May 06, 2009

Mr. Yoshiki Ogata, General Manager
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SUBJECT: FINAL SAFETY EVALUATION FOR TOPICAL REPORT MUAP-07012-P AND MUAP-07012-NP, REVISION 2 "LOCA MASS AND ENERGY RELEASE ANALYSIS CODE APPLICABILITY REPORT FOR US-APWR"

Dear Mr. Ogata:

On July 20, 2007, Mitsubishi Heavy Industries, Ltd. (MHI) submitted Topical Report MUAP-07012-P and MUAP-07012-NP, Revision 0, "LOCA Mass and Energy Release Analysis Code Applicability Report for US-APWR" to the staff for review. On January 26, 2009, a U.S. Nuclear Regulatory Commission (NRC) draft safety evaluation (SE), regarding our approval of MUAP-07012, was provided for your review and comments. By letter dated March 3, 2009, (MHI REF: UAP-HF-09069) MHI provided four comments on the draft SE.

The NRC staff has reviewed Topical Report MUAP-07012, Revision 2 and has found that this report is acceptable for use by MHI for United States-Advanced Pressurized Water Reactor design certification activities to the extent specified and under the limitations delineated in the topical report and in the enclosed SE. The SE defines the basis for acceptance of the topical report. Our acceptance applies only to the material provided in the subject topical report. We do not intend to repeat our review of the acceptable material described in the topical report.

We request that MHI publish an accepted version of this topical report. The accepted version of this topical report shall incorporate this letter and the enclosed SE after the title page. Also, the accepted version must include NRC requests for additional information and your responses.

The accepted version shall include an “-A-“ (designating accepted) following the topical report identification symbol. If future changes to the NRC’s regulatory requirements affect the acceptability of this topical report, MHI will be expected to revise the topical report appropriately, or justify its continued use.

Sincerely,

/RA/

Carey Bickett, Acting Chief
US-APWR Projects Branch
Division of New Reactor Licensing
Office of New Reactors

Docket No. 52-021

Enclosure:
As stated

The accepted version shall include an “-A-“ (designating accepted) following the topical report identification symbol. If future changes to the NRC’s regulatory requirements affect the acceptability of this topical report, MHI will be expected to revise the topical report appropriately, or justify its continued use.

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FINAL SAFETY EVALUATION BY THE OFFICE OF NEW REACTORS

TOPICAL REPORT MUAP-07012-P R2

"LOCA MASS AND ENERGY RELEASE ANALYSIS CODE APPLICABILITY REPORT FOR US-APWR"

MITSUBISHI HEAVY INDUSTRIES, LTD.

DOCKET NO. 50-021

1.0 INTRODUCTION

Topical Report MUAP-07012 (Reference 1) describes a portion of the methodology which Mitsubishi Heavy Industries (MHI) uses in the evaluation of the ability of the United States-Advanced Power Water Reactor (US-APWR) containment building to withstand the over-pressure and temperature that would be produced following a large loss of coolant accident (LOCA). Specifically the topical report presents methodology for calculating the steam, water and nitrogen releases from postulated reactor coolant piping breaks. These releases are then utilized in the model of the containment building to evaluate the increase in internal pressure and temperature. The containment building model is not part of the MUAP-07012 review. During the review, the U.S. Nuclear Regulatory Commission (NRC) staff issued a number of requests for additional information (RAIs). MHI's responses to these RAIs are included in Reference 1.

The methodology in MUAP-07012 is based on that which has been previously approved by the NRC staff. In particular, for the initial periods of discharge from the reactor, computer codes developed by Westinghouse (Reference 2), which have previously been approved by the staff, are utilized. For the final period of the discharge, MHI methodology uses the GOTHIC computer code which was developed by the Electric Power Research Institute and has also been utilized in approved containment evaluations (Reference 3). Revisions to the approved methodology were necessary because of special features of the US-APWR. The special features which would impact the discharge to the containment building during a LOCA include the advanced accumulators, the heavy neutron reflector within the reactor vessel and the in-containment refueling water storage pit (RWSP).

2.0 REGULATORY EVALUATION

General Design Criterion (GCD) 50 of the NRC's regulations requires that the containment structure and the containment heat removal systems be designed so that the containment structure can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from LOCA.

The mass flow rate of fluid from a postulated primary system piping break and the energy contained within that fluid are key factors which must be included in the evaluation of the adequacy of the containment structure. In reviewing MUAP-07012, the staff utilized the guidance within Standard Review Plan (SRP) 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)." Adherence to the guidance of SRP 6.2.1.3 ensures that the requirements of GDC 50 will be met for this area of the review.

