Enclosure 3
Position Paper on Radiological Source Term Methodology
(Redacted)

babcock & wilcox mPower

Position Paper on Radiological Source Term Methodology for B&W mPower™ Reactor

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B&W mPower™ Reactor Program
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RECORD OF REVISION

Revision No.	Date	Preparer	Description of Changes
000	June 2012	R. P. Martin J. D. Malloy	Initial Release
001	October 2013	R.P. Martin J.D. Malloy J.E. Chang	Due to design changes and changes in methodology, primarily Sections 1, 2, 3, and 4 have been updated.

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1. Introduction

Babcock & Wilcox (B&W) letter MPWR-LTR-12-00068 to the U.S. Nuclear Regulatory Commission (NRC), dated July 9, 2012, provided a position paper describing the methodology for calculating an accident source term for addressing reactor siting criteria for the Babcock & Wilcox mPower™ (B&W mPower) reactor design. The purpose of the following is to update this paper to reflect changes in the B&W mPower design and in the source term evaluation methodology.

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 robust B&W mPower defense-in-depth approach begins with a design strategy that precludes the possibility of fuel damage in design-basis events, and is complemented by defense-in-depth design features and characteristics to address postulated but unlikely conditions for which fuel damage is conservatively assumed to occur.

Considering the features in the B&W mPower reactor, the radiological consequences are expected to be significantly less than the total effective dose equivalent (TEDE) safety limits at a plant's exclusion area boundary (EAB) and low population zone (LPZ) expressed in 10 CFR Part 50 and Part 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," RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and NUREG-0800, Standard Review Plan (SRP), Chapter 15 (References 2 – 5). The source term methodology framework described herein is a roadmap for how B&W mPower intends to address radiological consequence analysis associated with design-basis plant safety analyses in Chapter 15 of the Design Control Document (DCD).

B&W mPower is committed to an application of the existing regulatory framework, with the [ ]. It is expected that the resulting quantification of the accident source term and its associated dose consequences will establish the EAB and LPZ well below the regulatory limits. B&W mPower also supports a reexamination by the NRC of the existing regulatory framework to reflect design advances being made by the small modular reactor (SMR) vendors (Reference 6), thus resulting in a more appropriate designation of the radiological source terms for SMRs.

Specifics on the application of the evaluation methodology to Chapter 15 transients and accidents will be presented in a separate topical report. Software and analytical treatments of computer code input will be applied in a manner consistent with industry precedence, meaning deterministic analysis will treat important phenomena conservatively. Realistic analysis will also be performed to quantify analytical margins. Further, the results from realistic analyses will serve a second purpose related to the sizing of the emergency planning zone (EPZ). While analysis of radiological consequences addresses both plant siting and the EPZ, B&W mPower recognizes that these are distinct evaluations. B&W mPower is participating with other SMR

vendors in ongoing discussions between the Nuclear Energy Institute (NEI) and the NRC on both design-basis source term and EPZ evaluation methodology issues.

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 (ALWRs) are very stringent with regard to the radiological impact on the public. To demonstrate the safety of the plant, the following basic objectives should be fulfilled per Reference 7:

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 postulated 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 mitigative actions under accident conditions,
    - minimizing and mitigating 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 applicable 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.

1.2 Radiological Consequence Requirements

As described in RG 1.183, an accident source term is intended to be representative of a major accident involving significant core damage and is typically postulated to occur in conjunction with a large break 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. The 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 low population zone, 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."
- General Design Criterion (GDC) 19 of Appendix A to 10 CFR 50, "Control room," as it relates to maintaining the control room (CR) in a safe condition under accident conditions by providing adequate protection against radiation, ensuring that radiation exposures shall not exceed 5 rem TEDE.

- 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.

2. General Design Overview

The B&W mPower reactor is an advanced pressurized light water reactor (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. This flange provides electrical feed-throughs for the 69 CRDMs. The CRDM motors and release mechanisms are physically located immediately above the control rod guide frames and the top of the CRDMs are below the lower vessel flange. As a result, the CRDMs are entirely within the RCS pressure boundary, which makes the rod ejection accident non-credible because there is no significant differential pressure to force rods out of the core.

