MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

February 20, 2014

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

Attention: Mr. Perry Buckberg

Docket No. 52-021
MHI Ref: UAP-HF-14012

Subject: **Submittal of Revision to Published Topical Report (PQD-HD-19005-A Revision 0) entitled Quality Assurance Program (QAP) Description for Design Certification of the US-APWR**

References: 1) Letter MHI Ref: UAP-HF-13281, "Submittal of Revision to Published Topical Report (PQD-HD-19005-A Revision 0) entitled Quality Assurance Program (QAP) Description for Design Certification of the US-APWR", dated December 9, 2013. (ML13350A510)

By letter dated December 9, 2013 (Reference 1), Mitsubishi Heavy Industries, Ltd. (MHI) submitted a revised version of Quality Assurance Program Description (QAPD) PQD-HD-19005-A, Rev.0, to the U.S. Nuclear Regulatory Commission (NRC).

The purpose of this letter is to forward a further revised version of PQD-HD-19005-A, Rev. 0, to the NRC for approval. The change to PQD-HD-19005-A, Rev. 0, is a change of organization in the Nuclear Energy Systems in MHI. MHI is making this revision for administrative purposes only, and as such, this revision does not change any QA program commitments documented and approved by the NRC in the Safety Evaluation Report dated January 24, 2008 and does not reduce any QA program quality or safety aspects of the Design Certification.

Since various Chapters of the US-APWR Design Control Document (DCD), Revision 4 reference Revision 5 of the QAPD, MHI will update the revision number of the QAPD in Revision 4 of the DCD. MHI will also update the organizational titles in Revision 4 of the DCD to be consistent with the change in organization in the QAPD. The markups to the DCD related to the QAPD revision are provided as an additional enclosure to this letter.

We make this administrative change to our QAPD effective February 1, 2014 since it does not reduce any QAP commitments or quality aspects of the design certification. We are requesting formal NRC review and approval as an administrative revision to our Quality Assurance Topical Report.

The QAPD is being submitted electronically on a compact disc (CD). This letter includes a copy of the non-proprietary version of the QAPD revision (Enclosure 1) and the non-proprietary DCD markup showing the updated to the QAPD revision number (Enclosure 2).

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

Q004
D081
NRO

Sincerely,

Yoshiki Ogata,
Executive Vice President
Mitsubishi Nuclear Energy Systems, Inc.
On behalf of Mitsubishi Heavy Industries, LTD.

Enclosures:

1. CD1: "Revised QAPD Topical Report (PQD-HD-19005, "Quality Assurance Program (QAP) Description for Design Certification of the US-APWR," Revision 7)"
2. DCD Markups for QAPD Revision 7

The file contained on the CD is listed in Attachment 1 hereto.

CC: P. Buckberg
J. Tapia

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Enclosure 2

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DCD Markups for QAPD Revision 7

February 2014

(Non-Proprietary)

ACRONYMS AND ABBREVIATIONS (CONTINUED)

MSS-SP	Manufacturer Standardization Society-Standard Practice
MSSV	main steam safety valve
MT	main transformer
MTC	moderator temperature coefficient
MTCDS	main turbine control and diagnostic system
MTCV	main turbine control valves
MTS	main turbine system
MTSV	main turbine stop valve
MTTR	mean time to repair
MV	medium voltage
N Center	Nuclear Energy Systems Engineering Center
N/A	not applicable
N/E	normal/emergency
N/ELS	normal/emergency lighting system
NaTB	sodium tetraborate decahydrate
NCIG	National Construction Issues Group
NDE	nondestructive examination
NDRC	National Defense Research Committee
NDS	non-radioactive drain system
NDTT	nil ductility transition temperature
NEC	National Electric Code
NEI	Nuclear Energy Institute
NEMA	National Electrical Manufacturer Association
N~~ESH~~ESD	Nuclear Energy Systems ~~Headquarters~~Division
NFPA	National Fire Protection Association
NFR	new fuel rack
NGHS	noble gas holdup system
NIS	nuclear instrumentation system
NIST	National Institute of Standards and Technology
NLS	normal lighting system
non-ECWS	non-essential chilled water system
non-ESW	non-essential service water
NPGS	nuclear power generating stations
NPS	nominal pipe size
NPSH	net positive suction head
NQA	nuclear quality assurance
NR	neutron reflector

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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 System Description and HFE Process, Revision 5, November 2011	1.5.2, 7.1, 7.1.1, 7.1.3, 7.5.1, 7.6.1, 7.8, 18.1.1, 18.1.5, 18.2.2, 18.7.2, 18.7.3, 18.9.2, 18.10.2
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 ~~6~~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

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NOTE(1): -P(proprietary) , -NP(non-proprietary), -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.

