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TOSHIBA CORPORATION

1-1,SHIBAURA 1-CHOME, MINATO-KU TOKYO 105-8001,JAPAN
PHONE:(03)3457-3734
FACSIMILE:(03)5444-9195

No. TOS-CR-DCA-0003
September 12, 2011

10 CFR 52.57(a)

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

Toshiba Corporation

Docket Number 52-044

Subject: Submittal of Responses to NRC Questions Regarding the Probabilistic Risk Assessment in Support of the Toshiba Corporation Application for Renewal of the Design Certification Rule for the U.S. Advanced Boiling Water Reactor

References:

1. Letter to U.S. Nuclear Regulatory Commission (NRC) Document Control Desk from Keisuke Kitsukawa, Technology Executive, Light Water Reactor Systems, Toshiba Corporation Power Systems Company dated October 27, 2010 (ML103080158)
2. Memorandum to Mark Tonacci, Chief – BWR Projects Branch/Division of New Reactor Licensing/Office of New Reactors/U.S. Nuclear Regulatory Commission from David Misenhimer, Project Manager - BWR Projects Branch/Division of New Reactor Licensing/Office of New Reactors/U.S. Nuclear Regulatory Commission dated July 21, 2011 (ML112010645)

On June 23, 2011, representatives of the Toshiba Corporation ("Toshiba") met with the U. S. Nuclear Regulatory Commission (NRC) staff to discuss Toshiba's probabilistic risk assessment (PRA) update in support of its application for renewal of the design certification rule for the U.S. Advanced Boiling Water Reactor dated October 27, 2010 (Reference 1). By memorandum dated July 21, 2011, the NRC staff documented the meeting and a summary of the NRC staff questions (Reference 2).

In support of the design certification renewal (DCR) application, attached are Toshiba's responses to the NRC questions regarding the PRA update.

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Please contact Mr. Robert W. Schrauder, Vice President – Licensing, US ABWR Projects & Technologies, Toshiba America Nuclear Energy Corporation ("TANE") at (704) 548-7640 if you have any questions regarding this information. TANE is a wholly owned U. S. subsidiary of Toshiba. For convenience, Toshiba has authorized TANE to communicate with the NRC with respect to this DCR application.

There are no commitments in this letter.

I declare under penalty of perjury under the laws of the United States of America that the foregoing is true and correct. Executed on September 12, 2011.

Kenji Arai

Kenji Arai
Senior Fellow
Nuclear Energy Systems & Services Division
Toshiba Corporation Power Systems Company

Attachment:

Responses to NRC Questions included with NRC Summary of June 23, 2011 Public Meeting on Toshiba DCR PRA Updates

Contact Information:
Mr. Robert W. Schrauder
Vice President – Licensing, US ABWR Projects & Technologies
Toshiba America Nuclear Energy Corporation
3545 Whitehall Park Drive, Suite 500
Charlotte, NC 28273

RSchrauder@tane.toshiba.com
(704) 548-7640
Facsimile: (704) 548-7701

cc: (electronic copies)

Michael R. Johnson
David B. Matthews
David Misenhimer
Mark Tonacci
U. S. Nuclear Regulatory Commission

Masahiro Horiguchi
Ryuji Iwasaki
Toshiba Corporation

Richard DiSalvo
Robert W. Schrauder
Dale R. Wuokko
Toshiba America Nuclear Energy

Daniel F. Stenger
Hogan Lovells US LLP

Robert Coward
Caroline Schlaseman
Randolph Trench
MPR Associates

M.A. McBurnett
Nuclear Innovation North America

Robert D. Quinn
Westinghouse Electric Company

Responses to NRC Questions included with NRC Summary of June 23, 2011 Public Meeting on Toshiba DCR PRA Updates

1. *Toshiba stated that the risk-beneficial design changes were not considered for further evaluation in the probabilistic risk assessment (PRA), because the design certification results would remain bounding. Though the statement may be true, it may not meet regulatory requirements. Standard Review Plan (SRP) 19.0 states that the PRA should realistically reflect the actual plant design. Also, risk-beneficial changes could significantly affect the results and risk insights (such as the risk profile, accident sequences, importance analyses, important operator actions, and the list of risk-significant systems/components). These would need to be reported in accordance with SRP 19.0 and the interim staff guidance on SRP 19.0 (DC/COL-ISG-03).*

