NRC Staff's Response to the ACRS Interim Letter Regarding the AP1000 Design

Draft SER and Design-Basis Compliance

Issue 1 - Automatic Depressurization System (ADS)-4 Squib Valve Function:

The most important safety function in the AP-1000 design is the automatic depressurization of the primary system. We have had discussions on the performance characteristics of the ADS-4 Squib valves. We agree with the staff that inspections, test, analyses, and acceptance criteria (ITAAC) should be used to assure that the combined license (COL) inspection and testing program verifies that these valves meet the design-basis specifications.

Staff Response

During the ACRS Full Committee meeting on March 3-6, 2004, the staff indicated that the ADS-4 squib valves are designed, constructed and tested with the requirements set forth in Section III of the Boiler and Pressure Vessel Code promulgated by the American Society of Mechanical Engineers (ASME) and are actuated by redundant and diverse instrumentation and control systems.

The staff also stated we have performed a sensitivity study by increasing the failure probability and the common cause failure probability of the ADS-4 squib valves by an order of magnitude. This sensitivity study indicated that the core damage frequency (CDF) increased by only a factor of three and was not large enough to impact the probabilistic risk assessment (PRA) conclusions and insights about the AP1000 design.

In addition, the staff indicated that AP1000 Design Control Document (DCD) Tier 1, Table 2.1.2-4, Item 12.a includes an ITAAC that ensures that the ADS-4 squib valves will perform their active safety-related function to change position. Item 12.a(iv) specifies that tests or type tests of ADS-4 squib valves will be performed to demonstrate the capability of the valves to operate under their design conditions. A test is defined as the actuation, operation, or establishment of specified conditions to evaluate the performance or integrity of as-built structures, systems, or components. A type test is defined as a test on one or more sample components of the same type and manufacturer to qualify other components of the same type and manufacturer. A type test is not necessarily a test of the as-built structures, systems, or components. Therefore, the staff concludes that the performance characteristics of the ADS-4 squib valves will be adequately verified.

Issue 2 - Assurance of Long-Term Cooling (Strainer Blockage):

The AP1000 appears to incorporate a robust design to prevent sump screen blockage. The design utilizes screen areas slightly larger than those of current pressurized water reactors (PWRs); locates the screens higher off the floor with a flow guard overhead; uses deeper water levels; uses much lower recirculation flow rates and consequent lower flow velocities approaching and entering the screen; and uses reflective insulation and high density non-safety coatings. Since the issue of ensuring long-term cooling is still under regulatory discussion, we *recommend that the AP1000 design for this be the subject of ITAAC to ensure that it complies with the generic regulatory resolution of this issue.*

Staff Response

DCD Tier 1, Table 2.2.3-4, Item 8.c contains an ITAAC that ensures that the passive core cooling system (PXS) provides reactor coolant system (RCS) makeup, boration, and safety injection during design basis events. This ITAAC includes verification of (1) the location of the plates above the containment recirculation screens, (2) the surface area of the in-containment refueling water storage tank (IRWST) and containment recirculation screens, (3) the location of the bottom of the containment recirculation screens, (4) the type of insulation used inside containment, and (5) the dry film density of the coatings used inside containment.

DCD Tier 2, Section 6.3.8.1, “Containment Cleanliness Program,” states that the COL applicant will develop a program to limit the amount of debris that might be left in the containment following refueling and maintenance outages. This program will limit the storage of outage materials (such as temporary scaffolding and tools) inside containment during power operation.

DCD Tier 2, Section 6.3.8.2, “Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA,” states that the COL applicant will perform an evaluation consistent with Regulatory Guide (RG) 1.82, Revision 3, to demonstrate that adequate long-term core cooling is available considering debris resulting from a loss-of-coolant accident (LOCA) together with debris that exists before a LOCA. As discussed in DCD Tier 2, Section 6.3.2.2.7.1, a LOCA in the AP1000 design does not generate fibrous debris attributable to damage to insulation or other materials. The evaluation will consider resident fibers and particles that could be present considering the plant design, location, and containment cleanliness program. The determination of the characteristics of such resident debris will be based on sample measurements from operating plants. The evaluation will also consider the potential for the generation of chemical debris (precipitants). The potential to generate such debris will be determined considering the materials used inside the AP1000 containment, the post-accident water chemistry of the AP1000, and the applicable research/testing.