A 25 consecutive start preoperational test, without loading, will be performed for each AAC GTG.

The AAC power system will be inspected and tested periodically based on manufacturer's recommendations and Reg 1.155 to demonstrate operability and reliability. The surveillance test interval does not exceed 3 months (Quarterly). During the quarterly test the AAC is started and brought to operating conditions. Additionally, during every refueling outage, the AAC generator is tested by performing a timed start and rated load capacity test. Following preoperational testing, the AAC power system will be maintained to meet or exceed 95% reliability as determined in accordance with NSAC-108 (Reference 8.4-2) or equivalent methodology to meet Criterion 5 of Section C.3.3.5, RG 1.155 (Reference 8.3.1-21). Testing and maintenance of the AAC is evaluated under the reliability assurance program and the maintenance rule program.

Procedures to cope with SBO are addressed in Section 13.5 and the training is addressed in Section 13.2. These include all operator actions necessary to cope with SBO for at least the duration in accordance with Subsection 8.4.2.1.1 and to restore normal long-term core cooling/decay heat removal once ac power is restored. This meets the requirement of Regulatory Position C.3.4 of RG 1.155.

The quality assurance of AAC GTG is controlled in accordance with DCD Chapter 17 and related topical report PQD-HD-19005 Revision ~~6~~7 (Reference 8.4-3). This meets the requirements of Regulatory Position C.3.5 of RG 1.155. | MIC-04-08-0 0006

8.4.3 Combined License Information

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

8.4.4 References

8.4-1 Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors, NUMARC 87-00, Revision. 1, August 1991.

8.4-2 Reliability of Emergency Diesel Generators at U.S Nuclear Power Plants, NSAC-108, September 1986.

8.4-3 Quality Assurance Program (QAP) Description For Design Certification of the US-APWR, PQD-HD-19005 Revision ~~6~~7, ~~May~~February 201~~3~~4. | MIC-04-08-0 0006

8.4-4 US-APWR Evaluation and Design Enhancement to Incorporate Lessons Learned from TEPCO's Fukushima Dai-ichi Nuclear Power Station Accident, MUAP-13002, Revision 0, March 2013.

ACRONYMS AND ABBREVIATIONS

AAC	alternate AC
ac	alternating current
CAP	corrective action program
CCF	common cause failure
CCW	component cooling water
CCWS	component cooling water system
CDF	core damage frequency
CFR	Code of Federal Regulations
COL	Combined License
COLA	Combined License Application
CS	containment spray
CSS	containment spray system
CVCS	chemical and volume control system
DAS	diverse actuation system
dc	direct current
DCD	Design Control Document
D-RAP	design reliability assurance program
DVI	direct vessel injection
ECCS	emergency core cooling system
EE	Energy & Environment
EFW	emergency feedwater
EFWP	emergency feedwater pit
EFWS	emergency feedwater system
EJ	engineering judgment
EP	expert panel
EPS	emergency power source
ESF	engineered safety features
ESW	essential service water
ESWS	essential service water system
FIRE	FIRE event
FLOOD	FLOOD event
FSS	fire protection water supply system
FV	Fussell Vesely
HSIS	human-system interface system
HVAC	heating, ventilation, and air conditioning
I&C	instrumentation and control
ITAAC	inspection, test, analyses, and acceptance criteria

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ACRONYMS AND ABBREVIATIONS (CONTINUED)

LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LPSD	low-power and shutdown
M/D	motor driven
MCC	motor control center
MFWS	main feedwater system
MHI	Mitsubishi Heavy Industries, Ltd.
MOV	motor operated valve
MSS	main steam supply system
NES~~H~~D	Nuclear Energy Systems ~~Headquarters~~Division
NRC	U.S. Nuclear Regulatory Commission
O-RAP	reliability assurance program during the operations phase
PAM	postaccident monitoring
PCMS	plant control and monitoring system
PRA	probabilistic risk assessment
QA	quality assurance
QAP	quality assurance program
QAPD	quality assurance program description
RAP	reliability assurance program
RAW	risk achievement worth
RCP	reactor coolant pump
RCS	reactor coolant system
RG	Regulatory Guide
RHR	residual heat removal
RHRS	residual heat removal system
RPS	reactor protection system
RTNSS	regulatory treatment of non-safety-related systems
RWSAT	refueling water storage auxiliary tank
RWS	refueling water storage
RWSP	refueling water storage pit
RWSS	refueling water storage system
SBO	station blackout
SDV	safety depressurization valve
SFP	spent fuel pit
SFPCS	spent fuel pit cooling and purification system
SG	steam generator
SGTR	steam generator tube rupture
SIS	safety injection system
SRP	Standard Review Plan

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17.3 Quality Assurance Program

The Senior General Manager of Nuclear Energy Systems Division (NESD) is responsible for the Design Certification Activities of US-APWR. The design activities performed by the Engineering and Designing Departments ~~Nuclear Plant Engineering Division and the Nuclear Plant Production Division~~ for the US-APWR standard plant design are subjected to the QA Program controls specified in "Quality Assurance Program (QAP) Description For Design Certification of the US-APWR (PQD-HD-19005 Rev.~~6~~7)" (Ref 17.5-4). MIC-04-17-0 0002 MIC-04-17-0 0002 MIC-04-17-0 0002

Subcontractors of the ~~Nuclear Plant Engineering Division and the Nuclear Plant Production Division~~Engineering and Designing Departments performing design activities in support of the US-APWR are also required to follow QAPD (PQD-HD-19005 Rev.~~6~~7).

For the Quality Assurance Program Description during the Design Certification phase for the US-APWR standard plant design, see Section 17.5.

The COL Applicant is responsible for the development of a Quality Assurance Program Description applicable to site-specific design activities and for plant construction and operation phases.

and III of the D-RAP and the O-RAP are the responsibility of the COL Applicant. The COL Applicant will specify the policy and implement procedures to address the specific plant operation and maintenance activities associated with the risk-significant SSCs identified by the D-RAP.

The non safety-related RAP SSCs would be subjected to the appropriate QA controls that are described in the Section 17.5 of the US-APWR DCD for the phase I of the D-RAP, and in Section 17.5 of the site specific COL for Phases II and III of the D-RAP.

17.4.3 Scope

The US-APWR D-RAP identifies risk-significant SSCs and considers risk insights and key assumptions from the aspect of plant operation, maintenance, and performance monitoring to be addressed to ensure safe, reliable plant operation or mitigate plant transients or other events that could present a risk to the public. The risk-significant SSCs are identified using PRA, deterministic, or other methods of analysis, including industry experience, and EPs. DCD Table 19.1-119 provides a comprehensive list of risk insights and key assumptions applicable to the US-APWR for activities such as Expert Panel D-RAP deliberations.

17.4.4 Quality Controls

a. Organization

MHI is responsible for Phase I of the D-RAP.

Manager, ~~US~~-APWR project: Manager, ~~US~~-APWR project is overall responsible for the establishment of and implementation of the US-APWR D-RAP. In this regard, Manager or his designated representative is responsible to assure all affected organizations are aware of the D-RAP, its purpose, and the requirements herein. MIC-04-17-0 0002

Director, Reactor and Plant Safety: Director, Reactor and Plant Safety, is responsible for the use of the PRA results and risk insights for the EP, and for the conduct and coordination of the EP. The Reactor and Plant Safety organization includes the risk and reliability organization.

Director, QA: Director, QA is responsible to assure proper implementation of QA program elements. This includes design control, procedures and instructions, records, corrective actions and audits pertaining to the D-RAP.