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 such that the RCP seal leak is precluded. The steam generator is provided with a single normal feedwater and steam outlet penetration. Feedwater penetrations near the upper tube sheet are provided for auxiliary cooling (CNX).

The integral design of the RCS eliminates all large bore reactor coolant piping. All reactor vessel penetrations below the top of the pressurizer are less than [ ] inches in diameter and are automatically isolated by redundant valves bolted directly to the reactor vessel. Five penetrations in the pressurizer region provide pressurizer spray and

paths to the code safety valves and the automatic depressurization valves (ADVs). RCS operating conditions are provided in Table 1.

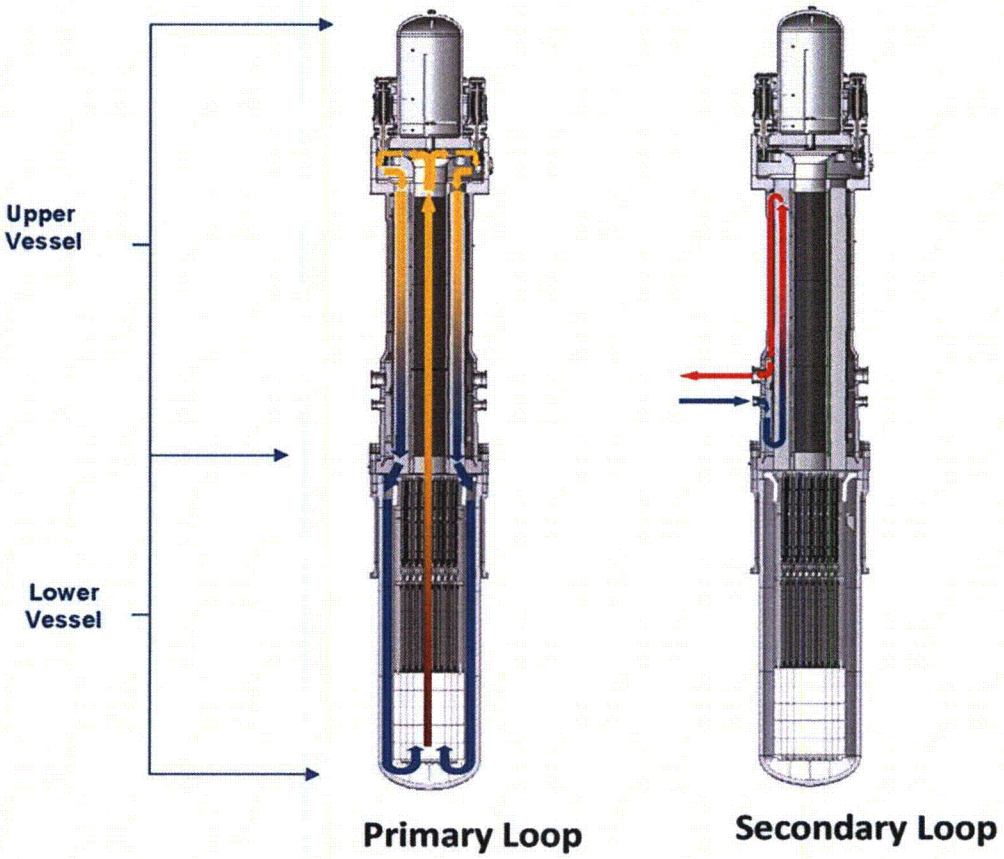

Figure 1. RCS General Arrangement

Table 1. General RCS Specifications

Operating power, MWt	530
Operating pressure, psia (MPa)	2060 (14.2)
Design pressure, psia (MPa)	2300 (15.9)
Cold leg temperature, °F (°C)	564 (296)
Hot leg temperature, °F (°C)	606 (319)

