UNITED STATES - ADVANCED PRESSURIZED WATER REACTOR COMPUTER PROGRAMS AND PIPING DESCRIBED IN DESIGN CONTROL DOCUMENT SECTIONS 3.9.1 AND 3.12 AUDIT REPORT

NRC Audit Staff:

- Dennis Galvin, Project Manager, NRO/DNRL/LB2
- Robert Hsu, Sr. Mechanical Engineer, NRO/DE/EMB
- John Wu, Mechanical Engineer, NRO/DE/EMB
- Yiu Law, Mechanical Engineer, NRO/DE/EMB
- Terri Spicher, Mechanical Engineer, NRO/DE/EMB
- Gary Swearingen, PNNL, NRC Contract Technical Reviewer

1.0 SUMMARY

An audit was conducted by the U.S. Nuclear Regulatory Commission (NRC) staff at the Mitsubishi Nuclear Energy Systems office in Arlington, Virginia from August 22-30, 2011. The NRC staff conducted the audit in accordance with the NRC Office of New Reactors (NRO) Office Instruction NRO-REG-108, "Regulatory Audits." The plan for this audit was documented and can be found in the Agencywide Document Access and Management System (ADAMS) under accession number ML112150252, dated August 16, 2011.

The purpose of the audit was to review the computer programs that are used for static, dynamic, and hydraulic transient analyses of the United States - Advanced Pressurized Water Reactor (US-APWR) design. The audit staff reviewed these programs in relation to the requirements of Title 10 of the *Code of Federal Regulations*(10 CFR) Part 50 Appendix B, American Society of Mechanical Engineers (ASME) NQA-1 Code, the methodology and criteria described in the ASME Code Section III and Design Control Document (DCD) Revision 3 in support of the US-APWR Design Certification. In addition, the review included piping and support design specification, stress analysis methodology and fatigue evaluation reports, and sample piping runs to verify and validate the methodology as documented in the DCD.

2.0 AUDIT SCOPE

The scope of the NRC staff's on-site audit included the following:

A. Computer Programs

The primary scope of this audit is to review computer programs that are used for static, dynamic, and hydraulic transient analyses as they relate to the US-APWR components and piping design. The audit will verify that the program performed the safety analysis in accordance with requirements of ASME Section III Code. The review includes, but is not limited to, computer programs listed in Table 1 of the audit plan.

The areas of review are shown in the list below:

1. Review individual program documents that describe author, source code, executable file(s), input deck(s), dated version, user's manual and theoretical formulations.

2. Review individual program verification reports.

3. Review individual program flow chart logic.

4. Review individual program verification and validation(V&V) benchmark package.

5. Review individual program input and output data and program limitations.

6. Perform test runs on individual program as required.

7. Review Quality Assurance (QA) procedure for control and maintenance of program.

8. Confirm that the analyses for ASME Section III, Class 1 components and piping will include the environmental effects in accordance with Regulatory Guide 1.207, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors."

B. Piping Review

During this portion of the audit, the NRC staff reviewed Mitsubishi Heavy Industries, Ltd. (MHI) documents related to:

1. ASME piping and support design specifications.

2. Piping stress analysis and fatigue reports including environmental effects and all input related supporting documents (e.g. design loads and transients, piping and instrumentation diagram, support drawings and information, computer codes information, and stress calculations).

3. The audit scope regarding the reactor coolant loop (RCL) piping, pressurizer (PZR) surge line, and main steam (MS) piping inside the containment vessel (CV) are shown in Figure 1, "Scope of NRC Audit for RCL Piping," Figure 2, "Scope of NRC Audit for PZR Surge Line," and Figure 3, "Scope of NRC Audit for MS Piping inside CV," of the audit plan respectively.

- The design inputs (supporting documents), as shown in Figure 1 through Figure 3 of the audit plan, are within the scope of the audit as reference information.
- Seismic analysis for RCL and seismic input for other piping systems are out of scope of the audit per MHI's request.

4. Environmental assisted fatigue evaluation methodology.

3.0 REGULATORY BASIS

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 3.9.1, "Special Topics for Mechanical Components" and Section 3.12, "ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and their Associated Supports," of NUREG-0800, the Standard Review Plan (SRP), and are summarized below.

- 10 CFR 50, Appendix A, General Design Criterion 1, which requires, in part, that components important to safety be designed, fabricated, erected, and, tested to quality standards commensurate with the importance of the safety functions to be performed.

