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David Terao

From: David Terao NRO
Sent: Friday, January 23, 2009 10:47 AM
To: Eileen McKenna
Cc: Patrick Donnelly; Timothy Steingass; Thomas Scarbrough; Charles Hammer; Eric Reichelt; Neil Ray; John Honcharik; Laura Dudes
Subject: CIB Responses to NNSA's Questions
Attachments: NNSA Qs and As.doc

Eileen,

As you requested in your memorandum dated January 8, 2009, attached are CIB's responses to the Chinese National Nuclear Safety Administration's questions on the AP1000 standard plant design in preparation for their upcoming visit to the NRC's offices the week of February 9, 2009. We had some difficulty understanding specifically what NNSA was seeking with some questions since the questions were quite broad. For those questions, we addressed the question generally at a high level and stated that we will provide more detailed information to address their specific concerns when we meet with them. The CIB 1 and 2 staff addressed the following questions:

3-2	5-5(2)
3-3	5-7
3-8	5-8
5-1	5-9
5-2	6-5
5-3	

Please let me know if any clarifying detail is needed or if any other questions need to be addressed. We are looking forward to meeting the NNSA representatives and discussing these issues further with them. Thank you!

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Send email to Terence? thanking John Tsao for his contributions with NNSA on (AP1000) LBB Qs.

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QUESTIONS FROM CHINESE NATIONAL NUCLEAR SAFETY ADMINISTRATION

Question 3-2 About the material toughness of LBB Piping

SRP 3.6.3 requires that "the piping material will not become susceptible to brittle cleavage-type failures over the full range of system operating temperatures (that is, the material is on the upper shelf of the Charpy Impact energy versus test temperature curve)", but the SER of NRC on AP1000 says, quoting from NUREG 1061, that LBB analysis for brittle materials may use fracture mechanics method than the limit load method for ductile materials. If the material is on the upper shelf of the fracture toughness curve, is there still any to do the fracture mechanics analysis?

NRC Response: In the NRC's safety evaluation report for AP1000 (NUREG-1793, page 3-51), the NRC states, "...The stability analysis of the LBB uses either a fracture mechanics analysis for brittle materials or a limit load analysis for ductile materials to determine a critical crack size for a postulated circumferential, through-wall crack under normal and seismic loads..."

The statement in Question 3-2 was characterized differently from the NRC's statement above. The intent of the NRC's statement above is to provide an applicant with the option of using different analytical methods to calculate the critical crack size of candidate pipes that have different material properties (i.e., brittle vs. ductile materials). For brittle materials, the elastic-plastic fracture mechanics method may be used to determine the critical crack size. For ductile materials, the limit load method may be used to determine the critical flaw size. If a pipe uses a material that is on the upper shelf of the fracture toughness curve, the critical crack size for that pipe will still need to be calculated to demonstrate its compliance with the margins recommended in SRP 3.6.3.

Owners of nuclear plants have calculated the critical crack size using both elastic-plastic fracture mechanics method and limit load method. The conservative critical crack size of the two methods is used to satisfy SRP 3.6.3 margins.

The intent of the cleavage-type failure statement in SRP 3.6.3 is to restrict the LBB application for those piping systems that are susceptible to brittle cleavage-type failures in a very high temperature range (above 700°F, see SRP 3.6.3.III.6). The LBB application in SRP 3.6.3 was developed to be used in the light water reactors. The operating temperature in light water reactors is much lower than the operating temperature in high temperature reactor designs.

Question 3-3 About the Limitation on Fatigue potentials for LBB candidates

SRP 3.6.3 requires that LBB candidates shall not possible fatigue failures. How should this be evaluated, for example, should the usage factor U=0.5, U=0.8 or something else for "having a potential of fatigue failure."

NRC Response: The cumulative usage factor calculated by the ASME Boiler and Pressure Vessel Code, Section III, NB-3200 may be used to determine whether a piping system would have a potential for fatigue failure. However, the usage factor does not provide all the necessary information to predict the potential for fatigue failures. For example, the usage factor will not be able to predict the high- and low-cycle fatigue from pipe vibration or thermal fatigue from fluid stratification in a pipe. Also, based on operating experience, pipes with high usage

factor (e.g., U=0.9) have not had fatigue failures. Therefore, a pipe with a high usage factor does not necessarily imply that it will have a high likelihood of fatigue failures.

SRP 3.6.3 does not provide a specific usage factor limit to determine the potential for fatigue failures. However, the ASME Code, Section III, NB-3000 requires that the cumulative usage factor be less than 1.0 for all Class 1 piping. This implies that the usage factor for the LBB piping must be less than 1.0.