Response:
Of 144 DCD design or process changes for the Toshiba ABWR design, only seven were concluded to be represented in the level of detail in the PRA model that could have an effect on the results. Of these, five were considered risk beneficial, and the remaining two (Evaluation of Common Cause Failures and Ultimate Heat Sink Design) were selected to be considered in the PRA update. The five beneficial changes are: (1) changed RCIC pump turbine from Terry type to integrated monoblock pump and turbine, (2) changed Class 1E I&C power distribution system from three to four safety-related divisions, (3) changed containment strainer design consistent with RG 1.82, Rev. 3, (4) changed the medium-voltage electrical distribution system to a dual voltage system, and (5) AIA (Aircraft Impact Assessment) addition of an alternate feedwater injection capability.

In assessing the highest importance contributors to risk, only RLU001DW (RCIC turbine lubrication system failures), RPM001DW (RCIC pump C001 fails to start), and RTU001DH (RCIC turbine mechanical failure) would be affected by the risk-beneficial changes. However, based on lack of experience with the new RCIC pump-turbine design, justifying a considerably lower failure probability was judged to not be appropriate. Therefore, the effect of the change on the PRA would be minor and was not included.

Toshiba concludes that the design changes for the Toshiba ABWR standard design are sufficiently minor and the Core Damage Frequency (CDF) already so low, that the general PRA insights and conclusions would not change if the risk-beneficial changes were modeled in the PRA. As described during the June 23, 2011 meeting, Toshiba has updated the ABWR PRA in several respects to consider more recent standards and information. For example, the PRA update used the latest initiating event data developed using NUREG/CR-6928 and NUREG/CR-6890; common cause failure (CCF) probabilities were updated; and a Low Power and Shut Down (LPSD) PRA was performed. Toshiba does not believe a full update of the PRA to current guidance in the SRP and ISG is required or necessary. Toshiba's updated PRA evaluates portions of the Level 1 and Level 2 PRA as needed to address changes from the current DCD. Toshiba believes this approach is appropriate for a DCD renewal. Under the criteria of 10 CFR 52.59, a DCD renewal application is generally not required to meet current regulatory requirements (except for amendments to the design certification) or to comply with the Staff's review criteria contained in the SRP. Toshiba's updated PRA will be as described in the June 23, 2011, meeting but Toshiba does not intend to perform a full update of the PRA to meet the current SRP guidance.

2. *Section 19D.8.6 of the original Advanced Boiling Water Reactor (ABWR) Design Control Document (DCD) states an action item that common cause failures (CCFs) of Reactor Building Cooling Water (RBCW), Reactor Service Water (RSW), High Pressure Core Flooder (HPCF), and Residual Heat Removal (RHR) trains will be included in the next revision of the PRA model. South Texas Project (STP) did a similar revision in their reconstituted PRA model. In accordance with the action item in Section 19D.8.6 of the original ABWR DCD, did Toshiba include in the PRA the CCFs of the HPCF and RHR trains?*

   Response:
   The statement at the end of Section 19D.8.6 of the GE Standard Safety Analysis Report (SSAR) is "These CCFs will be added to the plant model in any future revised basic quantification of the ABWR PRA." This was done in the Toshiba PRA update. Specifically, as shown in UTLR-0011, Figure 19D.6-3 pages 1 and 15 (was 19D.6-3a in the GE SSAR) for the High Pressure Core Flooder includes a new basic event HPCFCCFBC "HPCF B AND C TRAIN LEVEL CCF." Similarly, Figure 19D.6-4 pages 1, 11, and 24 (was 19D.6-4a in the GE SSAR) for the RHR Core Flooding Mode includes a new basic event WDCSCCFABC "RHR CORE FLOOD A, B, AND C TRAIN LEVEL CCF." This is also true for RHR Suppression Pool Cooling in Figure 19D.6-6 pages 1, 12, and 21. Reactor Building Cooling and Service Water in Figure 19D.6-14A page 1 includes four CCF Basic Events for RBCW and page 5 includes four RSW CCF Basic Events. Note that both the original and the updated fault trees already include HPCF CCFs for water level, pressure and flow instrument miscalibration.

