Enclosure 3
B&W mPower™ Approach to Satisfy GSI-191
MPWR-EPP-005009
(Redacted)

babcock & wilcox mPower

B&W mPower™ Approach to Satisfy GSI-191

MPWR-EPP-005009
Revision 000
06/27/2012
(Redacted Version)

a progressive energy solution

B&W mPower™ Reactor Program
Babcock & Wilcox mPower, Inc.
109 Ramsey Place
Lynchburg, VA 24501

Open Items	Safety Classification	Proprietary	Files Attached	Total No. Pages
Y	N	Y	N	17

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

Document No.	Title	Rev. No.
MPWR-EPP-005009	B&W mPower™ Approach to Satisfy GSI 191	**000**

SIGNATURES			
Prepared By:	John Malloy	John Malloy	6/28/12
	Printed Name	Signature	Date
Reviewed By:	Ronald Watson	Ronald Watson	6/28/12
	Printed Name	Signature	Date
Reviewed By:			
	Printed Name	Signature	Date
Approved By:	David Kanuch	David Kanuch	6-28-2012
	Printed Name	Signature	Date

Document No.	Title	Rev. No.
MPWR-EPP-005009	B&W mPower™ Approach to Satisfy GSI 191	**000**

RECORD OF REVISION

Revision No.	Date	Preparer	Description of Changes
000	06/28/12	John Malloy	Initial Revision

Document No.	Title	Rev. No.
MPWR-EPP-005009	B&W mPower™ Approach to Satisfy GSI 191	**000**

TABLE OF CONTENTS

List of Tables (NA)

List of Figures

Document No.	Title	Rev. No.
MPWR-EPP-005009	B&W mPower™ Approach to Satisfy GSI 191	**000**

1. SUMMARY

The purpose of this report is to provide information to demonstrate that Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on PWR Sump Performance," is not applicable to the B&W mPower design. Therefore, the B&W mPower program does not need to evaluate post-accident containment debris for the purpose of sizing sump screens and does not need to test screen designs based on the assumption of operation in a post-accident environment.

2. INTRODUCTION

GSI-191 was initiated by the Nuclear Regulatory Commission (NRC) to resolve a concern where, at a typical pressurized water reactor (PWR) nuclear power plant, in the event of a loss-of-coolant accident (LOCA), thermal insulation and other materials in the vicinity of the pipe break are dislodged by the impingement of a high-energy steam/water jet. In such instances, some of the debris could be transported to the containment sump, where it accumulates and causes a pressure drop (i.e., head loss) across the sump screens. This debris accumulation and the resulting pressure drop, could then challenge the plant's ability to provide adequate long-term cooling, resulting in potential non-compliance with the long-term cooling requirements in 10 CFR 50.46(b)(5). As a result of research on this issue, the NRC requested licensees provide sufficient information to assess the potential for debris blockage of flow paths during design basis accidents requiring recirculation and containment drainage (References 9.1, 9.2, 9.3, 9.4 and 9.5).

Resolution of GSI-191 typically requires in-depth evaluations of debris generation, debris characteristics, debris transport, debris head loss, net positive suction head, upstream effects, and chemical effects. However, the B&W mPower design does not rely on recirculation of water from either the refueling cavity or reactor vessel cavity sumps for either short-term or long-term cooling after an accident. This eliminates concerns raised by GSI-191 and eliminates the need to evaluate post-accident containment debris and its impact on sump debris screens inside containment.

3. B&W mPOWER DESIGN OVERVIEW

3.1 General

The B&W mPower plant utilizes an integral reactor with simplified, passive, safety features. These features provide long-term core and containment cooling using large inventories of water. The key safety system, the emergency core cooling system (ECC) is designed to allow [

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

3.2 Emergency Core Cooling System

The ECC is the primary fluid safety system in the B&W mPower plant and ensures adequate water is available in the reactor vessel to provide core cooling. The ECC is [

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

[

]

Figure 1 – Containment Building Elevations

[CCI per Affidavit 4(a)-(d) and/or SRI]

4. Postulated Loss of Coolant Accidents (LOCAs)

The design of the B&W mPower reactor incorporates reactor vessel penetrations at the following locations:

- [

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

Potential breaks in the piping connected to these penetrations results in different accident responses.

