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February 17, 2012

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. Jeffrey A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF-12045

Subject: **MHI's Response to US-APWR DCD RAI No. 892-6169 Revision 3 (SRP 14.03.03)**

Reference: 1) "Request for Additional Information No. 892-6169 Revision 3, SRP Section 14.03.03 – Piping Systems and Components – Inspections, Tests, Analyses, and Acceptance Criteria - Application Section: 14.3.3", dated January 18, 2012.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Response to Request for Additional Information No. 892-6169 Revision 3."

Enclosure 1 contains the response to the questions that are contained within Reference 1.

Please contact Mr. Joseph Tapia, General Manager of Licensing Department, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of the submittals. His contact information is below.

Sincerely,

Y. Ogata

Yoshiki Ogata,
Director- APWR Promoting Department
Mitsubishi Heavy Industries, LTD.

Enclosure:

1. Response to Request for Additional Information No. 892-6169 Revision 3

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NRO

CC: J. A. Ciocco
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Contact Information

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Enclosure 1

UAP-HF- 12045
Docket No. 52-021

Response to Request for Additional Information No. 892-6169
Revision 3

February 2012

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

02/17/2012

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 892-6169 REVISION 3**
SRP SECTION:	**14.03.03 – PIPING SYSTEMS AND COMPONENTS INSPECTIONS, TESTS, ANALYSES, AND ACCEPTANCE CRITERIA**
APPLICATION SECTION:	**14.03.03**
DATE OF RAI ISSUE:	**1/18/2012**

QUESTION NO.: 14.03.03-26

In Rev. 3 of US-APWR DCD Tier 1, Table 2.3.2, ITAAC Items 4) and 5), both as-designed and as-built pipe break hazard analysis reports are addressed. However, in DCD Tier 2 Chapter 14, Section 14.3.4.3 under Generic ITAAC, only as-built pipe break analysis report is discussed. The applicant is requested to correct this inconsistency between DCD Tier 1, Table 2.3.2 and DCD Tier 2 Section 14.3.4.3. Specifically, Bullet 3 of DCD Tier 2 Section 14.3.4.3 should address both as-designed and as-built pipe break hazard analysis reports.

ANSWER:

Tier 2 Section 14.3.4.3, "Generic ITAAC," bullets are revised to provide consistency with Tier 1 Table 2.3-2 ITAAC by addressing both as-designed and as-built pipe break hazard analyses.

The ASME Code Section III PSC reconciliation and LBB evaluation description is moved from "Generic ITAAC" to "ITAAC for specific systems" to be consistent with Tier 1 where they are found in Tier 1 Sections 2.4, 2.7 and 2.11, rather than Tier 1 Section 2.3 on generic ITAAC.

Impact on DCD

Tier 2 Section 14.3.4.3 is revised as discussed in the answer above and shown on the attached markup. (See Attachment-1.)

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Technical / Topical Reports

There is no impact on the Technical / Topical Reports.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

02/17/2012

US-APWR Design Certification

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RAI NO.:	**NO. 892-6169 REVISION 3**
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APPLICATION SECTION:	**14.03.03**
DATE OF RAI ISSUE:	**1/18/2012**

QUESTION NO.: 14.03.03-27

In Rev. 3 of the US-APWR DCD, the applicant did not provide the complete piping design information because piping design was not completed when the application was submitted. As defined in SECY-92-053, design acceptance critieria (DAC) are "a set of prescribed limits, parameters, procedures, and attributes upon which the NRC relies, in a limited number of technical areas, in making a final safety determination to support a design certification." RG 1.206 identified four areas: radiation protection, piping, instrumentation and control, and human factor engineering, in which the use of DAC is acceptable because 1) providing detailed design information is not desirable for applicants using technologies that change so rapidly that the design may have become obsolete between the time the agency certifies the design and the time a plant is eventually built (e.g., digital I&C systems and HFE); and 2) completing the final design is impractical given the unavailability of sufficient as-built or as-procured information (e.g., in the radiation shielding and piping areas).
The applicant is requested to make clear, in the US-APWR DCD, whether the piping design including the pipe break hazard analysis will be completed before design certification, or the use of DAC will be utilized. In addition, if DAC is used, a section needs to be included in DCD Tier 2 to discuss the possible DAC closure processes, as well as the COL items in which future COL applicants are responsible for.

