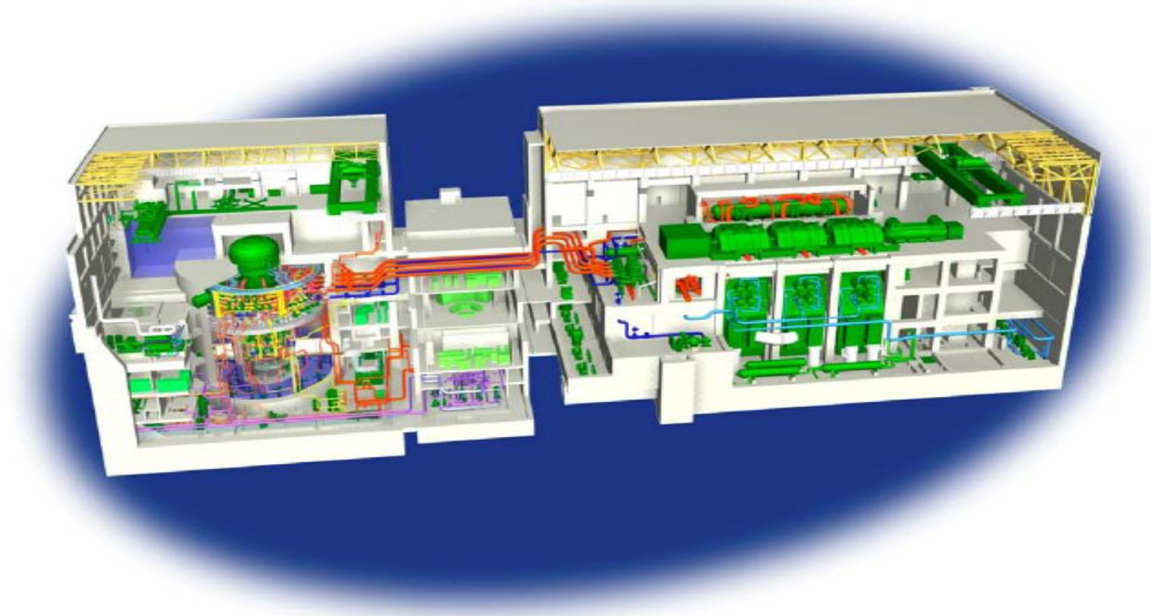
Toshiba ABWR
Design Certification Renewal



PRA Update June 23, 2011

Desired Outcomes

- The NRC has an understanding of what Toshiba has done and plans to do to assure the conclusions of the PRA done to support Appendix A to 10 CFR 52: Design Certification Rule for the U.S. Advanced Boiling Water Reactor remain valid for the renewal application; and
- Toshiba has an understanding of NRC feedback on the scope and content of the PRA update

Overview

- Background
- Toshiba PRA Update
  - Completed updates
  - Other PRA areas and future plans
- Conclusions

Background

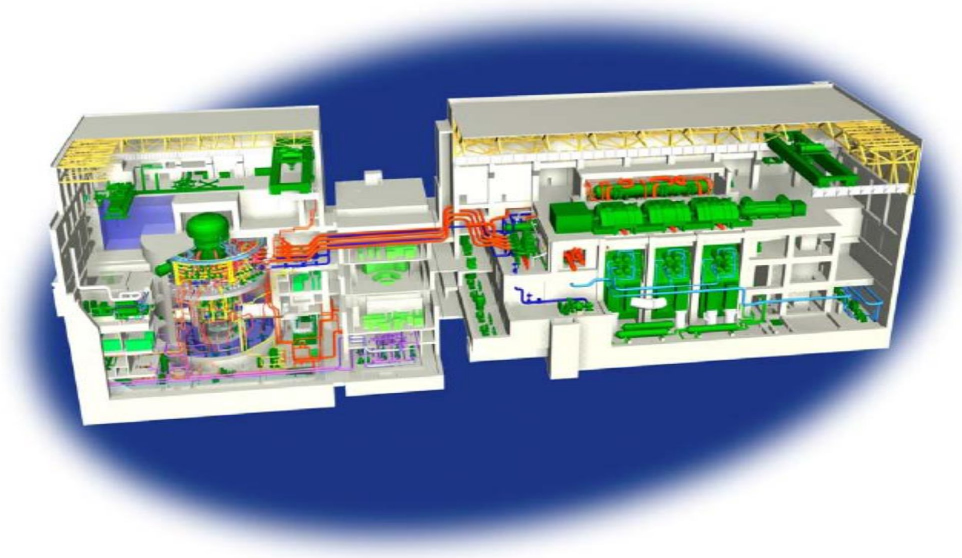
- ABWR Design Certification (DC) based on Design Control Document Revision 4 (DCDR4)
  - ABWR was designed to meet risk goals
  - DCDR4 reported very low CDF and risk values