Total coolant flow @ full power, millions of pounds per hour (kilograms per second)	31 (3906)

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% $_{235}U$, integral burnable poison rods (BPRs) consisting of UO_2 mixed with gadolinium oxide (Gd_2O_3) (to minimize assembly-to-assembly peaking), non-integral aluminum oxide (Al_2O_3)-boron carbide (B_4C) BPRs (to suppress the excess reactivity at the beginning of the cycle), and control rod guide tubes and are designed to meet the requirement for adequate shutdown margin with the most reactive control rod stuck out after reactor trip. Groups, or banks, of control rod assemblies allow control of core reactivity, including fine control for shaping axial power, while also providing adequate shutdown margin under all operating conditions.

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 two identical, redundant trains with all components located inside containment. Each train includes two sets of high pressure and low pressure ADVs, an intermediate pressure injection tank (IPIT), and one of the two compartments of the refueling water storage tank (RWST). The ADVs are connected to the pressurizer region of the RCS and the IPIT and RWST are connected to the lower reactor vessel by injection lines that utilize integral isolation valves bolted directly to the lower vessel.

On very high pressurizer pressure or very low pressurizer level, the high pressure ADVs will open to either depressurize the RCS or accelerate the depressurization of the RCS. As the RCS depressurizes, the IPITs will inject to control minimum water level. When RCS depressurization slows, the low pressure ADVs are opened to continue depressurization until water from the RWST can drain into the reactor vessel through redundant emergency pressure regulating valves (EPRVs). The EPRVs control water flow from the RWSTs to maximize exit quality through the ADVs and any potential pipe break. As a result, the ECC can maintain core cooling in either loss of heat sink or loss-of-coolant accidents (LOCAs) for a minimum of seven days. In all design basis accidents, the ECC provides sufficient water during depressurization to ensure that collapsed water level in the reactor vessel covers the reactor core to prevent film boiling throughout that seven day period.

2.1.3 Auxiliary Condenser System

The auxiliary condenser system (CNX) is a non-safety system consisting of a high pressure, air-cooled condenser and piping connecting the inlet of the condenser to the main steam line upstream of the containment isolation valve and condensate line connecting to the steam generator auxiliary injection ports. The CNX is initiated on loss

of normal feedwater flow by opening the redundant isolation valves on the condensate side of the condenser and by starting the fans providing air flow to the condenser. The fans are powered by redundant batteries, each set sized to ensure a minimum of eight hours of operation without alternating-current (AC) power. The system is maintained in a hot standby condition during normal operation to prevent water hammer and thermal stress issues.

Once activated, the condenser fans are controlled to maintain the RCS in hot standby conditions. The system has the capability to cool the RCS to safe shutdown conditions using battery powered RCI makeup pumps to provide sufficient makeup to counter reactor coolant shrinkage.

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,
- provide makeup and letdown to the RCS during startup and shutdown and if required during normal operation,
- low pressure residual heat removal,
- high pressure decay heat removal, and
- injection of sodium pentaborate for emergency shutdown.

The RCI automatically provides makeup and letdown based on pressurizer water level and will initiate high-pressure decay heat removal on high RCS pressure. Operator action may be taken to initiate low-pressure residual heat removal or soluble neutron poison injection when appropriate.

2.1.5 Component Cooling Water System

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

- RCI in-containment non-regenerative heat exchangers (used for normal coolant purification and high pressure decay heat removal),
- RCI low-pressure residual heat removal heat exchangers,
- RCPs, and
- spent fuel pool heat exchangers.

During normal operation, heat is transferred from CCW using mechanical chillers or using a cooling loop from the main cooling towers. When normal AC power is not available, a forced draft air-cooled heat exchanger is used as the heat sink.

2.1.6 Reactor Containment

The reactor containment is **[**

]. 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 water in a passive containment cooling tank (PCCT) located on the top dome of the containment. Under accident conditions, heat is removed from the hot steam and air inside containment through the containment dome structure to the water in the PCCT on the outside surface of the dome. Sufficient water is available to cool the containment atmosphere by evaporation without makeup to the PCCT for up to 14 days.