ANSWER:

As stated in UAP-HF-11135, "Revised Design Completion Plan for US-APWR Piping Systems and Components," (Accession Number: ML11136A234), MHI uses DAC for PSC design. This includes stress analysis, environmental fatigue analysis, LBB evaluation and pipe break hazard analysis. Design information will be available after design certification.

The US-APWR DCD is revised to describe a DAC closure process that is consistent with the NEI 08-01 endorsed by RG 1.215.

A new COL item is added in DCD Tier 2 Section 14.3.4.3 to provide a DAC ITAAC closure schedule and to state whether the standard DAC closure process will be used as described.

Tier 1 Table 2.3-2 ITAAC wording is revised to remove unnecessary exceptions from Table 2.3-2 ITAAC #1.a, b, #2.a, b and #3. Tier 1 wording is changed to align with the plan described in UAP-HF-11135 (ML11136A234) as well as the revised Tier 2 Section 14.3.4.3 and Appendix 14B.

Impact on DCD

Tier 1 Table 2.3-2 and Tier 2 Section 14.3.4.3 and Appendix 14B are revised and added, respectively, as described in the answer above and shown on the attached markups.(See Attachment-1.)

Impact on R-COLA

R-COLA will be affected by adding new COL item "COL 14.3(4)".

Impact on S-COLA

S-COLA will be affected by adding new COL item "COL 14.3(4)".

Impact on PRA

There is no impact on the PRA.

Impact on Technical / Topical Reports

There is no impact on the Technical / Topical Reports.

This completes MHI's response to the NRC's question.

Lines identified below for systems listed in Table 2.3-3 meet the LBB criteria or an evaluation is performed of the dynamic effects of a rupture of the line:

- Reactor coolant piping
- Reactor coolant piping branch piping with nominal diameter of 6 inches or larger, except for the steam piping for the pressurizer safety valves and power operated relief valves
- Main steam pipe in PCCV

Component Stress Analysis

Components, component supports, and core support structures are analyzed and designed to the requirements of the ASME Code Section III, based on Code classification and ASME Service Level. The requirements of the ASME Code Section III, Subsections NB (Class 1), NC (Class 2), or ND (Class 3) code are used in component stress analysis. Stress analysis of components considers design basis loads and load combinations applicable to each system. Component supports and their attachments for ASME Code Class 1, 2, and 3 components are designed in accordance with ASME Code Section III, Subsection NF up to the interface with a building's structure, with jurisdictional boundaries as defined by Subsection NF. The requirements of the ASME Code Section III, Subsection NG are used in core support structure stress analysis. Additionally, ASME Code Section III, Class 1 pressure boundary components are subject to fatigue usage evaluations over the design life of the plant.

Table 2.3-3 lists systems that have ASME Code Section III, Class 1, 2 and 3 components, component supports and core support structures.

1.a The ASME Code Section III, Class 1 piping systems and components (PSC), for systems identified in Table 2.3-3, are designed to retain their pressure integrity and functional capability under internal design and operating pressures and design basis loads.

1.b The usage factors for ASME Code Section III Class 1 PSC, for systems identified in Table 2.3-3, are evaluated for both air and reactor coolant environments.

2.a ~~Reactor coolant piping, pressurizer surge line piping and main steam piping in the PCCV, for systems identified in Table 2.3-3, are designed in accordance with the LBB method.~~Deleted. DCD_14.03.03-27

2.b Portions of the high-energy piping, for systems identified in Table 2.3-3, ~~except reactor coolant piping, pressurizer surge line piping and main steam piping in the PCCV,~~ are designed in accordance with the LBB method.

3. The ASME Code Section III, Class 2 and 3 PSC, for systems identified in Table 2.3-3, are designed to retain their pressure integrity and functional capability under internal design and operating pressures and design basis loads.

