of this test are applied for verification of the analysis methodology in the FIV assessment program.

Further information of the test is reported in Reference 1.5-2.

1.5.2.1.2 Reactor Vessel Lower Plenum 1/7 Scale Model Flow Test

The US-APWR reactor internals has been designed for a core consisting of 257 fuel assemblies of 14 ft length. The design of the reactor internals, which includes integrated lower core support system, has been proven through considerable operating experiences as the standard design of 14 ft core reactors in USA. Although there is no operating experience with 257 fuel assemblies, and the structure in the vessel lower plenum is simplified, the geometries of the lower plenum are similar to those of current 4-loop PWRs. Thus hydraulic and vibration characteristics are also similar to those of current PWR.

A reactor vessel lower plenum 1/7 scale model flow test was performed to obtain the following data related to the lower plenum design configuration and to confirm the design:

- Hydraulic characteristics such as core inlet flow distribution
- FIV of the structures in the lower plenum

The report on the test is given in Reference 1.5-3.

1.5.2.1.3 Neutron Reflector Reflooding Confirmation Test

The neutron reflector (NR) reflooding test is a separate test focused on the thermal hydraulic phenomena of a single neutron reflector flow hole with 1/1 scale in a large break loss of coolant accident (LBLOCA).

The test and calculation results were presented as a technical report that provided the additional information to the topical report (Reference 1.5-4), describing the code applicability for the LBLOCA.

1.5.2.2 Digital Instrumentation and Control System

Verification and validation (V&V) of the safety digital Instrumentation and control (I&C) system and the human ~~-~~ system interface (HSI) system have been ~~already conducted~~performed in Japan, and detailed descriptions of those tests given as technical and topical reports in Reference 1.5-5 through 1.5-8. As described in these reports, the safety I&C system consists of a digital platform, four train redundancy and safety ~~video~~visual display units. The I&C system has a diverse actuation system (DAS) as a diversity of a digital platform. MIC-04-01-00013

MIC-04-01-00013

The platform and integrated I&C and HSI systems have been qualified for service as safety equipment according to the guidelines of IEEE 7-4.3.2 and NUREG-0711 guidelines.

~~In order to evaluate whether the US-APWR design conforms to human factor engineering (HFE) design principles and enables plant personnel to successfully perform their tasks~~ MIC-04-01-00013

~~to achieve plant safety and other operational goals based on the V&V methodology, the HSI system V&V for plant personnel in USA was conducted in 2008 and the summary results were issued as a technical report in Reference 1.5-10.~~ MIC-04-01-0 0013

1.5.2.3 Gas Turbine Generator

The Class 1E gas turbine generator (Class 1E GTG) is used as emergency power source for the US-APWR.

A Class 1E GTG has not been used in conventional PWR plants for emergency power source. However, Class 1E GTG has some advantages compared with diesel generators from reliability and maintenance points of view. A Class 1E GTG qualification program is planned. This Class 1E qualification program consists of the following items:

- Evaluation of basic specification (electrical and mechanical)
- Evaluation of reliability
- Evaluation of seismic capability
- Verification of compliance with codes, regulations and standards
- Testing in accordance with IEEE 387 and other regulations or standards

Detail a of this Class 1E qualification program, including schedule, are provided in the technical report MUAP-07024 (Reference 1.5-9). This qualification program was started in early 2008 and its results were submitted as a technical report in Reference 1.5-11.

1.5.3 Combined License Information

No additional information is required to be provided by a COL Applicant in connection with this section.

1.5.4 References

1.5-1 The Advanced Accumulator, MUAP-07001-P (Proprietary) and MUAP-07001-NP (Non-Proprietary), Revision 5, Mitsubishi Heavy Industries, Ltd., June 2013.

1.5-2 APWR Reactor Internals 1/5 Scale Model Flow Test Report, MUAP-07023-P (Proprietary) and MUAP-07023-NP (Non-Proprietary), Revision 2, Mitsubishi Heavy Industries, Ltd., August 2011.

1.5-3 US-APWR Reactor Vessel Lower Plenum 1/7 Scale Model Flow Test Report, MUAP-07022-P (Proprietary) and MUAP-07022-NP (Non-Proprietary), Revision 0, Mitsubishi Heavy Industries, Ltd., June 2008.

1.5-4 Large Break LOCA Code Applicability Report for US-APWR, MUAP-07011-P (Proprietary) and MUAP-07011-NP (Non-Proprietary), Revision 3, Mitsubishi Heavy Industries, Ltd., December 2011.

1.5-5 Safety I&C System Description and Design Process, MUAP-07004-P (Proprietary) and MUAP-07004-NP (Non-Proprietary), Revision 7, Mitsubishi Heavy Industries, Ltd., May 2011.