- The DCDR4 ABWR PRA included the following analyses:
  - Internal Events (Full Power) PRA
  - Shutdown Risk Study
  - Fire Risk Analysis
  - Internal Flood PRA
  - Seismic Margins Analysis
  - Level 2 PRA and supporting analyses
  - Level 3 PRA

Background (continued)

- On Oct 27, 2010, Toshiba submitted application for Design Certification Renewal (DCR)
- Toshiba's renewal application includes some changes from the approved DCD
- Toshiba evaluated portions of the Level 1 and 2 PRA as needed to address changes
- This PRA update addressed Toshiba DCD changes and verified that risk estimates and insights remain valid
- Toshiba incorporated the PRA update information (e.g., Chapter 19) in its renewal application
- Toshiba provided the supporting PRA update documentation to the NRC

Toshiba PRA Update:
Completed Updates



Scope of Completed Updates

- Modeled Toshiba's Standard ABWR plant configuration
- Developed new Initiating Event Frequencies
- Developed new Component Reliability Data
- Modeled Common Cause Failure (CCF) of pumps for decay heat removal (DHR)
- Updated risk-significant human actions
- Updated contribution of each source term category to the Level 2 release frequency
- Developed Low Power and Shut Down PRA
- Replaced/removed non-Toshiba proprietary information

Toshiba Standard ABWR Configuration

- Changes from DCDR4 were evaluated to determine if there is a change to the PRA model

- Changes were qualitatively evaluated for risk:
  – Risk-beneficial changes not considered for further evaluation: previous ABWR PRA remains bounding
  – Potentially risk-adverse changes incorporated into the updated PRA model to quantify CDF and Large Release Frequency (LRF)
    - Only the Ultimate Heat Sink change from a spray pond to a basin-type cooling tower met this criterion

Initiating Event Frequencies

- Over 15 years have elapsed since original PRA and considerable new operating experience is available

- PRA update used latest initiating event data developed using NUREG/CR-6928 and NUREG/CR-6890

Component Reliability Data

- Only failure probabilities were changed in most cases
- In a few cases, failure modes were changed due to different methods of modeling some types of components
- 20-hour mission times were increased to 24-hour mission times

Common Cause Failures

- The CCF probabilities were updated due to new failure frequency data
- Reactor Building Cooling Water (RBCW) and Reactor Service Water (RSW) pump CCFs were added
- Most significant effect on CDF is due to the CCFs among all three divisions of RBCW/RSW leading to failure of both HPCF and RHR Core Flooding, given loss of all RBCW/RSW divisions. All other CCFs have relatively little effect on CDF.

