HITACHI

GE Hitachi Nuclear Energy

**Jerald G. Head**
Senior Vice President, Regulatory Affairs

3901 Castle Hayne Road
P.O. Box 780 M/C A-18
Wilmington, NC 28402
USA

T 910 819 5692
M 910 362-5692
jerald.head@ge.com

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This letter transmits **proprietary information** and **security-related information** in accordance with 10 CFR 2.390. Upon the removal of **Enclosures 4 and 6**, the balance of the letter may be considered non-proprietary public information.

M170046

Docket Number: 52-045

February 23, 2017

US Nuclear Regulatory Commission
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Subject: **Request for Additional Information Letter Number 8 Related to Chapter 6 for GE-Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application- GEH Response to RAI 06.03-2, Revision 2**

In regards to the Request for Additional Information (RAI) 06.03-2 transmitted in Reference 1, and the earlier GEH responses to RAI 06.03-2 submitted in References 2 and 3, please find a revised response in Enclosure 1. Other enclosures, listed below and described in the RAI response, provide the Advanced Boiling-Water Reactor (ABWR) Design Control Document (DCD) markups and a supporting technical report (proprietary and public versions).

The revised response is provided to address information discussed during a public NRC teleconference held on January 5, 2017.

Please contact me or Patricia Campbell (202-637-4239) if you have any questions.

Sincerely,

Patricia L. Campbell
for
Jerald G. Head
Senior Vice President, Regulatory Affairs

D106
NRO

Commitments: None.

References:

1. Letter from USNRC to Jerald G. Head, GEH, Subject: Request for Additional Information Letter Number 8 Related to Chapter 6 For GE-Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application, December 15, 2015.
2. Letter from Jerald G. Head, GEH, to USNRC, Subject: Request for Additional Information Letter Number 8 Related to Chapter 6 For GE-Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application-GEH Response to RAI 06.03-2, May 27, 2016.
3. Letter from Jerald G. Head, GEH, to USNRC, Subject: Request for Additional Information Letter Number 8 Related to Chapter 6 For GE-Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application-GEH Response to RAI 06.03-2, Revision 1, December 19, 2016.

Enclosures:

1. GEH Revised Response to RAI 06.03-2
2. Table of ABWR DCD Tier 2 Figures
3. ABWR DCD Revision 6 Markups (Public)
4. ABWR DCD Revision 6 Markups (Non-Public)
5. Technical Report NEDO-33878, Public Information
6. Technical Report NEDE-33878P, GEH Proprietary Information

cc: A. Muniz, NRC
DBR-0026220

# Enclosure 1

# M170046

# GEH Revised Response to RAI 06.03-2

GEH has revised its response to NRC Request for Addition Information (RAI) 06.03-2, Revision 1. The previous response was provided in a GEH letter dated December 19, 2016 (MFN 16-034 Revision 1). This revised response is provided to address information discussed in a public teleconference held on January 5, 2017. The RAI is repeated below and the updated response, which replaces the previous response, follows.

***NRC Request for Information RAI 06.03-2:***

*In RAI 06-03-1, in accordance with 10 CFR 52.59(a) (2014), the NRC staff requested GE Hitachi Nuclear Energy (GEH) to provide information showing that the ECCS suction strainer design complies with 10 CFR 50.46(b)(5) (1997). GEH responded in letters, dated April 8, 2015, and July 17, 2015. The staff's review of the applicant's response found the need for additional information as cited below.*

***A. Design and Analysis of ECCS Strainer***

1. *To enable making a safety determination with respect to 10 CFR 50.46(b)(5), the staff needs the applicant to provide its evaluation of ECCS strainer performance (e.g. head loss) and the results of any analysis and/or tests performed in support.*

2. *GEH updates the design to install reflective metal type insulation on the ASME Section III, Class 1 piping greater than 80 mm in the drywell. As pointed out by GEH, use of reflective metal type insulation improves the design by reducing the potential suction strainer debris load and clogging. However, the types and quantities of insulation debris being transported to the ECCS suction strainers and the core following a design basis accident are needed for evaluating the ECCS and core design. Staff needs a design basis debris load in the DCD to enable the staff to make the 50.46(b)(5) finding.*

3. *GEH's response deletes the following without providing an alternate description of the debris strainers: "The ABWR ECCS suction strainers will utilize a 'T' arrangement with conical strainers on the 2 free legs of the 'T'. This design separates the strainers so that it minimizes the potential for a contiguous mass to block the flow to an ECCS pump." Staff needs sufficient design detail to assess the performance of the system under the accident conditions. For example, staff would need to understand the strainer type, flow area, and hole size to assess strainer head loss. (See Regulatory Positions C.2.1.2.1 and C.2.1.2.2 of RG 1.82 rev 3.) This combined with the information in response to Question 1 would enable staff to assess the head loss due to debris and enable the staff to make a finding on compliance with 10 CFR 50.46(b)(5).*

4. *GEH's response deletes Tables 6C-1 and 6C-2 which provide debris analysis input parameters and results of ECCS debris strainer sizing analysis without providing alternate tables or references to calculation reports. Staff needs the type of information from these tables to be provided in the DCD to support the staff's safety finding.*

5. *GEH's response provides references to guidance documents, e.g., "Of the debris generated, the amount that is transported to the suppression pool shall be determined in accordance with Reference 6C-3 based on similarity of the Mark III upper drywell design." However, the response does not provide the ECCS debris strainer design input calculated using these guidance documents nor does it reference calculation reports providing such*

*information. Staff requests GEH provide the analysis documenting the implementation of this guidance be made available for staff audit and the summary of the inputs methods and results be placed on the docket.*

***B. Chemical Effects***

*The staff requests the following additional information about how the potential for chemical effects is addressed through design, to enable the staff to make the safety finding for 10 CFR 50.46(b)(5).*

1. *There is currently a limited understanding of chemical precipitation under the anticipated range of post-loss-of-coolant accident (LOCA) chemical and temperature conditions for Boiling Water Reactors. An investigation suggested corrosion products from steel and zinc may have contributed to head loss across a bed of mineral wool on the strainer in a test facility ("Influence of Corrosion Processes on the Protected Sump Intake after Coolant Loss Accidents," Nuclear Technology Annual Convention 2006, translated from German, ADAMS Accession No. ML083510156). Research on pressurized water reactor (PWR) chemical effects has shown aluminum, calcium silicate, concrete, and silicon-rich insulation materials can form chemical precipitates under conditions similar to BWR post-LOCA conditions. Since the amount of chemical precipitation would depend, in part, on the amount of the contributing materials in communication with the ECCS, the staff requests that you describe the use of the following materials in the ABWR containment and how the design establishes limits on the quantities of:*

   a. *Aluminum*
   b. *Metallic zinc, including galvanized steel*
   c. *Inorganic zinc-rich coatings*
      i. *All unqualified*
      ii. *Qualified and not top coated within 10D zone of influence of a piping system break location*
      iii. *Qualified and epoxy top coated within 4D zone of influence of a piping system break location*
   d. *Uncoated carbon steel*
   e. *Concrete*
      i. *Uncoated*
      ii. *Coated and within the zone of influence for the coating*
   f. *Insulation other than reflective metal insulation (i.e., fiberglass, calcium silicate, mineral wool, amorphous silica, etc.)*

2. *Since chemical precipitation depends on the temperature and chemical environment, provide the ranges and timing of pH, pool temperature, and boron concentration possible following a loss of coolant accident. Include the timing of the Standby Liquid Control System initiation. In addition, identify where this information is found in the DCD.*

3. *Identify and justify how the strainer and fuel assembly performance criteria will be met, considering chemical precipitates that may form under the conditions described above (bounding or plant-specific).*

### *C. Downstream Effects*

1. *Similar to question A.2 above, staff needs an in-vessel design basis debris load in the DCD that will establish the design basis limits for in-vessel testing. This will enable the staff to make the 50.46(b)(5) finding.*

2. *A justification for the acceptability of the core design with respect to core cooling in the presence of debris should be provided. GEH should provide testing results and/or analysis to support its design. Any justification for reliance on historical testing should be accompanied by a justification of the applicability of the referenced tests to the GE-7 fuel specified in the certified design.*

3. *GEH should provide the ABWR-specific acceptance criteria it relied upon in evaluating the test and/or analysis results.*

**GEH Revised Response:**

This response addresses the questions above and is updated to address discussions with and feedback provided during a public teleconference held on January 5, 2017. This updated response supersedes the response provided in GEH letter number MFN-16-034 Revision 1 (NRC ADAMS Accession Number ML16358A445). Note that there are references in this response that are GEH proprietary internal documents that are available for NRC audit at GEH facilities.

This revised response addresses the evaluation of downstream effects of debris bypass from the ECCS suction strainers on ECCS long-term recirculation capability performed by GEH. This evaluation is described in NRC guidance in Regulatory Guide 1.82 Revision 4. While Regulatory Guide 1.82 notes that Westinghouse Commercial Atomic Power (WCAP)-16406-P-A, Evaluation of Downstream Sump Debris Effects in Support of GSI-191, and its SE provide evaluation methods and criteria that the NRC considers acceptable, this reference is proprietary and is not available to GEH as input for this evaluation. Therefore, GEH developed a methodology to assess ECCS downstream effects from public documents and GE proprietary documents including the following:

- NEI 04-07, PWR Sump Performance Evaluation Methodology-Volume 1, Revision 0 (ML050550138)

- NEI 04-07, PWR Sump Performance Evaluation Methodology-Volume 2, Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02, Revision 0 (ML050550156)

- Safety Evaluation by the Office of Nuclear Reactor Regulation Topical Report (TR) WCAP-16406-P, Evaluation of Downstream Sump Debris Effects in Support of GSI-191 Pressurized Water Reactor Owners Group Project No. 694 Revision1. (ML073520295)

- NEDO-32686-A Vol. 4, Utility Resolution Guide for ECCS Suction Strainer Blockage, Revision October 1998, containing Technical Support Documentation GE-NE-T23-00700-15-21, Revision 1, Evaluation of the Effects of Debris on ECCS Performance (ML092530507)

- NEDC-32721P-A, Licensing Topical Report: Application Methodology for the GE Stacked Disk ECCS Suction Strainer, Revision 2 (GEH Proprietary Information)

- NUREG/ CR-2792, September 1982, An Assessment of Residual Heat Removal and Containment Spray Pump Performance Under Air and Debris Ingesting Conditions (ML100110155)

- NUREG/CR-6808, Knowledge Base for the Effect of Debris on Pressurized Water Reactor Emergency Core Cooling Sump Performance, February 2003 (ML030780733; ML030920540)

- ANSI/API 610, 11th Edition (September 2010), Centrifugal Pumps for Petroleum, Petrochemical and Natural Gas Industries

- ASME QME-1-2007, Qualification of Active Mechanical Equipment Used in Nuclear Power Plants

NEDE-33878P, ABWR ECCS Suction Strainer Evaluation of Long-Term Recirculation Capability, Rev. 0, (GEH Proprietary Information) was developed to address debris carried downstream of the ECCS strainers causing postulated downstream blockage or wear and abrasion. The three areas of concern evaluated are (1) blockage of system flow paths at narrow flow passages (e.g., ECCS sparger spray nozzles, some pump internal flow passages, and tight-clearance valves), (2) wear and abrasion of surfaces (e.g., pump running surfaces) and heat exchanger tubes and orifices, and (3) blockage of flow paths through fuel assemblies.

This response is developed using guidance in the topical reports listed below, as well as guidance in Regulatory Guide (RG) 1.82 (Revisions 3 and 4). In addition, as discussed with the NRC in a public meeting held May 7, 2015, GEH acknowledges that RG 1.1 has been withdrawn and replaced with RG 1.82. The DCD markups provided with the earlier response include changes to address the withdrawal of RG 1.1.

## A. Design and Analysis of ECCS Strainer

1. The ABWR ECCS strainers will be the patented GE optimized stacked disk design in accordance with NEDC-32721P-A Rev. 2 (NRC ADAMS Package ML031010392; public version). This strainer design was developed in response to NRC Bulletin 96-03 as a replacement of existing ECCS strainers with a large capacity passive strainer design. This new design utilizes disks whose internal radius and thickness vary over the height of the strainer. The selected variation in these parameters achieves an increased surface area compared to existing strainers of the same size to provide a higher capacity for debris capture. The new ABWR strainer will perform with a minimum head loss for the range of possible amounts of debris while fitting in the required volume. NEDC-32721P-A, which was reviewed and approved by the NRC, describes methods for sizing a stacked disk suction strainer and evaluating the head losses due to debris accumulation (note that an updated head loss correlation is used along with NEDC-32721P-A). The DCD will be updated to include a general description of the stacked disk strainer, and to remove obsolete information related to the T-shaped conical strainer, and outdated information such as the guidance to design for 50% plugging.

The ABWR ECCS suction strainers are the same design as an existing stacked disk design from the operating BWR fleet (105E2586 Rev. 02, Suction Strainer RHR). The ABWR specific debris load, flow rate and pool conditions are applied to demonstrate that the qualified strainer design (applied to ABWR Residual Heat Removal RHR, High Pressure Core Flooder HPCF and Reactor Core Cooling RCIC systems) will support ABWR certification renewal.

As noted above, the strainers have been designed in accordance with NEDC-32721P-A Rev. 2 (ML031010392). This licensing topical report documents the application methodology for the GE stacked disk ECCS suction strainer, including (1) hydraulic performance design methods and (2) procedures for calculation of loads for new strainer installation used in the structural analysis of the suppression pool penetration(s), the strainer supports, and the strainer itself. The ABWR ECCS suction strainer design and procurement specifications are based on NEDC-32721P-A. The applicable ECCS suction strainer design specification is GE specification 24A5822 Rev. 7. A similar fabrication specification is GE specification 24A5849 Rev. 8. These calculations are available for audit by the NRC staff at GEH facilities.