Directors, Design Engineering: Directors, Design Engineering, are responsible to implement this D-RAP and specifically to assure that the US-APWR is designed consistent with the risk insights and key assumptions for risk significant SSCs.

The risk and reliability organization is responsible to ask the related design engineering sections to review key assumptions and to feed back their comments to ensure key assumptions are realistic and achievable.

The risk and reliability organization is responsible to provide the D-RAP related inputs in the design process by participating in the design change process.

The risk and reliability organization is also responsible to involve in the design review.

b. Design Control

The list of risk-significant SSCs for the D-RAP as well as the associated risk insights and key assumptions shall be maintained by the risk and reliability organization. The list and changes thereof shall be approved by the EP and be provided to design engineering and QA staff working on the US-APWR project.

The risk and reliability organization shall ensure that the design engineers are provided the list of risk-significant SSCs for the D-RAP including the associated risk insights and key assumptions listed in DCD Table 19.1-119. The design engineers shall take into account the list of risk-significant SSCs for the D-RAP as well as the risk insights and key assumptions in their design activities. Based on this information, design engineers shall provide feedback to the risk and reliability organization in order to ensure that the risk insights and key assumptions are appropriately incorporated into the design, construction, and operational protocol.

c. Procedures and Instructions

Manager, ~~US~~ APWR project or his designated representative has prepared the procedures and instructions used in implementation of the D-RAP. Manager, ~~US~~ APWR project is responsible for development and verification of implementation of the D-RAP, and for assuring all affected MHI organizations are aware of the D-RAP. MIC-04-17-0 0002

d. Records

Records related to the D-RAP which are required to be maintained include the following:

- List of Risk-Significant SSCs
- EP meeting minutes/summaries
- Other quality assurance program records in accordance with the US-APWR QAPD (Ref. 17.4-2) for design certification.

e. Corrective action

RAP activities determined to be in error, deficient, defective, or nonconforming shall be entered into the corrective action program (CAP) and addressed appropriately. The CAP utilized to support the QAPD is used to implement the corrective actions related to the RAP.

f. Audit

Audit plans shall include for consideration, sampling the effectiveness of implementation of RAP implementation procedure. Audits shall consider several key aspects of the RAP including the identification of risk-significant SSCs, process, and controls that ensure that the plant will be designed and constructed in a manner that is consistent with the risk insights.

1) Reliability performance goals for risk-significant SSCs consistent with the existing maintenance and quality assurance processes on the basis of information from the D-RAP (for example, implementation of the maintenance rule following the guidance contained in RG 1.160 is one acceptable method for establishing performance goals provided that SSCs are categorized as HSS within the scope of the Maintenance Rule program), and

2) Performance and condition monitoring requirements to provide reasonable assurance that risk-significant SSCs do not degrade to an unacceptable level during plant operations.

17.4.10 References

17.4-1 "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Design," SECY 95-132, U.S. Nuclear Regulatory Commission, Washington, DC, May 1995.

17.4.2 "Quality Assurance Program (QAP) Description For Design Certification of the US-APWR (PQD-HD-19005 Rev.~~6~~7, ~~May~~February 201~~3~~4)" | MIC-04-17-0 0002

17.4-3 'Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,' "Domestic Licensing of Production and Utilization Facilities," Energy. Title 10, Code of Federal Regulations, Part 50.65, U.S. Nuclear Regulatory Commission, Washington, DC.

17.4-4 10 CFR 50.69 SSC Categorization Guideline NEI 00-04 Rev.0, Nuclear Energy Institute, July 2005.

17.4-5 Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants, NUREG/CR-6928, U.S. Nuclear Regulatory Commission, Washington, DC, February 2007.

17.4-6 Guide to the Collection And Presentation of Electrical, Electronic, Sensing Component, And Mechanical Equipment Reliability Data For Nuclear Power Generating Stations, IEEE Std. 500, Appendix D, Institute of Electrical and Electronics Engineers, New York, NY, 1984.

17.4-7 Analysis of Core Damage Frequency: Internal Events Methodology, NUREG/CR-4550 Volume 1, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, DC, January 1990.