3. *In accordance with SRP 17.4 and the interim staff guidance on SRP 17.4 (DC/COL-ISG-018), what components were added to the list of risk-significant structures, systems, and components (SSCs) in Appendix 19K as a result of the added CCF events? Were the entire trains added, or just the pumps? How about the other redundant components in the system trains, such as the motor-operated valves, check valves, breakers, flow/pressure transmitters and so on, which would also be risk-significant, because they would have similar risk assessment worth (RAW) and Fussell-Vesely (FV) values as the pumps?*

Response:
The pumps dominate CCF probabilities and were added to Table 19K-4. Room air conditioners, heat exchangers, and injection MOVs for RHR, injection MOVs and F005 valves for HPCF, and strainers for RSW were also added. As noted in the response to #2 above, the update incorporated train level CCF. As a result, there are no RAW or FV values identified for the individual components, and their significance to RAP (and inclusion in Table 19K-4) is based on judgment. As noted in the response to #5 below, Toshiba will add a COL item for an Expert Panel to identify other components for inclusion.

4. *In accordance with SRP 17.4 and the interim staff guidance on SRP 17.4 (DC/COL-ISG-018), were the list of risk-significant SSCs in Appendix 19K updated, as necessary, based on the results of the new low-power/shutdown PRA analysis?*

Response:
The RAW and FV values for SSCs contributing to LPSD risk are shown in Tables 23, 28, and 33 of UTLR-0013 for different SSC availability assumptions. Many risk-significant SSCs are also significant for operating plant conditions, such as the Combustion Turbine Generator, Emergency Diesel Generators, and RHR pump room air conditioners, and are already included in Table 19K-4. Due to the low LPSD CDF of 1.77E-09/yr, which is about 1% of the Level 1 operational CDF, risk significant contributors to LPSD CDF are negligible contributors to overall CDF.

5. *In accordance with SRP 17.4 and the interim staff guidance on SRP 17.4 (DC/COL-ISG-018), the list of risk-significant SSCs in Appendix 19K of the original ABWR DCD is not considered complete due to the limitations of the ABWR PRA model and the methodology used to identify the risk-significant SSCs. For example:*
   - *The ABWR PRA model is 15 years old and may not meet all current PRA standards.*
   - *Some components may not be identified as risk-significant due to the high-level modeling of some CCF events.*
   - *The limited fire, shutdown, level 2 modeling may fail to identify some risk-significant SSCs.*

   All trains of the RHR, HPCF, Reactor Water Cleanup (CUW), RSW, RBCW, and alternating current (AC) systems (rather than single trains as specified in Section 19K.9 of the original ABWR DCD) should be designated for RAP
   - *An expert panel was not used, which could compensate for the limitations of the PRA.*
   - *The original ABWR DCD states that the results in Appendix 19K are a "starting point" for the design reliability assurance program (D-RAP).*

   STP addressed this issue by augmenting the PRA techniques using a deterministic method and an expert panel to effectively populate the list of risk-significant SSCs. The staff found this acceptable. Does Toshiba plan to implement similar techniques (i.e., a deterministic method and use of an expert panel) to update the list of risk-significant SSCs in Appendix 19K or specify a COL information item for the COL applicants to augment the list of risk-significant SSCs?

Response:
Toshiba will add a COL item that requires the COL applicant to update the list of risk-significant SSCs (prior to entering the detailed design and construction phases) by augmenting the PRA techniques using an expert panel, industry operating experience, and a deterministic technique that is based on defense-in-depth.