By letter dated January 13, 2004, Westinghouse changed the design of the containment recirculation screens and the IRWST screens by increasing the fine screen area by at least a factor of 2 to 13 m^2 (140 ft^2) or more by using a folded screen design. An increased screen area will allow the screen to tolerate more debris, while lowering the water velocity at the screen face. Westinghouse also added a cross-connection pipe between the two containment recirculation screens. This design change was incorporated in DCD Tier 2, Section 6.3.2.2.7, “IRWST and Containment Recirculation Screens,” and Table 6.3-2, “Component Data - Passive Core Cooling System.” Based on the above design changes in concert with the cleanliness program in DCD Tier 2, Section 6.3.8.1, the minimal fibrous materials used in containment, and the other screen design features described in the DCD, the staff considers the capability of the containment recirculation screens and the IRWST screens to accommodate anticipated debris loadings to be acceptable. The staff finds that the COL action item in DCD Tier 2, Section 6.3.8.2 will capture any impact on the ability of the containment recirculation screens and the IRWST screens to accommodate anticipated debris loadings identified during the resolution of Generic Safety Issue (GSI)-191, “Assessment of Debris Accumulation on PWR Sump Performance,” and that those impacts can be addressed using programmatic means.

During the development of Revision 3 to RG 1.82, “Water Sources For Long-term Recirculation Cooling Following a Loss-of-coolant Accident,” the staff identified concerns related to additional debris that can be caused by chemical reactions in the containment. In a letter dated November 12, 2003, the staff asked Westinghouse to address the following issues:

1. To minimize potential debris caused by chemical reaction of the pool water with metals in the containment, exposure of bare metal surfaces (e.g., scaffolding) to containment cooling water through spray impingement or immersion should be minimized either by removal or by chemical-resistant protection (e.g., coatings or jackets).

2. In addition to debris generated by jet forces from the pipe rupture, the analyses should consider debris created by the resulting containment environment (thermal and chemical). Examples of this type of debris would be disbondment of coatings in the form of chips and particulates or formation of chemical debris (precipitants) caused by chemical reactions in the pool.

In its response, dated November 26, 2003, Westinghouse explained that the AP1000 is designed such that there should not be a need for temporary scaffolding during outages, and the large storage areas outside the containment will eliminate the need for the storage of outage material in containment. Westinghouse revised the COL action item in DCD Tier 2, Section 6.3.8.1 to have the containment cleanliness program limit the storage of outage material in containment. Also, Westinghouse explained that its preferred approach is to use materials that do not need coatings or have permanent coatings to minimize coating disbondment.

Westinghouse also revised the COL action item in DCD Tier 2, Section 6.3.8.2 to include the evaluation of chemical debris. The staff finds that this COL action item will capture any impact of chemical effects on the ability of the affected components to accommodate anticipated debris loadings identified during the resolution of GSI-191 and those impacts can be addressed using programmatic means. Therefore, the staff concludes that Westinghouse has resolved the staff's concerns related to additional debris that can be caused by chemical reactions in the containment.

Codes and Validation Testing

Issue 3 - Code Deficiencies:

When deficiencies such as these are identified in codes, they should lead to the consideration of research programs to correct the weaknesses and avoid resorting to a patchwork of ad hoc methods.

Staff Response

The AP1000 review helped to identify deficiencies in both NOTRUMP and the staff’s RELAP5 thermal-hydraulic codes. Neither code was found to adequately simulate integral test data from the AP1000 Advanced Plant Experiment (APEX-AP1000) facility. While each code has deficiencies unique to its own formulation and set of correlations used, phase separation in the upper plenum and hot leg with the resultant carryover of liquid to the ADS-4 was difficult to predict with much certainty. For AP1000, an acceptable solution was to conservatively bound the calculations. The TRACE code was not used in the AP1000 review, but is currently being assessed using APEX-AP1000 test data.