Enclosure

3.0 TECHNICAL EVALUATION

Following a LOCA caused by a large piping rupture of reactor system piping, the sequence of events including the recovery of reactor core cooling has been divided into four phases namely:

1. The blowdown period: During this period coolant is rapidly ejected from the reactor system into the containment building and the reactor core temperature increases because of the lack of cooling.

2. The refill period: During this period the reactor vessel lower plenum is refilled with emergency coolant to the bottom of the core. Little additional mass or energy release to the containment will occur and the core temperature will continue to increase.

3. The reflood period: During this period the core is recovered with water and cooled below the initial temperature that occurred before the start of the accident. As the heated fuel comes in contact with the rising water level, experiments of simulated core reflooding have shown that a large fraction of the emergency coolant which reenters the bottom of the core will be carried out of the top of the core along with the exiting steam as water droplets. Depending on the break location, these droplets will be heated by the primary system piping and the stored energy within the steam generators to be turned into steam before flowing into the containment building.

4. The post-reflood or long-term cooling period: Evaluations for this period should consider the core decay heat and the removal of the remaining sensible heat from the reactor system and steam generators. During the early portion of this period, the core will still be boiling at a rapid rate. Void fraction evaluations have shown that a two-phase mixture will extend above the core so that additional liquid will enter the steam generator tubes. The flow of a two-phase mixture through the reactor system will remove the remaining sensible heat and will provide for additional steam formation to be added to the containment building.

As discussed in Reference 2, the SATAN-VI computer code is used to determine the mass and energy addition rates to the containment during the blowdown phase of the accident. To obtain a conservatively high energy release from the core during the blowdown period, a core average channel is modeled assuming nucleate boiling for an extended period to maximize the energy release. By this means, a major portion of the available energy is removed from the core during the blowdown. With these assumptions, the core transfers more heat to the containment than would be predicted by a calculation suitable for emergency core cooling performance evaluation. This additional energy release from the core will increase the calculated containment pressure and therefore provides a margin of conservatism in the analyses.

MHI will assume a refill period of zero duration. This is the assumption which is recommended in SRP 6.2.1.3. The assumption is conservative since steaming to the containment would be reduced as the lower plenum filled. It is, therefore, conservative for containment analysis to assume that the lower plenum is filled at the end of the blowdown.

The analysis of the reflood phase of the accident is important with regard to pipe ruptures of the reactor coolant system cold legs, since the steam and entrained liquid carried out from the core for these break locations will pass through the steam generators. The entrained water leaving the core and passing through the steam generators will be evaporated.

MHI will utilize the WREFLOOD code, as described in Reference 2, to calculate mass and energy release to the containment during the reflood period. The WREFLOOD code utilizes a reactor coolant system hydraulic resistance model and an energy balance model. The hydraulic model determines the core flooding rate, whereas the energy balance model determines the core exit conditions and the energy addition from the steam generators. The rate of energy release to the containment during the reflood period will be proportional to the core flooding rate.

The effluent fraction of the core inlet flow rate is calculated from a correlation based on the results of the FLECHT experiments. The results of the FLECHT experiments show that liquid entrainment continues until the fuel is recovered with water to an elevation of approximately four feet from the top. For conservatism plants, using the methodology of Reference 2 have assumed that quenching of the core occurs at an elevation of two feet from the top. This is the assumption recommended by SRP 6.2.1.3 and is the assumption used by MHI. All the available sensible heat in the core and in the neutron shield is added to the containment during the reflood period. The metal of the reactor system primary system piping is lumped with the metal in the steam generators. Heat transfer assumptions are made, which maximize the flow of sensible heat into the reactor system fluid.

For the blowdown and reflood period, modifications were required to both the SATAN-VI and the WREFLOOD computer codes to account for the advanced accumulators of US-APWR. The advanced accumulators are designed to supply emergency coolant at a rapid rate at the start of the accident and then to supply emergency coolant at a slower rate, but for longer time during the core reflood. The modeling of these injection characteristics required programming changes to both SATAN-VI and WREFLOOD.

Essentially, the same advanced accumulator model which has been programmed into SATAN-VI and WREFLOOD has also been programmed into the WCOBRA/TRAC computer code. This code is being utilized by MHI to evaluate compliance with Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50.46 following a large break LOCA (Reference 4). The staff is performing a detailed review of this model as part of the 10 CFR 50.46 compliance review. Provided that the advanced accumulator model is found to be acceptable for the 10 CFR 50.46 compliance review, the staff concludes that the model is also acceptable for containment analysis. MHI performed a sensitivity study to determine the initial accumulator conditions and injection line resistance which produce conservative results for containment analyses. These conservative inputs are used for US-APWR safety analysis.