4. Safety-related SSCs are designed to be protected against or qualified to withstand the dynamic and environmental effects associated with analyses of postulated failures in high-energy piping and moderate-energy piping systems identified in Table 2.3-1 so that

Table 2.3-2 Piping Systems and Components Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 1 of 3)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria	
1.a The ASME Code Section III, Class 1 piping systems and components (PSC), for systems identified in Table 2.3-3, are designed to retain their pressure integrity and functional capability under internal design and operating pressures and design basis loads.	1.a.i An inspection of the stress report(s) for the ASME Code, Section III, Class 1 ~~piping and valves~~PSC, for systems identified in Table 2.3-3, will be performed.	1.a.i The stress report(s) exist and conclude that the design of the ASME Code Section III Class 1 ~~piping and valves~~PSC, for systems identified in Table 2.3-3, comply with the requirements of the ASME Code Section III.	DCD_14.03.03-27
	1.a.ii ~~An inspection of the stress report for the ASME Code, Section III, Class 1 components, except valves, for systems identified in Table 2.3-3, will be performed.~~Deleted.	1.a.ii ~~The stress report(s) exist and conclude that the design of the ASME Code Section III Class 1 components, except valves, for systems identified in Table 2.3-3, comply with the requirements of the ASME Code Section III.~~Deleted	DCD_14.03.03-27
1.b The usage factors for ASME Code Section III Class 1 PSC , for systems identified in Table 2.3-3, are evaluated for both air and reactor coolant environments.	1.b.i An analysis of the ASME Code, Section III, Class 1 ~~piping and valves~~PSC, for systems identified in Table 2.3-3, ~~except for reactor coolant loop piping and pressurizer surge line piping,~~ will be performed.	1.b.i Report(s) exist and conclude that the usage factors for ASME Code Section III Class 1 ~~piping and valves~~PSC, for systems identified in Table 2.3-3, ~~except reactor coolant piping and pressurizer surge line piping,~~ are evaluated for air and reactor coolant environments.	DCD_14.03.03-27
	1.b.ii ~~An analysis of the ASME Code, Section III, Class 1 components and reactor coolant piping and pressurizer surge line piping for, systems identified in Table 2.3-3, will be performed.~~Deleted.	1.b.ii ~~Report(s) exist and conclude that the usage factors for ASME Code Section III Class 1 components and reactor coolant piping and pressurizer surge line piping, for systems identified in Table 2.3-3, are evaluated for air and reactor coolant environments.~~Deleted.	DCD_14.03.03-27

Table 2.3-2 Piping Systems and Components Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 2 of 3)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria	
2.a ~~Reactor coolant piping, pressurizer surge line piping and main steam piping in the PCCV, for systems identified in Table 2.3-3, are designed in accordance with the LBB method.~~ Deleted.	2.a ~~A LBB analysis using the LBB method will be performed for each reactor coolant piping, pressurizer surge line piping and main steam piping in the PCCV, for systems identified in Table 2.3-3.~~ Deleted.	2.a ~~The results of the LBB analysis for each reactor coolant piping, pressurizer surge line piping and main steam piping in the PCCV, for systems identified in Table 2.3-3, conclude that the stress values conform to the LBB acceptance criteria using the LBB assumptions.~~ Deleted.	DCD_14.03.03-27
2.b Portions of the high-energy piping, for systems identified in Table 2.3-3, ~~except reactor coolant piping, pressurizer surge line piping and main steam piping in the PCCV,~~ are designed in accordance with the LBB method.	2.b A LBB analysis using the LBB method will be performed for portions of the high-energy piping, for systems identified in Table 2.3-3~~, except reactor coolant piping, pressurizer surge line piping and main steam piping in the PCCV~~.	2.b The results of the LBB analysis for portions of the high-energy piping, for systems identified in Table 2.3-3, ~~except reactor coolant piping, pressurizer surge line piping and main steam piping in the PCCV~~ conclude that the stress values conform to the LBB acceptance criteria using the LBB assumptions.	DCD_14.03.03-27
3. The ASME Code Section III, Class 2 and 3 PSC, for systems identified in Table 2.3-3, are designed to retain their pressure integrity and functional capability under internal design and operating pressures and design basis loads.	3.i An inspection of the stress report(s) for the ASME Code, Section III, Class 2 and 3 PSC, for systems identified in Table 2.3-3, ~~except for the accumulators, main steam piping in the PCCV, safety injection pumps, CS/RHR pumps, charging pumps, emergency feedwater pumps (motor driven), emergency feedwater pumps (turbine driven) and component cooling water pumps,~~ will be performed.	3.i The stress report(s) exist and conclude that the design of the ASME Code Section III Class 2 and 3 PSC, for systems identified in Table 2.3-3, ~~except for the accumulators, main steam piping in the PCCV, safety injection pumps, CS/RHR pump, charging pumps, emergency feedwater pumps (motor driven), emergency feedwater pumps (turbine driven) and component cooling water pumps,~~ comply with the requirements of ASME Code Section III.	DCD_14.03.03-27
	3.ii ~~An inspection of the stress report(s) for the accumulators, main steam piping in the PCCV, safety injection pumps, CS/RHR pumps, charging pumps, emergency feedwater pumps (motor driven), emergency feedwater pumps (turbine driven) and component cooling water pumps will be performed.~~ Deleted.	3.ii ~~The stress report(s) exist and conclude that the design of the accumulators, main steam piping in the PCCV, safety injection pump, CS/RHR pumps, charging pumps, emergency feedwater pumps (motor driven), emergency feedwater pumps (turbine driven) and component cooling water pumps comply with the requirements of ASME Code Section III.~~ Deleted.	