Two ongoing experimental studies may help to correct code deficiencies. First, the Air-water Test Loop for Advanced Thermal-hydraulic Studies (ATLATS) facility at Oregon State University has been modified so that visualization and phase separation in a (scaled) upper plenum can be observed and measured. Second, several separate effects tests are planned in the APEX-AP1000 facility to isolate and measure upper plenum pool entrainment. These new data, in addition to existing data from the Upper Plenum Test Facility (UPTF) program will be used to evaluate upper plenum entrainment / de-entrainment models in TRACE as part of the ongoing code development and assessment process. New thermal-hydraulic models will be developed and implemented into TRACE as deficiencies are identified. These experimental programs, along with the associated TRACE assessment can be described to the ACRS Thermal-Hydraulic Subcommittee at a future meeting.

Issue 4 - Range of Pi-Group Values:

We have yet to be shown a sufficient technical justification that a range of 0.5 to 2.0 for various scaling Pi-groups represents general adequate scaling. We, therefore, recommend that the staff undertake confirmatory research on pertinent scaling issues for relating test facilities to prototypic systems and verify that the Pi group range of 0.5 to 2.0 is appropriate.

Staff Response

Scaling evaluations in the AP1000 review assumed an acceptance range of 0.5 to 2.0 for various Pi-groups in identifying distortions between test facilities and the full-scale prototype. This range does not have a firm technical basis, but has been used as a de facto standard in scaling analyses. It should be noted, however, that the question regarding the appropriate acceptance range for Pi groups is generic and does not represent an issue that is specific only to AP1000.

For the AP1000, acceptability of this Pi group range was taken into account for selected parameters that were considered crucial in understanding the AP1000 passive safety system performance. For these crucial parameters, the staff’s scaling evaluation was not solely based on whether the Pi groups are within the range of 0.5 to 2.0, but also on other considerations. At the ACRS Thermal-Hydraulic Subcommittee meeting on February 14-15, 2002, we described a procedure to be followed which would examine the importance of a particular Pi group. The procedure included performing a sensitivity study using a non-dimensional, simplified model of a prototype for a particular period of a transient. The staff examined the ADS-4 blowdown period, confirming the importance of liquid entrainment to the ADS-4 during this period and that the critical Pi group for this period (which was approximately 2.0), was not sufficiently scaled. We believe that this is an example of how an acceptable Pi group range might be established. As a long term effort, the staff will work to develop and document a procedure to define an appropriate Pi group range for scaling integral test facilities.

Materials

ACRS Comment

Several items of concern relating to materials degradation were identified during our reviews. These ranged from quality assurance (QA) criteria for Alloy 52/152 weldments, to fracture toughness of high chromium nickel-base alloys under specific operating conditions, to the stress corrosion resistance of some alloys currently regarded as immune to such failure. The applicant believes the best alloys have been selected for these applications based on currently available information. Ongoing and future studies may suggest material and environmental changes that will be addressed at the COL stage.

Staff Response

The process to change the type of material or alloy used in the AP1000 design at the COL stage depends on who is requesting the change, and where the material or alloy is specified in the DCD. Section VIII of Appendices A, B, or C to Title 10, Part 52, of the *Code of Federal Regulations* (10 CFR Part 52) sets forth the process for generic changes to or plant-specific departures from the DCD. Section VIII is also divided into three paragraphs, which correspond to Tier 1, Tier 2, and operational information. Generic changes must be accomplished by rulemaking because the intended subject is the design certification rule itself. These rulemakings must include an opportunity for hearing with respect to the proposed change. Plant-specific departures could be either a Commission-issued order to one or more applicants or licensees; or an applicant or licensee-initiated departure applicable only to the applicant’s or licensee’s plant(s), similar to a 10 CFR 50.59 departure or exemption.

If the material or alloy is specified in Tier 1 information, and the change is generic, rulemaking is needed to amend the generic DCD and the process is governed by the standards in 10 CFR 52.63(a)(1). Departures from Tier 1 may occur if the Commission orders a licensee to depart from Tier 1, as provided in subparagraph A.3, or the applicant or licensee requests an exemption from Tier 1, as provided in subparagraph A.4.

