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June 2, 2010

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

Attention: Mr. Jeffrey A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF-10151

Subject: **MHI's Response to the NRC's Request for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR" on 4/15/2010 : "Scaling Analysis for US-APWR Small Break LOCA", UAP-HF-09568**

Reference: 1) "REQUEST FOR ADDITIONAL INFORMATION ON TOPICAL REPORT MUAP-07013-P, 'SMALL BREAK LOCA METHODOLOGY FOR US-APWR'," dated April 15, 2010.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") an official document entitled 'MHI's Response to the NRC's Request for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR" on 4/15/2010 : "Scaling Analysis for US-APWR Small Break LOCA", UAP-HF-09568'. In the enclosed document, MHI provides the all 18 (eighteen) items requested in Reference 1.

As indicated in the enclosed materials, this document contains information that MHI considers proprietary, and therefore should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential. A non-proprietary version of the document is also being submitted in this package (Enclosure 3). Any proprietary information that is written inside a bracket in the proprietary-version is replaced by the designation "[ ]" without any text, in the non-proprietary-version.

This letter includes a copy of proprietary version (Enclosure 2), a copy of non-proprietary version (Enclosure 3), and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the bases of MHI request that all materials designated as "Proprietary" in Enclosure 2 be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

Please contact Dr. C. Keith Paulson, Senior Technical Manager, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of this submittal. His contact information is provided below.

Sincerely,

Y. Ogata

Yoshiki Ogata
General Manager - APWR Promoting Department
Mitsubishi Heavy Industries, LTD.

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NRO

Enclosures:

1. Affidavit of Yoshiki Ogata

2. MHI's Response to the NRC's Request for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR" on 4/15/2010 : "Scaling Analysis for US-APWR Small Break LOCA", UAP-HF-09568 (proprietary)

3. MHI's Response to the NRC's Request for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR" on 4/15/2010 : "Scaling Analysis for US-APWR Small Break LOCA", UAP-HF-09568 (non-proprietary)

CC: J. A. Ciocco
C. K. Paulson

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ENCLOSURE 1

Docket No.52-021
MHI Ref: UAP-HF-10151

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, being duly sworn according to law, depose and state as follows:

1. I am General Manager, APWR Promoting Department, of Mitsubishi Heavy Industries, Ltd ("MHI"), and have been delegated the function of reviewing MHI's US-APWR documentation to determine whether it contains information that should be withheld from disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential.

2. In accordance with my responsibilities, I have reviewed the enclosed "MHI's Response to the NRC's Request for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR" on 4/15/2010 : "Scaling Analysis for US-APWR Small Break LOCA", UAP-HF-09568", and have determined that portions of the report contain proprietary information that should be withheld from public disclosure. Those pages containing proprietary information are identified with the label "Proprietary" on the top of the page and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]". The first page of the technical report indicates that all information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

3. The information in the report identified as proprietary by MHI has in the past been, and will continue to be, held in confidence by MHI and its disclosure outside the company is limited to regulatory bodies, customers and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and is always subject to suitable measures to protect it from unauthorized use or disclosure.

4. The basis for holding the referenced information confidential is that it describes the unique codes and files developed by MHI for the fuel of the US-APWR and also contains information provided to MHI under license from the Japanese Government. These codes and files were developed at significant cost to MHI, since they required the performance of detailed calculations, analyses, and testing extending over several years. The referenced information is not available in public sources and could not be gathered readily from other publicly available information. MHI knows of no way the information could be lawfully acquired by organizations or individuals outside of MHI and the Japanese Government.

5. The referenced information is being furnished to the Nuclear Regulatory Commission ("NRC") in confidence and solely for the purpose of supporting the NRC staff's review of MHI's Application for certification of its US-APWR Standard Plant Design.

6. Public disclosure of the referenced information would assist competitors of MHI in their design of new nuclear power plants without the costs or risks associated with the design of new fuel systems and components. Disclosure of the information identified as proprietary would therefore have negative impacts on the competitive position of MHI in

the U.S. nuclear plant market.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information and belief.

Executed on this 2nd day of June, 2010.

Y. Ogata

Yoshiki Ogata
General Manager- APWR Promoting Department
Mitsubishi Heavy Industries, LTD.