Following a large cold leg break, once the core has been reflooded and a static pressure balance has been reached between the water in the downcomer and the water in the core and upper plenum, additional emergency coolant added to the downcomer or to the cold legs will spill out of the break. With no additional flow into the core from the bottom, the core will soon begin to boil. Once boiling has begun, additional coolant will flow into the core from the downcomer to replenish that which is boiled away. As a result of boiling, the void fraction in the core will become higher than that in the downcomer. A two-phase level will therefore rise above the core so that a pressure balance is maintained. Reference 2 describes the Westinghouse FROTH methodology which calculates the two-phase level and the additional steaming which will occur if the two phase level reaches the steam generators. By this process, the remainder of the available steam generator energy is removed to be added to the containment building.

Instead of using the approved Westinghouse FROTH methodology described in Reference 2, MHI developed a reactor system model utilizing the GOTHIC computer code (Reference 5). The same void fraction model will be utilized by MHI in the GOTHIC reactor system model, as was approved by the NRC staff for the Westinghouse FROTH model. At the request of the NRC staff, MHI provided additional details of the GOTHIC reactor system model. Within GOTHIC, the reactor system is divided into sufficient detail using a multi-node approach such that the two-phase mixture from the core reaches the steam generators where it is all turned to steam and flows into the containment through one side of the assumed double-ended break in a cold leg. Excess emergency core cooling system (ECCS) water is spilled out of the other side of the break. The spilled ECCS water is assumed not to mix with the steam from the core and steam generators which is conservative. Heat transfer is evaluated from the steam generators and reactor system metal until all available sensible heat is removed and added to the containment. The advanced accumulators will still be injecting into the reactor system at the beginning of the post-reflood period. GOTHIC uses a simplified model of the advanced accumulators which holds the accumulator flow resistance essentially constant until they have emptied. The simplified modeling was determined to be appropriate, based on a more detailed analysis. The advanced accumulators will be emptied before peak containment pressure is reached for the limiting large break LOCA.

Release of the accumulator cover gas will occur after the water in the accumulators has depleted. The nitrogen gas release is conservatively assumed to occur at the end of reflood phase, which is before the time of peak containment pressure for the limiting case. This assumption is conservative, since the nitrogen gas affects the peak pressure magnitude. Although the nitrogen gas temperature will decrease with expansion as accumulator water is discharged, the temperature of the released gas is conservatively assumed constant at the initial value, which is the maximum operating temperature.

MHI will calculate the containment pressure and temperature for the functional design evaluation using the GOTHIC code. Reference 1 contains a brief description of the GOTHIC model. The details of the model have been submitted with the Design Control Document (DCD) for the US-APWR (MUAP-DC006). The staff will defer the review of the GOTHIC containment model to the DCD review.

During the blowdown period, the flow from the postulated reactor system piping rupture to the containment will be at sonic velocities. For this condition, the containment pressure will have little influence on the mass flow rate from the break. Flow from the ECCS pumps, which take suction from the in-containment RWSP, is not assumed during the blowdown period which is conservative for containment analysis. Changes in RWSP temperature, as well as containment pressure, will affect the steaming rate to the containment during core reflood. For this reason, MHI links the GOTHIC containment simulation with the reactor system simulation in WREFLOOD. Similarly, during the post reflood period, the multi-node reactor system model in GOTHIC is solved simultaneously with the GOTHIC containment model. Using a single computer run for both the containment and the reactor system analysis avoids the need for iteration between computer codes, improves accuracy, and is acceptable to the staff.

In the US-APWR design, water from the accumulators is injected into the cold legs, whereas water from the safety injection pumps is injected directly into the downcomer. The safety injection pumps take suction from the RWSP. In the WREFLOOD and GOTHIC reactor system models, MHI takes credit for mixing of the injected accumulator fluid with steam contained within the cold-leg piping. The mixing of steam and accumulator water reduces the amount of steam released to the containment and results in a lower calculated containment pressure. Experimental steam-water mixing studies were performed by EPRI (Reference 6) for water streams injected into steam filled piping test sections. The results of these tests showed that complete mixing occurred immediately, so that thermal equilibrium was obtained at a very short distance from the injection nozzle. Based on these tests, the staff accepted the assumption of complete steam-water mixing in the cold legs for the Westinghouse methodology in Reference 2 and finds this assumption also acceptable for MHI.

The EPRI steam-water mixing data is not applicable to the condition of the safety injection water which is pumped directly into the reactor vessel downcomer. This water may enter the reactor system below the downcomer water level and will not be susceptible to steam-water mixing. MHI will therefore not take credit for steam-water mixing in the downcomer or on any ECCS water stream that spills from the break. Since the accumulators will be emptied early during the course of the accident, steam-water mixing with injected ECCS water will not be credited during the long-term cooling phase of the accident.