The change processes for the different categories of Tier 2 information (Tier 2, Tier 2*, and Tier 2* with a time expiration) are set forth in paragraph VIII.B. The change processes for Tier 2 have the same elements as the Tier 1 change processes, but also have some different standards for plant-specific orders and exemptions. The process for generic Tier 2 changes tracks the process for Tier 1 changes. Departures from Tier 2 may occur in five ways. Specifically, (1) the Commission may order a plant-specific departure, as set forth in subparagraph B.3; (2) an applicant or licensee may request an exemption from a Tier 2 requirement as set forth in subparagraph B.4; (3) a licensee may make a departure without prior NRC approval in accordance with subparagraph B.5 [the “10 CFR 50.59-like” process]; (4) the licensee may request NRC approval for proposed departures that do not meet the requirements in subparagraph B.5 as provided in subparagraph B.5.d; and (5) the licensee may request NRC approval for a departure from Tier 2* information under subparagraph B.6.

The change process for technical specifications and other operational requirements in the DCD is set forth in paragraph VIII.C. This change process has elements similar to the Tier 1 and Tier 2 change process, but with significantly different change standards. The key to using this change process is to determine whether the proposed change or departure requires a change to a design feature described in the generic DCD. If a design change is required, the appropriate change process in paragraph VIII.A or VIII.B applies. However, if a proposed change to the technical specifications or other operational requirements does not require a change to a design feature in the generic DCD, then paragraph VIII.C applies. The language in paragraph VIII.C also distinguishes between generic (Section 16.1 of the DCD) and plant-specific technical specifications to account for the different treatment and finality accorded technical specifications before and after a license is issued.

In conclusion, any future material and environmental changes that would be requested during the COL stage would have to follow the change processes which are discussed in the paragraphs above.

Severe Accidents

ACRS Comment

The ACRS and the staff have questioned the technical justification for the aerosol removal coefficient (lambda) for containment. The issue has been addressed by Westinghouse using the STARNAUA code for the limiting sequence. We understand that the staff is using the MELCOR code to calculate the time dependence. We look forward to reviewing the staff's analysis.

Staff Response

As discussed in DCD Tier 2 Appendix 15B, the AP1000 design relies on natural aerosol removal processes for deposition in the containment such as gravitational settling and plateout on containment surfaces through diffusiophoresis and thermophoresis. In DCD Tier 2 Table 15B-1, Westinghouse provides the AP1000 aerosol removal coefficient values starting at the onset of gap release through the first 24 hours into a design-basis accident (DBA). The values range between 0.444 to 1.147 per hour and conservatively neglect steam condensation on the airborne particles, turbulent diffusion, and turbulent agglomeration.

Westinghouse's methodology includes the industry's MAAP code, an integrated accident analysis program, to establish thermal-hydraulic boundary conditions as an input to a deterministic aerosol code (STARNAUA). Westinghouse did perform some sensitivity studies to determine whether changing the parameters would affect the overall removal coefficients. However, Westinghouse's calculation is based on an analysis of a single T-H scenario (3BE-1 accident sequence) and uses a single aerosol model without providing an uncertainty analysis. It is the staff's opinion that the Westinghouse approach represents a single best-estimate result. The chosen 3BE-1 scenario is considered by staff to be acceptable for use in the aerosol deposition calculations because (1) it is representative of the "3BE" accident class, which is the dominant contributor to the core damage frequency for the AP1000; (2) the thermal hydraulic conditions for 3BE accidents are typical of most of the analyzed sequences; (3) the corresponding thermal hydraulic profiles for these depressurized and reflooded cases are sufficiently similar to each other within the 3BE accident class; and (4) the use of a fully depressurized, low pressure accident sequence in conjunction with the source term described in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," is appropriate because the release fractions for the source terms presented in NUREG-1465 are intended to be representative or typical of those associated with a low pressure core-melt accident. Although the choice of scenario is acceptable, the staff believes that a best estimate approach requires performing an evaluation of associated uncertainties.