ENCLOSURE 3

UAP-HF-10151

MHI's Response to the NRC's Request for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR" on 4/15/2010 : "Scaling Analysis for US-APWR Small Break LOCA", UAP-HF-09568

June 2010
(Non-Proprietary)

REQUEST S-1

Section 1.2 of the report states "In this report, quantitative scaling analyses based on the hierarchical two-tiered scaling (H2TS) methodology[1-7] were performed to complete the M-RELAP5 development and assessment which is required in the EMDAP." It further states that both top down and bottom up analyses are performed. One purpose of the top down analysis is to address system interactions. Yet, on the one hand the report contains arguments concerning US-APWR likeness to a traditional 4-loop Westinghouse PWR and at the same time, the report does not mention any specific system interactions whose scalability is of particular interest.

What specific new and/or unique system interactions is the top down scaling analysis intending to address?

RESPONSE

No known new responses or interactions are being investigated. The primary focus is on whether the new US-APWR specific design features cause significant changes in response relative to ROSA/LSTF which was scaled to a standard 4-loop Westinghouse plant.

Important phenomena and processes expected to take place in the US-APWR during SBLOCAs are similar to those expected in the conventional Westinghouse 4-loop PWRs. The accident progression can be divided into five phases: 1) blowdown, 2) natural circulation, 3) loop seal clearance, 4) boil-off, and 5) core recovery phases.

The new and improved features adopted in the US-APWR design, related to LOCA are as follows:

a) Advanced Accumulator
b) HHIS/DVI
c) Neutron Reflector
d) Gas Turbine Generator

a) Advanced Accumulator (Reference S-1-1)
The advanced accumulator primarily contributes to suppressing the PCT for cases with larger break sizes. The advanced accumulator is designed to start injecting emergency coolant passively when the RCS pressure falls below the set point pressure. The set point pressure is the same as that for the 4-loop PWR. For cases with larger break sizes, the RCS sufficiently depressurizes below the set point pressure. Therefore, there is no significant concern on the interaction between the RCS and the advanced accumulator operating behavior. In this regard, the advanced accumulator does not contribute to suppressing the PCT for cases with smaller break sizes, in which the HHIS suppresses the fuel cladding heat-up during the loop seal phase.

b) HHIS/DVI
The High Head Injection System (HHIS) injects emergency coolant through a Direct Vessel Injection (DVI). In the case of US-APWR SBLOCAs, PCT occurs in the 1 ft^2

break and the accumulator governs the core thermal-hydraulics. The effect of system interaction between the RCS and HHIS behaviors on the PCT is negligible.

c) Neutron Reflector
In comparison with large break LOCAs, the effect of the neutron reflector on the phenomena and processes during reflood phase is smaller, because core liquid level depression is not as large during an SBLOCA transient.

d) Gas Turbine Generator
The gas turbine generator has no direct interaction with the reactor responses.

Therefore, due to the inherent similarity of the US-APWR design to the 4-loop Westinghouse design as described above, the top-down scaling analysis is limited to a confirmatory approach. The scope is limited to the SBLOCA scenarios resulting in the highest peak cladding temperatures (PCT). The objective of the top-down scaling analysis is to show that the same phenomena and mechanism are active and dominant in the US-APWR and IET responses.

Reference:

S-1-1 Mitsubishi Heavy Industries, Ltd., The Advanced Accumulator, MUAP-07001 (R2), September 2008.

REQUEST S-2

In reference to the repeated argument of "similarity" between US-APWR and traditional PWRs, which in the past were analyzed, for purposes of scaling, as a simple loop with emphasis on local phenomena ranges and traditional non-dimensional parameters, such as Re, Nu, Bi, etc.

- What is new and different about the systems of US-APWR that requires a top down scaling analysis to ensure that the code can predict the new system interactions?
- What sequence of events and conditions are expected to vary from that of conventional PWRs due to US-APWR larger dimensions?
- Is it not possible to evaluate the sufficiency of the data base and the applicability of the code through a bottom up (local component and phenomena ranges) scaling analysis? What exactly does the top-down approach, based on a single IET, contribute to this goal?

RESPONSE

There are several new features in the US-APWR design (14 foot core, advanced accumulator, HHIS flow with DVI). While no significant changes in LOCA response are expected relative to the reference 4-loop Westinghouse plant, that outcome cannot be assumed a priori. The present scaling analysis was performed to conform to the requirements specified in Steps 6, 8, and 15 of the EMDAP as described in Regulatory Guide 1.203. In the corresponding sections of the Regulatory Guide there are references to reports INEL-96/0400 and INEL-96/0040. Where possible the methodologies in those reports were used as templates for this scaling analysis.