Risk-Significant Human Actions

- Limited to non-THERP-based actions
- Two U.S. BWR PRAs were examined to find similar operator actions
- Revised those actions with five highest Risk Achievement Worth (RAW) values
  - Initiation and control of condensate injection after reactor vessel depressurization
  - Control of feedwater injection during a non-isolation event
  - Control of feedwater injection during an isolation event
  - Control of water level and pressure during an anticipated transient without scram (ATWS) event
  - Initiation of the high pressure core flooder (HPCF) on low water level

PRA Update Internal Events Results

- Core Damage Frequency (CDF) < 1E-7/rx-yr

- Level 2 analysis concluded that Normal Containment Leakage (NCL) continues to be the majority of the release frequency

Conclusion:
PRA update CDF results are low and consistent with the DCDR4 PRA. The original conclusions from the ABWR Certification PRA remain applicable.

Low Power and Shut Down (LPSD)

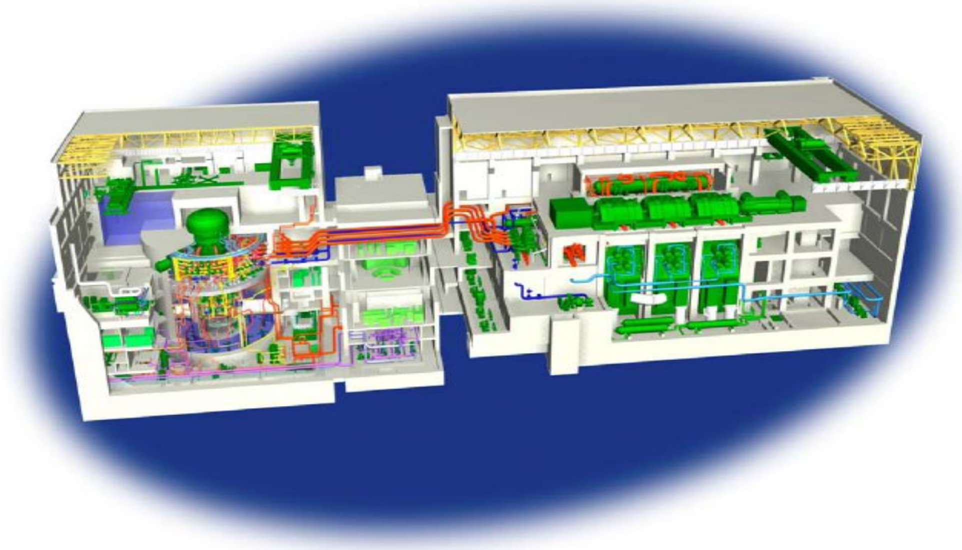
- Guidance, methodology, and expectations for LPSD have matured since DCDR4 PRA
- LPSD risk can be contributor to overall risk
- Toshiba performed an updated LPSD PRA
- Modes of operation considered
  - Mode 3 (hot shutdown) after the entry into RHR cooling
  - Mode 4 (cold shutdown)
  - Mode 5 (refueling)
  - Startup not evaluated since considered low risk due to very low decay heat levels late in shutdown
- Effects considered
  - Decay heat removal
  - Inventory control
  - Reactivity control
  - Electrical power (as subset of inventory control and decay heat removal)

Low Power and Shut Down (LPSD) (continued)

- Defined Plant Operating State (POS) characteristics
- Identified and analyzed Shutdown Initiating Events
- Analyzed Accident Sequences and Event Trees
- Developed systems models
- Quantified results:
  - CDF (assuming all systems in one Division unavailable): 1.77 E-9/rx-yr
  - CDF (realistic, assuming nominal unavailabilities): 4.21 E-10/rx-yr

Conclusion:
The Toshiba LPSD PRA is adequate for DC Renewal.

Toshiba PRA Update:
Other PRA Areas and Future Plans



Seismic

- PRA-Based Seismic Margin Assessment (SMA) consistent with guidance of NRC ISG-20
- SMA figure of merit is HCLPF (site independent): acceleration for which plant has High Confidence (95%) of Low Probability (<5%) of Failure
- NRC Review Level Earthquake (RLE) Goal:

  RLE = 1.67*SSE
- ABWR designed for a safe shutdown earthquake (SSE) level of 0.3g (➔ RLE = 0.5g)
- ABWR HCLPF > 0.5g

Conclusion:
The original seismic margins analysis is adequate for DC Renewal.