17.4-8 US-APWR Probabilistic Risk Assessment, MUAP-07030-P Rev. 3 (Proprietary), Mitsubishi Heavy Industries, June 2011.

17.5 Quality Assurance Program Description

During the Design Certification phase for US-APWR standard plant design, the MHI-~~NESH~~EE US-APWR Project Quality Assurance Program (QAP) is the top-level policy that establishes the quality assurance policy and assigns major functional responsibilities. The QAP provides for the methods and establishes the QAP and administrative control requirements described in "Quality Assurance Program (QAP) Description For Design Certification of the US-APWR (PQD-HD-19005 Rev.~~6~~7)" (MHI QAPD)(Ref 17.5-4), that meet 10 CFR Part 50, Appendix B and 10 CFR Part 52 for safety-related SSCs. The MHI QAPD is based on the requirements of ASME NQA–1-1994, "Quality Assurance Requirements for Nuclear Facility Applications," Parts I and II, as specified in Ref.17.5-4. | MIC-04-17-0 0002 | MIC-04-17-0 0002

Selected elements of the QAP description (QAPD) are also applied to SSCs that are nonsafety-related in accordance with their contribution to plant safety (Part III of Reference 17.5-4) or to meet NRC guidance that establishes applicable quality assurance requirements. The controls applied to nonsafety-related SSCs per the QAPD Part III are referred to as "augmented" quality assurance controls. The contribution of nonsafety-related SSCs to plant safety is determined by (1) the SSC's risk-significance as determined by the D-RAP as described in Section 17.4 and (2) the reliance on the SSC to address regulatory events, such as ATWS, fire protection and SBO. Specific program controls are applied to these nonsafety-related SSCs in a selected manner, targeted at those characteristics or critical attributes that render the SSC a significant contributor to plant safety.

The MHI QAPD for the Design Certification Phase has been prepared on the basis of the NRC approved QAP template (NEI, 06-14A Rev.4 and earlier revisions) (Ref 17.5-3) prepared by the Nuclear Energy Institute and has been evaluated against the SRP. The MHI QAPD provides the controls that implement the QAP. MHI performed a comparison of the MHI QAPD against the SRP (Mar. 2007) (Ref 17.5-2) and the draft SRP (Sept. 2006) (Ref 17.5-1) which was used as a reference for the MHI QAPD and determined that the MHI QAPD is satisfactory.

Business policies of MHI-~~NES~~EE establish high level responsibilities and authority for carrying out administrative functions which are outside the scope of the QAP. | MIC-04-17-0 0002

Procedures establish practices for certain activities which are common to all MHI-~~NES~~EE organizations performing those activities such that the activity is controlled and carried out in a manner that meets QAP requirements. Organization specific procedures establish detailed implementation requirements and methods, and may be used to implement the business policies of MHI-~~NES~~EE or be unique to particular functions or work activities. | MIC-04-17-0 0002 | MIC-04-17-0 0002

The COL Applicant is responsible for the development of a Quality Assurance Program Description for site-specific design activities and for plant construction and operation.

17.5.1 Combined License Information

COL 17.5(1) *The COL Applicant shall develop and implement a Quality Assurance Program Description for site-specific design activities and for plant construction and operation.*

17.5.2 References

17.5-1 "Draft Standard Review Plan (SRP) 17.5 dated September 22, 2006"

17.5-2 "Standard Review Plan (SRP) 17.5 March 2007"

17.5-3 "Quality Assurance Program Description (NEI 06-14A Rev.4 and earlier versions)"

17.5-4 "Quality Assurance Program (QAP) Description For Design Certification of the US-APWR (PQD-HD-19005 Rev.~~5~~7, ~~May~~February 201~~3~~4)" | MIC-04-17-0 0002

18.1.6 Combined License Information

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

COL 18.1(1) *Deleted*

COL 18.1(2) *Deleted*

18.1.7 References

18.1-1 HSI System Description and HFE Process, MUAP-07007-P (Proprietary) and MUAP-07007-NP (Non-Proprietary), Revision 5, November 2011.

18.1-2 Conditions of Licenses, NRC Regulations Title 10, Code of Federal Regulations, Part 50.54.