- 10 CFR 50 Appendix B, as it relates to quality of design control.

- 10 CFR 50.55a (c), as to the establishment of minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of reactor coolant pressure boundary(RCPB) components and other safety-related fluid systems of boiling- and pressurized-water reactor nuclear power plants by compliance with appropriate editions of published industry codes and standards.

4.0 OBSERVATIONS AND RESULTS - COMPUTER PROGRAMS

The review confirmed that the MHI's house and commercial codes listed in Enclosure 3,"Documents Made Available to the NRC Staff," which are used for the US-APWR design process, are compliant with 10 CFR 50, Appendix B and ASME NQA-1-1994.

Documentation for the house and commercial codes identified in the audit plan was available and reviewed, with the exception of the RELAP5 commercial code, which had already been audited in October 2010. The audit report for RELAP5 is available in ADAMS under accession number ML112630530. One additional house code US CHERRY was added during the course of the audit with many questions answered during the course of the audit. The information listed below was reviewed by the staff for each computer program.

- Individual program documentation.

- Individual program flow charts.

- Individual program verification reports.

- Individual program V&V benchmark packages.

- Individual program input and output data and program limitations.
- QA procedure for control and maintenance of computer programs

A list of program flowcharts, provided to the staff, is included in Enclosure 3. The flowcharts themselves are MHI proprietary information and are not available to the public. Enclosure 3 also contains tables that identify by computer code, the MHI QA documents for software control and design applications made available for audit.

Several house code error reports (e.g. NX NASTRAN) were reviewed for impact analysis, change control and implementation with no issues identified. The audit team requested that MHI provide copies of approval pages of translated documents (cover page and revision history) for the staff's review.

Test runs were satisfactorily conducted for the following nine house codes selected by the audit team: MCPEVALPRI, MCPEVALSI, EB3500, EVALPRI, EVALSEFAV, EVALIRAMJ, RIGHT, SABRINA. The audit confirmed that the contents of the input and output files, chosen at random by the audit team for each house code and already stored in the folder were the same as described in the verification report before the test run. After the test run, the audit confirmed that the newly produced output file matched the output file stored in the folder beforehand. All questions were answered satisfactorily before the end of the audit. Based on the total of nine test cases as a selected sample, no issues were identified.

The audit team asked about the QA controls in place for the Excel spreadsheet's use. MHI has responded that it uses Excel to calculate the force and moments applied on the lumped mass stick model. It is a simple calculation (a multiplication). After calculation by Excel, MHI has confirmed that the highest force value is correct by comparing with the hand calculation results.

The audit team noted that for several of the house codes, more recent revisions are effective. Initially this was confusing, as MHI was presenting a snapshot of its program records even as the codes are evolving. It was suggested that code current revision status be clearly communicated to future audit teams.

MHI described the hand calculation to convert GOTHIC results into stress analysis input. In the next revision of MUAP-09002, "Summary of Seismic and Accident Load Conditions for Primary Components and Piping," MHI will clarify the Excel spreadsheet usage (i.e., hand calculation) for the accident analysis.

MHI in-house computer codes are categorized into "new code" and "old code". For the "new" codes, which were developed for design of the ASME code-components, the development process, documentation and the quality assurance program were reviewed and found to be fully in compliant with ASME NQA-1 code requirements with appropriate verification and validation documentation. On the other hand, the "old" codes, which were previously developed for the design of Japanese domestic components, were later verified and validated for use in the design of ASME code-components. Many of the earlier supporting documents (e.g. software development plan) for the "old" codes were not available and, according to MHI, do not exist. For each of these "old" codes, MHI has completed the required verification and validation documentation. No technical issues were identified regarding these in-house computer codes in this category during the audit.

5.0 OBSERVATIONS AND RESULTS – PIPING AND SUPPORTS

The staff reviewed the documents listed in Enclosure 3 of this report, in whole or in part, and interviewed the applicant's technical staff.

During the Class 1 piping design specification review, the staff noted that the PZR surge nozzle, accumulator tank nozzle, residual heat removal (RHR) suction nozzle and charging nozzle are designed with thermal sleeves and the PZR spray is designed with a scoop. The thermal sleeves and scoops are not RCPB components. The staff noted that the attachment weld of thermal sleeves will be considered as part of the RCPB component. The applicant's technical staff concurred that the attachment weld is part of the RCPB and this issue will be addressed during inspection, test, analysis, and acceptance criteria (ITAAC) closure activity to verify the thermal fatigue analysis of the weld. The staff finds this acceptable.