A concrete internal structure limits the volume around the lower reactor vessel and forms the refueling cavity. The walls of this structure support the refueling machine, the upper vessel mover and a bridge crane for moving the reactor vessel internals and other miscellaneous components. The floor of the refueling cavity is immediately below the lower vessel flange. The lower vessel is located in a much smaller reactor vessel cavity. While isolated during refueling to prevent water ingress, this region is open during normal operation. In the event of a LOCA, water will collect initially in the reactor vessel cavity before flooding the refueling cavity. **[**

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2.1.7 Class 1E DC Power Supply

The Class 1E DC/UPS (DC) system provides reliable power for safety-related equipment including the reactor protection system (RPS), engineered safety features actuation system (ESFAS), nuclear instrumentation, solenoid and motor-operated valves, emergency lighting, and control room emergency heating, ventilation, and air conditioning (HVAC) equipment. The Class 1E DC systems supply 125V DC power to DC loads and inverters. Class 1E UPS systems supply uninterruptible 120V AC power to AC loads via inverters.

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 two separate and independent standby diesel generators that are available to power the 480V essential

bus during a loss of the normal and preferred AC power supplies. The diesels are located in the reactor service building (RSB) and are designed to provide power after a design basis event (e.g., safe shutdown earthquake).

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 CNX, RCI and CCW. The interaction of these systems is schematically shown on Figure 2. The response of the nuclear island to various plant upsets illustrates the benefits of the system defense-in-depth approach. Some of these scenarios are outlined briefly below.