No expected changes to the sequence of events or specific changes in mechanisms have been identified.

(1) *What is new and different about the systems of US-APWR that requires a top down scaling analysis to ensure that the code can predict the new system interactions?*

As described in the Response to REQUEST S-1, the top down scaling analysis was performed as a confirmatory analysis limited to the SBLOCA scenarios resulting in the highest PCT. It clarifies that there are no new system interactions in the US-APWR during an SBLOCA transient relative to the 4-loop Westinghouse plant used as the reference design for the ROSA facility.

(2) *What sequence of events and conditions are expected to vary from that of conventional PWRs due to US-APWR larger dimensions?*

The accident progression, observed phenomena and processes during the US-APWR SBLOCAs are similar to those in the conventional Westinghouse 4-Loop PWRs. Effects due to the US-APWR larger dimensions are addressed in previous MHI RAI responses[S-2-1, S-2-2] and in the scaling analysis report[S-2-3], which indicates that there are no significant effects caused by the larger dimensions in the US-APWR design.

(3) Is it not possible to evaluate the sufficiency of the data base and the applicability of the code through a bottom up (local component and phenomena ranges) scaling analysis? What exactly does the top-down approach, based on a single IET, contribute to this goal?

The applicability and scale-up capability of the thermal-hydraulic models in M-RELAP5 and RELAP5-3D have been well assessed and validated in various studies using experimental test data obtained in SET and IET facilities scalable to the 4-Loop PWRs. In the present scaling analysis, MHI demonstrated that the SBLOCA experiments in ROSA are sufficiently scaled to the representative US-APWR SBLOCA. In other words, even though the ROSA facility was originally scaled to the conventional 4-Loop PWRs, the test data and code validation using the test data are applicable to the US-APWR SBLOCAs. RELAP5-3D, the basis of M-RELAP5 has been established based on the past various code validations, and these past works show that the code is applicable also to the US-APWR SBLOCAs.

Using the bottom-up scaling, each response mechanism is studied separately and it is not possible to assess the relative importance of the mechanisms. In the top-down scaling, all the relevant processes during a phase are assessed together and through numerical magnitude ranking of the ψ groups the relative importance of each mechanism is clarified. Comparing the numerical ranking of the ψ groups between the test facility and the plant provides a better understanding of how well the system level processes are represented in the test facility. In the top-down scaling two IETs from the ROSA facility were studied to see how well the system level processes were represented over a range of break sizes.

References:

S-2-1 Mitsubishi Heavy Industries, Ltd., MHI's Partial Responses to the NRC's Requests for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR", UAP-HF-09002-P (R0), January 2009.

S-2-2 Mitsubishi Heavy Industries, Ltd., MHI's 2nd Response to the NRC's Request for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR" on 09/08/2009, UAP-HF-09512-P (R0), November 2009.

S-2-3 Mitsubishi Heavy Industries, Ltd., Scaling Analysis for US-APWR Small Break LOCAs, UAP-HF-10152, June 2010.

REQUEST S-3

In section 1.2 of the scaling report it is stated that *"Specifically, the IET and SET facilities and experimental data are evaluated by the top-down and bottom-up approaches to respond to Step 6 in Element 2 of EMDAP 'Perform Scaling Analysis and Identify Similarity Criteria', which demonstrates whether similar thermal-hydraulic behaviors expected in the US-APWR are also observed in the scaled test facilities."*

This implies that multiple facilities are included in the scaling analysis. The Regulatory Guide 1.203 (on page 13) also refers to multiple facilities. However, the text of this scaling report seems to imply that the SBLOCA "scaling analysis" is limited to a single facility and a single test.

If that is the case, what demonstrates that the data base is sufficient, being that a single test does not contain all of the ranges of phenomena expected in the prototype?

RESPONSE

The top down scaling analysis for SBLOCA is looking at data from two break sizes from tests in the ROSA/LSTF facility. The top-down scaling analysis results show that for the two break sizes studied, in general the same thermal-hydruaulic mechanisms are occurring with the same phenomena dominant for important responses. Based on the results of the top-down scaling analysis, the responses in the ROSA/LSTF tests appear to be in the correct range for simulating the US-APWR response (Reference S-3-1). This is determined by comparing the individual evaluated dimensionless groups from the two facilities and the numerical rankings of the dimensionless groups.