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (Sheet 32 of 36)

COL ITEM NO.	COL ITEM
COL 14.2(11)	*The COL holder for the first plant is to perform the first plant only tests and prototype test. For subsequent plants, either these tests are performed, or the COL Applicant provides a justification that the results of the first-plant only tests are applicable to the subsequent plant and are not required to be repeated. [14.2.8]*
COL 14.2(12)	*The COL holder makes available approved test procedures for satisfying testing requirements described in Section 14.2 to the NRC approximately 60 days prior to their intended use. [14.2.3, 14.2.11, 14.2.12.1]*
COL 14.3(1)	*The COL Applicant provides the ITAAC for the site specific portion of the plant systems specified in Subsection 14.3.5, Interface Requirements. [14.3.4.6,14.3.4.7]*
COL 14.3(2)	*The COL Applicant provides proposed ITAAC for the facility's emergency planning not addressed in the DCD in accordance with RG 1.206 (Reference 14.3-1) as appropriate. [14.3.4.10]*
COL 14.3(3)	*The COL Applicant provides ITAAC for the facility's physical security hardware not addressed in the DCD, in accordance with RG 1.206 (Reference 14.3-1) as appropriate, and provides abstracts describing the specific inspections, tests and analysis for the facility's physical security hardware ITAAC not addressed in the DCD. [14.3.4.12]*
COL 14.3(4)	*The COL Applicant provides a DAC closure schedule and declares whether the standard approach is used for closure of DAC ITAAC, as described by Appendix 14.B.1 [14.3.4.3]* DCD_14.03.03-27
COL 15.0(1)	*In the COLA, if the site-specific χ/Q values exceed DCD χ/Q values, then the COL Applicant is to demonstrate how the dose reference values in 10 CFR 50.34 and 10 CFR 52.79 and the control room dose limits in 10 CFR 50, Appendix A, General Design Criterion 19 are met for affected events using site-specific χ/Q values. Additionally, the Technical Support Center (TSC) dose should be evaluated against the habitability requirements in Paragraph IV.E. 8 to 10 CFR Part 50, Appendix E, and 10 CFR 50.47(b)(8) and (b)(11).*
COL 16.1(1)	*Adoption of RMTS is to be confirmed and the relevant descriptions are to be fixed.*
COL 16.1(2)	*Adoption of SFCP is to be confirmed and the relevant descriptions are to be fixed.*
COL 16.1(3)	*Deleted*
COL 16.1_3.3.1(1)	*Deleted*
COL 16.1_3.3.2(1)	*Deleted*
COL 16.1_3.3.5(1)	*The time delay values in SR 3.3.5.3 are to be confirmed based on the plant specific transmission system performance.*
COL 16.1_3.3.6(1)	*Deleted*
COL 16.1_3.4.17(1)	*Deleted*

ACRONYMS AND ABBREVIATIONS

ABVS	auxiliary building ventilation system
ac	alternating current
AHU	air handling unit
ALARA	as low as reasonably achievable
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
ARMS	area radiation monitoring system
ASSS	auxiliary steam supply system
ASTM	American Society for Testing and Materials
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
BTP	branch technical position
C/V	containment vessel
CAGS	compressed air and gas system
CCW	component cooling water
CCWS	component cooling water system
CFR	Code of Federal Regulations
CFS	condensate and feedwater system
CHS	containment hydrogen monitoring and control system
CIS	containment isolation system
CIV	containment isolation valve
COL	Combined License
COLA	Combined License Application
CPS	condensate polishing system
CRDM	control rod drive mechanism
CRDS	control rod drive system
CRE	control room envelope
CS	containment spray
CSS	containment spray system
CSF	condensate storage facilities
CVCS	chemical and volume control system
CVVS	containment ventilation system
CWS	circulating water system
DAC	design acceptance criteria DCD_14.03.03-27
DAS	diverse actuation system
dc	direct current
DCD	Design Control Document