An updated method was implemented for sizing and qualifying the ABWR ECCS strainers due to non-conservatisms noted in the methodology presented in NEDC-32721P-A Rev. 2. The evaluation, using the ABWR design parameters, demonstrates that the head loss across the strainer under the design basis debris load (discussed below) is within an acceptable limit such that the required Net Positive Suction Head (NPSH) can be supplied to the ECCS pumps. The evaluation (DBR-0017510, ABWR Suction Strainer Performance Evaluation) and the updated head loss correlation calculations are available to the NRC for audit at GEH facilities.

The NPSH for the ABWR RHR pump was assessed in accordance with RG 1.82 under calculation 31113-0E11-2113 Rev. 1. The NPSH for the ABWR HPCF pump total head was assessed in accordance with RG 1.82 under calculation 31113-0E22-2106 Rev. 0. The NPSH for the ABWR RCIC pump was assessed in accordance with RG 1.82 under calculation 31113-0E51-2121 Rev. 0. While these calculations are not part of the ABWR certified design basis, these calculations are available for audit by the NRC staff. The results of these analyses demonstrate that the original certified design values for NPSH are met as reflected below.

**Net Positive Suction Head – ECCS Pumps**

| Pump | DCD Reference for Calculations | NPSH Required | NPSH Available |
|---|---|---|---|
| RCIC Pumps | Table 5.4-1a | 7.3m | 7.65m |
| RHR Pumps | Table 6.2-2b | 2.4m | 2.75m |
| HPCF Pumps | Table 6.2-2c | 2.2m | 2.55m |

2. The ABWR DCD is updated, under revision 1 of this response, to include the following table that lists the types and quantity of debris determined in accordance with Utility Resolution Guidance (URG) NEDO-32686-A (ML092530482). The spherical zone of influence (ZOI) was used to determine quantities of pipe insulation (both RMI and Nukon fiber). The note indicates that no chemical effects are listed due to the ABWR design (see the section below on Chemical Effects for more details).

**ECCS Strainer Debris Load**

| Debris Type | Strainer Load |
|---|---|
| Sludge / corrosion prod. | 90.7 kg (200 lbm) |
| Inorganic Zinc (IOZ) | 21.3 kg (47 lbm) |
| Epoxy Coated IOZ | 38.6 kg (85 lbm) |
| Rust Flakes | 22.7 kg (50 lbm) |
| Dust / Dirt | 68.0 kg (150 lbm) |
| Reflective Metal Insulation | 35.8 $m^2$ (385 $ft^2$) |
| Nukon Fiber Insulation | 23.4 kg (51.6 lbm) |

*NOTE: No chemical effects are included based on the ABWR design

- The ABWR has substantially reduced the amount of piping in the drywell relative to earlier designs and consequently the quantity of insulation required. Furthermore, there is no equipment in the wetwell spaces that requires insulation or other fibrous materials.

- The non-thermal insulation debris values are as recommended by NEDO-32686-A. The sludge load of 200 lbm is equivalent to 100 lbm/year assuming a two-year cycle. This value was chosen to envelope the survey results reported in URG Section 3.2.4.3.2 in which the median sludge generation rate for operating BWRs was found to be 88 lbm/year. The ABWR design includes many improvements over the conventional BWRs that will help to minimize the generation of sludge. Specifically, the ABWR design includes the following improvements:

  - The suppression pool is equipped with a stainless steel liner on the normally wetted surfaces, and many interfacing systems utilize stainless steel pipe, which reduces the generation of carbon steel corrosion products.
  - The ABWR suppression pool is enclosed in a concrete compartment and protected from the drywell environment, unlike some containment designs from the BWROG survey which have debris sources above the pool that can fall directly in by gravity.
  - The Suppression Pool Cleanup System (SPCU) is run periodically during operation to remove suspended impurities, and a method for maintaining suppression pool cleanliness is required by DCD Section 6.2.7.3.

  For these reasons, there is reasonable assurance that the ABWR will have significantly less sludge generation than the operating fleet of BWRs, and the selection of 200 lbm (90.7 kg) total is reasonable.

3. The conical strainer design is obsolete and has been updated to the GE stacked disk strainer as explained in the response to item A.1. Design details related to the stacked disk strainer performance and sizing methodology can be found in the report NEDC-32721P-A applying an updated head loss correlation. The ECCS suction strainer configuration utilized for the ABWR applications is shown on DWG 105E2586 R4).

   Type: GE stacked disk passive suction strainer
   Flow Area: Each strainer has perforated area 36 $m^2$ (388 $ft^2$) with 20 disks [combined surface area of 216 $m^2$ (2328 $ft^2$) for three (3) RHR, two (2) HPCF and one (1) RCIC strainer]
   Hole Size: 3.2 mm (0.125 inch) diameter

   GEH calculation 0000-0080-3039R1 (updated under DBR-0017510) applies ABWR RHR suction strainer parameters to the updated methodology provided in NEDC-32721P-A. This drawing and the supporting sizing calculations are available for NRC audit at the GEH facilities.

4. DCD markups included in earlier response updated Table 6C-1 to include the design basis debris load shown above for Response A.2. This information, combined with the

methodology in NEDC-32721P-A, applying an updated head loss correlation, provides the necessary inputs to design a strainer that complies with 10CFR50.46(b)(5).

5. DCD markups included in earlier response show the updates explained in response to item A.1 to reference the report NEDC-32721P-A (with a note explaining the updated head loss correlation), which provides the strainer design methodology. As noted above, the evaluation described in response to item A.1 is available for NRC audit.

   When applying NEDC-32721P-A, an updated head loss correlation is used to address an issue identified in a letter to the NRC (MFN 08-286, NRC ADAMS Accession Number ML080850242). A note is being included with the reference to NEDC-32721P-A in Table 1.6-1, as shown on the DCD markups with this revised RAI response. The updated head loss correlation is in the information that the NRC may audit, as discussed above.

**B. Chemical Effects**

1. The material discussion of "Engineered Safety Feature Materials" for the ABWR are given in DCD Chapter 6, Section 6.1. It covers metallic materials and organic materials. Steel is used to line the containment thus isolating the concrete from degradation and preventing dissolution. In that the RAI asks for the use of specific materials these are addressed individually as follows:

   a. Aluminum:
      Aluminum will not be installed in the ABWR containment. RMI insulation will be stainless steel construction. Aluminum cable trays shall not be permitted for use inside the Reinforced Concrete Containment Vessel (RCCV). Therefore, aluminum will not be a material of concern.
   b. Metallic Zinc:
      Use of exposed metallic zinc is limited to galvanized steel in ladders, ductwork, unistruts, cable trays, conduit and grating and will not make any significant contribution to corrosion products in the suppression pool. Therefore, metallic zinc will not be a material of concern.
   c. Inorganic Zinc-Rich (IOZ) Coatings:
      i. All unqualified:
         As stated in the DCD, only small amounts of unqualified IOZ coatings associated with small size equipment will be present. These include electrical trim, face plates and valve handles.
      ii. Qualified and not top-coated:
         None. Zinc rich primer is always coated with a qualified epoxy top coat.
      iii. Qualified and epoxy top coated within 4D zone of influence of a piping system break location:
         The quantity of epoxy coated IOZ is bounded by the guidance of NEDO-32686-A, Section 3.2.2.2.2.1.1 Table 3.
   d. Uncoated carbon steel:
      Uncoated carbon steel will not be used in the ABWR containment.
      Therefore, carbon steel will not be a material of concern.
   e. Concrete:
      i. Uncoated:
         None. As stated, the concrete containment is isolated from the coolant by the steel liner. A steel liner plate is located at the pressure boundary of the

containment and on top of diaphragm slab. Uncoated concrete is not used in the ABWR containment. Therefore, concrete will not be a material of concern.

ii. Coated and within the zone of influence for the coating:
None. The quantity of debris generated due to jet impingement on coatings is addressed by Item C above. Therefore, concrete and concrete dissolution products will not be a material of concern. As mentioned above, all concrete is protected from jet impingement, but a quantity of 150 lbs. of dirt and dust is assumed per the guidance of NEDO-32686-A, Section 3.2.2.2.3, Item 2.

f. Insulation other than RMI
The only type of pipe insulation permitted, aside from RMI, is Nukon fiber. The ABWR limits the amount of Nukon by restricting it to pipe sizes of 80 mm and smaller. A calculation is performed per the URG guidance of NEDO-32686-A to determine the quantity of Nukon that is assumed to reach the suppression pool during a LOCA (see Response A.2). This calculation is available to the NRC for audit at GEH facilities. Also, see response item C.2, which provides an evaluation for chemical and downstream effects.

2. Per the DCD Chapter 6, Section 6.1.1.2: "The post-LOCA ESF coolant, which is high- purity water, comes from one of two sources. Water in the 304L stainless steel-lined suppression pool is maintained at high purity (low corrosion attack) by the Suppression Pool Cleanup (SPCU) System. Since the pH range (5.3 - 8.6) is maintained, corrosive attack on the pool liner (304L SS) will be insignificant over the life of the plant. ESF coolant may also be obtained from the condensate storage tank, if available." The Standby Liquid Control System (SLC) is credited to mitigate ATWS events (discussed in DCD Section 15.8), but does not operate during a design basis LOCA. Therefore, for the purpose of suction strainer design, sodium pentaborate is not a contributing factor affecting pool chemistry.

3. The strainer shall be designed as described in response to Part A of this RAI response. The debris loading is based on the values given in the Table shown in response to Item A.2. The downstream effects on the fuel assemblies are discussed more directly in response to Item C of the RAI. The term "chemical effects" refers to the possibility that interactions of materials in the containment environment will generate chemical precipitate debris that may contribute to blockage and head loss. As noted in DCD markup for Table 6C-1, no chemical effects are considered based on the ABWR design.

   It is not expected that interactions of materials will generate chemical precipitation debris in the ABWR containment environment.

   - Reactive materials such as aluminum, phosphates and calcium silicate will not be installed in the ABWR containment.

   - Zinc chemical debris that could result from corrosion of inorganic zinc coating was assumed to transport to the suction strainers but will cause minimal head loss because the calculated chemical debris quantity is small relative to the strainer area.

   - Concrete does not include particle generation by the LOCA jet. This is based on the ABWR mitigating design feature of isolating the concrete from the coolant by the steel liner.

- There are no potential chemical reactor products (or precipitates) resulting from boron injection into the primary system as a design basis accident mitigating system.

**C. Downstream Effects**

1. As stated in DCD Section 6C.1, the ABWR commits to following the guidance related to ECCS blockage in RG 1.82 (sections pertaining to BWRs) and in Topical Report NEDO-32686-A, "Utility Resolution Guidance for ECCS Suction Strainer Blockage," (the URG).

   The possibility of debris clogging flow restrictions downstream of the strainers is assessed in NEDE-33878P, ABWR ECCS Suction Strainer Evaluation of Long-Term Recirculation Capability, Rev. 0. This technical report and a public version are submitted to the NRC with this updated RAI response. The results of this evaluation, described in DCD 6C.3.3, ensure adequate long-term ECCS performance.

   The design basis suppression pool debris load is provided in Table 6C-1 of the DCD. A percentage of this debris is assumed to bypass the ECCS strainers and interact with downstream components such as pumps, valves and heat exchangers. The downstream evaluation includes several conservative assumptions:

   - It was assumed that all debris other than NUKON and RMI passed through the strainer and was available in the ECCS
     - This assumption is conservative with no credit for sludge, dust/dirt, or rust flakes settling in the suppression pool or on the ECCS strainers
   - The downstream evaluation also determined the design basis debris source term for ECCS components assuming all NUKON and RMI bypassed the suction strainers (SP debris concentration)
     - NUREG/CR-6808 assumes model where 23% of NUKON generated are fines passing through strainer
     - NUREG/CR-6808 shows 4.3% of typical RMI debris (shards) generated by large break LOCA are ¼ inch or less
   - All debris that was introduced into the SP remains in uniform suspension within the pool and is available for ingestion into (circulation through) the ECCS for the entire duration of the ECCS mission time.
     - This assumption is conservative because the evaluation does not credit sludge, dust/dirt, or rust flakes settling in the suppression pool or on the ECCS strainers during LOCA recovery

The ABWR RHR strainer perforated plates contain 3.2 mm (1/8 inch) diameter holes in the plates resulting in approximately 40% open area. This allows the suction strainer to filter debris larger than this nominal size.

NEDE-33878P, ABWR ECCS Suction Strainer Evaluation of Long-Term Recirculation Capability, Rev. 0, evaluates downstream effects based on the design basis debris source term and maximum debris particle size downstream the suction strainer. The maximum dimension (length, width and/or thickness) of non-deformable particulates that may pass through the strainer is limited to the cross-sectional flow area of the penetration (hole) in the strainer.

NEDE-33878P, ABWR ECCS Suction Strainer Evaluation of Long-Term Recirculation Capability, Rev. 0, evaluates downstream effects as listed below. Where appropriate, typical ABWR ECCS components were evaluated for tolerance to LOCA generated debris (debris source term and maximum dimension) for the duration of the component mission time. The limiting debris size may be listed as a procurement / design requirement in the applicable system / component specification (i.e. ASME Purchase Specification for Vertical Can Type ASME Pumps). This will allow a generic evaluation to remain valid for site-specific details (i.e. specific pump and heat exchanger procurement).

- <u>Blockage of Flow Paths in Equipment</u>
  - The orifice sizes in ECCS components were compared with maximum dimension of debris downstream of suction strainer with clogging not credible if orifice size is greater
  - The fluid velocity through ECCS flow paths was compared with settling velocity for debris type with clogging not credible if flow maintains debris in solution
- <u>Wear and Abrasion of Surfaces</u>
  - The ECCS materials were reviewed for wear and degradation when exposed to debris laden water and compared to industry standards and operating experience with ECCS components under bounding debris ingesting conditions. Significant wear and degradation affecting the ECCS safety function is not credible.
- <u>Blockage of Flow Clearances through Fuel Assemblies</u>
  - Limiting dimensions in the reactor vessel, core and fuel assemblies were reviewed against the design basis debris source term and maximum dimension of debris downstream of suction strainer to assess the risk of flow blockage on long term cooling degradation. All internal flow paths that influence long-term cooling were addressed for the potential for plugging these paths. This failure mode is not credible.
  - The flow blockage associated with core grid supports, mixing vanes, and debris filter, and its effect on fuel rod temperature were considered. This failure mode is not credible.