The LBB concept was initiated in the late 1970's and implemented in the NRC regulations in the mid-1980's. At the time of the LBB development, many nuclear plants had been in operation for several years and, therefore, operating experience regarding thermal and vibrational fatigue problems in certain piping systems was available. On the basis of this operating experience, NRC recognized that certain piping systems had problems with flow-induced vibrational fatigue and thermal fatigue due to stratification (e.g., pressurizer surge line in PWRs). The NRC did not allow the use of LBB for those piping systems that were susceptible to significant vibrational or thermal fatigue.

In general, the NRC reviews plant operating parameters such as water chemistry, flow velocity, operating temperatures, and steam quality, as well as their effects on plant operating procedures. The NRC's safety evaluation report for AP1000 (NUREG-1793, page 3-53) provides additional information regarding how the NRC evaluates various degradation mechanisms, including fatigue, prior to approving the use of LBB.

For new reactor plant designs such as the AP1000, there is no actual operating experience yet. To evaluate the potential of fatigue failures in new reactor plants, the designer should be able to predict the potential for thermal and vibrational fatigue based on fluid dynamic analyses. These analyses would use system and operational parameters such as mass flow rate, fluid velocity, pressure drops in the piping system, piping layout, pump speed, and fluid temperature in the pipe. Because the AP1000 plant is a light-water reactor and is similar in many respects to currently operating Westinghouse PWRs from an operational standpoint, the staff found that operating experience from existing PWRs could be extrapolated to the AP1000 design as it relates to identifying candidate piping systems that might be susceptible to fatigue. Therefore, the screening of piping systems for fatigue issues is more qualitative than quantitative.

5-1 NRC published Generic Letter (GL) 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves." To respond to this letter, the AP1000 designer should take measures to solve the problem of the susceptibility to bonnet overpressurization, pressure locking, and thermal binding so as to ensure the safety function of safety-related valves. Please introduce what kind of valve and which valves that NRC paid attention on. Related to this issue, were there any tests performed to validate the satisfied performance of the valve? If any, please provided the detailed information, and if necessary, please illustrate by drawings.

NRC Response: In GL 95-07, the NRC staff focused on resolving concerns with potential pressure locking and thermal binding of gate valves operated by power actuators (for example, motor, air, and hydraulic). Extensive evaluations including some modifications were performed on power-operated gate valves to preclude pressure locking and thermal binding of gate valves at U.S. operating nuclear power plants. The NRC staff will be prepared to discuss any specific questions and provide additional details on the regulatory requirements for ensuring the design-basis capability of safety-related power-operated valves, the pressure locking and thermal binding phenomena for gate valves, operating

experience with pressure locking and thermal binding at U.S. nuclear power plants, research activities to study the phenomena, regulatory communications related to this issue, industry actions in response to this issue, results of NRC and industry activities to address this issue, and expectations for new reactors to prevent pressure locking and thermal binding of power-operated valves.

5-2. According to the description of AP1000 PSAR_ the explosively actuated valves are used in the ADS system, IRWST and recirculation sump. Till now, we have not gotten any associated qualification information, it is said that the valve qualification will be finished around Sept. 2010. For the qualification of these valves, are there any requirements raised by NRC? Since the explosively actuated valves are important safety related valve, we care about the valve performance when the accident happens. Please introduce what's NRC most attention paid on such valves, what is the most important affect of this valve. Please explain which parts or component of the valve are critical for the function, and list the fail mode of these explosively actuated valve to make sure that the valve will perform the safety function successfully during or after the accident.

NRC Response: The NRC requires that the squib valves be adequately designed, constructed, and tested to demonstrate they will fulfill their necessary safety function. This includes requirements to meet the ASME Code to ensure adequate structural integrity, to have a safety-related power actuation system, and to undergo qualification testing to demonstrate that the valves will reliably perform their safety function under the most limiting design operational and accident conditions. The NRC plans to review and observe some qualification testing that is currently scheduled for later in 2009. Ultimately, the NRC will need to be satisfied that all aspects of the qualification have been properly performed, including verifying that the valves are fully capable of opening on demand under the necessary fluid temperature, differential pressure, and flow conditions.

5-3. The applicant explains that explosively actuated valve has been successfully used in BWR in USA, the further question is that: Are explosively actuated valve used in BWR important safety related valve? If they are safety related valve, how much is the inspection test frequency? Does it also follow the requirement of ASME OM-one times every two years_for detail see ASME OM_? In AP1000, since the explosively actuated valve is important safety related valve, whether NRC raised the requirement to increase the inspection test frequency?

NRC Response: The squib valves used in currently operating US BWRs perform a safety-related function in actuating the Stand-by Liquid Control System, which is necessary for shutting down the reactor during certain postulated events. The inspection test frequency is determined by the ASME OM Code, which is that at least 20% of the squib charges must be tested by firing them every two years, with all charges required to be tested within the qualified life, not to exceed 10 years. If any charges fail, all charges in that batch are required to be replaced with those from another batch that has been tested. The squib valves in the AP1000 design will be required to be tested according to the ASME OM Code requirements.