ACRONYMS AND ABBREVIATIONS (CONTINUED)

MFRV	main feedwater regulatory valve
MHI	Mitsubishi Heavy Industries, Ltd.
MSIV	main steam isolation valve
MSRVBV	main steam relief valve block valve
MSS	main steam supply system
MSSV	main steam safety valve
NaTB	sodium tetraborate decahydrate
NIS	nuclear instrumentation system
NPS	nominal pipe size
NRC	U.S. Nuclear Regulatory Commission
NRCA	non-radiological controlled area
NSSS	nuclear steam supply system
OHLHS	overhead heavy load handling system
PCCV	prestressed concrete containment vessel
PCMS	plant control and monitoring system
PERMS	process effluent radiation monitoring and sampling system
PMWS	primary makeup water system
PRA	probabilistic risk assessment
PSMS	protection and safety monitoring system
PSS	process and post-accident sampling system
PSWS	potable and sanitary water system
PWR	pressurized-water reactor
RCA	radiological controlled area
RCCA	rod cluster control assembly
RCDT	reactor coolant drain tank
RCP	reactor coolant pump
RCS	reactor coolant system
RG	Regulatory Guide
RHRS	residual heat removal system
RO	reactor operator
RPS	reactor protection system
RSS	remote shutdown system
RTS	reactor trip system
RTD	resistance temperature detector
RWSP	refueling water storage pit
SBO	station blackout
SCIS	secondary side chemical injection system

DCD_14.03.03-27

- Seismic loads (GDC 2)
- Flood, wind, and tornado (GDC 2)
- Rain and snow (GDC 2)
- Pipe rupture (GDC 4)
- Codes and standards (GDC 1, "Quality Standards and Records")
- Containment integrity (GDC 16, "Containment Design")
- As-built reconciliation

14.3.4.3 ITAAC for Piping Systems and Components

Section 2.3 of Tier 1, which addresses piping systems and components, is prepared in accordance with the guidance in RG 1.206 (Reference 14.3-1), SRP 14.3 (Reference 14.3-2), and SRP 14.3.3 (Reference 14.3-7). The ITAAC in this section address piping system design and components, along with dynamic qualification, welding, fasteners, and safety classification of SSCs, covering matters such as the following:

- Piping design criteria, structural integrity, and functional capability of safety-related and risk-significant piping
- ASME Code Class 1, 2, and 3 piping and supports
- Buried piping and instrumentation lines
- Interaction of non-seismic piping with seismic Category I piping
- Any safety-related and risk-significant piping designed to industry standards other than the ASME Code
- Analysis methods, modeling techniques, pipe stress analysis criteria, pipe support design criteria, high-energy line break criteria, and the leak before break (LBB) approach, as applicable

Generic ITAAC – which apply to all ASME Class 1, 2, and 3 piping systems and high-energy and moderate-energy piping systems – provide for as follows: DCD_14.03-03-26

- ~~Requiring the existence of a design report to assure that the ASME Code Class 1 piping system and components are designed to retain their pressure boundary integrity and functional capability under internal design and operating pressures and design basis loads.~~
- Requiring the existence of a~~n ASME Code certified~~ stress report to assure that the ~~as-built~~ ASME Code Class 1, 2, and 3 piping systems and components are

designed to retain their pressure boundary integrity and functional capability under internal design and operating pressures and design-basis loads.

- Requiring the existence of an as-designed pipe break hazard analysis report that documents dynamic effects analysis results for high-energy piping systems and environmental effects analysis for the high-energy and moderate-energy piping systems. DCD_14.03.03-26

- Requiring the existence of a pipe break analysis report that documents that ~~the~~ as-built high-energy piping and moderate-energy piping systems are installed in accordance with the as-designed pipe break hazard analysis~~SSCs that are required to be functional during and following a safe shutdown earthquake have adequate high-energy pipe break mitigation features~~.

- Requiring the existence of an LBB evaluation report that documents that ~~the as-built~~ piping stress values~~and piping materials~~ comply with the LBB acceptance criteria for the systems to which LBB is applied.