In the bottom-up scaling analysis, data from several test facilities are being used. In each case the applicability of the test data to the US-APWR conditions is assessed and found to cover the range of conditions expected in the US-APWR.

Reference:

S-3-1 Mitsubishi Heavy Industries, Ltd., Scaling Analysis for US-APWR Small Break LOCAs, UAP-HF-10152, June 2010.

REQUEST S-4

In Section 6.1.2.3 it states *"...the governing conservation equations, (6.1-1) and (6.1-2), is nondimensionalized by dividing by the reference quantity of the parameter, e.g. the initial value ..."* The text further states that the reference time is chosen to make a particular nondimensional coefficient (Φ_6) equal to unity.

Please provide the criterion for the selection of each of the other "reference quantities".

RESPONSE

The top-down scaling analysis of the blowdown phase reported in Reference S-4-1 was a direct implementation of the approach used in INEL-96/0040, including the assumption that there was no saturated fluid outside the pressurizer. This assumption did not apply during the duration of the blowdown phase in the larger break sizes investigated for the US-APWR. The analysis of the blowdown phase has been redone using equations consistent with saturated fluid outside the pressurizer.

The overall approach to selecting reference parameters is based on making the individual dimensionless variable terms of order unity. For the SBLOCA analysis the RCS inventory is generally the metric of highest interest because it strongly affects core cooling and ultimately PCT. When the accident scenario is broken into phenomenologically based phases the reference mass or in some cases reference time can be selected to highlight the mass inventory response to the initial system mass or a component mass. In the analysis referred to in the request the reference parameters were selected such that Φ_6 would be equal to unity at the time the pressurizer emptied. In this case $\phi_6 = \frac{\dot{m}_0 t_0}{M_0}$ where $\dot{m}_0$ is the average surge line flow during the phase, t_0 is the phase duration, and M_0 was the initial liquid inventory in the pressurizer. With these definitions, making Φ_6 equal to unity provided a direct comparison of the times for the pressurizer to empty in ROSA and the US-APWR.

Reference:

S-4-1 Mitsubishi Heavy Industries, Ltd., Scaling Analysis for US-APWR Small Break LOCAs, UAP-HF-09568, December 2009.

REQUEST S-5

Equation (6.1-9) defines the coefficient Φ6 and the reference time.

- How was the pressurizer mass flow used as the reference chosen?
- How was the reference pressurizer mass flow calculated?
- Since the rest of the system is apparently subcooled, the break flow might serve as a better reference value. Please address the merits of using the pressurizer mass flow as a reference as opposed to the break flow.

RESPONSE

The parameters are addressed as follows in Reference S-5-1:

(1) How was the pressurizer mass flow used as the reference chosen?

The use of pressurizer mass flow was based on directly implementing the methodology used in INEL-96/0040 used to analyze the 1-inch break in the AP600 design.

(2) How was the reference pressurizer mass flow calculated?

The reference pressurizer mass flow was calculated as the average surgeline mass flow while the pressurizer was draining.

(3) Since the rest of the system is apparently subcooled, the break flow might serve as a better reference value. Please address the merits of using the pressurizer mass flow as a reference as opposed to the break flow.

If the remainder of the RCS was subcooled the pressurizer and break volumetric flows would be essentially equal and the mass flows would be related by the ratio of the densities in the pressurizer and the cold leg.

Subsequently it was determined that the remainder of the RCS was not subcooled during the blowdown phase for the larger break sizes studied in the US-APWR. In the hot leg and upper plenum flashing was decoupling the break flow from the pressurizer surge line flow. The analysis of the blowdown phase has been updated to account for flashing outside the pressurizer in the revised version of the scaling report (Reference S-5-2).

References:

S-5-1 Mitsubishi Heavy Industries, Ltd., Scaling Analysis for US-APWR Small Break LOCAs, UAP-HF-09568, December 2009.

S-5-2 Mitsubishi Heavy Industries, Ltd., Scaling Analysis for US-APWR Small Break LOCAs, UAP-HF-10152, June 2010.