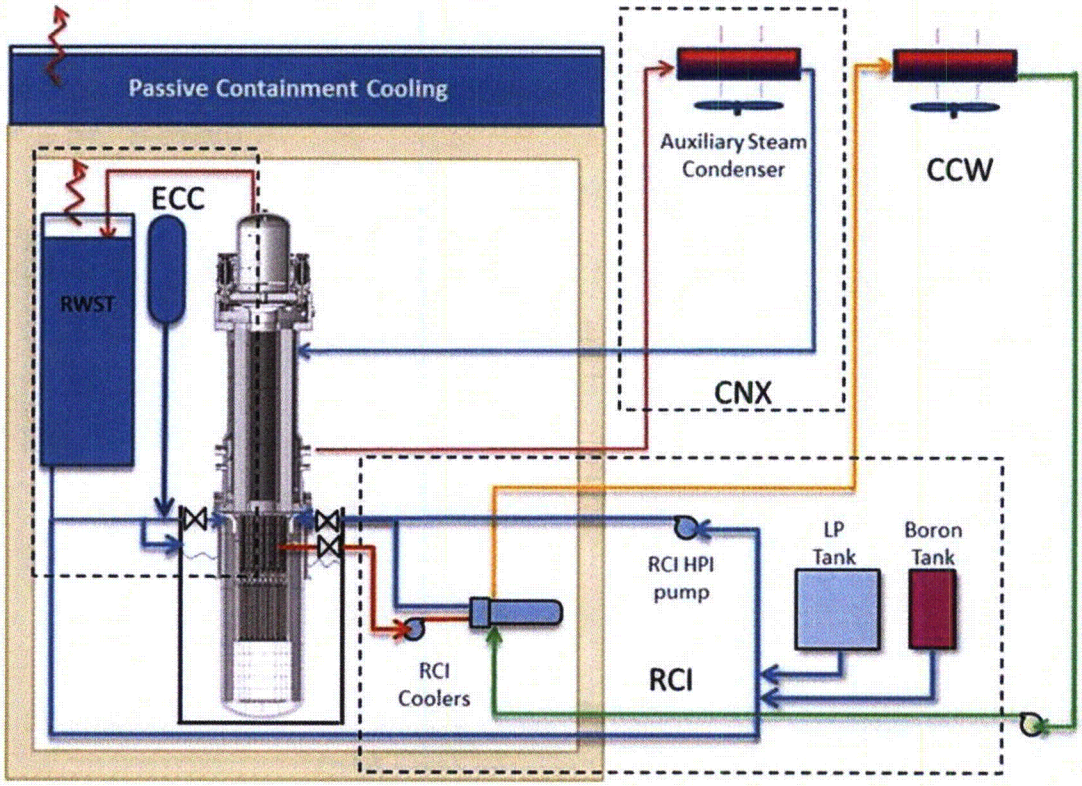

Figure 2. System Defense-in-Depth Approach

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 that prevent adequate turbine bypass will result in a reactor trip. This will automatically result in closure of the main steam and feedwater isolation valves and initiation of plant cooling using CNX. If the CNX system is ineffective or unavailable, RCI high pressure decay heat removal will be initiated automatically.

2.2.2 Loss of Normal Feedwater Flow

Normal plant response – Loss of feedwater flow will initiate a reactor trip, closure of the main steam and feedwater isolation valves, and initiation of CNX operation. The plant control system will control CNX fan speed to maintain the RCS in hot standby conditions until the operator restores feedwater flow or begins orderly plant shutdown.

Faulted plant response – If CNX does not provide adequate heat removal, RCI high pressure decay heat removal will be automatically initiated. RCI will continue to cool the RCS and transition to low pressure residual heat removal. If RCI is not available, ECC will be initiated to depressurize the RCS and begin long-term decay heat removal.

2.2.3 Loss of Off Site Power

Normal plant response – Loss of the grid will result in closure of the turbine throttle valve to prevent overspeed, leading to a rise in steam pressure and the opening of the turbine bypass valves. Reactor power will then be reduced until 20% power is reached. The plant bus will simultaneously isolate from the grid to maintain voltage in the plant. Turbine load will drop to match station power requirements with excess steam being sent directly to the condenser.

Faulted plant response – Loss of feedwater flow will initiate a reactor trip, closure of the main steam and feedwater isolation valves, and initiation of CNX operation. The plant control system will control CNX fan speed to maintain the RCS in hot standby conditions until the operator restores feedwater flow or begins orderly plant shutdown. Fans are powered by DC-powered motors connected to redundant non Class 1E batteries, each with an eight hour operating capacity. When the standby diesels are started, the auxiliary power system will recharge the batteries to ensure continuous operation.

Station blackout response – The station blackout is a loss of offsite power with failure to transition the plant to island mode and a failure of both standby diesels to start. If CNX is unable to adequately remove heat and auxiliary power cannot be recovered, the ECC will be initiated on high RCS pressure. This will result in RCS depressurization and long-term decay heat removal using water in the RWST compartments.

2.2.4 Loss-Of-Coolant Accident

The B&W mPower plant is limited to small breaks ([ ] inches and smaller) due to the integral design of the RCS. RCI makeup will be initiated on low water level and flow progressively increased as the error signal between desired and actual water level increases. If water level in the pressurizer continues to fall, then a reactor trip signal will be generated and if level continues to fall, letdown lines will be isolated at the integral isolation valves. A significant leak on a makeup line will be isolated by redundant check valves (makeup line integral isolation valves).

Breaks in piping leading to the ADVs and code safety valves will result in unmitigated inventory loss to the containment. When water level drops below the pressurizer heaters, high pressure ADVs will be opened to ensure RCS depressurization. This is followed by automatic injection from the IPITs and opening of the low pressure ADVs. When the RCS pressure drops below the static head of the RWST, water will begin to drain into the reactor vessel. Steam, with some water, is vented back to the RWST where the steam is released to containment. When water level in the RWST drops sufficiently, flow to the reactor vessel will be controlled by the EPRVs to prevent overfill of the reactor.