- o Bypass debris was assessed for the potential blockage of coolant flow at the entrance to the fuel assemblies as described in NEDC-33302P, Fiber Insulation Effects with Defender Lower Tie Plate (available for NRC audit in GEH facilities)

NEDO-32686, Volume 4, includes Report GE-NE-T23-00700-15-21 March 1996 (Rev. 1), "Evaluation of the Effects of Debris on ECCS Performance," which provides a description of extensive testing of the GE stacked disk passive suction strainers. This evaluation shows that adequate core cooling provided during a LOCA will not be compromised by the presence of rust, epoxy paint chips, sand, iron oxide sludge, and fibrous debris in the ECCS system or reactor core. It was concluded that there is no safety concern for the potential failure of the ECCS pumps, inadequate cooling capacity from the RHR heat exchangers, plugging of the core spray header nozzles, plugging of the containment spray nozzles, corrosion or chemical reaction with other reactor materials, or fuel bundle flow blockage.

For this debris analysis, the particles evaluated are rust, paint chips, sand, and fibrous debris of random sizes and shapes. The rust chips are of low strength and will fracture into even smaller pieces upon interaction with other components. Similarly, the epoxy paint is also relatively brittle and will breakup as well. The sand will not melt or form a large enough agglomeration to significantly block flow. The glass fibers are fragile with virtually no mechanical strength. The rust, paint, and fiberglass debris that pass through the ECCS suction strainers will be subjected to the ECCS flow rates and turbulence that will cause disintegration into particles of even smaller sizes.

The ABWR ECCS stacked disk suction strainer design ensures core cooling with the presence of debris in the ECCS suction strainers.

ABWR consists of three divisions of ECCS, each of which includes one high pressure and one low pressure makeup system. The high-pressure configuration consists of two motor driven high pressure core flooders (HPCF) each with its own independent sparger discharging inside the shroud and the steam driven Reactor Core Isolation Cooling system (RCIC) which discharges into the feedwater injection line. The low pressure ECCS utilizes three residual heat removal (RHR) pumps in the post LOCA Low Pressure Flooding (LPFL) mode. The RHR system provide core and containment cooling following a postulated LOCA.

Limiting LOCA conditions (MSL break) were used to evaluate the RHR suction strainer under design debris loading when determining NPSH margin as described in GEH ABWR Suction Strainer Performance Evaluation (DBR-0017510). A debris load fraction for fiber debris (Nukon) of 57% is applied to the RHR strainer (43% applied to the HPCF strainer). This single RHR strainer is also subjected to the entire debris load from other sources (sludge, IOZ, epoxy coated IOZ, rust flakes, dust/dirt). In addition, the entire inventory of RMI debris is assumed to be collected in one RHR suction strainer. The head loss associated with this design basis debris accumulation is added to the clean strainer and remaining piping system resistance to estimate RHR pump NPSH margin.

This analysis reflects a strainer gap fill ratio less than 1.0 indicating that the gaps between the strainer discs are not filled under the design debris loading. This supports the conclusion that the passive suction strainer will filter debris ensuring that downstream ECCS system components or reactor fuel will not be impacted significantly. This bounding strainer design

(RHR) was shown to satisfy the NPSH requirements for HPCF and RCIC pumps since the ECCS strainer sizing calculations assumed all ECCS Strainers (RHR, RCIC and HPCF) are of the RHR size. Conservatisms used in the assessment provide additional support for conclusion that core cooling will be maintained.

- Two RHR strainers are assumed available for capturing insulation debris for the limiting design condition. The debris-laden flow from the suppression pool will be injected into the vessel only after the initial inventory of the ECCS piping, which is clean, is swept and injected into the vessel. Therefore, any suppression pool water will be further diluted by this clean initial injection.
- Although not credited, the HPCF (and RCIC) pumps initially inject from the condensate storage tank, which is a clean source of water.
- The diversity of ECCS delivery points (injection inside the core shroud above the fuel by the HPCF and injection outside the core shroud in the annulus by the RHR and RCIC) helps to maintain the core flooded and reduce the consequences of a blockage in the fuel assembly.
- The ABWR design also has additional features not utilized in earlier designs that could be used in the highly improbable event that all suppression pool suction strainers were to become plugged. The Alternate AC Independent Water Addition (ACIWA) mode of RHR allows water from the Fire Protection System to be pumped to the vessel and sprayed in the wetwell and drywell from diverse water sources to maintain cooling of the fuel and containment

This bypass debris is also assessed for the potential blockage of coolant flow at the entrance to the fuel assemblies. Tests have been performed to simulate clogging of the Defender Lower Tie Plate (DLTP) with a small concentration of fiber insulation material. This evaluation concludes that significant BWR fuel bundle inlet clogging does not result in GNF2 fuel heat-up after the LOCA re-fill from ECCS injection. These conclusions apply to other BWR fuel bundles (i.e. ABWR GE P8x8R) with equivalent degree of inlet resistance as used in this evaluation.

The ABWR response during LOCA is different from a typical BWR which reduces impact of debris on fuel coolability:

- The core flow rate decreases quickly due to the rapid coast down of the RIPs following a reactor scram resulting from a LOCA in containment.
    - This results in an early boiling transition upon reactor scram. The reduction in the heat transfer results in an increase in the fuel cladding temperature. The decrease in core power caused by increased voiding and reactor scram results in a rapid reduction in the cladding temperature. The cladding temperature excursion is short-lived with the peak clad temperature occurring prior to ECCS injection.

- The elevations of potential large pipe break locations are above the top of the active core.

- o The location of the pipe break in conjunction with the actuation of the ECCS, results in a nearly continuous two-phase cooling of the core. The typical extended core uncovery phase of the BWR LOCA does not occur in the ABWR. Thus, the peak cladding temperature occurs before the ECCS actuation and is independent of the ECCS performance.

The ABWR evaluation examines the effects of bundle inlet clogging that reduces the available inlet flow from natural circulation phenomena following initial core refill when the core region is covered by a two-phase mixture. During this post-LOCA period, the reduced inlet flow results in increased bundle voiding and higher velocities such that the heat transfer is sufficient to remove the decay heat. Once the bundle decay heat has decreased and insufficient voids exist to maintain the level in the bundle above the top of the fuel channel, adequate cooling from the upper plenum spillover will exist. Thus, the evaluation concludes that for significant bundle inlet clogging following initial core refill, BWR fuel bundle cooling is assured.

The ABWR design provides reasonable assurance that downstream effects, from debris bypassing the ECCS suction strainers, will not have a deleterious effect on critical components such as fuel rods, valves, and pumps downstream of the suction strainers. This reasonable assurance is based on the following:

- The relative reduced likelihood of debris generation compared to operating BWRs (restricted access to the containment, the suppression pool cleanup system, the operational program for suppression pool clean-up)
- Minimal LOCA-generated debris (elimination of recirculation piping, minimal fibrous insulation)
- Inconsequential impacts of chemical effects
- ABWR design features that minimize the transport of accident-generated debris.
- Suction strainer design
- Diversity of ECCS delivery locations

2. The justification of the acceptability of the ABWR core design with respect to core cooling in the presence of debris is provided in C.1 above.

3. The justification of the acceptability of the ABWR core design with respect to core cooling in the presence of debris is provided in C.1 above. The ABWR design provides reasonable assurance that downstream effects, because of debris bypassing the ECCS suction strainers, will not have a deleterious effect on critical components such as fuel rods. Therefore, COL items associated with Evaluation of Post LOCA Fuel Bundle Blockage (COL 4.1a) and Debris evaluation of ECCS Strainers (COL 6.12) have been removed.

**Impact on DCD:**

The following ABWR DCD Revision 6 tables and sections are revised, as shown in the markups provided in Enclosure 2, of this revised response. Note that certain of the DCD markups are on pages from draft Revision 7 for better clarity in showing the changes from the previous RAI response versions. Note that the DCD markups include changes to certain sections that were changed in

previous responses. If no changes are included in this response, the previous DCD markups remain valid.

Tier 1:

- None

Tier 2:

- Table 1.6-1: Added NEDE-33878P, "ABWR ECCS Suction Strainer Evaluation of Long-Term Recirculation Capability," Rev. 0, and NEDO-33878, "ABWR ECCS Suction Strainer Evaluation of Long-Term Recirculation Capability," Rev. 0, to ABWER referenced reports

- Appendix 6C: 6C.2 Added description of GE stacked disk strainer

  6C.3.3 Updated design considerations for downstream effects to describe evaluation and conclusion from for NEDE-33878P, "ABWR ECCS Suction Strainer Evaluation of Long-Term Recirculation Capability," Rev. 0.

  6C.7 Added reference 6.3-5 to 6C.7 References for NEDE-33878P, "ABWR ECCS Suction Strainer Evaluation of Long-Term Recirculation Capability," Rev. 0, and NEDO-33878, "ABWR ECCS Suction Strainer Evaluation of Long-Term Recirculation Capability," Rev. 0

- Section 14.2.12.1.8 Removed testing requirement for 50% blockage of RHR suction strainer due to elimination of this requirement

- Section 14.2.12.1.10 Removed testing requirement for 50% blockage of HPCF suction strainer due to elimination of this requirement

Tier 2 -Section 21

- Figure 1.2-13i Updated wetwell arrangement plan to reflect GE stacked disk strainer design that supersedes the T-type conical strainer design (Chapter 21, Volume 1; contains security-related information)

- Figure 5.4-9 (Sh. 1 of 2) Note 10- Replaced requirement for 50% blockage of RCIC suction strainer with strainer head loss at design value

- Figure 5.4-11 (Sh. 2 of 2) Numerous entries- Replaced requirement for 50% blockage of RHR suction strainer with strainer head loss at design value

- Figure 6.3-1 (Sheet 1 of 2) — Note 4- Replaced requirement for 50% blockage of HPCF suction strainer with strainer head loss at design value

# Enclosure 2

# M170046

# Table of ABWR DCD Tier 2 Figures

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## Table of Changes to DCD Figures

| Figure | Change Associated with Revised Response to RAI 06.03-2 |
|---|---|
| Figure 1.2-13i, Wetwell, Arrangement Plan at Elevation -8200 mm<br>(Chapter 21, Volume 1; contains security-related information) | • Updated wetwell arrangement plan to reflect GE stacked disk strainer design that supersedes the T-type conical strainer design |
| Figure 5.4-9 Reactor Core Isolation Cooling System PFD (Sheet 1 of 2)<br>(Chapter 21, Volume 2) | • Note 10- Replaced requirement for 50% blockage of RCIC suction strainer with strainer head loss at design value |
| Figure 5.4-11 Residual Heat Removal System PFD (Sheet 2 of 2)<br>(Chapter 21, Volume 2) | • Numerous entries- Replaced requirement for 50% blockage of RHR suction strainer with strainer head loss at design value |
| Figure 6.3-1 High Pressure Core Flooder System PFD (Sheet 1 of 2)<br>(Chapter 21, Volume 2) | • Note 4- Replaced requirement for 50% blockage of HPCF suction strainer with strainer head loss at design value |

# Enclosure 3

# M170046

# ABWR DCD Revision 6 Markups (Public)

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Revised DCD Markups are shown in a red box with blue background, as this box.

## Table 2.4.1 Residual Heat Removal System (Continued)

| Inspections, Tests, Analyses and Acceptance Criteria | | |
|---|---|---|
| **Design Commitment** | **Inspections, Tests, Analyses** | **Acceptance Criteria** |
| 4. continued<br>c. The RHR pumps have sufficient NPSH. | 4. continued<br>c. Inspections, tests and analyses will be performed upon the as -built RHR System. ~~NPSH tests of the pumps will be performed in a test facility. The analyses will consider the effects of:~~<br>– Pressure losses for pump inlet piping and components.<br>– Suction from the suppression pool with water level at the minimum value.<br>– ~~50% blockage of pump suction strainers.~~<br>– Design basis fluid temperature (100°C).<br>– Containment at atmospheric pressure. | 4. continued<br>c. The available NPSH exceeds the NPSH required by the pumps. |

Delete

required

Analytically derived values for blockage of pump suction strainers based upon the as-built system.

ADD
Inspections of the as-built system will be performed to obtain piping system dimensions and other necessary information. The required NPSH of procured pumps will be determined by an inspection of the vendor specifications.
The analysis will consider the effects of:

- Confirm vertical and horizontal separation between the SRV Quencher and RHR Suction Strainer

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## Table 2.4.2 High Pressure Core Flooder System (Continued)

| Inspections, Tests, Analyses and Acceptance Criteria | | |
|---|---|---|
| **Design Commitment** | **Inspections, Tests, Analyses** | **Acceptance Criteria** |
| 3. continued<br>d. The HPCF System flow in each division is not less than a value corresponding to a straight line between a flow of 182 $m^3$/h at a differential pressure of 8.12 MPa and a flow of 727 $m^3$/h at a differential pressure of 0.69 MPa. | 3. continued<br>d. Tests will be conducted on each division of the as-built HPCF System in the HPCF high pressure flooder mode. Analyses will be performed to convert the test results to the conditions of the Design Commitment. | 3. continued<br>d. The converted HPCF flow satisfies the following:<br>The HPCF System flow in each division is not less than a value corresponding to a straight line between a flow of 182 $m^3$/h at a differential pressure of 8.12 MPa and a flow of 727 $m^3$/h at a differential pressure of 0.69 MPa. |
| e. The HPCF System has the capability to deliver at least 50% of the flow rates in item 3d with 171°C water at the pump suction. | e. Analyses will be performed of the as-built HPCF System to assess the system flow capability with 171°C water at the pump suction. | e. The HPCF System has the capability to deliver at least 50% of the flow rates in item 3d with 171°C water at the pump suction. |
| f. System flow into the reactor vessel is achieved within 16 seconds of receipt of an initiation signal and power available at the emergency busses. | f. Tests will be conducted on each HPCF division using simulated initiation signals. | f. The HPCF System flow is achieved within 16 seconds of receipt of a simulated initiation signal. |
| g. The HPCF pumps have sufficient NPSH available at the pumps. | g. Inspections, tests and analyses will be performed upon the as-built system. ~~NPSH tests of the pumps will be performed in a test facility. The analyses will consider the effects of:~~<br>– Pressure losses for pump inlet piping and components.<br>– Suction from the suppression pool with water level at the minimum value.<br>– ~~50% minimum blockage of the pump suction strainers.~~ | g. The available NPSH exceeds the NPSH required by the pumps. |

Delete

required

Analytically derived values for blockage of pump suction strainers based upon the as-built system.