5-8 For inservice inspection of Class 2 and 3 components, is it required that ultrasonic examination system is qualified in accordance with requirements of ASME Section XI, Appendix VIII.

NRC Response: For Class 3 components, no volumetric examinations are required. For Class 2 components such as welds and bolting/studs, the procedures, equipment, and personnel must be qualified in accordance with ASME Section XI, Appendix VIII. Refer to Mandatory Appendix I, Article I-2000 for exclusions.

5-9 Please introduce Inservice Testing (IST) and IST program of snubbers and their review requirements.

NRC Response: The NRC regulations in 10 CFR 50.55a require U.S. operating nuclear power plants to meet the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code) with some additional requirements in performing inservice testing of pumps, valves, and snubbers. The NRC staff will be prepared to discuss any specific questions and provide additional details on the regulatory requirements for periodically assessing the operational readiness of pumps, valves, and dynamic restraints at nuclear power plants; IST provisions in the ASME OM Code; operating experience with IST programs at U.S. nuclear power plants; research activities related to valve qualification and performance; regulatory communications related to IST activities; industry actions in response to IST regulatory communications; results of NRC and industry activities to improve IST programs; NRC review of IST program descriptions in Design Certification and Combined License (COL) applications, including IST programs for snubbers; and NRC inspection of IST programs developed and implemented at new reactors following COL issuance.

3-8. In Subsection 5.3.4.6 of Sanmen NPP PSAR, the screening criterion for PTS in 10 CFR 50.61 is used, but the criterion is based on the risk evaluation of the old NPPs. AP1000 is newly designed and adopts a direction vessel injection that is different from the old NPPs. Is the screening criterion for PTS in 10 CFR50.61 is still valid for AP1000 or not?

5-7 The screening criterion for PTS in 10 CFR50.61 is used for AP1000, the criterion is based on the risk evaluation of the old NPPs, but AP1000 is newly designed, please explain why the criterion is valid for AP1000.

NRC Response to Questions 3-8 and 5-7: NRC did not believe it was necessary to revisit the PTS issue for AP1000 even though some of the transients affecting PTS are different. The reasons include the following:

- The original PTS rule as stated in 10 CFR 50.61 is based on detailed risk analysis and the consideration of large number of transients. Current PTS analysis used highly conservative assumptions in arriving at PTS screening criteria. These conservatisms were discussed in detail in the preparation of the proposed PTS revision 10CFR50.61a. Because of the differences in AP1000 transients with current PWRs, NRC decided not to use the proposed revision to the PTS rule in 10 CFR 50.61a for the AP1000.

- In addition to the in-built conservatisms of 10 CFR 50.61, the AP1000 vessel will

    a. Be built with much better materials, specifically, beltline materials will have very low Cu, Ni, P.
    b. Have a diameter and thickness that are similar to current PWRs.
    c. Will not have any axial or longitudinal welds in the beltline region. Welds are the major source of flaws.

d. Have an additional neutron shield that will reduce the projected cumulative fluence compared to current PWR vessels.

In summary, RT_{PTS} values for AP1000 for 60 years are projected to be well below 270^0 F, which is the current PTS screening criterion for forgings.

Therefore, considering the AP1000 vessel materials and projected cumulative fluence up to 60 years as described above, the use of current PTS screening criteria is justified. The staff does not believe any additional studies on the applicability of the current PTS rule for the AP1000 is warranted.

5-5(2) Is the flywheel's 125% over speed running test reasonable, which is operated alone in air in the canned motor pump model test and product test? Whether the US criteria require the flywheel should run the 125% over speed test in the running medium together with the rotor or not?

NRC Response: Currently, the U.S. criteria for testing the flywheel is to perform a spin test at the design overspeed (typically 125% of normal operating speed). Therefore, the AP1000 flywheel 125% overspeed test is reasonable and can be performed in any medium. The spin test is to verify the integrity and capability to withstand design overspeed of the fabricated flywheel.

6-5. As regards the downstream effects of sump screen, comparing AP1000 with traditional PWR design, the traditional PWR will only permit debris small than the sump screen size get into the reactor coolant system, but according to the design of AP1000, the debris larger than sump screen size (0.125") can get into the reactor coolant system through the break during and after LOCA if the break is located in the RCS main pipe, that will exceed the envelopment of the downstream effect analysis after LOCA for traditional PWR. Whether AP1000 have provided specific downstream effect analysis report?

NRC Response: Westinghouse has provided a technical report (TR 26) that includes an analysis of the downstream effects of the estimated larger debris that could enter a postulated break of the direct vessel injection piping. The staff has identified several issues with the Westinghouse analysis and is requesting additional information regarding both the quantity and characterization of the debris and its effects on downstream components.