2.2.5 Anticipated Transient Without Scram

Any operating transient that generates RCS parameters outside of the acceptable operating envelope will result in the initiation of a scram signal in the four-channel RPS. The scram signal will open the [ ], the CRDM latches will move releasing the control rods and allowing them to fall into the reactor core.

In addition to the redundant methods of dropping rods into the reactor, the operator has two additional long-term methods of taking the core subcritical. The first is to use the motors within the CRDMs to insert the control rods. The second is to use the RCI high pressure injection pumps to inject sodium pentaborate from a tank in the RSB into the reactor vessel. Either method will provide acceptable shutdown margins when the RCS is cold.

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. If the defense-in-depth systems are incapable of doing this because the initiating event is too severe or because of multiple failures, the ECC system

depressurizes the plant and provides long-term core cooling while the PCCT provides long-term heat rejection to the atmosphere. These safety features provide cooling for at least seven days without operator action. Both sources are capable of being replenished by operator action to maintain core and containment cooling indefinitely.

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, Standard Review Plan (SRP). This information is condensed from that appearing in Reference 8. 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 system 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	SRP section is applicable to the B&W mPower reactor. The design employs a conventional main steam system.
15.1.5.A	Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR	SRP section is applicable to the B&W mPower reactor. The design employs a conventional main steam system.
15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	SRP section is applicable to the B&W mPower reactor. The design employs small lines carrying reactor coolant outside containment, such as instrument lines.
15.6.3	Radiological Consequences of Steam Generator Tube Failure	SRP section is applicable to the B&W mPower reactor. The design employs an OTSG which is similar to operating PWR steam generators of the same type.
15.6.5.A	Radiological Consequences of a Design Basis Loss-of-Coolant Accident Including Containment Leakage Contribution	SRP section is applicable to the B&W mPower reactor. This design employs a containment that is similar in design to other PWRs.
15.7.3	Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	SRP section is applicable to the B&W mPower reactor. This design includes presence of liquid containing tanks.
15.7.4	Radiological Consequences of Fuel Handling Accidents	SRP section is applicable to the B&W mPower reactor. This design will include refueling activities, during which a fuel handling accident could occur.
15.7.5	Spent Fuel Cask Drop Accidents	SRP section is applicable to the B&W mPower reactor. The design will include the spent fuel cask handling system.

Analysis of 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 Sections 15.7.4 and 15.7.5 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 paper.

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

- A pre-accident activity concentration – This refers to the consequence of defective fuel providing a leak path for fission product releases. During normal operation, the defective fuel gap is filled with steam and/or water, equilibrating with the coolant pressure, and the fission product release rate depends on the size of defect. A bounding leakage rate from the primary to the secondary system, coupled with coolant activity prescribed by RG 1.183 and SRP Chapter 15 will be used. This will serve as the basis for the corresponding B&W mPower Technical Specification.
- An accident-induced iodine spike – The reduction in power and pressure accompanying a reactor trip and system blowdown from either a LOCA or operation of the ADVs results in a spike in I-131. During power reduction, a portion of the fuel cools down, and the liquid water is forced into the defect fuel gap. The decay heat is still high enough to evaporate the water into steam, and some fission products leak out with the steam. When the pressure starts to drop, the higher pressure inside the fuel cladding begins to push the water- and steam-carrying fission products out of the cladding through the defect hole.
- In accordance with the guidance in NRC RIS 2006-04 (Reference 9), all halogen, noble gas, and alkali radionuclides appearing in the RCS and/or secondary coolant are considered released. Other radionuclides are assumed to remain within the liquid phase (RCS and secondary coolant). The release shall be modeled to last until the plant is cooled down.
- A loss-of-offsite-power (LOOP) coincident with the event or with a reactor trip (if more restrictive) is assumed for the design-basis radiological evaluations. A LOOP is not considered a single, active failure, but an addition to a single, active failure.
- The design-basis radiological evaluations credit safety-related structures, systems, or components (SSCs) to mitigate the radiological consequences of a LOCA or an AOO. However, nonsafety-related SSCs are assumed operational if the assumption results in a more limiting radiological consequence.
- Departure from nucleate boiling (DNB)-induced cladding failure – based on system and subchannel analysis evaluating DNB ratio (DNBR) relative to the 95/95 DNBR limit.
- Fuel centerline melt (FCM)-induced fuel failure.