6. *Please discuss why operational data (such as EPRI report 1003113, "An Analysis of Loss of Decay Heat Removal Trends and Initiating Event Frequencies (1989-2000)") was not used to estimate the frequency of inadvertent RCS draindown events. Inadvertent draindown events are often a key contributor to BWR shutdown risk in operating BWRs, evolutionary, and passive designs. This is discussed in RG 1.200.*

Response:
The LPSD assessment does consider draindown events. The following is an excerpt from Table 3 and the draindown assessment is further discussed in 3.2.4 of UTLR-0013:

Interfacing Systems Drainage (Maintenance Related or Valve Failure)		
CRD system	DCD 19.L.6.3 and Table 19L-6 qualitatively state that multiple valve failures must occur for each path resulting in a very low probability and low flow (Three Paths).	Qualitative or quantitative screening used to exclude paths from evaluation
CUW system	DCD 19L.6.4 and Table 19L-7 discuss 7 paths. There is no quantitative analysis. Based on design features, qualitatively assumes low risk significance.	5 of 7 paths screened out using qualitative or quantitative criteria
RHR System LOCA in connected system (RHR)	DCD 19L.6.5 and Table 19L-8 discuss 9 paths. There is no quantitative analysis. Based on design features, qualitatively assumes low risk significance.	2 of 9 paths screened out using qualitative or quantitative criteria
Over drain of refuel well event	Beyond ending time period of analysis. LOCA isolation already addressed in DCD section 19L.	No change to DCD

Nine paths were identified that did not screen out as negligible probability, and these paths were included in the fault tree model as potential initiating events. For draindown via RHR, only automatic isolation was modeled and manual isolation was not credited for conservatism.

Human Error Probabilities were modeled with specific values, as described on page 22 of UTLR-0013.

Toshiba considers that the design specific treatment of LPSD draindown events is appropriate and more representative than using available operational data for past design BWRs.

7. *PRA Technical Adequacy – Recently, Toshiba has done quite a number of PRA updates. Although it is not required, has Toshiba conducted an independent review and/or PRA peer review? (RG 1.200, ASME PRA Standard, COL/DC-ISG-003 Item 4)*

   Response:
   The PRA update was reviewed independently by two organizations that are separate from the preparing organization, but a peer review per ASME guidance was not performed and, as noted in the question, is not required.

8. *Internal Events (IEs) PRA - Slide 7 "Scope of Completed Updates" does not mention digital instrumentation and controls (I&C). Are there any updates on digital I&C including software failures and common cause failures? This affects PRA Quality.*

   Response:
   As part of the process for determining which design changes should be included in the PRA update, the revised I&C architecture was considered. It was concluded that the level of detail in the PRA model was such that it would not be affected. This does result in the differences in terminology between the PRA and DCD (e.g., the updated PRA still uses terminology EMUX in place of ECF), but the fault tree logic remains representative. The PRA model did already include common cause failures (e.g., APRMS, SLU Bypass Units, and Digital Trip Units).

9. *PRA-based Seismic Margin Analysis - Has Toshiba re-evaluated the plant-level high confidence of low probability of failure (HCLPF) capacity based on the accident sequences generated by the updated IEs PRA? Have the accident sequences generated by the low power and shutdown models been included the plant-level HCLPF capacity evaluation? Refer to RG 1.200, ASME PRA Standard, DC/COL-ISG-020 Section 5.1.*

   Response:
   The Seismic Margins Analysis (SMA) has not been updated by Toshiba, and therefore new HCLPF values have not been generated for the new sequences. The Level 1 PRA update essentially involves the inclusion of the ultimate heat sink system and updated failure data, and Toshiba believes that the plant HCLPF goal of 1.67 times the SSE can be met by requiring the ultimate heat sink components to be designed for the needed seismic fragility. Therefore, a new SMA will not be required.

   The Seismic Margins Analysis was not performed for the low power and shutdown (LPSD) sequences as part of the original ABWR Design Certification, and Toshiba does not plan to carry out any new seismic margins analysis of the LPSD sequences. Components important to LPSD risk are already addressed because of their importance to the at-power PRA.

10. *Internal Flooding PRA -As mentioned on Slides 7 and 9, it is observed that the internal flooding PRA was developed about 15 years ago, and recently Toshiba has developed new frequencies for internal events. Has Toshiba considered updating the internal flood risk assessment with the recent studies and insights on internal flood frequencies? Refer to RG 1.200, ASME PRA Standard.*

Response:
Toshiba does not consider it necessary to update the internal flooding risk assessment.