REQUEST S-6

Table 6.1-2 defines Φ_6 as a *"Ratio of integrated mass flow to a reference mass."*

Does this mean that the reference pressurizer mass flow is an average of some sort? How is the reference mass flow defined for the plant?

RESPONSE

In Reference S-6-1, the reference mass flow rate is the average surge line mass flow rate from the time of break initiation to the time when the pressurizer was empty. For the plant this was evaluated by determining the time when the pressurizer was empty and averaging the surge line mass flow from break initiation up to that time.

Reference:

S-6-1 Mitsubishi Heavy Industries, Ltd., Scaling Analysis for US-APWR Small Break LOCAs, UAP-HF-09568, December 2009.

REQUEST S-7

Table 6.1-1 has a column named *"Reference Parameters"* If one calculates the reference time so that $\Phi_6 = 1.0$, one gets different values than those in the table.

Please describe how the reference values were determined and provide their values as used in the scaling analysis.

RESPONSE

In Table 6.1-1 there was a numerical error. The process for selecting reference values is discussed in the responses to REQUEST S-4. The equations used to analyze the blowdown phase have been updated to account for saturated fluid outside the pressurizer. The results are described in the updated version of the scaling report (Reference S-7-1).

Reference:

S-7-1 Mitsubishi Heavy Industries, Ltd., Scaling Analysis for US-APWR Small Break LOCAs, UAP-HF-10152, June 2010.

REQUEST S-8

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RESPONSE

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REQUEST S-9

At the end of Section 6.2.4, the following statements are made: *"... ROSA/LSTF was designed so that the test facility is scalable to the reference plant (Westinghouse-designed 4-loop PWR) which is also scalable to the US-APWR"* and *"Consequently, it can be judged that the ROSA/LSTF is sufficiently scalable to the US-APWR ..."*

- What is the definition of "scalable" in this context?
- What is the definition of "sufficient" as used here?

RESPONSE

The description of "scalable" is discussed in the responses to S-14 and S-18. In the revised scaling report (Reference S-9-1), the "sufficiently" was removed.

Reference:

S-9-1 Mitsubishi Heavy Industries, Ltd., Scaling Analysis for US-APWR Small Break LOCAs, UAP-HF-10152, June 2010.

REQUEST S-10

Section 6.3.2.3 states, *"Each of the physical parameters in the governing conservation equations, (6.3-16) and (6.3-17), is nondimensionalized' by dividing by a reference quantity of the parameter, e.g. the initial value."*

- Please define and discuss the criterion used to select these reference values?

RESPONSE

In Reference S-10-1, [

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Following a more detailed internal review, the equations were simplified to account for the fact that the downcomer remained full during the phase. When these equations were nondimensionalized individual reference values were selected for each variable to ensure that each term was of order unity. The updated equations are in the updated version of the report (Reference S-10-2).

References:
S-10-1 Mitsubishi Heavy Industries, Ltd., Scaling Analysis for US-APWR Small Break LOCAs, UAP-HF-09568, December 2009.
S-10-2 Mitsubishi Heavy Industries, Ltd., Scaling Analysis for US-APWR Small Break LOCAs, UAP-HF-10152, June 2010.

REQUEST S-11

Section 6.3.2.3 states *"the equations are mathematically solved to obtain the temporal derivatives of the core and upper plenum liquid levels and the liquid level at the loop seal clearing"*

- Does this mean that an analytical solution was obtained? Please provide the "solution" referred to in this statement.

RESPONSE

The equations were solved by combining each term in the right hand side of the equations in time series analysis, each term was evaluated using the calculated or measured results (References S-11-1 and S-11-2).

References:
S-11-1 Mitsubishi Heavy Industries, Ltd., Scaling Analysis for US-APWR Small Break LOCAs, UAP-HF-09568, December 2009.
S-11-2 Mitsubishi Heavy Industries, Ltd., Scaling Analysis for US-APWR Small Break LOCAs, UAP-HF-10152, June 2010.

REQUEST S-12

Equation 6.3-18 contains numerous non-dimensional mass flow terms, and various non-dimensional level terms. The definitions of starred variables just below this equation suggest that all of these mass flows are normalized with respect to one reference flow (same for all), and all of the levels are normalized with respect to one reference level, and the same is true of the non-dimensional areas.