The staff noted that the post-reflood steam release to the containment, as calculated by the GOTHIC reactor system simulation, is oscillatory as compared with the previously approved FROTH code methodology in WCAP-10325 which MHI references. In addition, after approximately 30,000 seconds, the steam release to the containment was less than that which would be expected from continued core boil-off. The staff was concerned that the oscillations may be an artifact of the GOTHIC reactor system model, which might cause unrealistic steam condensation to be calculated within the core and lower plenum of the reactor vessel. The staff did not understand why the long-term steam release rate predicted by GOTHIC would be less than that resulting from decay heat boiling, since the applicant’s assumptions for downcomer condensation are designed to conservatively eliminate steam condensation after accumulator injection. The staff notes that SRP 6.2.1.3 recommends the following: “Steam from decay heat boiling in the core should be assumed to flow to the containment by the path which produces the minimum amount of mixing with ECCS injection water.” For a postulated break at the reactor system pump suction, compliance with this recommendation for US-APWR would mean that all steam from the core would be assumed to flow directly to the containment without passing any of the ECCS injection locations.

In response to staff questions, MHI revised the mass and energy release model for the post-reflood period. The revised model will eliminate core oscillations by the use of a lag controller in the GOTHIC reactor system model and improvement in the model initialization which begins at the end of the reflood period. Late in the analysis, after the time when essentially all available sensible heat is removed from the reactor system and steam generators, MHI will calculate steam release to the containment using a boil-off calculation for the core decay heat which will pass all the available steam release directly to the containment. The staff concludes that the revised mass and energy model is in conformance with SRP 6.2.1.3 and is, therefore, acceptable. The revised mass and energy release model is predicted to result in a slightly higher peak containment pressure and slightly higher containment pressures during the long term cooling period. MHI believes that the resulting peak containment pressures will still be well below the US-APWR design pressure and that the long-term pressure response will be less than 50 percent of the peak containment pressure after 24 hours as recommended by SRP 6.2.1.1.A.

4.0 CONCLUSION

The NRC staff has reviewed MUAP-07012 "LOCA Mass and Energy Release Analysis Code Applicability Report for US-APWR." The staff concludes that the applicant's proposed methodology follows the guidance given in SRP 6.2.1.3 and is, therefore, in conformance with the NRC's regulations, and is acceptable for referencing provided that the following conditions are met:

- The staff did not review the models for the advanced accumulator that the applicant added to SATAN VI and WREFLOOD for the mass and energy release calculations. These models are essentially the same as programmed into WCOBRA-TRAC for the ECCS evaluation (Reference 4) which is also under review, by the staff. In the WCOBRA-TRAC review, the advanced accumulator model is being reviewed in detail. The staff will consider the advanced accumulator (Reference 7) to be acceptable for the mass and energy calculations when it is approved for the ECCS evaluation.

- The staff reviewed the methodology in MUAP-07012 for application to the US-APWR mass and energy release determination for containment functional design evaluation. Other safety analysis applications will require additional justification and staff review.

- The staff's acceptance of the methodology in MUAP-07012 is for the revised methodology for which a lag constant has been applied during the post-reflood period to dampen the core inlet oscillations predicted by the GOTHIC reactor system model. This methodology utilizes the direct steaming model after the time when essentially all available sensible heat is removed from the reactor system and steam generators.

The GOTHIC containment model for US-APWR is briefly described in MUAP-07012. The staff did not review the details of that model as part of the MUAP-07012 review. The containment modeling for US-APWR will be reviewed with Chapter 6.2 of the DCD.

5.0 REFERENCES

- "LOCA Mass and Energy Release Analysis Code Applicability Report for US-APWR," MUAP-07012-NP, Revision 2, ADAMS Accession No., ML081610448, May 2008.

- "Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Version," WCAP-10325-P-A, May 1983.

- Letter from Gerald T. Bischof (Virginia Electric and Power Company) to United States Nuclear Regulatory Commission dated November 6, 2006, Transmittal of Approved Topical Report DOM-NAF-3 NP-A, "GOTHIC Methodology for Analyzing the Response to Postulated Pipe Ruptures inside Containment." ADAMS Accession No., ML063190467, November 2006.

- 7 -

- "Large Break LOCA Applicability Report for US-APWR," MUAP-07011-NP, Revision 0, ADAMS Accession No., ML072140123, July 2007.

- "GOTHIC Containment Analysis Package Technical Manual, Version 7.2a (QA)," NAI 8907-06 Rev 16, January 2006.

- Lilly, G.P. and L.E. Hochreiter, "Mixing of Emergency Core Cooling Water with Steam: 1/3 Scale Test and Summary," EPRI 294-2, Electric Power Research Institute, June 1975.

- "The Advanced Accumulator," MUAP-07001-NP, Revision 1, ADAMS Accession No., ML070740602, February 2007.

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