The staff used an alternative thermal-hydraulic code (MELCOR), which includes the MAEROS aerosol mechanics code, as an input to a Monte Carlo sampling (200 runs) of 13 parameters known to affect aerosol settling and depletion. Engineering judgement was used for the choice of parameters, as well as for the range and distribution of their values, after several discussions between the staff and the contractor. The resultant distribution of possible aerosol removal coefficients has a 95-percent level of confidence, as depicted in Figure 1.

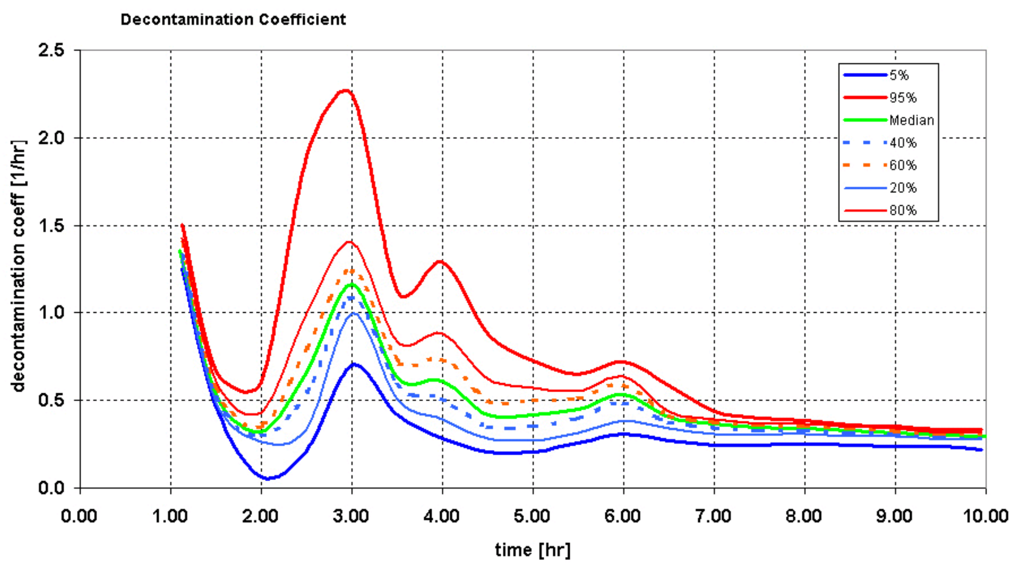

Figure 1: Uncertainty bands of aerosol removal rates (lambdas)

Although the Westinghouse calculated aerosol removal coefficient values lie within the staff's uncertainty analysis upper and lower bounds for portions of the early part of the accident, they generally do not lie below the median value calculated by the staff. While the staff does find that gravitational settling, thermophoresis and diffusiophoresis are physical processes that occur in the AP1000 containment and credit may be taken for aerosol removal through these processes, the staff is not approving the specific Westinghouse calculated aerosol removal coefficient values. These values are an intermediate product used in the dose analysis, and are not subject to any regulations.

The staff performed an independent dose analysis with the median aerosol removal coefficient values from the staff's uncertainty analysis, along with other analysis parameters and the bounding hypothetical atmospheric dispersion factors provided by Westinghouse, and the results are within the dose criteria of 10 CFR 50.34 and General Design Criterion (GDC)-19. The staff performed a sensitivity analysis which results in calculated doses that remain below the regulatory acceptance criteria for aerosol removal coefficients as low as those given in the 20^{th} percentile in the staff's uncertainty analysis. Thus, while the staff and Westinghouse diverge on values for the intermediate steps in the dose calculations, the staff agrees with the overall conclusion that the AP1000 design results in acceptable doses.

Issue 5 - In-Vessel Retention/Fuel-Coolant Interactions (FCI):

The assessment of in-vessel retention has not included exothermic intermetallic reactions which have been shown by some prototypic experiments to be important. If these factors are properly accounted for, the associated energetics of any resulting ex-vessel steam explosions are likely to be greater than has been currently evaluated. We would like to review the FCI models used and see additional justification that the initial conditions related to intermetallic reactions will not give rise to an energetic FCI that could fail containment.