ADD
Inspections of the as-built system will be performed to obtain piping system dimensions and other necessary information. The required NPSH of procured pumps will be determined by an inspection of the vendor specifications.
The analysis will consider the effects of:

- Confirm vertical and horizontal separation between the SRV Quencher and HPCF Suction Strainer

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## Table 2.4.4 Reactor Core Isolation Cooling System (Continued)

| Inspections, Tests, Analyses and Acceptance Criteria | | |
|---|---|---|
| **Design Commitment** | **Inspections, Tests, Analyses** | **Acceptance Criteria** |
| 3. continued<br>j. The RCIC System pump has sufficient NPSH. | 3. continued<br>j. Inspections, tests, and analyses will be performed based upon the as-built system. ~~NPSH tests of the pump will be performed at a test facility. The analyses will consider the effects of:~~<br>(1) Pressure losses for pump inlet piping and components.<br>(2) Suction from suppression pool with water level at the minimum value.<br>(3) ~~50% blockage of pump suction strainers.~~<br>(4) Design basis fluid temperature (77°C).<br>(5) Containment at atmospheric pressure. | 3. continued<br>j. The available NPSH exceeds the NPSH required by the pump. |
| k. The RCIC System operates for a period of at least 2 hours under conditions of no AC power availability and no other simultaneous failures, accidents, or other design basis conditions. | k. Inspections and analyses of the as-built RCIC and supporting systems will be performed to determine RCIC capability. | k. The RCIC System can operate for a period of at least 2 hours under conditions of no AC power availability and no other simultaneous failures, accidents, or other design basis conditions. |
| l. The RCIC can be started by local operation of the RCIC System components outside the MCR. | l. Tests will be conducted locally on RCIC System components required for system operation. | l. RCIC System components required for system operation can be actuated locally. |
| 4. If a system initiation signal occurs during the full flow test mode, the RCIC System automatically aligns to the RPV water makeup mode. | 4. Test will be conducted using simulated initiation signals. | 4. The RCIC System automatically aligns to RPV water makeup mode from test mode upon receipt of an initiation signal. |

Delete

required

(6) Confirm vertical and horizontal separation between the SRV Quencher and RCIC Suction Strainer

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Analytically derived values for blockage of pump suction strainers based upon the as-built system.

ADD
Inspections of the as-built system will be performed to obtain piping system dimensions and other necessary information. The required NPSH of procured pumps will be determined by an inspection of the vendor specifications.
The analysis will consider the effects of:

## Table 1.6-1 Referenced Reports (Continued)

| Report No. | Title | Tier 2 Section No. |
|---|---|---|
| NEDC-30851P-A | W. P. Sullivan, "Technical Specification Improvement Analyses for BWR Reactor Protection System,"~~,~~ March 1988. | 19D.6 |
| NEDE-31096-A | "GE Licensing Topical Report ATWS Response to NRC ATWS Rule 10CFR 50.62,"~~,~~ February 1987. | 19B.2 |
| NEDE-31152-P | "GE Bundle Designs,"~~,~~ December 1988. | 4.2 |
| NEDO-31331 | Gerry Burnette, "BWR Owner's Group Emergency Procedure Guidelines,"~~,~~ March 1987. | 18A |
| NEDC-31336 | Julie Leong, "General Electric Instrument Setpoint Methodology,"~~,~~ October 1986. | 7.3 |
| NEDC-31393 | "ABWR Containment Horizontal Vent Confirmatory Test, Part I,"~~,~~ March 1987. | 3B |
| NEDO-31439 | C. VonDamm, "The Nuclear Measurement Analysis & Control Wide Range Neutron Monitoring System (NUMAC-WRNMS),"~~,~~ May 1987 | 20.3 |
| NEDC-31858P | Louis Lee, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control System,"~~,~~ 1991 | 15.6 |
| NEDE-31906-P | A. Chung, "Laguna Verde Unit I Reactor Internals Vibration Measurement,"~~,~~ January 1991. | 7.4 |
| NEDO-31960 | Glen Watford, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology,"~~,~~ June 1991. | 4.4 |
| NEDC-32267P | "ABWR Project Application Engineering Organization and Procedures Manual,"~~,~~ December 1993. | 17.1 |
| NEDO-32686-A | "Utility Resolution Guide for ECCS Suction Strainer Blockage,"~~,~~ October 1998. | 6C |
| NEDC-32721P-A | "Application Methodology for the General Electric Stacked Disk ECCS Suction Strainer," Rev. 2, March 2003 (using an updated head loss correlation). | 6C |
| NEDO-33875 | "ABWR US Certified Design Aircraft Impact Assessment, Licensing Basis Information and Design Details for Key Design Features," Rev. 0, November 2016. | 19G |
| NEDE-33875P | "ABWR US Certified Design Aircraft Impact Assessment, Licensing Basis Information and Design Details for Key Design Features," Rev. 2, November 2016. | 19G |

ADD: NEDE-33878P, "ABWR ECCS Suction Strainer Evaluation of Long-Term Recirculation Capability," Rev. 0, February 2017 (GEH Proprietary Information); NEDO-33878, "ABWR ECCS Suction Strainer Evaluation of Long-Term Recirculation Capability," Rev. 0, February 2017. [Tier 2 Section No.] 6C

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Experiences related to identified regulatory or industry developed resolutions were eliminated to avoid repetition except for selected experiences that have a nuisance potential for reoccurring. Lead system engineers classified the more complex experiences.

GEH also reviewed international operating experience related to the ABWR plants built overseas to determine if any design changes were required. Recent ABWR plant design, licensing, preoperational and startup experiences were also reviewed for applicability to the ABWR Certified Design. It was determined that the previously submitted changes for the ABWR DCD Certification Renewal addressed the international operating experience and there are no additional changes required for the ABWR Certified Design.

Reference to t

| **Feature** | |
|---|---|
| Fine Moti | [illegible] |
| Internal R | [illegible] |
| Multiplexing | 7A.2 |
| Digital/Solid-State Control | 7A.7 |
| Lower Drywell Flooder | 9.5.1.2 |

~~Add: GEH also reviewed international operating experience related to the ABWR design. Experiences related to the ABWR licensing effort in the UK were reviewed for applicability to the ABWR Certified design. The UK Office of Nuclear Regulations (ONR) issued Regulatory Issues (RI) and Regulatory Observations (RO) as a result of the UK's Generic Design Assessment (GDA) of the ABWR during UK licensing review. These RIs and ROs were systematically reviewed and evaluated by ABWR subject matter experts for applicability to the ABWR standard design. The conclusion of the evaluation is that none of the RIs and ROs requires a design change to the ABWR standard design. The RIs and ROs are either unique to the UK licensing process, are already addressed in the ABWR standard design, or are the result of unique UK licensing regulations.~~

## 1.8.4 COL License Information

### 1.8.4.1 SRP Deviations

The SRP sections to be addressed by the COL applicant are indicated in the comments column of Table 1.8-19 as "COL Applicant". Where applicable the COL applicant will provide the information required by 10CFR50.34(g) similar to Tables 1.8-1 through 1.8-18 (see Subsection 1.8.1).

### 1.8.4.2 Experience Information

The experience information to be addressed by the COL applicant are indicated in the comment column of Table 1.8-22 as "COL Applicant" (see Subsection 1.8.3).

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ADD: Note that RG 1.1 has been withdrawn and replaced by RG 1.82 Revision 4.

## Table 1.8-20 NRC Regulatory Guides Applicable to ABWR

| RG No. | Regulatory Guide Title | Appl. Rev. | Issued Date | ABWR Applicable? | Comments |
|---|---|---|---|---|---|
| 1.1 | Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps | 0 | 11/70 | Yes | |
| 1.3 | Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors | 2 | 6/74 | Yes | |
| 1.4 | Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors | 2 | 6/74 | No | PWR only |
| 1.5 | Assumptions Used for Evaluating the Potential Radiological Consequences of a Steamline Break Accident for Boiling Water Reactors | 0 | 3/71 | Yes | |
| 1.6 | Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems | 0 | 3/71 | Yes | |
| 1.7 | Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident | 2 | 11/78 | Yes | |
| 1.8 | Personnel Selection and Training | -- | -- | -- | See Table 17.0-1 |
| 1.9 | Selection, Design, Qualification, and Testing of Emergency Diesel-Generator Units Used As Class 1E Onsite Electric Power Systems at Nuclear Plants | 3 | 7/93 | Yes | |
| 1.11 | Instrument Lines Penetrating Primary Reactor Containment | 0 | 3/71 | Yes | |
| 1.12 | Instrumentation for Earthquakes | 1 | 4/74 | No | NA |
| 1.13 | Spent Fuel Storage Facility Design Basis | 1 | 12/75 | Yes | |
| 1.14 | Reactor Coolant Pump Flywheel Integrity | 1 | 8/75 | No | PWR only |
| 1.16 | Reporting of Operating Information —Appendix A Technical Specifications | 4 | 8/75 | --- | COL Applicant |
| 1.20 | Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing | 2 | 5/76 | Yes | |
| 1.21 | Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light Water Nuclear Power Plants | 1 | 6/74 | Yes | |

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**Table 1.8-20 NRC Regulatory Guides Applicable to ABWR (Continued)**

| RG No. | Regulatory Guide Title | Appl. Rev. | Issued Date | ABWR Applicable? | Comments |
|---|---|---|---|---|---|
| 1.77 | Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors | 0 | 5/74 | No | PWR only |
| 1.78 | Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release | 0 | 6/74 | Yes | |
| 1.79 | Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors | 1 | 9/75 | No | PWR only |
| 1.81 | Shared Emergency and Shutdown Electric Systems for Multi-Unit Power Plants | 1 | 1/75 | Yes | |
| 1.82 | Water Sources for Long-Term Recirculation Cooling Following Loss-of-Coolant Accident | 4 ~~3~~ | 03/2012 ~~11/2003~~ | Yes | |
| 1.83 | Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes | 1 | 7/75 | No | PWR only |
| *[1.84* | *Design and Fabrication Code Case Acceptability, ASME Section III, Division 1* | *27* | *11/90* | *Yes]*(1) | |
| 1.85 | Materials Code Case Acceptability, ASME Section III, Division 1 | 27 | 11/90 | Yes | |
| 1.86 | Termination of Operating Licenses for Nuclear Reactors | 0 | 6/74 | ---- | COL Applicant |
| 1.87 | Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors (Supplement to ASME Section III Code Cases 1592, 1593, 1594, 1595, and 1596) | 1 | 6/75 | No | |
| 1.88 | Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records | | Super-ceded | | See Table 17.0-1 |
| *[1.89* | *Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants* | *1* | *6/84* | *Yes]*(2) | |
| 1.90 | Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons | 1 | 8/77 | --- | COL Applicant |
| 1.91 | Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants | 2 | 2/78 | Yes | |
| *[1.92* | *Combining Modal Responses and Spatial Components in Seismic Response Analysis* | *1* | *2/76* | *Yes]*(1) | |

**Table 1.9-1 Summary of ABWR Standard Plant COL License Information (Continued)**

| Item No. | Subject | Subsection |
|---|---|---|
| 3.27 | Reactor Internals Vibration Analysis, Measurement and Inspection Programs | 3.9.7.1 |
| 3.28 | ASME Class 2 or 3 Quality Group Components with 60-Year Design Life | 3.9.7.2 |
| 3.29 | Pump and Valve Testing Program | 3.9.7.3 |
| 3.30 | Audits of Design Specifications and Design Reports | 3.9.7.4 |
| 3.31 | Not Used | 3.9.7.5 |
| 3.32 | Not Used | 3.9.7.6 |
| 3.33 | Not Used | 3.9.7.7 |
| 3.34 | Not Used | 3.9.7.8 |
| 3.35 | Not Used | 3.9.7.9 |
| 3.36 | Not Used | 3.9.7.10 |
| 3.37 | Equipment Qualification | 3.10.5.1 |
| 3.38 | Dynamic Qualification Report | 3.10.5.2 |
| 3.39 | Qualification by Experience | 3.10.5.3 |
| 3.40 | Environmental Qualification Document (EQD) | 3.11.6.1 |
| 3.41 | Environmental Qualification Records | 3.11.6.2 |
| 3.42 | Surveillance, Maintenance, and Experience Information | 3.11.6.3 |
| 3.43 | Radiation Environment Conditions | 3I.3.3.1 |
| ~~4.1a~~ | ~~Fuel Design for ECCS Strainer Bypass~~ | ~~4.2.5.1~~ |
| 4.1b | Reactor Core Seismic and LOCA Structural Acceptance | 4.2.5.2 |
| 4.1 | Thermal Hydraulic Stability | 4.3.5.1 |
| 4.2 | Power/Flow Operating Map | 4.4.7.1 |
| 4.3 | Thermal Limits | 4.4.7.2 |
| 4.4 | CRD Inspection Program | 4.5.3.1 |
| 4.5 | CRD and FMCRD Installation and Verification During Maintenance | 4.6.6.1 |
| 5.1 | Leak Detection Monitoring | 5.2.6.1 |
| 5.2 | Plant Specific ISI/PSI | 5.2.6.2 |
| 5.3 | Reactor Vessel Water Level Instrumentation | 5.2.6.3 |
| 5.4 | Fracture Toughness Data | 5.3.4.1 |
| 5.5 | Materials and Surveillance Capsule | 5.3.4.2 |