- ~~Requiring the existence of a report that documents the results of an as-built reconciliation confirming that the piping systems are built in accordance with the ASME Code certified stress report.~~ DCD_14.03.03-26

ITAAC for specific systems typically verify the following:

- Reconciliation confirming that the as-built piping systems and components are built in accordance with the ASME Code Section III design report DCD_14.03.03-26

- Existence of an LBB evaluation report that documents that the as-built piping and piping materials comply with the as-designed LBB acceptance criteria

- Requirements such as piping and component safety classification

- Fabrication, especially pressure-boundary weld quality

- Hydrostatic testing

- Equipment seismic and dynamic qualification

- Design qualification of valves

Such ITAAC also address the verification of applicable dynamic qualification records and vendor test records, as well as performance of appropriate in-situ tests. All of these matters are addressed for safety-related systems, and appropriate ones are addressed for non-safety systems.

These ITAAC for the individual systems are covered in each plant system ITAAC such as Sections 2.4, 2.7 and 2.11 of Tier 1.

Design acceptance criteria (DAC) are used for piping system and component design. The DAC closure process is described in Appendix 14B. The COL Applicant provides a DAC closure schedule and declares whether the standard approach is used for closure of DAC ITAAC, as described by Appendix 14.B.1. DCD_14.03.03-27

14.3.4.4 ITAAC for Reactor Systems

Section 2.4 of Tier 1, which addresses reactor systems identified in Table 14.3-3, is prepared in accordance with the guidance in RG 1.206 (Reference 14.3-1), SRP 14.3 (Reference 14.3-2), and SRP 14.3.4 (Reference 14.3-8). ITAAC for reactor systems are provided to verify the following:

- Important input parameters used in the transient and accident analyses for the facility design
- Net positive suction head for key pumps
- The design pressures of the piping systems that interface with the reactor coolant boundary to validate intersystem LOCA analyses

ITAAC are also specified to verify the following top-level design aspects of reactor systems:

- Functional arrangement
- Seismic and ASME Code classification
- Weld quality and pressure boundary integrity
- Valve qualification and operation
- Controls, alarms, and displays
- Logic and interlocks
- Equipment qualification for harsh environments
- Interface requirements with other systems
- Numeric performance values
- Class 1E electrical power sources and divisions, if applicable
- System operation in various modes

ITAAC for the reactor system fluid systems follow NRC guidelines for fluid systems ITAAC in Appendix C.II.1-A of RG 1.206 (Reference 14.3-1), including those for figure content and ITAAC style.

Section 2.14, of Tier 1 provides a general description of the preoperational and startup test programs and the major program documents that define how the initial test program is to be conducted and controlled. This section also describes the key elements of the initial test program.

No ITAAC are necessary for the initial test program because all ITAAC are to be completed prior to fuel load.

14.3.5 Chapter 3 of Tier 1, Interface Requirements

Chapter 3 of Tier 1 focuses on the interface requirements of the safety-significant design attributes. The interface requirements in Chapter 3 of Tier 1 define the safety-significant design attributes and performance characteristics that assure that the site-specific portion of the design is in conformance with the certified design. The site-specific portions of the design are those portions of the design that are dependent on characteristics of the site.

Chapter 3 of Tier 1 also identifies the scope of the design to be certified by specifying the systems that are completely or partially out of scope of the certified design. Thus, interface requirements are defined for: (a) systems that are entirely outside the scope of the design, and (b) the out-of-scope portions of those systems that are only partially within the scope of the standard design based on the above methodology.

14.3.6 Combined License Information

COL 14.3(1) *The COL Applicant provides the ITAAC for the site specific portion of the plant systems specified in Subsection 14.3.5, Interface Requirements. [14.3.4.6, 14.3.4.7]*

COL 14.3(2) *The COL Applicant provides ITAAC for the facility's emergency planning not addressed in the DCD in accordance with RG 1.206 (Reference 14.3-1) as appropriate. [14.3.4.10]*

COL 14.3(3) *The COL Applicant provides ITAAC for the facility's physical security hardware not addressed in the DCD, in accordance with RG 1.206 (Reference 14.3-1) as appropriate, and provides abstracts describing the specific inspections, tests and analysis for the facility's physical security hardware ITAAC not addressed in the DCD. [14.3.4.12]*

COL 14.3(4) *The COL Applicant provides a DAC closure schedule and declares whether the standard approach is used for closure of DAC ITAAC, as described by Appendix 14.B.1 [14.3.4.3]* DCD_14.03. 03-27

14.3.7 References

14.3-1 Combined License Applications for Nuclear Power Plants (LWR Edition). Regulatory Guide 1.206, U.S. Nuclear Regulatory Commission, Washington, DC, June 2007.