The results of the flood PRA in the Design Certification PRA demonstrated CDF values significantly below the goal. The low CDF values are attributable to the three divisional design and significant built-in flood barriers that prevent most floods from affecting more than one division. The low CDF values are also attributable to the many features that were added to prevent and mitigate floods in the other areas of the plant. The changes to the Level 1 PRA update are unrelated to the flood, and therefore Toshiba believes that the flood PRA results continue to be low relative to the goal.

11. *Internal Fire PRA - Would ABWR fire PRA include fire propagation modeling? Refer to NUREG/CR-6850.*

    Response:
    Since cable layout details are not available at this stage of the design, the proposed approach for the fire PRA involves use of relatively large fire areas. In these fire areas, a conservative full burn out assumption will be used, and if the CDF results are acceptable, then fire propagation analyses will not be carried out. Such analyses will be considered only if results with the full burn out assumption are not acceptable.

12. *Level 2 PRA - During the presentation, it was observed that Toshiba would not evaluate the large release frequency. Thus, would Toshiba consider addressing the NRC safety goal of conditional containment failure probability as discussed in SECY-90-016 and the Commission's SRM dated June 26, 1990?*

    Response:
    SECY-90-016 identifies that "The overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation where a large release is defined as one that has a potential for causing an offsite early fatality." Also, it states that the "The containment design is to assure that the containment conditional failure probability is less than one in ten when weighted over credible core damage sequences." The SRM agrees with these goals, although it notes the definition of a large release requires further work and the 0.1 conditional containment failure probability is not a requirement in itself.

    As shown in the PRA update, the goal for the mean frequency of a large release is met, since the CDF for internal events is about 1 in 10,000,000 per year. Regarding the containment conditional failure probability, scenarios with normal containment leakage or proper operation of the COPS rupture disks are not considered failures. UTLR-0016 reports that about 87% of the fractional contribution to source term results comes from normal containment leakage. For scenarios where containment is challenged, the rupture disks have a high reliability, except for accident Class II where a rupture disk failure is part of the scenario but has a negligible probability. Based on this information, the conditional containment failure probability is much less than one in ten, and therefore the goal identified in SECY-90-016 is met.

13. *For Severe Accident Management: Was Toshiba considering providing information to the BWR Owners Group regarding the role of the design's severe accident mitigation features in addressing*

long-term station blackout scenarios, and how this information might be used to improve the Severe Accident Guidelines? For example, did Toshiba consider using the AC-Independent Water Addition System to assure long-term debris cooling after suppression pool water provided to the Lower Drywell through the passive flooder has evaporated? Severe accident management is not governed by any specific regulations, but is governed by voluntary initiatives put in place by Nuclear Energy Institute, the Owners Groups, and the individual utilities.

Response:
Following the meeting held on June 23, 2011, regarding the ABWR probabilistic risk assessment for the Design Certification Renewal Application, the NRC Near-Term Task Force reviewing the Fukushima Dai-Ichi accident issued recommendations for Commission approval addressing severe accident management and the strengthening of long-term station blackout mitigation capability at all operating and new reactors. Therefore, although Toshiba is not a member of the BWR Owners Group, the collective industry's response to the Near-Term Task Force recommendation is expected to include an in-depth review of the different types of severe accident mitigation design features available to improve the response to long-term station blackout scenarios.

14. *Is Toshiba planning to use the state-of-the-art severe accident tools (such as MAAP4 and MACCS2) to compute realistic source terms in a SAMDA analysis? SAMDA is required for design certification applicants under 10 CFR 51.55. New and significant information regarding risk has been developed using the state-of-the-art tools mentioned above.*

Response:
Toshiba does not plan to update the severe accident analysis. Toshiba concluded that calculating updated generic source terms was not needed because the conclusions would not change. This determination is based on the low ABWR CDF and conditional containment failure probability, our confirmation that the dose probabilities from the updated PRA were bounded by those of the original SAMDA evaluation, and the extremely low required cost to justify a SAMDA-based improvement.