffusiophoresis. Of these natural depletion processes, gravitational deposition is often the dominant mechanism for large control volumes such as those typically used to simulate the containment, although phoretic effects may be significant in some cases (e.g., diffusiophoresis during water condensation). Particle diffusion is generally a relatively unimportant deposition process. The deposition velocities are calculated using the well-known aerosol equations presented in the MELCOR Reference Manual. The deposition rate for a heat sink for an aerosol section (calculated in sec^{-1}) is given as:

$$K_{j,l} = \frac{A_j}{V}(v_{grav} + v_{diff} + v_{therm} + v_{diffusio}) \qquad \text{(Equation 1)}$$

Where

- $K_{j,l}$ = Deposition rate for structure j, for aerosol section i
- A_j = Area of the structure
- V = Volume of the compartment
- v_{grav} = Gravitational (settling) velocity
- v_{diff} = Brownian diffusion velocity

- v_{therm} = Thermophoretic diffusion velocity
- $v_{diffusio}$ = Diffusiophoresis velocity

The above deposition rates are for a section (bin) of aerosols, and for one structure. The results for all structures and all sections can be integrated using control functions and tabular functions, for each radionuclide group.

MELCOR is utilized to evaluate the natural depletion process of aerosols using the models of aerosol fallout and deposition mechanisms, for the entire containment. In particular, it is intended to characterize the time constant for aerosol depletion (deposition due to the four processes described above) by a first order equation that relates the depletion rate of aerosol product in the containment to the mass of aerosol in a compartment as:

$$\frac{dm}{dt} = \overline{S} - \lambda . m \qquad \text{(Equation 2)}$$

Where

- m = Instantaneous mass of airborne containment aerosol (or fission product vapor)
- $\overline{S}$ = Source rate of aerosol (or fission product vapor) entering the containment
- λ = First order depletion constant of aerosol (due to fallout and deposition)

MELCOR is used to model the aerosol depletion process in the mPower containment, and to determine the overall instantaneous value of λ as a function of time using Equation 2. The focus is on determining the expected distribution of possible values for the decontamination coefficient, λ, for a LOCA scenario with no injection from the RWST into the vessel. The LOCA conditions establish thermal hydraulic boundary conditions for the containment environment.

In order to simulate the gap and early in-vessel release phases in RG 1.183 using MELCOR, [

] [CCI per Affidavit 4(a)-(d)]

4.7 Containment Leakage

Airborne activity in containment is assumed to leak to the environment at 0.1 percent volume per day, a value expected to be incorporated into the mPower technical specifications. RG 1.183 allows the technical specification leakage rate to be reduced by 50 percent after the first 24 hours of the accident for PWRs. This reduction will be credited for mPower if it can be supported by an evaluation of the post-accident containment pressure profile. If a reduction of 50 percent cannot be justified, a smaller reduction may be credited.

5. Conclusions

Preliminary analyses of the mPower design have shown that the biggest challenge to maintaining core coolant in the reactor vessel, thus preserving fuel integrity, is a [

]

[CCI per Affidavit 4(a)-(d)]

6. References

The following references are cited within the text:

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2. U.S. Nuclear Regulatory Commission, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” Regulatory Guide 1.183, Revision 0, July 2000.
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