

April 24, 2007

LICENSEE: MITSUBISHI HEAVY INDUSTRIES (MHI)
FACILITY: US-APWR STANDARD DESIGN PRE-APPLICATION REVIEW
SUBJECT: SUMMARY OF MARCH 1, 2007, PUBLIC MEETING ON RADIATION DOSE ASSESSMENT, SEVERE ACCIDENT MITIGATION, AND PROBABILISTIC RISK ASSESSMENT (PRA) FOR THE US-APWR.

On March 1, 2007, a Category 1 public meeting was held between the U.S. Nuclear Regulatory Commission (NRC) staff and representatives of MHI at NRC Headquarters, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland. The purpose of the meeting was to discuss PRA, severe accident mitigation, and radiation dose assessment methodologies for the US-APWR design certification. MHI announced its intention to submit a design certification application for the US-APWR in December 2007. A list of attendees is provided as Enclosure 1. MHI presented handouts that are shown in Enclosures 2, 3, and 4 and can be accessed through the Agencywide Documents Access and Management System (ADAMS) accession numbers ML070670194, ML070670192, and ML070670188.

Dose Assessment

MHI opened the meeting by discussing the radiation dose analysis that will be used in the US-APWR design certification application. MHI presented the dose analysis for normal operation and accident conditions, and discussed the methods for radiation shielding protection. MHI plans to use the latest revisions to NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants," and Draft Regulatory Guide DG-1145, "Combined License Applications for Nuclear Power Plants," for the dose analysis and calculation codes that have already been approved by the NRC. The PWR-GALE code will be used to determine the atmospheric dispersion factors that will bound the atmospheric conditions at the plant sites. MHI stated that at the Combined Operating License (COL) stage, the COL applicants will validate the design certification analysis and compare the analysis to the site specific atmospheric dispersion factors. MHI presented the radioactive gaseous and liquid effluent pathways that will be analyzed using the atmospheric dispersion model.

When discussing the radiation dose analysis for accident conditions, MHI informed the NRC staff that it plans to use the RADTRAD and MicroShield codes to analyze several accidents. These accidents consist of the loss-of-coolant accident, steam generator tube rupture, main steam line break, fuel handling accident, rod ejection accident, and spent fuel cask drop accident. Next, MHI discussed the safety features available to mitigate these types of accidents and the fission product pathways that resulted from these particular accidents. MHI also discussed the applicable NRC regulations and regulatory guidance that would be used, including 10 CFR 50.67. The NRC staff informed MHI that 10 CFR 100.21 should be used instead of 10 CFR 50.67, since 10 CFR 50.67 only applies to current operating plants.

MHI then presented its methods and acceptance criteria for radiation shielding protection. Discussions were held on the acceptance criteria, radiation sources, and shielding design. During the acceptance criteria discussion, MHI stated that it would follow the requirements of 10 CFR 50.67 for personnel exposure; however, the NRC staff noted that 10 CFR 50.34 was the correct regulation that should be used instead of 10 CFR 50.67. MHI presented several codes that will be used for radiation shielding design. The ORIGEN2 Code will be used for source term calculations for core inventory and spent fuel, the MicroShield code will be used for almost all of the contained sources besides neutron source, G3 Code will be used for scattered gamma-ray dose calculation, MCNP Code will be used for complicated geometries, and the DORT and ANSIN codes will be used for the reactor pressure vessel and reactor coolant system (RCS) equipment.

Severe Accident Treatment and Mitigation

MHI discussed the severe accident analysis and its plans for submitting this analysis as part of the design certification application in December 2007. MHI plans to comply with the Three Mile Island requirements contained in 10 CFR 50.34(f), develop an appropriate PRA model, and resolve medium and high priority generic safety issues.

MHI identified eight issues associated with severe accidents. They are hydrogen mixing and combustion, core debris coolability, steam explosion, high pressure melt ejection and direct containment heating, temperature induced steam generator tube rupture, molten core concrete interaction, equipment survivability, and early and late containment overpressure failure. MHI also presented plans for mitigating these types of severe accidents. The hydrogen mixing and combustion accident will be mitigated by using a large dry containment that provides adequate strength to contain most hydrogen burns. Combustible gas will be controlled by the use of hydrogen monitors and gas igniters. Core debris coolability will be addressed by providing sufficient reactor cavity floor area for debris spreading. The reactor cavity floor will be made of basalt concrete. This feature will serve to mitigate any molten core concrete interaction.

The dedicated severe accident depressurization valves will be used to reduce RCS pressure to avoid high pressure melt ejection and direct containment heating. These dedicated severe accident valves can also be used to reduce the likelihood of temperature induced steam generator tube rupture. The firewater injection system and a drain line to the reactor cavity can also be used to reduce RCS pressure. The large dry containment, the decay heat removal system, and the backup firewater pump will serve to mitigate the effects of containment overpressure.