Flooding

- Internal flooding analyzed for all postulated flood sources
  - Screening analysis: Only floods in turbine building, control building, and reactor building are of potential concern
  - ABWR's three divisional design minimizes consequences of a divisional flood
  - Features incorporated to mitigate Turbine Building and Control Building floods based on feedback from original flood PRA include:
    - Flood detectors
    - Pump trips/valve isolations on flood detection
  - Internal flood CDF is very low
- External flooding to be analyzed by COL applicant

Conclusion:
The original Internal Flood PRA is adequate for DC Renewal.

Original ABWR DC Fire Risk Analysis

- Used EPRI Fire Induced Vulnerability Evaluation (FIVE) methodology
  - FIVE provides prescriptive procedures
  - Evaluated plant fire areas and frequencies accounting for combustible material contained within
  - Internal event fault and event trees used to calculate bounding CDF
  - Assumed fire disables all systems in the area, but fire does not propagate beyond divisional fire barriers
- Fires in each area analyzed showed CDF risks less than the screening value of 1.0E-6/rx-yr
- Much progress in fire risk evaluation in recent years
- NUREG/CR-6850 is widely accepted methodology

Toshiba Plan for Updating Fire Risk Analysis

- Perform a conservative and simplified fire PRA using NUREG/CR-6850 methodology, consistent with level of details available at this stage of the design
- Use EPRI's FRANX code for quantification
- Steps for performing the Fire PRA are as follows:
  - Plant Partitioning: Use the broad-based high level plant partitioning from the FIVE methodology as the starting point. Make finer breakdown of specific fire zones if needed
  - Equipment Selection:
    - Components from internal events PRA
    - Components from available fire hazard analysis
    - Components from a Multiple Spurious Operations Review
    - Instruments needed to support modeled operator actions

Toshiba Plan for Updating Fire Risk Analysis (continued)

- Steps for performing the Fire PRA (continued):
  - Cable Selection: Plant-specific data not available; use available cable data
  - Qualitative Screening: Screen out areas that cannot contribute to fire risk
  - Plant Response Model: Develop a Plant Response Model in FRANX code
  - Ignition Frequencies: Use frequencies based on NUREG/CR- 6850 (Supplement 1)
  - Quantitative Screening: No screening to be performed
  - Fire Scenario Selection/Detailed Fire Modeling:
    - Initial approach is to consider large fire compartments and full burn-out
    - If results are not acceptable, finer compartmentalization may be needed
    - Perform conservative main control room analysis with credit for fire suppression and remote shutdown panel

Toshiba Plan for Updating Fire Risk Analysis (continued)

- Steps for performing the Fire PRA (continued):
  - Circuit Failure Analysis: Use conservative circuit failure probabilities initially and refine evaluations as needed
  - Fire Human Reliability Analysis (HRA): Plant-specific information not available; make conservative estimates
  - Fire PRA Quantification: Quantify results using the FRANX and CAFTA models
  - Uncertainty and Sensitivity Analysis: Propagate numerical uncertainties and perform sensitivity studies

Conclusion:
Toshiba will update the Fire PRA as described.