1.5-6 Safety System Digital Platform -MELTAC-, MUAP-07005-P (Proprietary) and MUAP-07005-NP (Non-Proprietary), Revision 8, Mitsubishi Heavy Industries, Ltd., July 2011.

1.5-7 Defense-in-Depth and Diversity, MUAP-07006-P-A (Proprietary) and MUAP-07006-NP-A (Non-Proprietary), Revision 2, Mitsubishi Heavy Industries, Ltd., September 2008.

1.5-8 ~~HSI~~Human-System Interface System Description ~~and HFE Process~~, MUAP-07007-P (Proprietary) and MUAP-07007-NP (Non-Proprietary), Revision ~~5~~6, Mitsubishi Heavy Industries, Ltd., ~~November 20011~~May 2014. MIC-04-01-00013

1.5-9 Qualification and Test Plan of Class 1E Gas Turbine Generator System, MUAP-07024-P (Proprietary) and MUAP-07024-NP (Non-Proprietary), Revision 3, Mitsubishi Heavy Industries, Ltd., September 2012.

1.5-10 Deleted~~Human System Interface Verification and Validation (Phase 1a), MUAP-08014-P (Proprietary) and MUAP-08014-NP (Non-Proprietary), Revision 1, Mitsubishi Heavy Industries, Ltd., May 2011.~~ MIC-04-01-00013

1.5-11 Initial Type Test Result of Class 1E Gas Turbine Generator System, MUAP-10023-P (Proprietary) and MUAP-10023-NP (Non-Proprietary), Revision ~~5~~7, Mitsubishi Heavy Industries, Ltd., December 201~~2~~3. MIC-04-01-00008

Table 1.6-1 Material Referenced as Topical Reports

Report Number(1)	Title	DCD Section Number(2)	
MUAP-07001-P MUAP-07001-NP	The Advanced Accumulator, Revision 5, June 2013	1.5.1, 6.3.2	
MUAP-07006-P-A MUAP-07006-NP-A	Defense-in-Depth and Diversity, Revision 2, September 2009	1.5.2, 7.1, 7.1.3, 7.1.4, 7.3.1, 7.5.1, 7.8, 7.8.1, 7.8.2, 7.8.3, 7.9.2, 16(B 3.3.6)	
MUAP-07007-P MUAP-07007-NP	~~HSI~~Human-System Interface System Description ~~and HFE Process~~, Revision ~~5, November 2011~~6, May 2014	1.5.2, 7.1, 7.1.1, 7.1.3, 7.5.1, 7.6.1, 7.8, 18.1.1.1, 18.7.1, 18.7.2~~18.1.1, 18.1.5, 18.2.2, 18.7.2, 18.7.3, 18.9.2, 18.10.2~~	MIC-04-01-00013
MUAP-07008-P-A MUAP-07008-NP-A	Mitsubishi Fuel Design Criteria and Methodology, Revision 2, July 2013	4.2.1, 4.2.2, 4.2.3, 4.3.1, 4.3.2, 4.4.1, 4.4.2, 4.4.4, 15.0.2, 15.4.8	
MUAP-07009-P-A MUAP-07009-NP-A	Thermal Design Methodology, Revision 0, August 2013	4.4.1, 4.4.2, 4.4.4, 15.0.2, 15.3.1, 15.3.3, 15.4.1, 15.4.3, 15.4.8	
MUAP-07010-P-A MUAP-07010-NP-A	Non-LOCA Methodology, Revision 4, July 2013	6.2.1, 15.0.2, 15.1.1, 15.1.2, 15.1.3, 15.1.4, 15.1.5, 15.2.1, 15.2.2, 15.2.6, 15.2.7, 15.2.8, 15.3.1, 15.3.3, 15.4.1, 15.4.2, 15.4.3, 15.4.8, 15.5.2, 15.6.1, 15.6.3	
MUAP-07011-P MUAP-07011-NP	Large Break LOCA Code Applicability Report for US-APWR, Revision 3, December 2011	1.5.2, 15.0.2, 15.6.5	
MUAP-07012-P-A MUAP-07012-NP-A	LOCA Mass and Energy Release Analysis Code Applicability Report for US-APWR, Revision 2, June 2008	6.2.1	
MUAP-07013-P-A MUAP-07013-NP-A	Small Break LOCA Methodology for US-APWR, Revision 2, October 2010	6.2.1, 15.0.2, 15.6.5	
MUAP-07034-P MUAP-07034-NP	FINDS: Mitsubishi PWR Fuel Assemblies Seismic Analysis Code, Revision 4, June 2013	4.2.3	
PQD-HD-19005	Quality Assurance Program (QAP) Description For Design Certification of the US-APWR, Revision ~~5~~7, ~~May~~February 201~~3~~4	8.4.2, 8.4.4, 17.3, 17.4.4, 17.4.10, 17.5, 17.5.2~~, 18.1.2, 18.1.3, 18.1.4, 18.1.5, 18.1.7, 18.10.1, 18.10.5~~	MIC-04-01-00011