- Is this interpretation correct? Are the reference mass flow, reference level, and reference area the same for all like-variables? What is the rationale for this approach?
- Is this also true of the other non-dimensional parameters of other governing equations and in other phases of the transient?

RESPONSE

(1) Is this interpretation correct? Are the reference mass flow, reference level, and reference area the same for all like-variables? What is the rationale for this approach?

In this case the same reference values were used. The reference mass flow was the break flow since that is the only flow changing the liquid inventory in the RCS. The sum of the individual mass flows was normalized since the sum represented the net reactor vessel mass change. The formulation of the equations has been updated to account for the fact that the downcomer remains full during the phase. In the updated equations separate reference values were used for each variable.

(2) Is this also true of the other non-dimensional parameters of other governing equations and in other phases of the transient?

No, in general we used separate reference values for each variable in each equation.

REQUEST S-13

In Section 6.3.2.3, the reference time is defined as the difference between the timing of two events, neither of which is determined by a simple calculation. This reference time itself is not identified with any one specific process, but rather the result of competing processes.

- If it is necessary to know a priori, t_1 and t_2, in order to do the quantification of the nondimensional coefficients required for the analysis, how are the plant numbers evaluated?
- How sensitive are the analysis results to the reference time?

RESPONSE

(1) If it is necessary to know a priori, t_1 and t_2, in order to do the quantification of the nondimensional coefficients required for the analysis, how are the plant numbers evaluated?

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(2) How sensitive are the analysis results to the reference time?

The reference time appears in the ψ_1 term as $\psi_1 = \frac{\dot{m}_{0a} t_0}{\rho_0 L_{0a} A_{0a}}$. This makes the ψ_1 group directly proportional to the reference time. Since the reference times agree within 2.5% it does not have a significant impact on the calculated ψ values.

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REQUEST S-14

Section 6.3.2.4 states "In the scaling analysis for AP1000, it is shown that an acceptable range for the facility/plant scaling ratios is from 0.5 to 2.0." The specific document cited is NUREG-1793, Section 21.5.7, 2004.

NUREG-1793, Section 21.5.7 suggests that for the relationship between AP600 and AP1000 there is a range for which the AP600 test data base is applicable to AP1000 scaling analysis. It does not demonstrate (as implied by the word "shown" in the US-APWR scaling report) or offer any explanation for why the range was chosen. We can easily demonstrate (attached draft reference) that this overly simplified criterion does not ensure similarity at all.

References cited in S-14 and S-15:

1. Ortiz, M. G. "On Top-Down Scalability Criteria" Draft manuscript attached
2. Ortiz, M.G. and Gavrilas, M. "PUMA Scaling Distortion Analysis: A Method" Presented at the NRC 19th Annual Regulatory Information Conference (RIC)
3. Banerjee, S., M. G. Ortiz, T. K. Larson, D. L. Reeder. "Scaling in the safety of next generation reactors," Nuclear Engineering and Design, vol186, 1998, pp 111-133.
4. Banerjee, S., M. G. Ortiz, T. K. Larson, D. L. Reeder. "Scaling In The Safety Of Next Generation Reactors," Eighth International Topical Meeting on Nuclear Reactor Thermal-Hydraulics, Kyoto, Japan, September 30 -October 4, 1997, Proceedings Volume 1 pp 508-527
5. Ortiz, M. G., S. Banerjee, T. K. Larson. ``A Systems Approach To The Scaling Analysis Of Integral Test Results," Presented at The Fifth International Topical Meeting on Nuclear Thermohydraulics, Operations & Safety (NUTHOS-5), Beijing, China, April 13-18, 1997. Published in the Conference Proceedings, pp Q1-Q7

RESPONSE

The issue of when is scaling good enough has not been addressed in a definitive way for reactor safety experiments investigating LOCA response. There is no regulatory guidance for assessing the applicability of scaled facility experimental data to a plant design. In recent years during the licensing process for the Westinghouse AP1000 and the General Electric ESBWR, several criteria were used by the vendors and NRC staff and presented to the ACRS. For the AP1000, the acceptability criteria used were that the ratio of ψ groups for the plant and experimental facility, for a specific mechanism should be between ½ and 2 (Reference S-14-1). This range was used by Westinghouse and the USNRC and presented to the ACRS. For the ESBWR, General Electric used a range of 1/3 to 3. In a July 2003 meeting of the ACRS Subcommittee on Thermal Hydraulic Phenomena, it was stated by one of the members that the use of the ½ to 2 criteria for the AP1000 established a tradition. Other members noted that the ranking of the ψ groups was more important than the numerical value of the ratios (Reference S-14-2).