Level 2 & 3 PRA and Severe Accident Analysis

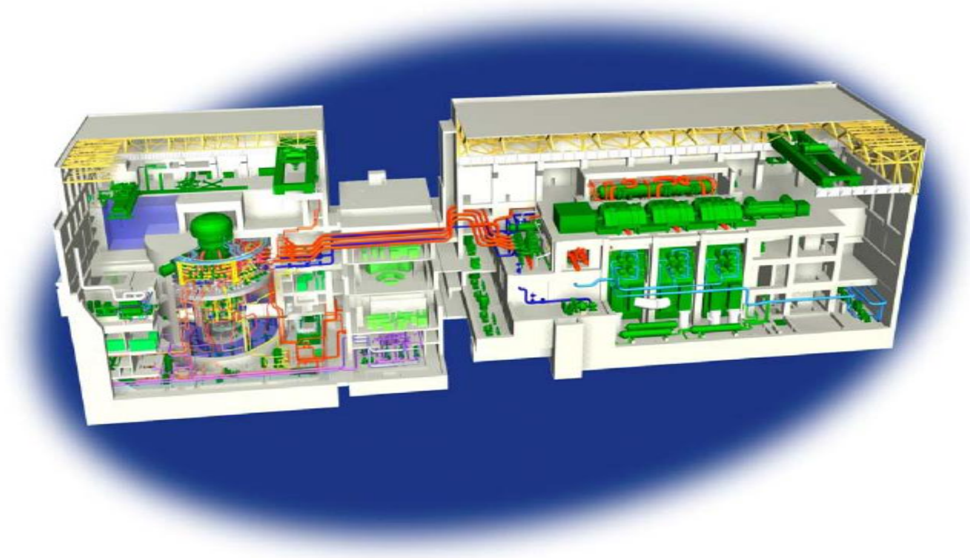
- Level 2 & 3 PRA and extensive severe accident analyses were performed for the original DC
- Following NRC review, to reduce radioactive release frequency, features were added to the original design, such as:
  - Passive Lower Drywell Flooder
  - Containment Overpressure Protection System (COPS)
- All results demonstrate extremely low risk from radioactive release following a core melt event
- All NRC severe accident goals are met with substantial margin and the design is judged to be safe even after accounting for uncertainty

Level 2 & 3 PRA and Severe Accident Analysis (continued)

- No major advancement has been made in severe accident tools to significantly reduce the uncertainty
- Assessment of Severe Accident Mitigation Design Alternatives (SAMDA): PRA update results yield total cumulative exposure risk for Toshiba's ABWR design lower than the original total cumulative exposure risk

Conclusion:
No update of Level 2 & 3 PRA or Severe Accident Analyses is needed.

Conclusions



Conclusions

- PRA update performed to date, with additional fire PRA work, is sufficient. No update required for flooding, seismic margins, or deterministic severe accident analysis.
- Internal Events Core Damage Frequency and Large Release Frequency – slight decrease
- SAMDA – conclusions unchanged
- Overall conclusions unchanged – risk considerably below risk goals

Conclusion:
PRA Updates already carried out, plus the planned Fire PRA update, are adequate for DC Renewal.

Abbreviations

- ABWR Advanced Boiling Water Reactor
- ATWS Anticipated Transient Without Scram
- BWR Boiling Water Reactor
- CCF Common Cause Failure
- CDF Core Damage Frequency
- COPS Containment Overpressure Protection System
- DC Design Certification
- DCD Design Control Document
- DCDR4 Design Control Document (Revision 4)
- DCR Design Certification Renewal
- DHR Decay Heat Removal
- EPRI Electric Power Research Institute
- FIVE Fire Induced Vulnerability Evaluation
- HCLPF High Confidence of Low Probability of Failure
- HPCF High Pressure Core Flooder (System)
- HRA Human Reliability Analysis

Abbreviations (Continued)

- ISG — Interim Staff Guidance
- LPSD — Low Power and Shut Down
- LRF — Large Release Frequency
- NCL — Normal Containment Leakage
- NRC — Nuclear Regulatory Commission
- POS — Plant Operating State
- PRA — Probabilistic Risk Assessment
- RAW — Risk Achievement Worth
- RBCW — Reactor Building Cooling Water (System)
- RHR — Residual Heat Removal (System)
- RLE — Reference Level Earthquake
- RSW — Reactor Service Water (System)
- SAMDA — Severe Accident Mitigation Design Alternatives
- SSE — Safe Shutdown Earthquake
- THERP — Technique for Human Error Rate Prediction

TOSHIBA

Leading Innovation