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. In this scenario, the main steam line pressure quickly reaches the setpoints for actuation for main steam and feedwater isolation and turbine trip. A reactor trip will follow shortly afterward, most likely on low feedwater flow. This event is expected to be the limiting AOO event for radiological consequences. The dose acceptance criteria for such events are defined in 10 CFR 50, Appendix I, for the summation of radioactive releases during normal operation and the annual average radioactive releases due to an AOO event based on realistic assumptions. The RCS and secondary coolant radionuclide concentrations correspond to normal operating conditions. At zero power level when the turbines are off line, the opening of a relief or safety valve can cause return to power, but the outcome would be bounded by the steam line break transient discussed in the following section.

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, one not expected to occur during the life of the NPP. Because pipe failures inside containment or within the space between the reactor services building and the containment are radiologically bounded by similar failures outside of the RSB, analysis of this class of events focuses on those bounding scenarios.

The limiting accident is expected to be a double-ended break of the main steam line downstream of the main steam isolation valve (MSIV). As with the inadvertent opening of a steam generator relief or safety valve, there is isolation of main steam and feedwater and a turbine trip, soon followed by a reactor trip. The atmospheric releases occur via the main steam relief train, a consequence of a plant cooldown without the main condenser.

The evaluation objectives include the determination of maximum departure of nucleate boiling (DNB)-induced cladding failure, and maximum FCM-induced fuel failure, that can be accommodated, independently, for an MSLB accident without exceeding 10 percent of the dose acceptance criterion at any receptor.

The MSLB is not expected to challenge the cladding damage or fuel melt criteria in the B&W mPower reactor. As such, the concurrent iodine spike leads to the bounding doses at the receptors of interest.

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 instrument, sample, radwaste, or RCI lines. This event is expected to be bounded by the LOCA involving the double-ended severing of the largest pipe connected to the RPV.

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 two iodine spike scenarios described previously. For a time, while the plant is at full power, the steam release is via the condenser and vent stack. After the condition is detected, the secondary side is isolated, the CNX is put into operation, and any additional release is via the steam generator main steam relief train.

3.5 Radiological Consequences of Fuel Handling Accidents

The fuel handling accidents (FHA) may occur in the containment as well as in the spent fuel pool area of the RSB. The B&W mPower reactor fuel is essentially identical to current large PWR fuels except for fuel assembly height. Further, the damaged fuel will be covered with at least 23 feet of water as provided in Regulatory Position 2 of Appendix B to RG 1.183. As such, the B&W mPower FHA will be similar to current large PWR FHA and will comply with RG 1.183.

3.6 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 [

].

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 [

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4.1.1 [

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4.1.2 [

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4.1.3 [

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[1] This non-safety system is not credited in Chapter 15 safety analyses.

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4.1.4 [

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4.1.5 [

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4.2 Evaluation Methodology

The release of radioactivity 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 and coolant systems,
- the progression of core damage (if any),
- the timing and fraction of radionuclide release from the fuel,
- the physical and chemical forms of released radioactive materials,
- the retention of radionuclides in the primary cooling system, and
- the containment performance relative to radioactivity mitigation (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, containment bypass).

The radiological consequence evaluation for the accidents discussed in this section will be calculated employing [ ]

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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 B&W mPower core isotopic inventory.

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

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4.4 Timing of Release, Release Fractions, 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 after the start of the MHA

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, five 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 and the elemental and organic components of the iodine release, all other fission products and actinides are assumed to be in particulate form.

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4.5 Alternative Time for Onset of Gap Release

In accordance with RG 1.183, Section 3.3, an alternative time for the onset of the gap release phase may be proposed based on facility-specific calculations using suitable analysis codes or on an accepted topical report.