14.3-2 'Inspections, Tests, Analyses, and Acceptance Criteria,' "Initial Test Program and ITAAC – Design Certification," Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants. NUREG-0800, SRP 14.3,

APPENDIX 14B

DCD_14.03.03-27

14B Design Acceptance Criteria ITAAC Closure Process

US-APWR standard design uses DAC to specify the limits, parameters, procedures, and attributes associated with final design and analysis of piping systems and components. These DAC are identified in DCD Tier 1 Section 2.3 and provided with ITAAC to verify their completion prior to initial fuel load.

DAC ITAAC will be closed using the process described in this appendix. Following closure of DAC ITAAC, ITAAC for related as-built SSC will be closed to verify that their respective principal performance characteristics and safety functions conform to the certified design. RG 1.206 (Reference 14.3.B-1), "Combined License Applications for Nuclear Power Plants (LWR Edition)," Section C.III.5 provides DAC ITAAC closure guidance.

14B.1 Design Acceptance Criteria ITAAC Closure Options

There are three options available to close DAC ITAAC. Design information used to close DAC ITAAC represents a level of detail similar to that which would have been provided during design certification review if DAC ITAAC had not been used. The three options for DAC ITAAC closure are:

- Closure through amendment of design certification rule - A design certification rule amendment request is submitted to the NRC to provide the design and analysis information needed to close the DAC ITAAC and the DAC ITAAC are deleted from the DCD. ITAAC for as-built SSCs will remain or be modified, as appropriate, to demonstrate that the as-built facility conforms to the final design and analysis information.

- Closure through the COLA review process - A COL application contains the required design and analysis information needed to close the DAC ITAAC. ITAAC for as-built SSCs will remain or be modified, as appropriate, to demonstrate that the as-built facility conforms to the final design and analysis information.

- Closure after COL issuance - The NRC issues a COL with DAC ITAAC still open and inspects DAC ITAAC closure as part of the construction inspection process. DAC ITAAC closure is accomplished using the normal ITAAC closure process.

Regarding the first option, this method resolves DAC with finality for all COL applicants that subsequently reference the amended standard design.

The second or third option may be applied only by the first licensee following completion of the required design and analysis information needed to close DAC ITAAC. Subsequent licensees may use the standard plant design and analysis information approved for closure of DAC ITAAC by the first licensee. This does not include DAC that are dependent upon site-specific parameters. As discussed by RG 1.206 Section C.III.5 (Reference 14.3.B-1), the licensee and NRC may use the design centered review approach to close DAC ITAAC for subsequent licensees.

Topical reports may be submitted to the NRC to support DAC ITAAC closure using any of the three options. The NRC may issue a safety evaluation in conjunction with a closure letter or inspection report conclusion that DAC ITAAC acceptance criteria have been satisfied.This allows subsequent COL applicants or licensees to reference NRC closure documents to close DAC ITAAC. DCD_14.03.03-27

14B.2 Design Acceptance Criteria ITAAC for Piping Systems and Components Design

Piping systems and components DAC ITAAC verify final design at a level of detail adequate for procurement and construction. This includes stress analyses of piping systems and components listed in Tier 1 Table 2.3-3, environmental fatigue analyses of piping systems and components listed in Tier 1 Table 2.3-3, leak-before-break (LBB) analyses of piping systems and components listed in Tier 1 Table 2.3-3, and pipe break hazard analyses for safety-related SSC. These are verified by separate DAC ITAAC listed in Tier 1 Table 2.3-2. Piping systems and components design and analysis and other information used to close DAC ITAAC will be made available for NRC review, inspection, and audit as soon as reasonably practicable. Information will be made available to the NRC throughout the process to facilitate review, inspection, and audit and to allow early identification and timely resolution of their concerns.

ASME Code prescribes procedures and requirements for completing piping systems and components design and analysis. Stress reports, which consist of modeling, methodology, sizing calculation, analysis and evaluations, are prepared in accordance with ASME Code, Section III, Sub-article NCA-3550 (Reference 14.3B-2), but not necessarily certified by the registered professional engineer, to ensure that the appropriate code design requirements for each ASME Code Section III class have been met. Design information, including stress reports, will be provided to the NRC for review, inspection, and audit as the information becomes available, in order to ensure that the closure of the DAC ITAAC can be completed in a timely manner after the DAC ITAAC closure notification letter is submitted.