NOTE(1): -P(**p**roprietary) , -NP(**n**on-**p**roprietary), -A (approved)
(2): If actual section number is indicated as x.y.z.a.b, a x.y.z level is used for the DCD Section Number. (ex. When actual section number is 6.3.2.1.2, only 6.3.2 is used in Table.

Table 1.6-2 Material Referenced as Technical Reports (Sheet 1 of 6)

Report Number(1)	Title	DCD Section Number(2)
MUAP-07004-P MUAP-07004-NP	Safety I&C System Description and Design Process, Revision 7, May 2011.	1.5.2, 7.1, 7.1.1, 7.1.2, 7.1.3, 7.1.4, 7.2.1, 7.2.2, 7.2.3,7.3.1, 7.3.3, 7.4.1, 7.4.2, 7.4.3, 7.5.1, 7.5.3, 7.6.3, 7.7.1, 7.7.2, 7.8.1, 7.8.2, 7.9.1, 7.9.2, 7.9.3, 14.2.12 16(1.1) ,16(B3.3.1), 16(B 3.3.2), 16(B 3.3.3) 16(B3.3.4), 16(B3.3.5)
MUAP-07005-P MUAP-07005-NP	Safety System Digital Platform -MELTAC-, Revision 8, July 2011.	1.5.2, 7.1, 7.1.3, 7.2.1, 7.3.1, 7.7.2, 7.8.1, 7.9, 7.9.1, 7.9.2, 7.9.3 16(1.1), 16(B3.3.1), 16(B3.3.2), 16(B3.3.3), 16(B3.3.4), 16(B3.3.5)
MUAP-07014-P MUAP-07014-NP	Defense-in-Depth and Diversity Coping Analysis, Revision 5, September 2011.	7.5.1, 7.8, 7.8.1, 7.8.3, 7.9.2 , 18.6.1, 16 (B 3.3.6)
MUAP-07015	FMEA of Control Rod Drive Mechanism Control System, Revision 0, December 2007.	4.6.2
MUAP-07016-P MUAP-07016-NP	US-APWR Fuel System Design Evaluation, Revision 3, August 2010.	4.2.1, 4.2.3
MUAP-07017	Software Program Manual, Revision 4, May 2011.	7.1.3, 7.1.4
MUAP-07018-P MUAP-07018-NP	US-APWR Fuel System Design Parameters List, Revision 0, December 2007.	4.2.2
MUAP-07019-P MUAP-07019-NP	Qualification of Nuclear Design Methodology using PARAGON/ANC, Revision 0, December 2007.	4.3.2, 4.3.3
MUAP-07020	Validation of the MHI Criticality Safety Methodology, Revision 0, December 2007.	4.3.3
MUAP-07021-P MUAP-07021-NP	US-APWR Incore Power Distribution Evaluation Methodology, Revision 0, December 2007.	4.3.2
MUAP-07022-P MUAP-07022-NP	US-APWR Reactor Vessel Lower Plenum 1/7 Scale Model Flow Test Report, Revision 0, June 2008.	1.5.2
MUAP-07023-P MUAP-07023-NP	APWR Reactor Internals 1/5 Scale Model Flow Test Report, Revision 2, August 2011.	1.5.2, 3.9.2
MUAP-07024-P MUAP-07024-NP	Qualification and Test Plan of Class 1E Gas Turbine Generator System, Revision 3, September 2012.	1.5.2, 8.3.1, 9.5.4, 16 (B3.8.1), 19.1.4
MUAP-07025-P MUAP-07025-NP	Small Break LOCA Sensitivity Analyses for US-APWR, Revision 3, March 2011.	15.6.5
MUAP-07026-P MUAP-07026-NP	Mitsubishi Reload Evaluation Methodology, Revision 3, August 2013.	15.0.0, 16 (5.6.3, B3.1.3, B3.1.8, B3.1.9)
MUAP-07027-P MUAP-07027-NP	Comprehensive Vibration Assessment Program for US-APWR Reactor Internals, Revision 3, November 2013.	3.9.2, 3.9.5

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Table 1.6-2 Material Referenced as Technical Reports (Sheet 2 of 6)