PRA methodology

The last presentation given by MHI pertained to the PRA analysis for the US-APWR design certification and PRA model. The PRA model will be based upon available design information. For the areas where specific information will be provided at the COL stage, the PRA model will determine the limiting boundary conditions and operating assumptions.

The NRC staff and MHI discussed the technical elements that comprise the Level 1, 2, and 3 analysis. The US-APWR Level 1 PRA discussion on technical elements focused on internal events at full and low power operation and shutdown, internal flooding, internal fire, seismic events, and limited external events. Likewise, the discussion on level 2 PRA technical elements

involved internal floods and fires, and low power and shutdown events. The NRC staff questioned MHI on the PRA analysis and on the application of Regulatory Guide 1.200. The NRC staff stated that the PRA analysis for the US-APWR needs to meet the Commission's safety goals. The NRC staff informed MHI that DG-1145 will be revised soon and this revision would impact the US-APWR PRA analysis. MHI stated that it plans to submit the PRA Level 1 and 2 Technical Elements as part of the design certification application in December 2007, and submit the Level 3 PRA analysis in March 2008.

Members of the public were in attendance, but Public Meeting Feedback forms were not received. Please direct any inquiries to me at 301-415-1544, or srm2@nrc.gov.

/RA/

Stephen Raul Monarque
AP1000 Projects Branch 1
Division of New Reactor Licensing
Office of New Reactors

Project No. 751

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OFFICE	NRO/DNRL/NWE1/LA	NRO/DNRL/NWE1/PM	NRO/DNRL/NWE1/BC
NAME	KGoldstein	SMonarque	SCoffin
DATE	04/ 24 /2007	04/ 24 /2007	04/ 24 /2007

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Public Meeting on U.S.-APWR
March 1, 2007
Attendance Lists

Name	Affiliation
H. Bickel	Talisman
Stephen R. Monarque	NRC
Alan Levin	AREVA
Lynn Mrowca	NRC
Nick Saltos	NRC
Michelle Hart	NRC
Mark Beaumont	Washington Group International
Ed Fuller	NRC
David Bessette	NRC/Research
Hien Le	NRC
Hiroyuki Naito	Nuclear and Industrial Safety Agency
Yumi Kawanago	Mitsubishi Heavy Industries (MHI) - MNES
Yoshinori Takechi	MHI
Hiromasa Nishino	MHI
Dennis Buschbaum	TXU Power
Takahiro Imamura	MNES
Hiroshi Goda	MNES
Masayuki Kambara	MHI
Kiyoshi Yamauchi	MHI
Toshisada Kato	MHI
C. Keith Paulson	MHI
Alfred Torri	RMA
Makoto Toyama	MHI
Katsuya Kuroiwai	MHI
Futoshi Tanaka	MHI

Name	Affiliation
Katsunori Kawai	MHI
Paul Gaukler	PWSP
Charles Brinkman	Westinghouse
Terry Heames	Alion Serence
Harold Rothstein	Washington Group
Ellis W. Merschoff	Talisman International, LLC
Todd Ansehmi	Anion Science and Technology
Ryuji Zwasaki	Toshiba
John H. Bickel	Talisman International, LLC
Josh McGuire	NRC

Enclosure 2

US-APWR Pre-Application Review Dose Evaluation

(ML070670194)

Enclosure 3

US-APWR Pre-Application Review Severe Accident Treatment and
Mitigation

(ML070670192)

Enclosure 4

US-APWR, Pre-Application Review Overview PRA Methodology

(ML070670188)

cc:

Mr. Glenn H. Archinoff
AECL Technologies
481 North Frederick Avenue
Suite 405
Gaithersburg, MD 20877

Ms. Michele Boyd
Legislative Director
Energy Program
Public Citizens Critical Mass Energy
and Environmental Program
215 Pennsylvania Avenue, SE
Washington, DC 20003

Ms. Anne W. Cottingham
Assistant General Counsel
Nuclear Energy Institute
1776 I Street, NW, Suite 400
Washington, DC 20006

Mr. Marvin Fertel
Senior Vice President
and Chief Nuclear Officer
Nuclear Energy Institute
1776 I Street, NW
Suite 400
Washington, DC 20006-3708

Mr. Ray Ganthner
AREVA, Framatome ANP, Inc.
3315 Old Forest Road
P.O. Box 10935
Lynchburg, VA 24506-0935

Mr. Jerald S. Holm
Framatome NP, Inc.
3315 Old Forest Road
P.O. Box 10935
Lynchburg, VA 24506-0935

Masayuki Kambara
1700 K Street, NW
Suite 440
Washington, DC 20006

Masahiko Kaneda
Mitsubishi Heavy Industries, Ltd.
16-5, Konan 2-Chome, Minato-Ku
Tokyo, 108-8215 JAPAN