Staff Response

For the AP1000 design, Westinghouse considered a melt configuration in the lower head whereby a layer of oxidic melt is sandwiched between a layer of heavy metal melt at the bottom and a layer of light metal melt at the top. The staff analyzed the same configuration in its independent assessment of the in-vessel retention capability of the AP1000 design, and concluded that the lower head would likely fail on the side (due to the focusing effect of the light metallic layer) in that configuration. The staff's analysis did not include exothermic intermetallic reactions. However, the staff recognized that if there is a significant quantity of heavy metals that can form the bottom layer, the vessel failure at the RPV bottom location cannot be ruled out. Therefore, for the ex-vessel FCI analysis, the staff considered a bottom failure scenario whereby metallic melt at a higher superheat may be released. For the calculation of FCI loads, the initial and boundary conditions of importance are melt composition, melt temperature (superheat), water temperature (subcooling), melt flow rate, and geometry of vessel and containment. The staff performed a reasonably large number of sensitivity analysis bounding these parameters, and found that the ex-vessel FCI for AP1000 is of no greater concern than that for AP600. The ex-vessel FCI analysis for AP600 indicated that the containment integrity would not be challenged by the FCI load, although the auxiliary structures might experience some damage. As far as the ex-vessel FCI is concerned, the bottom failure will, in fact, have some mitigative effects, in that the increased distance between the explosion zone in the center of the cavity and the cavity wall will dampen pressure propagation. The staff plans to provide the ACRS a copy of its independent evaluation to support the ACRS Full Committee Meeting on June 3, 2004.

Issue 6 - Organic Iodine Production:

The acidification of containment water as a result of radiolysis of organic material could give rise to significant airborne fission product iodine in gaseous organic form. We need to review how Westinghouse and the staff have dealt with this potential.

Staff Response

Westinghouse committed, in the enclosed letter dated April 30, 2004, to provide the staff additional information regarding the pH of the water film on the inside of the containment wall where acidification could produce organic iodine. The staff plans to provide this information to the ACRS along with the staff's evaluation of Westinghouse's conclusion that the iodine in the film will not re-evolve.

Issue 7 - Catastrophic Failure of the Steel Containment

There is experimental evidence that a free-standing steel containment can fail in a catastrophic manner when its failure pressure is exceeded. Such a failure mode can lead to very rapid depressurization and, potentially, to resuspension of fission products that have been previously deposited or settled out. While the surrounding concrete structure of the AP1000 design may impede such a catastrophic depressurization, we would, nevertheless, like to see a sensitivity study on the fission product source term to assess the potential maximum effect on the risk of latent fatalities as compared to the Safety Goal.

Staff Response

Catastrophic failures of the AP1000 containment are possible, but the frequency of such events is so small that potential fission product resuspension would not have a noticeable impact on risk metrics such as large release frequency or compliance with the Commission's safety goals. Catastrophic failures can result from early energetic loads, such as fuel coolant interactions, or gradual over-pressure in the intermediate or late time-frames. The majority of early containment failures (~5E-9/year) are associated with a conservative assumption that failure of external reactor vessel cooling (ERVC) will result in early containment failure from ex-vessel phenomena (DSER Section 19.1.3.2.2.2). However, deterministic calculations of ex-vessel phenomena indicate that containment integrity will be maintained (DSER Sections 19.2.3.3.3 through 19.2.3.3.5). Thus, events involving ERVC failure would more likely result in either late or no containment failure. An additional 10 percent of early containment failures (<1E-9/year) occur as a result of hydrogen detonations. Although the source terms for these releases could be increased by resuspension, this would not have a noticeable impact on risk given the low accident frequency. Intermediate and late containment failures have a combined frequency of about 2E-10/year in the baseline PRA, and 5E-9/year if all events involving ERVC failure are included. Even if the source terms for these release categories were increased as a result of resuspension, this would not noticeably impact risk because of the low accident frequency.

It is important to recognize that even if an event progressed to an intermediate or late release, it would likely involve a vented release rather than a catastrophic containment failure, since the AP1000 design will include the capability to vent the containment (DSER Section 19.2.3.3.8) and severe accident management guidance on use of the vent (DSER Section 19.2.5). The non-safety-related containment spray function could also be effective in reducing re-suspended fission products following containment failure. However, the availability of this function cannot be ensured following catastrophic containment failure.