The staff identified that the RHR suction and return lines in Figure 3-8,Figure 3-9,Figure 3-10, and Figure 3-11 of "Class 1 Piping ASME Design Specification (Excluding Reactor Coolant Loop Piping)" indicate that a single isolation valve is used. However, DCD 3.12 states that double isolation valves are used for eliminating thermal oscillation. The staff asked the applicant to clarify the deviation between design specification and DCD. MHI's technical staff agreed to address this issue and will address this deviation in its response to Request for Additional Information (RAI) 3.12-29.

The staff reviewed the applicant's design basis document (Class 1 piping design specification) which indicated that the surge line stratification was analyzed during heatup with system temperature difference (PZR temperature minus reactor coolant system temperature) ΔT of 145°F in order to qualify the requirements for environment fatigue analysis. The staff noted that the above limitation has to be stated or identified in the DCD to ensure the licensed document to be consistent with design document. The applicant concurred that the above specified limitation will be incorporated in its response to RAI 3.12-25.

The staff reviewed the steam generator (SG) lower lateral supports drawing and computer model in the analysis. The staff noted that this support is a non-linear type support and identified that the applicant stated that the RCL dynamic analysis is performed using time-history direct integration, the time-history modal, or response spectra methods in DCD Appendix 3C. The staff noted that the SG lower lateral supports were modeled as linear type supports in the soil structure interaction model. The staff also noted that time-history direction integration should be used to address non-linear seismic analysis. The staff asked the applicant to clarify the methodology used for the RCL safe shutdown earthquake (SSE) analysis. MHI's technical staff concurred that time-history direct integration will be used for the RCL SSE final analysis and the DCD will be revised to clarify its position. MHI will provide the revised DCD and clarification in its response to RAI-3.12-26.

The staff reviewed the PZR surge Line stress analysis report. In this analysis report, MHI did consider four design basis pipe break cases. The PZR surge line seismic analysis used individual support motion methods with three percent damping for conservatism. The staff acknowledged that the seismic input (response spectra) has not been approved by the NRC staff yet. The staff reviewed the methodology and all other input and finds this acceptable.

The staff reviewed "PIPESTRESS Version 3.6.2 Software results of installation tests." In the test report, the applicant used PIPESTRESS computer code to perform 14 benchmark problems from NUREG/CR-1677, "Piping Benchmark Problems," and NUREG/CR-6414, "Piping Benchmark Problems for the Applicant AP600 Standardized Plant." The staff concluded that the PIPESTRESS computer program is benchmarked and acceptable. The applicant also performed a three-dimensional finite element (FE) analysis for the PZR surge line thermal stratification to demonstrate that the ASME code piping equation for the thermal qualification is more conservative than the actual FE thermal stress results.

The staff reviewed the stress calculation "Design Accident Loads for Reactor Coolant Loop Components and Component Support on Service Level C and D." The staff asked MHI to justify the convergence of the integration time interval of 0.0001 second. MHI's technical staff stated that a study for the convergence was performed to ensure the result is acceptable and that the 10 percent result difference acceptance criterion was used to determine the convergence of the time-history integration. The applicant using a 10 percent difference as the convergence tolerance for the time history, is consistent with American Society of Civil Engineers (ASCE) Standard ASCE 4, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary," Section 3.2.2.1(c). The staff finds this approach to be acceptable.

The staff also reviewed the applicant's environmental fatigue evaluation package, including RCL piping survey for environmental fatigue evaluation. The applicant performed an environmental assessment to determine which components have exceedance in fatigue cumulative usage factor (CUF). The applicant will perform its refined CUF evaluation for those components exceeding a CUF limit of 1.0. The staff finds this approach to be acceptable.

The staff also reviewed the supporting document; "Design Specification for Safety Related Pipe Supports inside the PCCV for Decoupled Analysis Method." The staff identified that the support cannot be welded at the elbow location. MHI clarified that this welding location will not be used for the US-APWR. The staff considered this acceptable, and it can be verified during the ITAAC closure.

6.0 CONCLUSION

The NRC staff determined that the audited materials supported the goals of the audit. Based on the audit discussions, MHI has submitted revised documentation to address the staff's concerns related to Section 3.9.1 and Section 3.12. The review of the submitted information is outside the scope of this audit report.