18.1-3 Wood, R. T., et al., Advanced Reactor Licensing: Experience with Digital I&C Technology in Evolutionary Plants, NUREG/CR-6842, March 2004.

18.1-4 Operators' Licenses, NRC Regulations Title 10, Code of Federal Regulations, Part 55.

18.1-5 Training and Qualification of Nuclear Power Plant Personnel, NRC Regulations Title 10, Code of Federal Regulations, Part 50.120.

18.1-6 Quality Assurance Program (QAP) Description for Design Certification of the US-APWR, PQD-HD-19005, Revision ~~6~~7, Mitsubishi Heavy Industries, Ltd., ~~May~~February 201~~3~~4. MIC-04-18-0 0004

18.1-7 Human Factors Engineering Program Review Model, NUREG-0711, Revision 2, U.S. Nuclear Regulatory Commission, Washington, DC, February 2004.

18.1-8 Specific Exemptions, NRC Regulations Title 10, Code of Federal Regulations, Part 50.12.

18.1-9 Petition for Rulemaking, NRC Regulations Title 10, Code of Federal Regulations, Part 2.802.

18.1-10 An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Regulatory Guide 1.174, Revision 1, November 2002.

18.1-11 General Design Criteria for Nuclear Power Plants, NRC Regulations Title 10, Code of Federal Regulations, Part 50, Appendix A.

18.1-12 US-APWR HSI Design, MUAP-09019-P (Proprietary) and MUAP-09019-NP (Non-Proprietary), Revision 2, September 2012.

- Identification of design solutions to address significant HEDs along with an indication of their current status (implemented or scheduled to be implemented)
- Determination of the HFE Program activities that must be re-performed to satisfy the requirements of the limited reapplication of the HFE analysis processes, as described in Sections 18.3 through 18.6
- Verification of the implementation of the design solutions resolving HEDs including how the change complies with the V&V evaluation criteria

HEDs are not considered in isolation and, to the extent possible, their potential for interaction with other HED disposition activities are considered when developing and implementing solutions. For example, if the HSI for a single plant system is associated with many HEDs, then the set of design solutions are coordinated to enhance overall performance and avoid incompatibilities between individual solutions. Approaches that develop design solutions to some HEDs before all have been identified from a particular verification or validation activity are acceptable provided that the potential interactions between HEDs are specifically considered prior to implementing the design solutions.

18.10.3 Results

The V&V Phase 1 results are documented in Reference 18.10-6 Part 1, and Reference 18.10-7, Part 3. The Phase 2 results, which include V&V program staffing and resources, the detailed procedures for conducting the V&V program, the V&V program data, analysis, and results, identification, and resolution of HEDs, and the major conclusions from these activities along with their bases, will be documented in a results summary report in accordance with Reference 18.10-5.

Phase 1 verification and validation activities for the US-Basic HSIS, as documented in References 18.10-6 and 18.10-7, are not credited for the US-APWR HSIS verification and validation, as required by NUREG-0711 Section 11. Phase 1 V&V activities are considered part of the US-Basic HSIS design process. Compliance to NUREG-0711 Section 11, relies on the Phase 2 V&V program which will be conducted in accordance with Reference 18.10-5.

18.10.4 Combined License Information

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

COL 18.10(1) *Deleted*

COL 18.10(2) *Deleted*

18.10.5 References

18.10-1 Quality Assurance Program (QAP) Description for Design Certification of the US-APWR, PQD-HD-19005, Revision ~~6~~7, Mitsubishi Heavy Industries, Ltd., ~~May~~February 201~~3~~4. MIC-04-18-0 0004

ATTACHMENT 1

Docket No. 52-021
MHI Ref: UAP-HF-14012

FILES CONTAINED ON CD 1

CD 1: **"Revised QAPD Topical Report (PQD-HD-19005, "Quality Assurance Program (QAP) Description for Design Certification of the US-APWR," Revision 7)"**

Contents of CD

	File Name	Size	Sensitivity Level
1.	PQD-HD-19005 Revision 7.pdf	640 KB	Non-Proprietary