For the US-APWR scaling study, we are using the ranking of ψ groups in ROSA and US-APWR as the primary metric for scalability. We are using the ½ to 2 range for the ratio of ψ groups between facilities metric as a heuristic guideline. The range provides a convenient way to assess the scalability of individual phenomena. Applying the range criterion also helps ensure that the individual mechanisms are reasonably similar between the facilities (Reference S-14-3).

References:

S-14-1 U.S. Nuclear Regulatory Committee, 'Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design,' NUREG-1793, Section 21.5.7, September 2004.

S-14-2 U.S. Nuclear Regulatory Commission Advisory Committee on Reactor Safeguards, Thermal-Hydraulic Phenomena Subcommittee, July 8, 2003, Transcript ACRST-3243.

S-14-3 Mitsubishi Heavy Industries, Ltd., Scaling Analysis for US-APWR Small Break LOCAs, UAP-HF-10152, June 2010.

REQUEST S-15

The documents suggest that the nondimensional coefficients of US-APWR appear to have been derived in a fundamentally different fashion than how it was done for AP600 and AP1000 scaling analysis. The US-APWR analysis appears to use a single reference variable to create the starred variables of several like variables (one mass-flow, one mass, one area are reference to several flows, masses and areas of the system); for the AP600 analysis, an individual reference value is chosen for each variable, so as to make all the starred variables of order one. The only common reference is the reference time, which is chosen according to the timing of the process of interest. These are the qualities that allow, in AP600, the ranking of processes based on the magnitude of their π values, and the direct comparison between facilities based on these π values. Moreover, the identification of potential distortions is made not only by comparing magnitudes of π s but how they stack against each other in the same equation and different facilities. There are several public domain references that describe these methods in some detail (attached is a presentation given at the NRC 19th Annual Regulatory Information Conference (RIC) and available online, the first 3 references at the end of that presentation are also available without proprietary restrictions)

- Since the approach in the scaling analysis of US-APWR seems to depart significantly from the AP-1000 scaling in the various aspects described above, please explain in more specific detail, how the single ROSA SB-CL-18 facility and test case demonstrates that the experimental database is sufficiently diverse that the expected plant-specific response is bounded and the EM calculations are comparable to the corresponding tests in non-dimensional space.

References cited in S-14 and S-15:

1. Ortiz, M. G. "On Top-Down Scalability Criteria" Draft manuscript attached
2. Ortiz, M.G. and Gavrilas, M. "PUMA Scaling Distortion Analysis: A Method" Presented at the NRC 19th Annual Regulatory Information Conference (RIC)
3. Banerjee, S., M. G. Ortiz, T. K. Larson, D. L. Reeder. "Scaling in the safety of next generation reactors," Nuclear Engineering and Design, vol186, 1998, pp 111-133.
4. Banerjee, S., M. G. Ortiz, T. K. Larson, D. L. Reeder. "Scaling In The Safety Of Next Generation Reactors," Eighth International Topical Meeting on Nuclear Reactor Thermal-Hydraulics, Kyoto, Japan, September 30 -October 4, 1997, Proceedings Volume 1 pp 508-527
5. Ortiz, M. G., S. Banerjee, T. K. Larson. "A Systems Approach To The Scaling Analysis Of Integral Test Results," Presented at The Fifth International Topical Meeting on Nuclear Thermohydraulics, Operations & Safety (NUTHOS-5), Beijing, China, April 13-18, 1997. Published in the Conference Proceedings, pp Q1-Q7

RESPONSE

The top-down scaling analysis approach used for US-APWR does not depart significantly from the methodology used for the AP600 and AP1000. In fact the methodology developed for the AP600, as described in INEL-96/0040, was used as the template for US-APWR analysis. The perception that the methodology was different appears to result from a few instances where a single reference parameter was used for more than one variable. In those instances, the analyses have been redone with unique reference values for each variable.