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## Table 1.9-1 Summary of ABWR Standard Plant COL License Information (Continued)

| Item No. | Subject | Subsection |
|---|---|---|
| 5.6 | Plant Specific Pressure-Temperature Information | 5.3.4.3 |
| 5.7 | Testing of Mainsteam Isolation Valves | 5.4.15.1 |
| 5.8 | Analyses of 8-hour RCIC Capability | 5.4.15.2 |
| 5.9 | ACIWA Flow Reduction | 5.4.15.3 |
| 5.10 | RIP Installation and Verification During Maintenance | 5.4.15.4 |
| 5.11 | Program for Surveillance and Venting of Accumulated Gases | 5.4.15.5 |
| 6.1 | Protection Coatings and Organic Materials | 6.1.3.1 |
| 6.2 | Alternate Hydrogen Control | 6.2.7.1 |
| 6.3 | Administrative Control Maintaining Containment Isolation | 6.2.7.2 |
| 6.4 | Suppression Pool Cleanliness | 6.2.7.3 |
| 6.5 | Wetwell-to-Drywell Vacuum Breaker Protection | 6.2.7.4 |
| 6.5a | Containment Penetration Leakage Test (Type B) | 6.2.7.5 |
| 6.6 | ECCS Performance Results | 6.3.6.1 |
| 6.7 | ECCS Testing Requirements | 6.3.6.2 |
| 6.7a | Limiting Break Results | 6.3.6.3 |
| 6.8 | Toxic Gases | 6.4.7.1 |
| 6.9 | SGTS Performance | 6.5.5.1 |
| 6.9a | SGTS Exceeding 90 Hours of Operation per Year | 6.5.5.2 |
| 6.10 | PSI and ISI Program Plans | 6.6.9.1 |
| 6.11 | Access Requirement | 6.6.9.2 |
| ~~6.12~~ | ~~ECCS Suction Strainer~~ | ~~6C.5.1~~ |
| 7.1 | Cooling Temperature Profiles for Class 1E Digital Equipment | 7.3.3.1 |
| 7.1a | Spent Fuel Pool Level Instruments | 7.5.3.1 |
| 7.2 | APRM Oscillation Monitoring Logic | 7.6.3.1 |
| 7.3 | Effects of Station Blackout on HVAC | 7.8.1 |
| 7.4 | Electrostatic Discharge on Exposed Equipment Components | 7.8.2 |
| 7.5 | Localized High Heat Spots in Semiconductor Material for Computing Devices | 7.8.3 |
| 8.1 | Diesel Generator Reliability | 8.1.4.1 |
| 8.2 | Periodic Testing of Offsite Equipment | 8.2.4.1 |
| 8.3 | Procedures When a Reserve or Unit Auxiliary Transformer is Out of Service | 8.2.4.2 |

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The thermal-mechanical design process emphasizes that:

(1) The fuel assembly provides substantial fission retention capability during all potential operational modes.

(2) The fuel assembly provides sufficient structural integrity to prevent operational impairment of any reactor safety equipment.

The fuel assembly and its components are designed to withstand:

(1) The predicted thermal, pressure and mechanical interaction loadings occurring during startup testing, normal operation, and anticipated operational occurrences

(2) Loading predicted to occur during handling

(3) Incore loading predicted to occur from an operational basis earthquake occurring during normal operating conditions

Operating limits are established to ensure that actual fuel operation is maintained within the fuel rod thermal-mechanical design bases. These operating limits define the maximum allowable fuel pellet operating power level as a function of fuel pellet exposure. Lattice local power and exposure capabilities are applied to transform the maximum allowable fuel pellet power level into Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits.

The detailed design bases for each of the fuel assembly damage, failure and coolability criteria defined in Section II.A of Standard Review Plan 4.2 (except control rod reactivity; see Subsection 4.2.1.2) are provided in Section 4B.3 of Appendix 4B.

Note:
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~~The potential for debris bypass of ECCS suction strainers for the ABWR DCD GEH P8x8R fuel design will be bounded by the BWROG GSI-191 committee fuel blockage test program results and application bases (Reference COL Information Item 6C.5.1) since the testing will be evaluated for 10x10 fuel. The ABWR does not have core uncovery during a LOCA and therefore has greater margin than the BWROG test program for fuel debris blockage.~~

### 4.2.1.2 Control Rods

The control rod is designed to have:

(1) Sufficient mechanical strength to prevent displacement of its reactivity control material

(2) Sufficient strength to prevent deformation that could inhibit its motion

The detailed design bases for the control rod are provided in Appendix 4C.

The approval in Reference 4.2-2 contains the following conditions:

(1) [*The license/applicant must provide a plant-specific analysis of combined seismic and LOCA loading using NRC-approved methodology or another acceptable method to demonstrate conformance to the structural acceptance requirements described in Appendix A of Standard Review Plan Section 4.2.*

(2) *The license/applicant must provide an acceptable post-irradiation surveillance program or endorse the approved GEH fuel surveillance program.*

*For the reference fuel design, the second condition is satisfied by the fuel surveillance program described in Section 4B.2(3) of Appendix 4B (see also Reference 4.2-3).*]*

#### 4.2.3.2 Control Rods

##### 4.2.3.2.1 Evaluation Results

The control rod evaluations described in Section 4C.3 have been completed for the reference control rod. The evaluations demonstrate that the criteria of Appendix 4C are satisfied for the reference $B_4C$ control rod.

### 4.2.4 Testing, Inspection, and Surveillance Plans

GEH has an active program of surveillance of fuel, both production and developmental. [*The NRC has reviewed the GEH program and approved it in Reference 4.2-3.*]*

### 4.2.5 COL License Information

#### 4.2.5.1 ~~Evaluation of Post-LOCA Fuel Bundle Blockage due to ECCS Strainer Debris Bypass~~

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~~The COL shall evaluate the consequences of debris loading on the fuel bundles to confirm the acceptability of fuel design for application to ABWR. The potential for debris bypass of ECCS suction strainers will be evaluated per COL Information Item 6C.5.1. The fuel design shall be compared against the BWROG GSI-191 committee fuel blockage test program results and application bases (Reference COL Information Item 6C.5.1) to identify and evaluate key differences that could affect blockage of flow through the bundle.~~

#### 4.2.5.2 Reactor Core Seismic and LOCA Structural Acceptance

The COL applicant shall provide the NRC a confirmatory plant-specific analysis of the reactor core combined seismic and LOCA loading using NRC-approved methodology or another acceptable method to demonstrate conformance to the structural acceptance requirements described in Appendix A of Standard Review Plan Section 4.2 for the fuel referenced in the COL application. This analysis will use as input the site-specific ground motion and the fuel characteristics of the plant's initial core load.

(6) The turbine trip throttle valve (part of C002) limit switch activates when fully closed and closes F004 and F011.

(7) High reactor water level (Level 8) closes F037, F012, F045 and, subsequently, F004 and F011. This level signal is sealed in and must be manually reset. It will automatically clear if a low reactor water level (Level 2) reoccurs.

(8) High turbine exhaust pressure, low pump suction pressure, 110% turbine electrical overspeed, or an isolation signal actuates the turbine trip logic and closes the turbine trip and throttle valve. When the signal is cleared, the trip and throttle valve must be reset from the control room.

(9) Overspeed of 125% trips the mechanical trip, which is reset at the turbine.

(10) An isolation signal closes F035, F036, F048, and other valves as noted in Items (6) and (8).

(11) An initiation signal opens F001 and F004, F037, F012 and F045 when other permissives are satisfied, starts the gland seal system, and closes F008 and F009.

(12) High- and low-inlet RCIC steamline drain pot levels respectively open and close F058.

(13) The combined signal of low flow plus pump discharge pressure opens and, with increased flow, closes F011. Also see Items (5), (6) and (7).

### 5.4.6.2.2 Equipment and Component Description

#### 5.4.6.2.2.1 Design Conditions

Operating parameters for the components of the RCIC System are shown in Figure . The RCIC components are:

(1) One 100% capacity turbine and accessories.

(2) One 100% capacity pump assembly and accessories.

(3) Piping, valves, and instrumentation for:

(a) Steam supply to the turbine

(b) Turbine exhaust to the suppression pool

(c) Makeup supply from the condensate storage tank to the pump suction

(d) Makeup supply from the suppression pool to the pump suction

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ADD: Note that RG 1.1 has been withdrawn and replaced by information in RG 1.82, but the analysis in Table 5.4-1a is not changed.

(e) Pump discharge to the feedwater line, a full flow test return line, a minimum flow bypass line to the suppression pool, and a coolant water supply to accessory equipment

The basis for the design conditions is ASME B&PV Code Section III, Nuclear Power Plant Components.

Analysis of the net positive suction head (NPSH) available to the RCIC pump in accordance with the recommendations of Regulatory Guide 1.1 is provided in Table 5.4-1a.

#### 5.4.6.2.2.2 Design Parameters

Design parameters for the RCIC System components are given in Table 5.4-2. See Figure  for cross-reference of component numbers.

#### 5.4.6.2.3 Applicable Codes and Classifications

The RCIC System components within the drywell, including the outer isolation valve, are designed in accordance with ASME Code Section III, Class 1, Nuclear Power Plant Components. The RCIC System is also designed to Seismic Category I.

The RCIC System component classifications and those for the condensate storage system are given in Table 3.2-1.

#### 5.4.6.2.4 System Reliability Considerations

To assure that the RCIC System will operate when necessary and in time to prevent inadequate core cooling, the power supply for the system is taken from reliable immediately available energy sources. Added assurance is given by the capability for periodic testing during station operation.

Evaluation of reliability of the instrumentation for the RCIC System shows that no failure of a single initiating sensor either prevents or falsely starts the system.

In order to assure HPCF or RCIC availability for the operational events noted previously, certain design considerations are utilized in design of both systems.

(1) Physical Independence

The two systems are located in separate areas of the reactor building. Piping runs are separated and the water delivered from each system enters the reactor vessel via different nozzles.

(2) Prime Mover Diversity and Independence

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**Table 5.4-2 Design Parameters for RCIC System Components (Continued)**

| |
|---|
| (a) ~~Pump NPSH requirements are satisfied when strainer is 50% plugged, and particles over 2.4 mm diameter are restrained from passage into the pump and feedwater sparger.~~ |

Pump NPSH requirements are satisfied given the strainer design methods described in Appendix 6C.

water is drawn from the suppression pool, pumped through an RHR heat exchanger and delivered to the suppression pool. On two of the loops (B&C), a portion of the water returned to the suppression pool may be passed through wetwell spray headers. These two loops also have a manual feature for providing drywell spray. Water from the RCWS is pumped through the heat exchanger shell side to exchange heat with the processed water. Three cooling loops are provided, each being mechanically and electrically separate from the other to achieve redundancy. A piping and instrumentation diagram (P&ID) is provided in Section 5.4. The process diagram, including the process data, is provided for all design operating modes and conditions.

All portions of the CCS mode are designed to withstand operating loads and loads resulting from natural phenomena. All operating components can be tested during normal plant operation so that reliability can be assured. Construction codes and standards are covered in Subsection 5.4.7.

The LPFL mode is automatically initiated from ECCS signals or manually initiated. The SPC mode is started manually or automatically. The RHR System must be realigned for suppression pool cooling by the plant operator after the reactor vessel water level has been recovered. The RHR pumps are already operating. Suppression pool cooling is initiated in any of the three loops by manually closing the LPFL injection valve and opening the pool return valve. For automatic initiation of suppression pool cooling, all three RHR loops are initiated. In the event that a single failure has occurred, and the action which the plant operator is taking does not result in system initiation, then the operator will place the other totally redundant system into operation by following the same initiation procedure. If the operator chooses to utilize the containment sprays, he must close the LPFL injection valves and open the spray valves. The drywell spray mode may be initiated manually only after a high drywell pressure permissive occurs.

Preoperational tests are performed to verify individual component operation, individual logic element operation, and system operation up to the containment spray spargers. A sample of the sparger nozzles is bench tested for flow rate versus pressure drop to evaluate the original hydraulic calculations. Finally, the spargers are tested by air and visually inspected to verify that all nozzles are clear (see Subsection 5.4.7.4 for further discussion of preoperational testing).

### 6.2.2.3 Design Evaluation of the Containment Cooling System

#### 6.2.2.3.1 System Operation and Sequence of Events

In the event of the postulated LOCA, the short-term energy release from the reactor primary system will be dumped to the suppression pool. Subsequent to the accident, fission product decay heat will result in a continuing energy input to the pool. The RHR SPC mode will remove this energy which is released into the primary containment system, thus resulting in acceptable suppression pool temperatures and containment pressures.

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ADD: Note that RG 1.1 has been withdrawn and replaced by information in RG 1.82, but the analyses in Tables 6.2-2b and 6.2-2c are not changed.

In order to evaluate the adequacy of the RHR System, the following is assumed:

(1) With the reactor initially operating at 102% of rated power, a LOCA occurs.

(2) A single failure of a RHR heat exchanger is the most limiting single failure.

(3) The ECCS flows assumed available are 2 HPCF, 1 RCIC, and 2 LPFL (RHR).

(4) Containment cooling is initiated after 30 minutes. This is a conservative assumption given that the RHR system design provides pool cooling during the LPFL mode of RHR which, for a large pipe break, can occur in 3 to 5 minutes.

Analysis of the net positive suction head (NPSH) available to the RHR and HPCF pumps in accordance with the recommendations of Regulatory Guide 1.1 is provided in Tables 6.2-2b and 6.2-2c, respectively.

General compliance for Regulatory Guide 1.26 may be found in Subsection 3.2.2.

#### 6.2.2.3.2 Summary of Containment Cooling Analysis

When calculating the long-term post-LOCA pool temperature transient, it is assumed that the initial suppression pool temperature and the RHR service water temperature are at their maximum values. This assumption maximizes the heat sink temperature to which the containment heat is rejected and thus maximizes the containment temperature. In addition, the RHR heat exchanger is assumed to be in a fully fouled condition at the time the accident occurs. This conservatively minimizes the heat exchanger heat removal capacity. Even with the degraded conditions outlined above, the maximum temperature is maintained below the design limit specified in Subsection 6.2.2.1.