Report Number(1)	Title	DCD Section Number(2)	
[*MUAP-07028-P MUAP-07028-NP*]*	[*Probability of Missile Generation From Low Pressure Turbines, Revision 2, June 2013.*]*	3.5.1, 10.2.3	DCD_14.03.07-94
[MUAP-07029-P MUAP-07029-NP]*	[*Probabilistic Evaluation of Turbine Valve Test Frequency, Revision 3, June 2013.*]*	3.5.1, 10.2.3	DCD_14.03.07-94
MUAP-07030	US-APWR Probabilistic Risk Assessment, Revision 3, June 2011.	6.2.5, 7.1.3, 7.5.1, 17.4.7, 19.0, 19.1.4, 19.2.3 19B	
MUAP-07031-P MUAP-07031-NP	Subcompartment Analysis for US-APWR Design Confirmation, Revision 1, October 2009.	6.2.1	
MUAP-07032-P MUAP-07032-NP	Criticality Analysis for US-APWR New and Spent Fuel Storage Racks, Revision ~~1~~2, ~~December,~~March 20~~09~~14.	9.1.1	MIC-04-01-00012
MUAP-07033-P MUAP-07033-NP	Mechanical Analysis for US-APWR New and Spent Fuel Racks, Revision ~~0~~1, March~~,~~ 20~~09~~14.	9.1.2	MIC-04-01-00012
MUAP-07035	Structural Analysis for US-APWR Reactor Coolant Pump Motor Flywheel, Revision 0, December 2007.	5.4.1	
MUAP-07036	Justification for Deviations Between NUREG-1431 Revision 3.1 and US-APWR Technical Specifications, Revision 2, November 2009.	16	
MUAP-08001-P MUAP-08001-NP	US-APWR Sump Strainer Performance, Revision 8, June 2013	5.2.3, 6.2.2, 6.3.2	
MUAP-08007-P MUAP-08007-NP	Evaluation Results of US-APWR Fuel System Structural Response to Seismic and LOCA Loads, Revision 2, December 2010.	4.2.3	
MUAP-08009	US-APWR Test Program Description, Revision 1, October 2009.	1.8.1, 14.2.1, 14.2.2, 14.2.3, 14.2.4, 14.2.5, 14.2.6, 14.2.13, 14.3.4	
MUAP-08011-P MUAP-08011-NP	US-APWR Sump Debris Chemical Effects Test Results, Revision 3, June 2013.	6.2.2	
MUAP-08012-NP	US-APWR Sump Strainer Stress Report, Revision 2, June 2013.	6.2.2	
MUAP-08013-P MUAP-08013-NP	US-APWR Sump Strainer Downstream Effects, Revision 5, June 2013.	6.2.2, 6.3.2	
~~MUAP-08014-P MUAP-08014-NP~~	~~Human System Interface Verification and Validation (Phase 1a), Revision 1, May 2011.~~	~~1.5.2, 18.1.1, 18.1.5, 18.2.3, 18.2.1,18.7.2,18.10.2, 18.10.3~~	MIC-04-01-00013
MUAP-08015	US-APWR Equipment Qualification Program, Revision 1, November 2009.	1.8.1, 3.11, 3.11.2, 3.11.4, 3.11.5, 3.11.6, 3.11.7, 3D.1.7, 3D.3, 7.1.3, 7.5.1, ~~1.8.1,3.11.2,3D.3,~~ 19.1	MIC-04-01-00010 DCD_19-516S01
MUAP-09001-P MUAP-09001-NP	Summary of Design Transient, Revision 0, February 2009.	3.9.1	

Table 1.6-2 Material Referenced as Technical Reports (Sheet 3 of 6)