Reconciliation of applicable as-built safety-related piping systems and components is verified by as-built ITAAC to demonstrate that the as-designed information is reconciled with as-built information. Reconciliation analysis results will be documented and made available for NRC inspection or audit.

ASME Code Section III Class 1 (Reference 14.3.B-2) piping and components are evaluated for fatigue effects from various thermal and pressure transients and other cyclic events, including earthquakes and thermal stratification. ASME Code Section III Class 1, Subsection NB-3653 (Reference 14.3.B-2) provides criteria that are to be used for all ASME Code Section III Class 1 piping and components greater than 1 inch nominal pipe size (NPS). Class 1 piping and components of one inch NPS and smaller are analyzed using Subsection NC rules. To close associated DAC ITAAC, demonstrate that Class 1 piping systems and components fatigue usage factor evaluation was performed considering both air and reactor coolant environment effects on fatigue in accordance with RG 1.207 guidance (Reference 14.3.B-3). This evaluation will be documented and made available for NRC review, inspection, and audit as the information becomes available.

Leak-before-break (LBB) evaluation uses the same methodology that is used for development of the bounding analysis curves (BAC) in accordance with DCD Tier 2 Appendix 3B. LBB applies to austenitic stainless steel piping used in the reactor coolant pressure boundary (RCPB) and carbon steel piping used in the main steam system (MSS) inside the prestressed concrete containment vessel (PCCV). DAC ITAAC verify that stress values for the applicable RCPB and MSS piping systems conform to the LBB acceptance criteria. Evaluation requirements are discussed in DCD Tier 2 Subsection 3.6.3. LBB evaluation results will be documented and made available for NRC review, inspection, and audit as the information becomes available. DCD_14.03.03-27

ITAAC for as-built piping systems and components verify that base metal welds, weldments, and safe end materials and specification for piping to which LBB applies are identified and that as-built material and material specifications for piping satisfy the bounding LBB analysis. To close these as-built ITAAC, provide an LBB evaluation report to confirm that the final bounding LBB analysis considers plant-specific and generic degradation mechanisms that affect as-built piping systems, confirm that the actual plant-specific stress analysis is based on final as-built plant piping layout and material properties and welds, and confirm toughness (J-R curves), tensile strength (stress-strain curves), yield and ultimate strength, and welding process and methods actually used.

To close pipe break hazard analysis DAC ITAAC, use the pipe break hazard analysis report to verify that dynamic effect analyses were performed for high-energy piping systems and environmental effect analyses were performed for both high-energy and moderate-energy piping systems. The pipe break hazard analysis report confirms that criteria used to postulate pipe breaks, analytical methods used to analyze pipe breaks, and the method used to determine adequacy of pipe break analysis results are appropriate. The pipe break hazard analysis report provides assurance that high-energy and moderate-energy pipe break analyses are complete and that, for each postulated piping failure, the reactor can be shut down safely and maintained in a safe, cold shutdown condition without offsite power. Report content is discussed in DCD Tier 2 Subsection 3.6.2.6.

Following NRC review of the pipe break hazard analysis report and supporting analyses, the NRC may decide to review design features intended to mitigate pipe break consequences. The appropriate information will be available to the NRC so that their issues can be identified and resolved prior to as-built installation of the protective features. Upon completion of protective feature installation, associated as-built ITAAC will verify that as-built principal performance characteristics and safety functions of protective features exist and are constructed as designed.

Piping systems and components design and analyses approved for the first standard US-APWR plant will be available for use by subsequent plants under the "one issue, one review, one position" approach to closure. As-built ITAAC will be closed following completion of DAC ITAAC and installation of piping systems and components and pipe break mitigation features.

14B.3 References

DCD_14.03.03-27

14B-1 Combined License Applications for Nuclear Power Plants (LWR Edition). Regulatory Guide 1.206, U.S. Nuclear Regulatory Commission, Washington, DC, June 2007.

14B-2 Rules for Construction of Nuclear Facility Components, American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code Section III, The American Society Of Mechanical Engineers.

14B-3 Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components due to the Effects of the Light-Water Reactor Environment for New Reactors, Regulatory Guide 1.207, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.