It should be noted that, when evaluating this long-term suppression pool transient, all heat sources in the containment are considered with no credit taken for any heat losses other than through the RHR heat exchanger. These heat sources are discussed in Subsection 6.2.1.1.3.

It can be concluded that the conservative evaluation procedure described above clearly demonstrates that the RHR System in the SPC mode limits the post-LOCA containment temperature transient.

#### 6.2.2.3.3 Severe Accident Considerations

The containment spray features of the RHR System can reduce the amount of radioactive material released to the environment in the event core damage occurs. The benefits provided by the sprays are condensing steam, scrubbing of fission products in the containment airspace, and supplying water to ex-vessel core debris. The conditions for activation of the containment sprays are described in the Emergency Procedure Guidelines in Appendix 18A.

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#### 6.3.1.2.4 Automatic Depressurization System

The ADS utilizes a number of the reactor safety/relief valves (SRVs) to reduce reactor pressure during small breaks in the event of HPCF failure. When the vessel pressure is reduced to within the capacity of the low pressure system, these systems provide inventory makeup so that acceptable post-accident temperatures are maintained.

## 6.3.2 System Design

A more detailed description of the individual systems, including individual design characteristics of the systems, is provided in Subsections 6.3.2.1 through 6.3.2.4.

The following discussion provides details of the combined systems; in particular, those design features and characteristics which are common to all systems.

### 6.3.2.1 Schematic Piping and Instrumentation Diagrams

The P&IDs for the ECCS are identified in Subsection 6.3.2.2. The process diagrams which identify the various operating modes of each system are also identified in Subsection 6.3.2.2.

### 6.3.2.2 Equipment and Component Descriptions

The starting signal for the ECCS comes from four indepe
drywell pressure and low reactor water level. The ECCS i
no operator action during the first 30 min following the a
of the systems is provided in Table 6.3-2.

ADD. Note that RG 1.1 is withdrawn and replaced by RG 1.82, which recommends no credit for containment pressurization during the transient.

Electric power for operation of the ECCS is from regular AC power sources. Upon loss of the regular power, operation is from onsite emergency standby AC power sources. Emergency sources have sufficient capacity so that all ECCS requirements are satisfied. Each of the three ECCS functional groups identified in Subsection 6.3.1.1.3(1) has its own diesel generator emergency power source. Section 8.3 contains a more detailed description of the power supplies for the ECCS.

Regulatory Guide 1.1 prohibits design reliance on pressure and/or temperature transients expected during a LOCA for assuring adequate NPSH. The requirements of this Regulatory Guide are applicable to the HPCF, RCIC and RHR pumps.

The BWR design conservatively assumes 0 kPaG containment pressure and maximum expected temperatures of the pumped fluids. Thus, no reliance is placed on pressure and/or temperature transients to assure adequate NPSH.

Requirements for NPSH are given in Tables 6.2-2c (HPCF), 5.4-1a (RCIC) and 6.2-2b (RHR). Vessel pressure versus system flow curves are given in Figures 6.3-4 (HPCF), 6.3-5 (RCIC) and 6.3-6 (RHR).

# 6C Containment Debris Protection for ECCS Strainers

## 6C.1 Background

NRC Bulletin No. 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers," references NRC guidance and highlights the need to adequately accommodate suppression pool debris in design by focusing on an incident at the Perry Nuclear Plant. Similar concerns were later identified throughout the industry and documented by subsequent bulletins and generic letters including NRC Bulletin 95-02, NRC Bulletin 96-03, Generic Letter 97-04, and Generic Letter 98-04. GEH reviewed the concerns addressed by these bulletins/letters and has determined that the ABWR design satisfactorily accommodates suppression pool debris for a number of reasons as discussed in the following:

The ultimate concern raised by the Perry incident was the deleterious effect of debris in the suppression pool and how it could impact the ability to draw water from the suppression pool during an accident. To address this concern, the ABWR design has committed to following the guidance provided in Regulatory Guide 1.82 as well as NEDO-32686-A (Utility Resolution Guide for ECCS Suction Strainer Blockage), and additional guidance as described below.

The ABWR is designed to inhibit debris generated during a LOCA from preventing operation of the Residual Heat Removal (RHR), Reactor Core Isolation Cooling (RCIC) and High Pressure Core Flooder (HPCF) systems.

## 6C.2 ABWR Mitigating Features

The ABWR has substantially reduced the amount of piping in the drywell relative to earlier designs and consequently the quantity of insulation required. Furthermore, there is no equipment in the wetwell spaces that requires insulation or other fibrous materials. The ABWR design conforms with the guidance provided by the NRC for maintaining the ability for long-term recirculation cooling of the reactor and containment following a LOCA.

The Perry incident was not the result of a LOCA but rather debris entering the Suppression Pool during normal operation. The arrangement of the drywell and wetwell/wetwell airspace on a Mark III containment (Perry) is significantly different from that utilized in the ABWR design. In the Mark III containment, the areas above the suppression pool water surface (wetwell airspace) are substantially covered by grating with significant quantities of equipment installed in these areas. Access to the wetwell airspace (containment) of a Mark III is allowed during power operations. In contrast, on the ABWR the only connections to the suppression pool are the 10 drywell connecting vents (DCVs), and access to the wetwell or drywell during power operations is prohibited. The DCVs will have horizontal steel plates located above the openings that will prevent any material falling in the drywell from directly entering the vertical leg of the DCVs. This arrangement is similar to that used with the Mark II connecting vent pipes. Vertically oriented trash rack construction will be installed around the periphery of the horizontal steel plate to intercept debris. The trash rack design shall allow for adequate flow

from the drywell to wetwell. In order for debris to enter the DCV it would have to travel horizontally through the trash rack prior to falling into the vertical leg of the connecting vents. Thus the ABWR is resistant to the transport of debris from the drywell to the wetwell.

In the Perry incident, the insulation material acted as a septa to filter suspended solids from the suppression pool water. The Mark I, II, and III containments have all used carbon steel in their suppression pool liners. This results in the buildup of corrosion products in the suppression pool which settle out at the bottom of the pool until they are stirred up and re-suspended in the water following some event (SRV lifting). In contrast, the ABWR liner of the suppression pool is fabricated from stainless steel which significantly lowers the amount of corrosion products which can accumulate at the bottom of the pool.

A further mitigating feature for the ABWR is that the insulation installed on the ASME Section III, Class 1 piping greater than 80 mm in the drywell, i.e., the large bore piping, is reflective metal type (RMI). Use of RMI minimizes the fibrous insulation source term from the upper drywell used in the suction strainer design. This use of RMI is a significant factor in design that reduces the potential suction strainer debris load and further reduces the potential for suction strainer clogging.

Since the debris in the Perry incident was created by roughing filters on the containment cooling units a comparison of the key design features of the ABWR is necessary. In the Mark III design more than 1/2 of the containment cooling units are effectively located in the wetwell airspace. For the ABWR there are no cooling fan units in the wetwell air space. Furthermore the design of the ABWR Drywell Cooling Systems does not utilize roughing filters on the intake of the containment cooling units during plant operation.

In the event debris enters the suppression pool and does not settle on the pool bottom, the Suppression Pool Cleanup System (SPCU) will remove the suspended debris during normal plant and SPCU operation. The SPCU is described in Section 9.5.9 and shown in Figure 9.5-1. The SPCU is designed to provide a continuous cleanup flow of 250 $m^3/h$. This flow rate is sufficiently large to effectively maintain the suppression pool water at the required purity. The SPCU system is intended for continuous operation and the suction pressure of the pump is monitored and an alarm is provided on low pressure. Early indication of any deterioration of the suppression pool water quality will be provided if significant quantities of debris were to enter the suppression pool and cause the strainer to become plugged resulting in a low suction pressure alarm.

INSERT 6C-1

The ABWR design also has additional features not utilized in earlier designs that could be used in the highly improbable event that all suppression pool suction strainers were to become plugged. The alternate Alternating Current independent water addition (ACIWA) mode of RHR allows water from the Fire Protection System to be pumped to the vessel and sprayed in the wetwell and drywell from diverse water sources to maintain cooling of the fuel and containment. The wetwell can also be vented at low pressures to assist in cooling the containment.

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The suction strainers design at Perry preceded and did not meet the current regulatory requirements. The ABWR ECCS suction strainers are patented GE optimized stacked disk design in accordance with NEDC-32721P-A Rev.2. This strainer design was developed in response to NRC Bulletin 96-03 as a replacement of existing ECCS strainers with a large capacity passive strainer design. This strainer design utilizes disks whose internal radius and thickness vary over the height of the strainer. The selected variation in these parameters achieves an increased surface area compared to existing strainers of the same size to provide a higher capacity for debris capture. The ABWR strainer will perform with a minimum head loss for the range of possible amounts of debris while fitting in the required volume. To avoid debris clogging the flow restrictions downstream of the strainers, the size of the holes in the perforated sheets is chosen by considering specific flow paths of ECCS equipment and piping (for example, the containment spray nozzle and the ECCS pump seal cooling flow orifices). The strainers will have holes no larger than 3.175 mm (0.125 inch).

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## 6C.3 Design Considerations

### 6C.3.1 RG 1.82 Improvement

All ECCS strainers will at a minimum be sized to conform with the guidance provided in Reg Guide 1.82 for the most severe of all postulated breaks.

The following clarifying assumptions will also be applied and will take precedence:

(1) The debris generation model ~~shall be consistent with Methods 1, 2, or 3 from the zone of influence approach in~~utilizes spherical zones of influence (ZOI) in accordance with the Utility Resolution Guidance, Reference 6C-3.

(2) Of the debris generated, the amount that is transported to the suppression pool shall be determined in accordance with Reference 6C-3 based on similarity of the Mark III upper drywell design. This approach is conservative due to the ABWR containment improvements over the Mark III as discussed in Section 6C.2.

(3) The debris in the suppression pool will be assumed to remain suspended until it is captured on the surface of a strainer.

Suction Strainer sizing is based on satisfying NPSH requirements at runout flow~~, plus margin,~~ with the design basis debris in the suppression pool accumulated on the suction strainers.

The sizing of the suction strainers assumes that the insulation debris in the suppression pool is ~~proportionally~~ distributed to the pump suctions based on the maximum debris load fraction assuming flow rates of the operating systems at limiting runout conditions. The strainers assumed available for capturing insulation debris for the limiting design condition are ~~two~~one RHR ~~suction strainers and a single~~loop, one HPCF loop, and the~~or~~ RCIC system~~suction strainer~~.

### 6C.3.2 Chemical Effects

~~The chemical effects of the post-LOCA environment on debris shall be evaluated to assess the extent to which chemical reaction products contribute to blockage of the ECCS strainers. The evaluation shall be submitted by the COL Applicant and shall demonstrate that the effects of chemical reaction products from post-LOCA debris shall not prevent long-term cooling of the core (COL 6.12).~~The ABWR design has been reviewed for the potential generation of chemical precipitates which may contribute to strainer head loss following a LOCA. In general, the ABWR design features preclude the materials and environmental conditions which are most problematic for generation of chemical precipitate debris that may contribute to blockage and head loss.

The primary containment will not contain reactive materials such as aluminum, phosphates, or calcium silicate, and minimizes zinc by prohibiting it except for a small amount in galvanized steel and inorganic primers. Inorganic zinc primers are top coated with an epoxy layer that

NEDE-33878P, ABWR ECCS Suction Strainer Evaluation of Long-Term Recirculation Capability, Rev. 0, (Reference 6C-5) evaluated the impact of debris downstream of the ECCS strainers causing blockage or wear and abrasion. The three areas of concern evaluated are (1) blockage of system flow paths at narrow flow passages (e.g.ECCS sparger spray nozzles), (2) wear and abrasion of surfaces (e.g., pump running surfaces and heat exchanger tubes and orifices, and (3) blockage of flow paths through fuel assemblies.

This assessment concludes that ECCS flow paths and components downstream of the strainers, including fuel assemblies, are not susceptible to failure from debris blockage, particulate ingestion, abrasive effects and long term degradation and can perform required safety functions during the required mission time.

prevents exposure to the LOCA environment. Coatings are qualified as described in Subsection 6.1.2. The debris load described in Table 6C-1 accounts for coatings that are destroyed during a LOCA.

An important consideration in the generation of corrosion products is the post-LOCA environment which, for some plant designs, can be of an acidic nature due to the use of boric acid in the primary coolant. The ABWR does not utilize boric acid. The Standby Liquid Control System is capable of injecting a sodium pentaborate solution, however this system is not used during a LOCA. Standby Liquid Control is only used to mitigate ATWS events as described in Appendix 15E. Consequently, the post-LOCA environment inside containment is relatively pH neutral with a flat time history throughout the event as described in Section 6.1.1.2.

### 6C.3.3 Downstream Effects

The effects of debris ~~passing through the strainers shall be~~being transported from the suppression pool are evaluated for interactions with downstream components such as pumps, valves, and heat exchangers and also for the potential blockage of coolant flow at the entrance to the fuel assemblies. ~~The evaluation shall be submitted by the COL Applicant and shall demonstrate that the effects of debris bypass of the strainer shall not prevent long-term cooling of the core (COL 6.12).~~The ABWR design includes several mitigating features that reduce the likelihood of such adverse debris interactions. These include:

- Minimal opportunity for debris generation in the wetwell. High energy breaks are restricted to the drywell, and debris generated there must pass through trash racks and vertical/horizontal vents before reaching the suppression pool.
- Diverse ECCS delivery locations, which include injection both inside and outside the core shroud.
- Bypass flow paths which exist around the debris filters of the fuel assemblies.
- The Suppression Pool Cleanup System will minimize the quantity of latent debris in the suppression pool. A suppression pool cleanliness program will be developed (Subsection 6.2.7.3) to minimize the quantity of latent debris.
- The suction strainers themselves, which capture any particles greater than the hole size of the perforated strainer plates.