Report Number(1)	Title	DCD Section Number(2)	
MUAP-09002-P MUAP-09002-NP	Summary of Seismic and Accident Load Conditions for Primary Components and Piping, Revision ~~3~~4 ~~July~~November 2013.	~~3.7.2,~~ 3.9.2, 3.9.3, 3C.5	MIC-04-01-00002
MUAP-09010-P MUAP-09010-NP	Summary of Stress Analysis Results for Reactor Coolant Loop Piping, Revision 3, March 2011.	3B.3.2.4	
MUAP-09013-P MUAP-09013-NP	Summary of Stress Analysis Results for Main Steam Piping inside Containment, Revision 2, March 2011.	3B.3.2.4	
MUAP-09014-P MUAP-09014-NP	Thermal-Hydraulic Analysis for US-APWR Spent Fuel Racks, Revision ~~0~~1, ~~June~~March 20~~09~~14.	9.1.2	MIC-04-01-00012
MUAP-09016	US-APWR Reactor Vessel Pressure and Temperature Limits Report, Revision 3, February 2013.	5.3.2, 16 (5.6.4)	
MUAP-09017-P MUAP-09017-NP	Justification for 20 Years Inspection Interval for Reactor Coolant Pump Flywheel, Revision 0, July 2009.	5.4.1	
MUAP-09018-P MUAP-09018-NP	Calculation Methodology for Reactor Vessel Neutron Flux and Fluence, Revision 1, October 2009.	4.3.2	
MUAP-09019~~-P MUAP-09019-NP~~	Human Factors Engineering Program Management Plan, Revision 4, May 2014.~~HSI Design, Revision 2, October 2012.~~	~~18.1.1,~~ 18.1.2, 18.1.3, 18.1.4, 18.1.5, 18.2.1, 18.2.2~~18.3.3, 18.4.2, 18.4.3, 18.6.1, 18.6.3, 18.7.2~~	MIC-04-01-00013
MUAP-09020-P MUAP-09020-NP	Function Assignment Analysis for Safety Logic System, Revision 2, May 2011.	7.3.1	
MUAP-09021-P MUAP-09021-NP	Response Time of Safety I&C System, Revision 3, August 2013.	7.9.2, 16(B3.3.1), 16(B3.3.2)	
MUAP-09022-P MUAP-09022-NP	US-APWR Instrument Setpoint Methodology, Revision 3, July 2013.	7.2.1, 7.2.2, 7.3.2, 7.5.1, 16(B3.3.1), 16(B3.3.2), 16(B3.3.3), 16(B3.3.4), 16(B3.3.5), 7.8.1	
MUAP-09023-P MUAP-09023-NP	Onsite AC Power System Calculation, Revision ~~0~~1, ~~March~~August 201~~0~~3.	8.3.1	MIC-04-01-00009
MUAP-09025-P MUAP-09025-NP	CFD Analysis for Advanced Accumulator, Revision 3, June 2013.	6.3.2	
MUAP-10002-P MUAP-10002-NP	Damping Ratio of SC Structure, Revision 0, March 2011.	3.7.1, 3.7.2, 3.7.3	
MUAP-10003	US-APWR Physical Security Hardware ITAAC Abstracts, Revision 1, March 2011.	14.3.4	
MUAP-10006	Soil-Structure Interaction Analyses and Results for the US-APWR Standard Plant, Revision 3, November 2012.	3.7.1, 3.7.2, Appendix 3H (3H.1), Appendix 3I (3I.1)	
MUAP-10008-P MUAP-10008-NP	Staffing and Qualifications Implementation Plan, Revision ~~2, October 2012~~4, May 2014.	18.1.1, 18.1.5, 18.5.1, 18.5.2, 18.5.3	MIC-04-01-00013

Table 1.6-2 Material Referenced as Technical Reports (Sheet 4 of 6)

Report Number(1)	Title	DCD Section Number(2)	
MUAP-10009-P MUAP-10009-NP	Human-System Interface Design Implementation Plan, Revision 4, May 2014.~~HSI Design Implementation Plan, Revision 2, October 2012.~~	18.1.1, 18.1.3, 18.1.5, ~~18.6.3,~~ 18.7.2, 18.7.3	MIC-04-01-00013
MUAP-10012-P MUAP-10012-NP	Human Factors Verification and Validation Implementation Plan, Revision ~~2, October 2012~~4, May 2014.	18.1.1, 18.1.3, 18.1.5,~~-18.6.3~~ 18.10.1, 18.10.2, 18.10.3	
MUAP-10013	Design Implementation Implementation Plan, Revision ~~2, October 2012.~~4, May 2014	18.1.3~~18.1.1~~, 18.1.5, ~~18.6.3,~~ 18.11.2, 18.11.3	
~~MUAP-10014~~	~~Human Performance Monitoring Implementation Plan, Revision 2, October 2012.~~	~~18.1.1, 18.1.5, 18.6.3, 18.12.2~~	
MUAP-10017-P MUAP-10017-NP	US-APWR Methodology of Pipe Break Hazard Analysis, Revision 3, May 2012.	3.6.2	
MUAP-10018-P MUAP-10018-NP	US-APWR Containment Performance for Pressure Loads, Revision ~~0~~1, ~~June~~August 201~~0~~3.	3.8.1, 19.2.4	DCD_19-529S01
MUAP-10019-P MUAP-10019-NP	Calculation Methodology for Radiological Consequences in Normal Operation and Tank Failure Analysis, Revision 1, March 2011.	11.2.3, 11.3.3	
MUAP-10020-P MUAP-10020-NP	Safety-Related Air Conditioning, Heating, Cooling, and Ventilation Systems Calculations, Revision 2, March 2013.	6.2.9, 6.5.7, 9.4.8	
MUAP-10022	Evaluation on Jet Impingement Issues Associated with Postulated Pipe Rupture, Revision 2, May 2012.	3.6.2	
MUAP-10023-P MUAP-10023-NP	Initial Type Test Result of Class 1E Gas Turbine Generator System, Revision ~~5~~7, December 201~~2~~3.	1.5.2, 3.10.2, 8.3.1, 16 (B3.8.1)	MIC-04-01-00008
MUAP-10024	Structural Design Criteria for US-APWR Access Building, Revision 1, November 2011.	3.7.2	
MUAP-11002	Turbine Building Model Properties, SSI Analyses, and Structural Integrity Evaluation, Revision 3, February 2013.	3.7.2	
MUAP-11003-P MUAP-11003-NP	Summary of Stress Analysis Results for Pressurizer Surge Line. Revision 1, March 2011.	3B.3.2.4	
MUAP-11005	Research Achievements of SC Structure and Strength Evaluation of US-APWR SC Structure Based on 1/10th Scale Test Results, Revision 1, December 2012.	3.8.3	
MUAP-11007	Ground Water Effects on SSI, Revision 2, November 2012.	3.7.2, 3.8.5	
MUAP-11012-P MUAP-11012-NP	US-APWR RCCA Insertion Limit Load Test Report, Revision 0, March 2011.	3.9.5	
MUAP-11013	Containment Internal Structure Design and Validation Methodology, Revision 2, February 2013	3.8.3	