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 adequate coolant mixture around the reactor core which precludes potential cladding damage during a LOCA. These features have been implemented 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:

- low core average linear heat rate of [ ] (vs. 18.7 or higher for conventional PWRs),

- high RCS volume per unit power of **[** **]** (vs. ~0.08 m^3/MW for conventional PWRs),
- maximum RCS break size of **[** **]** inches (vs. 36 inches for conventional PWRs),
- no Class 1 piping below the top of the pressurizer,
- high and low pressure ADVs to assure rapid primary system depressurization, assuring gravity-driven coolant injection from the RWST,
- IPIT providing additional coolant inventory during blowdown, and
- RWST coolant volume sufficient for at least seven days.

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

4.5.2 Evaluation Methodology Crediting Alternative Time

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. However, it is conservatively assumed that fuel damage does occur, resulting in the RG 1.183 releases, **[**

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4.6 Evaluation Methodology Crediting Containment Deposition

Regulatory Guide 1.183, Section 3.2 allows credit for reduction in airborne activity in the containment by natural deposition on containment surfaces. For B&W mPower, the MELCOR (Version 1.8.6) computer program (Reference 12) is being used to study the deposition rates. MELCOR is a fully integrated, computer simulation code intended to model the progression of accidents in light water reactors. MELCOR will be utilized to evaluate the natural depletion process of aerosols using the models of aerosol sedimentation and other deposition mechanisms (i.e., Brownian diffusion to surfaces, thermophoresis, and diffusiophoresis), for the entire containment. **[**

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4.7 Containment Leakage

Airborne activity in containment is assumed to leak to the RSB at 0.1 percent volume per day, a value expected to be incorporated into the B&W 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 B&W mPower if 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.

4.8 Activity Holdup in Reactor Service Building

As the containment is fully enclosed by the RSB, any containment leakage would enter the RSB before being released to the environment. The RSB is divided into two areas; the annular area outside the containment and the rest of the RSB. [

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4.9 Scoping Study

A scoping study of LOCA doses has been performed to understand the relative contributions of the various elements of the proposed methodology and to qualitatively demonstrate the effectiveness of the B&W mPower defense-in-depth features. **[**

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5. Conclusions

Preliminary analyses of the B&W mPower design have shown that the biggest challenge to maintaining core coolant in the reactor vessel, thus preserving fuel integrity, is a pipe break in a relief, safety, or automatic depressurization line [ ]. B&W mPower safety systems are designed such that uncovery of the core does not occur for this limiting design basis accident. The radiological consequences of an event that does not result in core uncovery and subsequent fuel damage are limited to a release of the activity in the primary coolant.

However, the DCD Chapter 15 evaluation of the radiological consequences of the limiting design basis accident (the MHA) will conservatively use the core nuclide release fractions, release timing and release chemical forms specified in RG 1.183, [ ]. Atmospheric dispersion factors that bound a majority of potential sites in the U.S will be used along with a containment leak rate of 0.1% per day. Deposition rates of radionuclides inside containment will be calculated using the MELCOR code. The natural deposition models in NUREG/CR-6604 will also be considered and [ ]

The results of this modified RG 1.183 analysis are expected to show that [ ] the TEDE safety limits expressed in 10 CFR Part 50 and Part 100.

Scoping study results demonstrate the significant safety benefit achieved by the B&W mPower design, which by effectively eliminating breaks below the top of the pressurizer and providing large quantities of ECC inventory inside containment, achieves at least seven days of core cooling and thus substantially reduces doses, [ ]

While radiological consequences analyses are performed to support both plant siting, i.e., EAB and LPZ, and EPZ, B&W mPower recognizes that these are distinct evaluations. A separate technical report is planned to address EPZ sizing evaluation methods, involving best-estimate and risk-informed techniques. B&W mPower is participating with other SMR vendors in ongoing discussions between the NEI and the NRC on EPZ evaluation methodology issues.

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