## 6C.4 Discussion Summary

In summary, the ABWR design includes the necessary provisions to prevent debris from impairing the ability of the RCIC, HPCF, and RHR systems to perform their required post-accident functions. Specifically, the ABWR design does the following:

(1) The design is resistant to the transport of debris to the suppression pool.

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(2) The suppression pool liner is stainless steel, which significantly reduces corrosion products.

(3) Plant Housekeeping and Foreign Material Exclusion (FME) procedures assure pool cleanliness prior to plant operation and over plant life such that no significant debris are present in the suppression pool.

(4) Periodic SPCU operation maintains suppression pool cleanliness. Low SPCU pump suction pressure can provide early indication of debris present in the suppression pool and permit the plant operator to take appropriate corrective action.

(5) The equipment installed in the drywell and wetwell minimize the potential for generation of debris.

(6) The ECCS suction strainers meet the current regulatory requirements.

## 6C.5 Strainer Sizing Analysis Summary

A preliminary analysis was performed to assure that the above requirements could be satisfied using strainers compatible with the suppression pool design as shown by Figure 1.2-13i.

Each loop of an ECCS system utilizes a single stacked disk suction strainer. The strainer design conforms to the methodology defined in Reference 6C-4. The strainer has a central core of varying radius such that the flow through the entire central region is maintained at constant velocity. The constant velocity core minimizes head loss where velocities are the greatest. A number of perforated disks of varying internal diameter and whose thickness may vary with radius surround the central core.

All of the debris is assumed to deposit on the strainers. The debris load is characterized by the methods in Reference 6C-3, and quantities are summarized in Table 6C-1. The distribution of debris volume to the strainer regions was determined as a fraction of the proportional loop flow splits. The strainer sizing is calculated based on the strainer flow rate and debris load. The head loss is calculated by a method based upon Reference 6C-4 which uses empirical correlations to test data. The methodology considers losses through a clean strainer and factors in the effects of the debris bed taking into account the thickness of the bed, and the type of debris (fiber, RMI, sludge, etc.). Consideration is given to whether the quantity of debris is sufficient to fully engulf the gaps between the strainer disks, as this has an influence on the head loss correlation.

By making realistic assumptions, the following additional conservatisms are likely to occur, but they were not applied in the analysis. No credit in water inventory was taken for water additions from feedwater flow or flow from the condensate storage tank as injected by RCIC or HPCF. Also, for the long term cooling condition, when suppression pool cooling is used instead of the low pressure flooder mode (LPFL), the RHR flow rate decreases from runout (1130 m3/h) to rated flow (954 m3/h), which reduces the pressure drop across the debris.

## 6C.6 COL License Information

### 6C.6.1 ~~Debris Evaluation for ECCS Suction Strainer~~Deleted

~~An evaluation shall be submitted by the COL Applicant that demonstrates that chemical effects and the effect of debris bypass of the strainers does not prevent long-term cooling of the core (COL 6.12). The evaluation shall be based on the research and recommendations of the BWR Owner's Group GSI-191 committee.~~

## 6C.7 References

6.C-1 Debris Plugging of Emergency Core Cooling Suction Strainers, USNRC Bulletin No. 93-02, May 11, 1993.

6C-4

6.C-2 Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident, USNRC Reg. Guide 1.82 Rev. ~~3~~4.

6.C-3 Utility Resolution Guidance for ECCS Suction Strainer Blockage, NEDO-32686-A, October, 1998.

~~6.3-4~~ Application Methodology for the General Electric Stacked Disk ECCS Suction Strainer, NEDC-32721P-A, March 2003 (using an updated head loss correlation).

Add: 6C-5 NEDE-33878P, "ABWR ECCS Suction Strainer Evaluation of Long-Term Recirculation Capability," Rev. 0, February 2017 (GEH Proprietary Information); NEDO-33878, "ABWR ECCS Suction Strainer Evaluation of Long-Term Recirculation Capability," Rev. 0, February 2017.

NO CHANGE THIS PAGE

### Table 6C-1 ECCS Strainer Debris Load

| Debris Type | Strainer Load |
|---|---|
| Sludge / corrosion prod. | 90.7 kg (200 lbm) |
| Inorganic Zinc (IOZ) | 21.3 kg (47 lbm) |
| Epoxy Coated IOZ | 38.6 kg (85 lbm) |
| Rust Flakes | 22.7 kg (50 lbm) |
| Dust / Dirt | 68.0 kg (150 lbm) |
| Reflective Metal Insulation | 35.8 $m^2$ (385 $ft^2$) |
| Nukon Fiber Insulation | 23.4 kg (51.6 lbm) |

(d) Proper operation of RCIS software, including verification of gang rod selection and verification logic, rod withdrawal sequence restrictions, rod worth minimizer, and banked position reference rod pull sequence functions

(e) Proper communication with interfacing systems such as the power generation control system, the automatic power regulator, and the automatic rod block monitor

(f) Proper operation of automated thermal limit monitor (ATLM) to generate a rod block signal based on LPRM and control rod position input data that simulate a condition of fuel operating thermal limits violation

(g) Capability of RCIS continued operation under the condition with different subsystems of RCIS being bypassed

(h) Proper functioning of the RCIS bypass interlock logic to preclude a bypass state that could render the RCIS inoperational as specified in the appropriate design documents

(i) Proper operation of single-failure design feature of the RCIS by verifying that the RCIS is capable of continued operation with one channel disabled, that one channel can cause a rod block, and that two channels must be in agreement to cause normal RCIS functioning of control rod movements

### 14.2.12.1.8 Residual Heat Removal System Preoperational Test

(1) Purpose

To verify the proper operation of the Residual Heat Removal (RHR) System under its various modes of operation: core cooling, shutdown cooling, wetwell and drywell spray, suppression pool cooling, and supplemental fuel pool cooling.

(2) Prerequisites

The construction tests have been successfully completed, and the SCG has reviewed the test procedure and approved the initiation of testing. The reactor vessel shall be intact and capable of receiving injection flow from the various modes of RHR. Reactor Building Cooling Water System, Instrument Air System, Fuel Pool Cooling and Cleanup System, Leak Detection System, RCIC System, Suppression Pool Water System, Nuclear Boiler System, Process Computer System, Electric Power Distribution System, Process Computer System and other required interfacing systems shall be available, as needed, to support the specified testing and the appropriate system configurations. Additionally, ~~RHR pump suctionline shall be installed with a 50% plugged temporary strainer throughout the test. Also,~~ the suppression pool water shall be of a quality acceptable prior to injection testing with flow from the suppression pool to the reactor.

NOTE: Remove validation of 50% clogged strainer requirement

NOTE: Remove validation of 50% clogged strainer requirement

### 14.2.12.1.10 High Pressure Core Flooder System Preoperational Test

(1) Purpose

To verify the operation of the High Pressure Core Flooder (HPCF) System, including related auxiliary equipment, pumps, valves, instrumentation and control, is as specified.

(2) Prerequisites

The construction tests have been successfully completed, and the SCG has reviewed the test procedure and approved the initiation of testing. ~~A temporary strainer shall be installed with 50% plugged in the pump suction throughout this test.~~ The suppression pool and condensate storage tank shall be available as HPCF pump suction sources and the reactor vessel shall be sufficiently intact to receive HPCF injection flow. The Instrument Air System, Makeup Water Condensate System, Residual Heat Removal System, Remote Shutdown System, Reactor Building Cooling Water System, and appropriate electrical power sources shall be available as needed, to support the specified testing and the appropriate system configurations.

(3) General Test Methods and Acceptance Criteria

Performance shall be observed and recorded during a series of individual component and integrated system tests. This test shall demonstrate that the HPCF System operates properly as specified by Subsections 6.3.2.2.1 and 7.3.1.1.1.1 and applicable HPCF System design specification through the following testing:

(a) Correct implementation and operation of the HPCF System software-based controls and instrumentation. This test shall check the system behavior against the functional, performance and interface requirements as specified by the appropriate design documents and the Hardware/Software System Specification (HSSS).

(b) Verification of various component alarms for proper alarm actuation by practically operating the detector of the alarm generating source or using the simulated signal and alarm reset.

(c) Proper operation of all motor-operated valves including opening and closing with the operating switch, valve status indication and travel timing, if applicable.

(d) Proper operation of HPCF pumps and motors during continuous run tests.

(e) Acceptable pump NPSH under the most limiting design flow conditions.

Provided under MFN 16-065

## Safety Issues Index (Continued)

| Title | | NRC Priority | Tier 2 Subsection |
|---|---|---|---|
| 105 | Interfacing Systems LOCA at BWRs | High | 19B.2.45<br>COL App. |
| 106 | Piping and Use of Highly Combustible Gases in Vital Areas | Medium | 19B.2.46 |
| 118 | Tendon Anchorage Failure | Resolved | 19B.2.48 |
| 124 | Auxiliary Feedwater System Reliability | Resolved | 19B.2.51 |
| 128 | Electrical Power Reliability | Resolved | 19B.2.52 |
| 142 | Leakage Through Electrical Isolators in Instrumentation Circuits | Medium | 19B.2.53 |
| 143 | Availability of Chilled Water Systems | High | 19B.2.54 |
| 145 | Actions to Reduce Common Cause Failures | Resolved | 19B.2.55<br>COL App. |
| 153 | Loss of Essential Service Water in LWRs | High | 19B.2.57<br>COL App. |
| 155.1 | More Realistic Source Term Assumptions | Resolved | 19B.2.58 |
| | Add Insert 9 | | |
| **Human Factors Issues** | | | |
| HF.1.1 | Shift Staffing | Resolved | 18.8.2 |
| HF.4.4 | Guidelines for Upgrading Other Procedures | High | 18.8.1<br>18E.1.7 |
| HF.5.1 | Local Control Stations | High | 18.8.11 |
| HF.5.2 | Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation | High | 18.8.9 |
| **Issues Resolved With No New Requirements** | | | |
| A-17 | Systems Interaction in Nuclear Power Plants | Resolved | 19B.2.59 |
| A-29 | Nuclear Power Plant Design for Reduction of Vulnerability to Industrial Sabotage | Resolved | 19B.2.60<br>COL App. |
| B-5 | Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments | Resolved | 19B.2.61 |
| C-8 | Main Steamline Leakage Control Systems | Resolved | 19B.2.61.1 |
| 29 | Bolting Degradation or Failure in Nuclear Power Plants | Resolved | 19B.2.62 |
| 82 | Beyond Design Basis Accidents in Spent-Fuel Pools | Resolved | 19B.2.63 |
| 113 | Dynamic Qualification Testing of Large Bore Hydraulic Snubbers | Resolved | 19B.2.64 |

Provided under MFN 16-065

# Insert 9
# Safety Issues Index

| Title | | NRC Priority | Tier 2 Subsection |
|---|---|---|---|
| New Generic Issues | | | |
| 186 | Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants | TBD | 19B.2.74 |
| 189 | Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion during a Severe Accident | TBD | 19B.2.75 |
| 191 | Assessment of Debris Accumulation on PWR Sump Performance | TBD | 19B.2.76 |
| 193 | BWR ECCS Suction Concerns | TBD | 19B.2.77<br>Tier 1<br>Table 2.4.1 Item 4c,<br>Table 2.4.2 Item 3g,<br>Table 2.4.4 Item 3j |
| 199 | Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States for Existing Plants | TBD | 19B.2.78<br>COL App. Items 2.3.1.2 and 2.3.2.22. |

Provided under MFN 16-065

(Figures 12.3-56 through 12.3-73) as well as the specific area radiation channels for each building, the detector map location, the channel sensitivity range, and the local alarm areas (Tables 12.3-3 through 12.3-7).

### References

19B.2.72-1 NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident", U.S. NRC, May 1980.

19B.2.72-2 NUREG-0737, "Clarification of TMI Action Plan Requirements", U.S. NRC, November 1980.

## 19B.2.73 III.D.3.3(2): Set Criteria Requiring Licensees to Evaluate Need for Additional Survey Equipment

### Issue

NUREG-0660 (Reference 19B.2.73-1) is a guideline to improve nuclear power plant worker radiation protection to allow workers to take effective action to control the course and consequences of an accident, as well as to keep exposures as low as reasonably achievable (ALARA) during normal operation and accidents.

### Acceptance Criteria

This issue required the NRR to set criteria requiring licensees to evaluate in their plants the need for additional survey equipment and radiation monitors in vital areas and requiring, as necessary, installation of area monitors with remote readout. The NRR evaluated the need to specify the minimum types and quantities of portable monitoring instrumentation, including very high dose rate survey instruments. Operating reactors were reviewed for conformance with Standard Review Plan (SRP) Section 12.3.4, "Area Radiation and Airborne Radioactivity Monitoring Instrumentation". The NRR revised the SRP Sections 12.5 and 12.3.4 to incorporate additional monitor requirement criteria.

### Resolution

Item III.D.3.3(2) which concerns licensees evaluate the need for additional radiation survey equipment is resolved in Subsection 12.3.4. This item also concerned the need to specify the minimum types and quantities of portable monitoring instrumentation, including very high dose rate survey instruments. As noted in Subsections 12.5.2, 19A.2.39 and 19A.3.5, COL applicants will provide the portable instruments in operating reactors that accurately measure radio-iodine concentration in plant areas under accident conditions.

### References

19B.2.73-1 NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident", U.S. NRC, May 1980.

19B.2.73-2 NUREG-0737, "Clarification of TMI Action Plan Requirements", U.S. NRC, November 1980.

Insert 10

### 19B.2.76 191: Assessment of Debris Accumulation on PWR Sump Performance

A study was deemed to be required to determine whether PWR ECCS sumps are adequate to ensure proper ECCS operation. Based on the existence of an action plan to address the safety concerns, the issue was considered nearly-resolved in September 1996. It was later given a HIGH priority ranking in SECY-98-166.

The staff's Technical Assessment concluded that GSI-191 was a credible concern for the population of domestic PWRs, and that detailed plant-specific evaluations were needed to determine the susceptibility of each U.S.-licensed PWR to ECCS sump blockage.

#### Acceptance Criteria

Not applicable. Issue does not apply to ABWRs.