Table 1.6-2 Material Referenced as Technical Reports (Sheet 5 of 6)

Report Number(1)	Title	DCD Section Number(2)
MUAP-11014-P MUAP-11014-NP	Over Temperature ΔT and Over Power ΔT Reactor Trip Functions and Setpoint Determination Process, Revision 0, June 2011.	7.2.1
MUAP-11017-P MUAP-11017-NP	Hydraulic Test of the Full Scale US-APWR Fuel Assembly, Revision 0, May 2011	4.2
MUAP-11018	Containment Internal Structure: Stiffness and Damping for Analysis, Revision 1, February 2013	3.8.3
MUAP-11019	Containment Internal Structure: Design Criteria for SC Walls, Revision 1, January 2013	3.8.3
MUAP-11020	Containment Internal Structure: Anchorage and Connection Design and Detailing, Revision 1, February 2013	3.8.3
MUAP-12002-P MUAP-12002-NP	Sliding Evaluation and Results, Revision 1, January 2013	1.8.1, 3.8.5
MUAP-12006	SC Wall Fabrication, Construction and Inspection, Revision 0, February 2013	1.8.1, 3.8.3
MUAP-13001-P MUAP-13001-NP	US-APWR Criticality Evaluation Following Small Break LOCAs, Revision 0, January 2013	15.6.5
MUAP-13002	US-APWR Evaluation and Design Enhancement to Incorporate Lessons Learned from TEPCO's Fukushima Daiichi Nuclear Power Station Accident, Revision 0, March 2013	1.9, 3.4.1, 8.4.1
MUAP-13005	Operating Experience Review Implementation Plan, Revision 1, May 2014.	18.1.3, 18.1.5, 18.2.2, 18.2.3
MUAP-13007-P MUAP-13007-NP	Functional Requirements Analysis and Function Allocation Implementation Plan, Revision 1, May 2014.	18.1.3, 18.1.5, 18.3.2, 18.3.3
MUAP-13009-P MUAP-13009-NP	Task Analysis Implementation Plan, Revision 1, May 2014.	18.1.1, 18.1.3, 18.1.5, 18.4.2, 18.4.3
MUAP-13011-P MUAP-13011-NP	Criticality Analysis for US-APWR Containment Racks, Revision 0, May 2013	9.1.1
MUAP-13012-P MUAP-13012-NP	Mechanical Analysis for US-APWR Containment Racks, Revision 0, May 2013	9.1.2

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Table 1.6-2 Material Referenced as Technical Reports (Sheet 6 of 6)

Report Number(1)	Title	DCD Section Number(2)
MUAP-13013-P MUAP-13013-NP	Thermal-Hydraulic Analysis for US-APWR Containment Racks, Revision 0, May 2013	9.1.2
MUAP-13014-P MUAP-13014-NP	Human Reliability Analysis Implementation Plan, Revision 1, May 2014.	18.1.3. 18.1.5, 18.2.1, 18.2.2, 18.6.2, 18.6.3

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NOTE(1): -P(**p**roprietary), -NP(**n**on-**p**roprietary)
(2): If actual section number is indicated as x.y.z.a.b, a x.y.z level is used for the DCD Section Number. (ex. When actual section number is 6.3.2.1.2, only 6.3.2 is used in Table.