Two ROSA tests are being used in the top-down scaling study (Reference S-15-1). The tests correspond to the two break sizes resulting in the highest PCT in the US-APWR SBLOCAs. In the 7.5-in break (Reference S-15-2), the PCT occurs at high pressure (~9 MPa) during the loop seal clearing phase, before the ECCS are activated. In the 1-ft^2 break (Reference S-15-3), the PCT occurs at low pressure (< 1 MPa) during the boil-off phase, after the ECCS are activated. These experiments cover a wide range of rewet conditions and include all the expected processes and mechanisms occurring in small break LOCAs.

References:

S-15-1 Mitsubishi Heavy Industries, Ltd., Scaling Analysis for US-APWR Small Break LOCAs, UAP-HF-10152, June 2010.

S-15-2 H. Kumamaru et al., 'ROSA-IV/LSTF 5% Cold Leg Break LOCA Experiment Run SB-CL-18 Data Report,' JAERI-M 89-027, March 1989.

S-15-3 JAEA, "Experimental Report on Simulated Intermediate Break Loss-of-Coolant Accident using ROSA/LSTF," March 2010 (*in Japanese, proprietary*).

REQUEST S-16

Section 6.1.3, Bottom-up Scaling Analysis, states:
"From the viewpoint of the bottom-up approach, the discharge flow characteristic out the break is important in determining the initial plant response. Since the US-APWR SBLOCA methodology employs a break flow model approved in Appendix K to 10 CFR 50 for its application to the licensing safety analysis, the break flow model in M-RELAP5 was not explicitly assessed using experimental test data. In addition, occurrence of dryout (DNB) is not expected during the blowdown phase which was confirmed in the spectrum analyses of US-APWR SBLOCAs. 6-' Therefore, there is no need to evaluate the breakflow model and relevant experimental data by using the bottom-up scaling approach."

- A conservative model implies an intended distortion in the plant model. How will this distorted behavior impact other phenomena and processes in its proximity?
- We still need to verify that the range of available data contains the expected response of the code calculation in nondimensional space. If it is not done, what makes the calculation believable?

RESPONSE

(1) A conservative model implies an intended distortion in the plant model. How will this distorted behavior impact other phenomena and processes in its proximity?

The break flow model used in M-RELAP5 is compliant with the Appendix K requirements, which means it is expected to over-predict the break flow for any RCS conditions upstream of the break during a LOCA. By over-predicting the break flow the calculated RCS inventory will be lower at any given pressure than the expected plant inventory would be. This artificially low RCS inventory will result in higher cladding temperatures due to less mass being available for core cooling. This conservative estimate of core thermal response is one of the expected and intended results of using the Appendix K model for break flow. Furthermore, spectral analyses for the break size are performed in order to cover an uncertainty of break mass flow rate prediction.

As shown in the bottom-up scaling study Section 6.1.3 in the updated Scaling Report, the tendency of M-RELAP5 over-prediction was attained for the ROSA/LSTF IB-CL-02 17% cold leg break test (Reference S-16-1). The over-prediction gives lower RCS inventory but the accumulator injection starts earlier due to the faster depressurization. From the safety analysis viewpoint, we used the IET data to verify the code applicability.

(2) We still need to verify that the range of available data contains the expected response of the code calculation in nondimensional space. If it is not done, what makes the calculation believable?

The updated Scaling Report discusses the range of available data in Chapter 8 using Figure 8-1. And the applicability of M-RELAP5 break flow model was verified using two ROSA/LSTF tests in Section 6.1.3 as mentioned above.

Reference:

S-16-1 Mitsubishi Heavy Industries, Ltd., Scaling Analysis for US-APWR Small Break LOCAs, UAP-HF-10152, June 2010.

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Also in page 6-18, the report states: "The heat transfer between the primary and secondary sides of the SG can also be an important phenomenon during the blowdown phase. In the top-down scaling analysis, however, the steam generator heat transfer was not explicitly addressed because the outflow from the pressurizer was adopted as the dominant factor including the effect of the steam generator heat transfer implicitly, as discussed in Section 6.1.2. Therefore, the heat transfer in the SG is not directly addressed by the bottom-up approach for the present study"

- The statement implies that the scaling analysis of one "important" phenomenon is discarded based on a subjective decision that chooses another phenomenon to be the focus of attention. Please, either demonstrate that the steam generator heat transfer is indeed irrelevant and requires no scaling analysis, or show that the experimental data covers the range of conditions expected to occur in the plant.

RESPONSE

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REQUEST S-18

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RESPONSE

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