4.1 RCS Makeup Line Breaks

[

Figure 2 – Integral Isolation Valves for Makeup and Letdown Lines

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

4.2 RCS Letdown Line Breaks

[

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

4.3 ADV/Safety Valve/Pressurizer Line Breaks

[

]

Figure 3 – Break Flow from LOCA on ADV Line

[CCI per Affidavit 4(a)-(d) and/or SRI]

4.4 Small Line Breaks (Instrument Lines)

[

]

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

4.5 Post-72 Hour Heat Removal

[

] [CCI per Affidavit 4(a)-(d) and/or SRI]

4.6 Summary

As discussed above, the ECC can maintain core cooling in the event of either a loss of heat sink or LOCA [

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

5. NON-LOCA DESIGN BASIS EVENTS

The ECC is not directly involved in non-LOCA design basis events unless they progress to a loss of heat sink event. Most non-LOCA initiating events will trigger non-safety system responses to control the plant. [

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

6. APPROACH TO MEET REGULATORY REQUIREMENTS ASSOCIATED WITH GSI-191

6.1 GSI-191 Background

The standard PWR design relies on aligning valves to allow decay heat removal pumps to recirculate water from a containment sump to provide core cooling following a LOCA. NRC concerns that such recirculation could be impaired if debris caused excessive blockage of the sump, reducing pump suction pressure below the net positive suction head needed for pump operation were the genesis for GSI-191. Subsequently, incidents involving the ECCs used for boiling water reactors indicated that the concern was not isolated to PWRs.

The functional requirements that are the regulatory concern identified in GSI-191 are found in 10 CFR Section 50.46, and in the general design criteria (GDCs) found in Appendix A of 10 CFR Part 50.

Specifically, 10 CFR 50.46 requires that the ECC have the capability to provide long-term cooling of the reactor core following a LOCA. To do so, the ECC must be able to remove decay heat so that the core temperature is maintained at an acceptably low value for the extended period of time required by the long-lived radioactivity remaining in the core. GDC 35 is cited in 10 CFR 50.46(d) and specifies additional ECCS requirements.

In addition to core cooling, containment heat removal or containment atmosphere cleanup may be accomplished in whole or in part by a system that includes recirculation from a sump. GDC 38 provides functional requirements for containment heat removal systems, and GDC 41 provides functional requirements for containment atmosphere cleanup. In many currently licensed PWRs, a containment spray system is employed to reduce the accident source term and, thereby, meet the limits of 10 CFR Part 100 or 10 CFR 50.67.

During the development of GSI-191, the NRC found that in addition to the concern about sump blockage by debris generated during a LOCA, the degree of blockage could be aggravated by debris located in containment (latent debris) that could be swept into the sump by water flow subsequent to the initial break flow. Later studies found that sump screen pressure loss could be further increased by chemical effects that result in reaction products collecting on the debris bed.

The resolution of GSI-191 culminated with the NRC issuing guidance in Generic Letter 2004-02 (Reference 9.5). The guidance specifies that designs that rely on recirculation through a sump for safety-related functions should follow specific design methodologies, including evaluation of design inputs testing to verify the design. For example, verification testing of sump screens is required to provide assurance that the pressure drop created by debris accumulation on the sump does not impair flow required for cooling the core. The NRC concluded that further verification testing is required to ensure that debris that bypasses the sump or enters the recirculation path through a break does not impair the cooling function.

6.2 B&W mPower Response to GSI-191

The B&W mPower design meets the functional requirements identified for GSI-191 [

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

7. CONCLUSIONS

In summary, the B&W mPower design addresses the concerns identified in GSI-191 as follows:

- [

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

8. DEFINITIONS, ABBREVIATIONS, AND ACRONYMS

8.1 Definitions

Refueling cavity – the containment space that is normally flooded during refueling activities. In discussion of accident scenarios, the refueling cavity also includes the space for inspection of the steam generator which is isolated from flooding by a movable partition during refueling.

Reactor cavity – the containment space surrounding the lower vessel. The reactor cavity is isolated to prevent flooding during refueling operations but has communication with the refueling cavity via ventilation openings during normal reactor operations.

8.2 Abbreviations and Acronyms

Acronym/ Abbreviation	Name
ADV(s)	automatic depressurization valve(s)
CCW	component cooling water system
CNX	auxiliary condenser system
ECC	emergency core cooling system
EPRI	Electric Power Research Institute
IIV	Integral isolation valve
IPIT	Intermediate pressure injection tank
ESF	engineered safety feature
LOCA	loss of coolant accident
NRC	United States Nuclear Regulatory Commission
PWR	pressurized water reactor
RCI	reactor coolant inventory and purification system
RCS	reactor coolant system
RSB	reactor service building
RWST	refueling water storage tank
SSC	structures, systems, and components
UHS	ultimate heat sink

9. REFERENCES

9.1 Draft RG 1.82 Rev. 4, July 2010

9.2 NEI 04-07 Volume 1, "Pressurized Water Reactor Sump Performance Evaluation Methodology" Revision 0, December 2004.

9.3 NEI-04-07 Volume 2, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02" Revision 0, December 2004.

9.4 NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactor" September 13, 2004.

9.5 Generic Letter 2004-02 Supplemental Response Content Guide, August 15, 2007.