Information in this table that is italicized and enclosed in square brackets with an asterisk following the closing bracket is a special category of information designated by the NRC as Tier 2. Any change to this information requires prior NRC approval.*

DCD_14.03.07-94

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (Sheet 36 of 38)

COL ITEM NO.	COL ITEM
COL 17.6(1)	*The COL Applicant must provide in its FSAR a description of the maintenance rule program, and its for implementation, for monitoring the effectiveness of maintenance necessary to meet the requirements of 10 CFR 50.65.*
COL 18.1(1)	*Deleted*
COL 18.1(2)	*Deleted*
COL 18.3(1)	*Deleted*
COL 18.3(2)	*Deleted*
COL 18.4(1)	*Deleted*
COL 18.4(2)	*Deleted*
COL 18.4(3)	*Deleted*
COL 18.5(1)	*Deleted*
COL 18.5(2)	*Deleted*
COL 18.6(1)	*Deleted*
COL 18.6(2)	*Deleted*
COL 18.7(1)	*Deleted*
COL 18.8(1)	*Deleted*
COL 18.9(1)	*Deleted*
COL 18.10(1)	*Deleted*
COL 18.10(2)	*Deleted*
COL 18.11(1)	*Deleted*
COL 18.11(2)	*Deleted*
COL 18.12(1)	~~*Deleted*~~*The COL Applicant is to develop the Human Performance Monitoring Program*
COL 19.3(1)	*The COL Applicant who intends to implement risk-informed applications will update and upgrade the information in the design-specific PRA to incorporate site-specific, as-built and as-operated information per 10 CFR 50.71(h)(1) for its intended uses and application. The COL Licensee will perform peer reviews of the site-specific PRA in accordance with requirements in PRA standards endorsed by the NRC prior to the use of the PRA to support risk-informed applications and will verify that the PRA model has the technical adequacy and detail to support the proposed licensee programs and applications.*
COL 19.3(2)	*Deleted*
COL 19.3(3)	*Deleted*

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Table 1.9.3-1 Conformance with Generic Issues (Sheet 16 of 24)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/ Sec.
New Generic Issue #186 Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants NRC priority: CONTINUE	This issue was identified when the concern was raised that licensees operating within the regulatory guidelines of GL 85-11 may not have taken adequate measures to assess and mitigate the consequences of dropped heavy loads. In April 1996, NRC Bulletin 96-02 was issued to alert licensees of potential high consequences that could result from a cask drop and to remind them of complying with existing regulatory guidelines on the control and handling of heavy loads. In nuclear plant operation, maintenance, and refueling activities, heavy loads may be handled in several plant areas. If these loads were to drop because of human error or crane failure, they could impact on stored spent fuel, fuel in the core, or on equipment that may be required to achieve safe shutdown or permit continued decay heat removal. In some instances, load drops at specific times, locations, and weights could potentially lead to offsite doses that exceed 10 CFR Part 100 limits. If a licensee elected to use long-term dry storage casks to store excess spent fuel, the large, heavy casks would have to be hoisted and transported to and from the spent fuel pool while the plant is at full power operation. A comprehensive analysis of U.S. nuclear industry crane operating experience from 1968 through 2002 was conducted by the NRC and documented in NUREG-1774. Some of the NRC's findings and observations were: 1) The human error rate for crane operating events increased significantly; 2) Load drop events between the period 1993-2002 increased over the period 1981-1992; 3) The number of below-the-hook crane events (mainly rigging deficiencies or failures) increased greatly; 4) Calculational methodologies, assumptions, and predicted consequences varied greatly from licensee to licensee for very similar accident scenarios; 5) The number of mobile crane events declined slightly; and	The concern of this generic issue is regarding load drops that have occurred in recent years and the possibility that such an event could someday result in the load being dropped onto a source of radioactive inventory, such as stored spent fuel, fuel in the core, equipment that is performing a decay heat removal function, or equipment that would be required for safe shutdown. Per the language of the generic issue, some of the events that have occurred could have been prevented by single failure proof crane design (i.e., load drops or hook and block assembly drops). Many of these “below the hook” events, however, were rigging errors that were strictly the result of manual operator faults, and would not have been prevented by single failure proof crane design. NRC adopted 4 recommendations for development of follow-up guidance. Two of the recommendations involve evaluation and endorsement of cranes and rigging equipment that would result in fewer mishaps. The other two recommendations involve NRC developing guidance on good practices for crane operations, load movements, and load drop calculations. For the US-APWR, design (and later, by the COL Applicant, operational procedures) for the containment polar and refueling cranes, spent fuel pit crane, and auxiliary building crane preclude the dropping of heavy loads. A critical load is defined in ASME NOG-1-2004 and referred to in this DCD as any lifted load whose uncontrolled movement or release could adversely affect a nuclear safety-related (SC-1) SSC in terms of its ability to perform a required safety function, or when uncontrolled movement or release could result in potential offsite exposure in excess of 10 CFR limits.	9.1.1 through 9.1.5 (descriptions of new and spent fuel handling and storage), chapter 13 (conduct of operations), 15.7.4 (fuel handling accident) 15.7.5 (spent fuel cask drop accident) ~~18.2 (human factors – operations organization), 18.4 (human factors – task analysis and hazards evaluation), 18.9 (human factors – procedural development)~~ MIC-04-01-00013