Shinji Kawanago
Mitsubishi Nuclear Energy Systems, Inc.
1700 K Street, NW
Suite 440
Washington, DC 20006

Dr. Gail H. Marcus
U.S. Department of Energy
Room 5A-143
1000 Independence Avenue, SW
Washington, DC 20585

Mr. John O'Neill
Pillbury, Winthrop, Shaw, Pittman
2300 N Street, NW
Washington, DC 20037

Mr. C. Keith Paulson
PO Box 355, Suite 301W
Pittsburgh, PA 15230

Dr. C. Keith Paulson
Mitsubishi Heavy Industries America, Inc.
4350 Nothorn Pike, Suite 301
Monroe, PA 15146

Yoshinobu Shibata
Vice President
U.S. Marketing and Operations Group
Nuclear Energy Systems Department
Mitsubishi Nuclear Energy Systems, Inc.
c/o C. Keith Paulson
4350 Northern Pike, Suite 301
Monroeville, PA 15146

Mr. Yoshinobu Shibata
Vice President
Mitsubishi Nuclear Energy Systems, Inc.
16-5, Konan 2-Chome
Minato-Ku
Tokyo, Japan

Mr. Robert E. Sweeney
IBEX ESI
4641 Montgomery Avenue
Suite 350
Bethesda, MD 20814

Mr. Ed Wallace
General Manager
Projects
PBMR Pty LTD
P.O. Box 9396
Centurion 0046
Republic of South Africa

Winston & Strawn
1400 L. Street, NW - 12th Floor
Washington, DC 20005-3502

Mr. Gary Wright
Director
Division of Nuclear Facility Safety
Illinois Emergency Management Agency
1035 Outer Park Drive
Springfield, IL 62704

Email

APH@NEI.org (Adrian Heymer)
bob.brown@ge.com (Robert E. Brown)
BrinkmCB@westinghouse.com (Charles Brinkman)
chris.maslak@ge.com (Chris Maslak)
ckpaulson@aol.com (C.K. Paulson)
CumminWE@Westinghouse.com (Edward W. Cummins)
cwaltman@roe.com (C. Waltman)
david.hinds@ge.com (David Hinds)
david.lewis@pillsburylaw.com (David Lewis)
DeLaBarreR@state.gov (R. DeLaBarre)
Dennis.Buschbaum@txu.com (Denny Buschbaum)
dlochbaum@UCSUSA.org (David Lochbaum)
dwoodla1@txu.com (Donald Woodlan)
erg-xl@cox.net (Eddie R. Grant)
frankq@hursttech.com (Frank Quinn)
gcesare@enercon.com (Guy Cesare)
greshaja@westinghouse.com (James Gresham)
james.beard@gene.ge.com (James Beard)
jclong@burkecounty-ga.gov (Jerry C. Long)
jgutierrez@morganlewis.com (Jay M. Gutierrez)
jim.riccio@wdc.greenpeace.org (James Riccio)
jjnesrsta@cpsenergy.com (James J. Nesrsta)
john.o'neil@pillsburylaw.com (John O'Neil)
Joseph.savage@ge.com (Joseph Savage)
Joseph_Hegner@dom.com (Joseph Hegner)
kcrogers@aol.com (K. C. Rogers)
KSutton@morganlewis.com (Kathryn M. Sutton)
lynchs@gao.gov (Sarah Lynch - Meeting Notices Only)
maria.webb@pillsburylaw.com (Maria Webb)
mark.beaumont@wsms.com (Mark Beaumont)
matias.travieso-diaz@pillsburylaw.com (Matias Travieso-Diaz)
media@nei.org (Scott Peterson)
mwetterhahn@winston.com (M. Wetterhahn)
mwl@nei.org (Melanie Lyons)
patriciaL.campbell@ge.com (Patricia L. Campbell)
paul.gaukler@pillsburylaw.com (Paul Gaukler)
PGunter@NIRS.org (Paul Gunter)
pshastings@duke-energy.com (Peter Hastings)
RJB@NEI.org (Russell Bell)
roberta.swain@ge.com (Roberta Swain)
ronald.hagen@eia.doe.gov (Ronald Hagen)
sandra.sloan@areva.com (Sandra Sloan)
sfrantz@morganlewis.com (Stephen P. Frantz)
steven.hucik@ge.com (Steven Hucik)
tansel.selekler@nuclear.energy.gov (Tansel Seleklek)
tom.miller@hq.doe.gov (Tom Miller)
tom.miller@nuclear.energy.gov (Thomas P. Miller)
trsmith@winston.com (Tyson Smith)

vijukrp@westinghouse.com (Ronald P. Vijuk)
waraksre@westinghouse.com (Rosemarie E. Waraks)
whorin@winston.com (W. Horin)