#### Resolution

This issue is specific to PWRs. GSI 193 addresses BWR ECCS Suction Concerns

Therefore, Issue 191 is resolved for ABWR.

#### References

19B.2.76-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-34)," U.S. NRC, June 2016.

### 19B.2.77 193: BWR ECCS Suction Concerns

#### Issue

This issue addressed the possible failure of low pressure emergency core cooling systems due to unanticipated, large quantities of entrained gas in the suction piping from suppression pools in BWR Mark I containments.

The swell/exclusion zone in the BWR Mark I torus after a LOCA is considered to be limited to less than one diameter of the down-comer pipe. The ABWR Containment is not as limiting as the Mark I and therefore this condition may not be present in the ABWR.

#### Acceptance Criteria

Not applicable. The ABWR containment differs from the Mark I Containment and the arrangement of the horizontal vents alleviates the problem identified for the Mark I Containment Downcomers. This issue is resolved for ABWR renewal application.

#### Resolution

For containment suppression pool LOCA analyses, an NRC SER for two GE topical reports (NEDO-30832 and NEDO- 31695-A) accepts the elimination of suppression pool local temperature limits with the proviso that the ECCS suction strainer inlet be below the quencher outlet. If this is not the case for a specific installation, the new strainers may need to be evaluated for the potential effect of air and or steam ingestion from an SRV quencher into the strainer as this could potentially affect ECCS pump/system performance.

NO CHANGE THIS PAGE
-Changes shown applied under
MFN 16-065 Supplement 1
[ML16323A006]

**Editorial Note: Replace the original text in MFN 16-065, Enclosure 2, Insert 10 with the following:**

## 19B.2.77 193: BWR ECCS Suction Concerns

### Issue

This issue addressed the possible failure of low pressure emergency core cooling systems due to unanticipated, large quantities of entrained gas in the suction piping from suppression pools in BWR Mark I containments.

The swell/exclusion zone in the BWR Mark I torus after a LOCA is considered to be limited to less than one diameter of the down-comer pipe. The ABWR Containment is not as limiting as the Mark I and therefore this condition ~~may~~ is not ~~be~~ present in the ABWR.

### Acceptance Criteria

Not applicable. The ABWR containment differs from the Mark I Containment and the arrangement of the horizontal vents alleviates the problem identified for the Mark I Containment Down-comers. Additionally, there is sufficient distance in the ABWR Design between the SRV Discharge Quencher and the ECCS suction filters to prevent steam ingestion into the ECCS Suction Piping (Figure 1.2-13i). This issue is resolved for ABWR renewal application.

### Resolution

~~For containment suppression pool LOCA analyses, an NRC SER for two GE topical reports (NEDO-30832 and NEDO- 31695-A) accepts the elimination of suppression pool local temperature limits with the proviso that the ECCS suction strainer inlet be below the quencher outlet.~~

Horizontal vent full scale testing for Mark III containments (Reference 19B.2.77-2) showed that the bubbles formed following vent clearing do not reach the containment outside wall during pool swell.

There is also sufficient distance between SRV Discharge Quencher and the ECCS suction filters to prevent steam ingestion into the ECCS Suction Piping (Figure 1.2-13i).

Tier 1, Tables 2.4.1, 2.4.2 and 2.4.4 (ITAAC) include a requirement for the respective ECCS pump suction Strainer for a verification of adequate vertical and horizontal separation between the ECCS suction strainer and the SRV quencher to prevent the potential effect of air and or steam ingestion. The acceptance criterion is based on Figure 1.2-13i.

Therefore, Issue 193 is resolved for ABWR with the addition of the ITAACs for verification of the distances in the as-built conditions. ~~actions identified for the COL~~

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-Changes shown applied under
MFN 16-065 Supplement 1
[ML16323A006]

~~Holder.~~

**References**

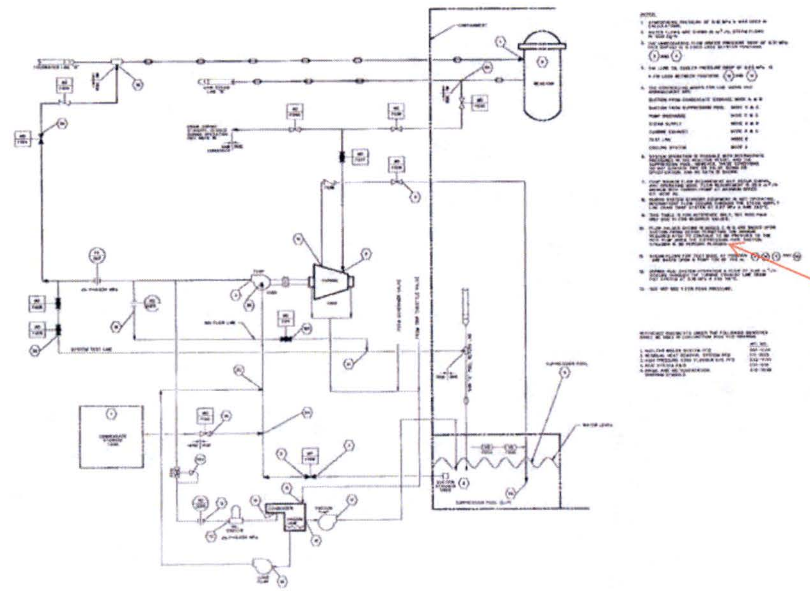

19B.2.77-1 NUREG-0933, "Resolution of Generic Safety Issues (Formerly entitled "A Prioritization of Generic Safety Issues" with Supplements 1-34)," U.S. NRC, June 2016.

~~19B.2.77-2 NEDC-32721P-A, "Application Methodology for the General Electric Stacked Disk ECCS Suction Strainer", Revision 2, March 2003~~

~~19B.2.77-3 NEDO-30832, "Elimination of Limit on BWR Suppression Pool Temperature For SRV Discharge with Quenchers", Revision 0, December 1984~~

~~19B.2.77-4 NEDO-31695-A, "BWR Suppression Pool Temperature Technical Specification Limits", Revision 0, May 1995~~

19B.2.77-2 NEDE-25273, "Scaling Study of the General Electric Pressure Suppression Test Facility Mark III Long-Range Program Task 2.2.1", Revision 0, March 1980.



REPLACE WITH THE FOLLOWING:

10. FLOW VALUES SHOWN IN MODES C & D ARE BASED UPON SUCTION PIPING DESIGN PERMITTING THE MINIMUM REQUIRED NPSH TO CONTINUE TO BE PROVIDED TO THE RCIC PUMP WHEN SUPPRESSION POOL SUCTION STRAINER IS 50 PERCENT PLUGGED HEAD LOSS IS AT THE DESIGN VALUE.

**Figure 5.4-9 Reactor Core Isolation Cooling System PFD (Sheet 1 of 2)**

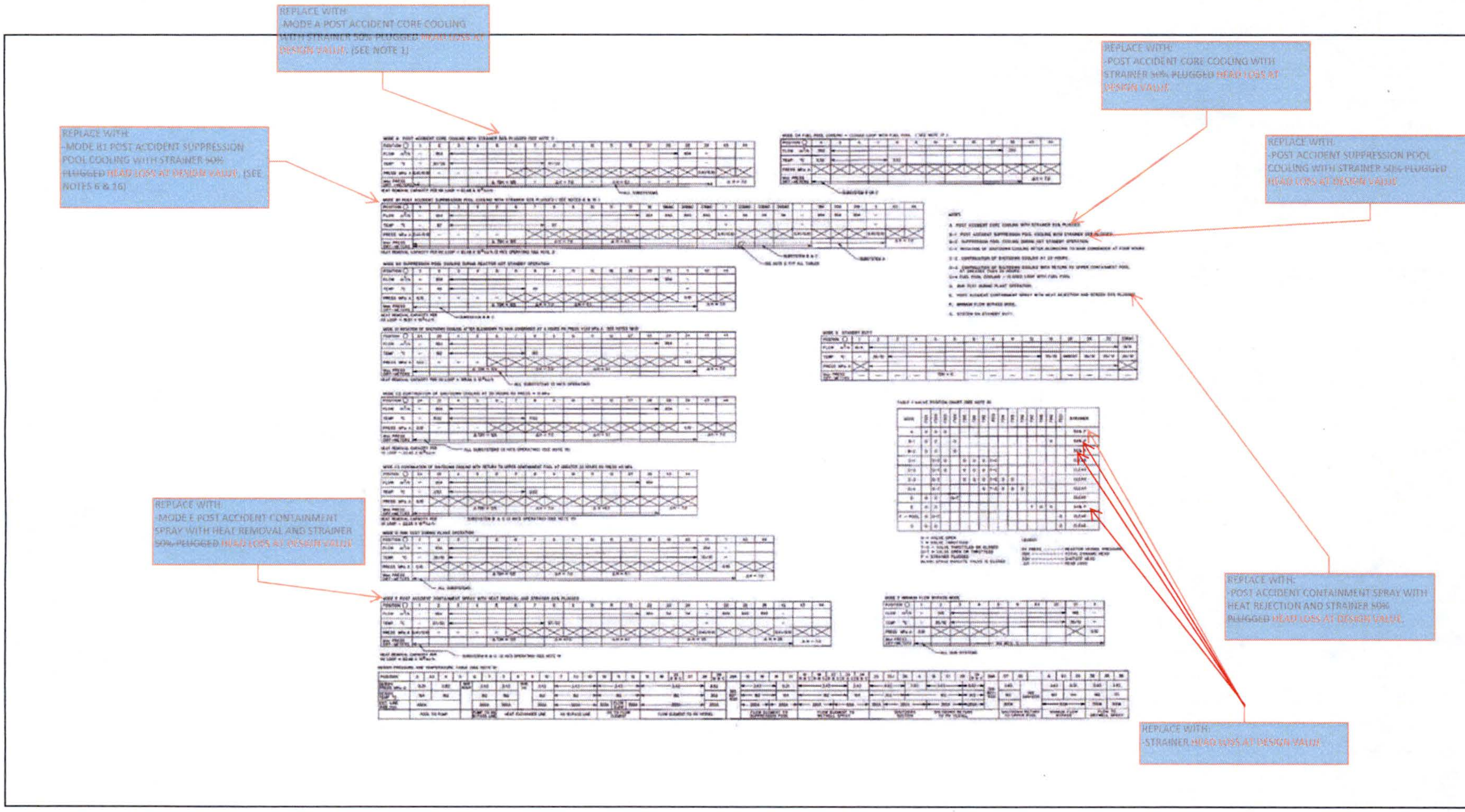



**Figure 5.4-11 Residual Heat Removal System PFD (Sheet 2 of 2)**

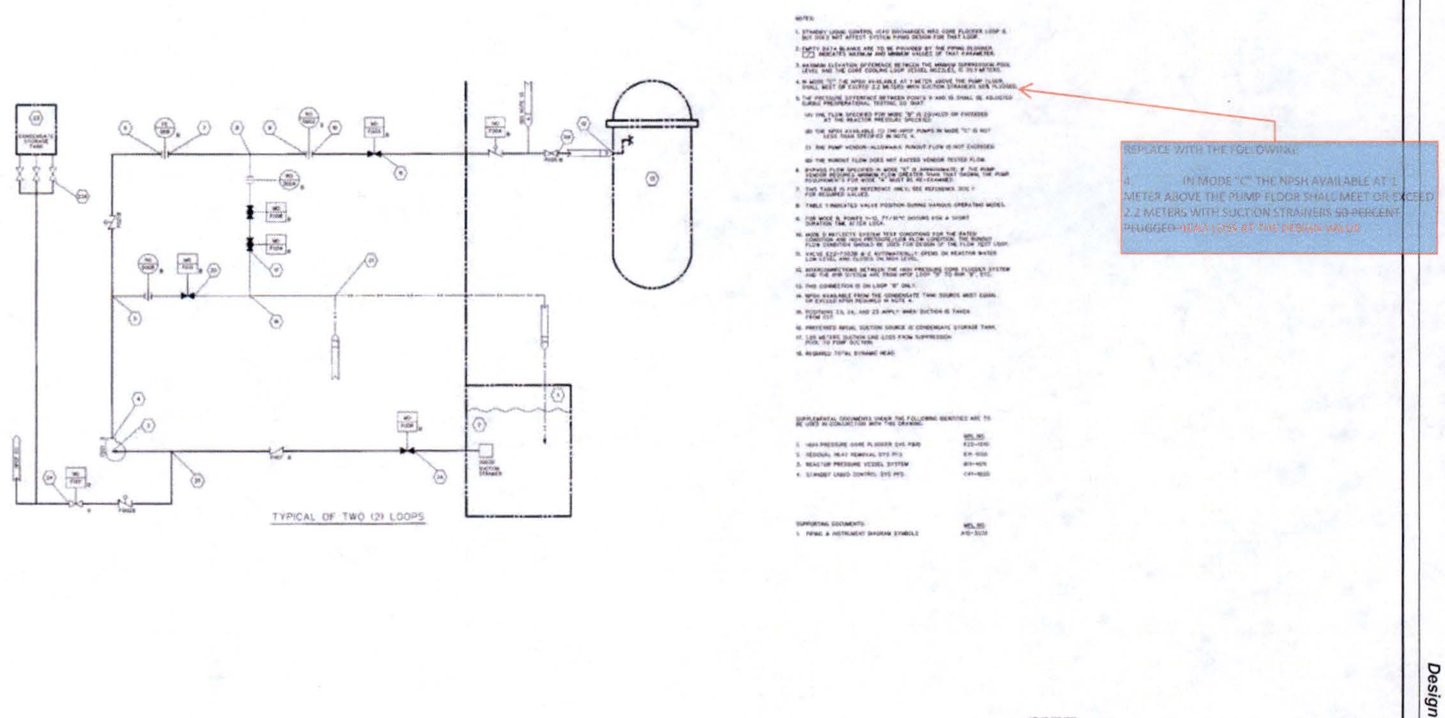



**Figure 6.3-1 High Pressure Core Flooder System PFD (Sheet 1 of 2)**