Table 1.9.3-1 Conformance with Generic Issues (Sheet 21 of 24)

Issue Number and Title	Summary	Status/Discussion	Addressed in DCD Chapter/ Sec.
New Generic Issue #186 Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants NRC priority: CONTINUE (continued)		• Defined safe load path and provisions to provide visual reference to the crane operator and/or signal person of the safe load path envelope • Specific steps and proper sequence to be followed for handling load • Precautions, limitations, prerequisites, and/or initial conditions associated with movement of the load • Slings and other equipment used to make a complete lifting device specified in the load handling procedures will conform to NUREG 0612 • Equipment layout drawings showing the safe load path will be used to define safe load paths in load handling procedures, and deviation from defined safe load paths will require a written alternative procedure approved by the COL Applicant's plant review board ~~In DCD chapter 18, human factors considerations and commitments appropriate to the DCD stage are described, applying to both design features and plant personnel. Of interest to this generic issue are sections 18.2 (human factors – operations organization), 18.4 (human factors – task analysis and hazards evaluation) and 18.9 (human factors – procedural development).~~ These considerations and commitments, as well as the other design and operational material presented above, are intended to prevent the types of events that are the subject of this generic issue. The US-APWR will, of course, also be attentive to any new operational guidance that comes from NRC's efforts related to this issue.	

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Table 1.9.3-2 Location of Description for Additional TMI-Related Requirements (Sheet 4 of 13)

50.34(f) Item	Action Plan Item	Requirement	Location in DCD
(2)(ii)	I.C.9	Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with [the Institute of Nuclear Power Operations (INPO)] and other industry efforts. (Applicable to construction permit applicants only)	N/A (COLA only)
(2)(iii)	I.D.1	Provide, for Commission review, a control room design that reflects state-of-the-art human factors principles prior to committing to fabrication or revision of fabricated control room panels and layouts.	7.5.1, 18.7~~18.7.2., 18.7.2.1, 18.7.2.2, 18.7.2.3, 18.7.2.4, 18.7.2.5, 18.7.2.6, 18.7.3.1, 18.7.3.2~~
(2)(iv)	I.D.2	Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded.	7.1.1.5.4, 7.5.1.4, 13.3, 18.7~~18.7.2.5~~

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Table 1.9.3-2 Location of Description for Additional TMI-Related Requirements (Sheet 5 of 13)

50.34(f) Item	Action Plan Item	Requirement	Location in DCD
(2)(v)	I.D.3	Provide for automatic indication of the bypassed and operable status of safety systems.	7.1.1.5.2, 8.1.5.3, Table 8.1-1, ~~18.7.3.2,~~ 18.7~~Table 18.7-1~~
(2)(vi)	II.B.1	Provide the capability of high-point venting of noncondensible gases from the RCS, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room, and their operation shall not lead to an unacceptable increase in the probability of LOCA or an unacceptable challenge to containment integrity.	5.4.12, 19.2.3.3.9

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Table 1.9.3-2 Location of Description for Additional TMI-Related Requirements (Sheet 11 of 13)

50.34(f) Item	Action Plan Item	Requirement	Location in DCD
(2)(xxv)	III.A.1.2	Provide an onsite Technical Support Center, an onsite Operational Support Center, and, for construction permit applications only, a near-site Emergency Operations Facility.	7.5.1.6, 13.3, 18.7~~18.7.3.1~~ (OSC is addressed as COL item.)
(2)(xxvi)	III.D.1.1	Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) accident source term[11] radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and the public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency.	Table 6.3-1(1/2)
(2)(xxvii)	III.D.3.3	Provide for monitoring of in-plant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions.	7.3.1.5, 7.7.1.11, Table 7.5-3, 11.5, 12.3
(2)(xxviii)	III.D.3.4	Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in an accident source term[11] release, and make necessary design provisions to preclude such problems.	6.4.2.5, 15.6.5.5

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