From: Belkys Sosa
To: Internet: creid@bechtel.com; Internet: langmanv@aecl.ca; Robert, Ion,
Date: 2/26/04 2:59PM
Subject: ACR-700 Phase 2 Meeting on DBA/SA - April 6-7, 2004

Vince,
The enclosed document is provided for guidance in preparation for the subject meeting (this document was originally provided as guidance for the PIRT panel meetings). In general, the meeting agenda should reflect that the purpose of the subject meeting is to discuss ACR compliance with SRP Chapter 15 in terms of event categorization and its associated acceptance criteria. Justification for any deviation from the SRP should also be discussed.

If you have any questions, please don't hesitate to contact me at 301-415-2375.

Thanks,
Belkys Sosa
Project Manager, ACR

CC: Bajorek, Stephen; Basu, Sudhamay; Carlson, Donald; Dennig, Robert; Drozd, Andrzej; Dudes, Laura; Flack, John; Kim, James; Lee, Jay; Lee, Richard; Palla, Robert; Stutzke, Martin; Sun, Summer

ACR-700 Reactor Pre-Application Review
Design Basis Accident and Acceptance Criteria
and
Severe Accidents

Design Basis Accidents

NRC's licensing review of the ACR-700 design certification will be guided by the regulatory requirements set forth in 10 CFR Part 52, Subpart B, "Standard Design Certification." Specifically, NRC will be reviewing technical information submitted by an applicant in accordance with 10 CFR 52.47 which makes reference to (1) 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," and (2) the site parameters postulated for the design and an analysis and evaluation of the design in terms of such parameters, among others.

The regulation in 10 CFR 50.34 requires an applicant to provide *the safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before release of radioactive material to the environment can occur. Special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents. In performing this assessment, an applicant shall assume a fission product release from the core into the containment assuming that the facility is operated at the ultimate power level contemplated.* A footnote to 10 CFR 50.34 states that *the fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purpose of site analysis or postulated from considerations of possible accidental events. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.*

The regulation also requires that an applicant performs an evaluation and analysis of postulated fission products release to determine that:

> *(1) An individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the postulated fission product release would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE).*
>
> *(2) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release during the entire period of its passage would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE).*

In reviewing the licensing application in performing an independent confirmatory evaluation, the NRC staff will be asking the following typical questions:

(1) Describe the safety features that are to be engineered into the ACR-700 design and those barriers that must be breached as a result of an accident before release of radioactive material to the environment can occur.

(2) Describe the major reactor accident and its progression hypothesized for the purpose of the radiological consequence analysis for the ACR-700 design.

(3) Describe fission product transport and mitigation phenomena that are credited in your radiological consequence analyses. Provide bases for methodology used in quantifying these phenomena.

(4) Provide the following fission product release characteristics (source term) including significant activation products (e.g., tritium) as a result of the major reactor accident postulated in response to item (2) above:

- Fission product inventory in the reactor core operated at the ultimate maximum proposed power level with the limiting condition which maximizes fission product releases.

- Times and rates of fission product release from the fuel following the major reactor accident. The fission product appearance rates in containment should be fractions of fission product inventory in the reactor core based on the maximum full power operation. Provide its technical bases, references, and/or justification.

- The isotopic quantities and the chemical forms of fission products released to the environment.

- Rates of fission product release to the environment during the entire period of accidents as a function of time.

(5) Provide hypothetical and bounding site meteorological parameters (atmospheric dispersion factors) to be used in the selection of a suitable site for a facility referencing the ACR-700 design.

(6) Provide the radiological consequence analysis for the major reactor accident postulated in response to item (2) above using the fission product release characteristics assumed in response to item (4) above, and the bounding site meteorological parameters assumed in response to item (5) above, to determine that the ACR-700 design will meet the dose acceptance criteria specified in 10 CFR 50.34(a)(1).

(7) For light-water-cooled reactors, the large-break LOCA is typically the maximum credible accident. However, the staff experience in reviewing license applications has indicated the need to consider other accident sequences of lesser consequence but higher probability of occurrence such as:

Steam generator tube rupture accident
Main steam line break accident
Spent fuel handling accident
Small-break LOCA

For the ACR-700 design, identify other design basis accident sequences (other than the major reactor accident postulated in response to item 2 above) of lesser consequence but higher probability of occurrence and rank them based on probability of occurrence.

(8) The dose criteria in 10 CFR 50.34(a)(1) are predicated on a major reactor accident involving core melt. For the design basis accidents of lesser consequences identified and ranked in item (7), provide dose acceptance criteria and rationale.

(9) Provide scenario-specific fission product release characteristics (source terms) as requested in item (4) above for those design basis accidents identified in response to item (7) above for the ACR-700.

(10) Provide the radiological consequence analyses as requested in item (6) above for those design basis accidents identified in response to item (7) above.

(11) Provide the following bounding thermal-hydraulic conditions as a function of time in the ACR-700 containment following the major reactor accident postulated in response to item (2) above:

Containment pressure
Gas temperature
Steam condensation rate on the wall
Steam quality
Total heat transfer rate
Containment air mixing
Stratification
Stagnation
Hydrogen separation

(12) Predict the following fission product aerosol characteristics and its behavior in the ACR-700 containment following the major reactor accident postulated in response to item (2) above:

Non-radioactive aerosol
Aerosol removal coefficients
Aerosol deposition mechanisms
Aerosol shape factor, size, and density (for dispersion calculations and for evaluation of control room habiltability per GDC19)
Aerosol concentrations in the containment
Iodine deposition and re-suspension

Severe Accidents

Severe core damage accidents (severe accidents) are low probability events beyond the design basis accident established in 10 CFR 50.34 that can lead to significant core damage and subsequent release of fission products from the reactor. The NRC has requirements to address severe accidents, such as anticipated transients without scram (10 CFR 50.62), station blackout (10 CFR 63), and combustible gas control (10 CFR 50.44); however a definitive set of regulatory requirements for addressing specific severe accident phenomena does not exist. Instead, the Commission has developed guidance and goals for resolving safety issues related to reactor accidents more severe than design basis accidents. Regulatory guidance pertaining to severe accidents can be found in Federal Register (50 FR 32138) dated August 8, 1985, "NRC Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," and SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor Designs," and corresponding SRM dated July 21, 1993. SECY-93-087 discusses the staff technical and policy issues pertaining to evolutionary and advanced light-water reactor design certification and it includes severe accident preventative and mitigative feature issues.

The staff will request AECL to identify and address severe accident issues relevant to ACR-700 design. The staff will assess this information and will use an approach similar to that taken by the staff in SECY-93-087 to establish design requirements/criteria for each.

Request for Clarification:

In many places AECL uses the following note (e.g., 108-03500-AB-002 Rev 1 Page 1-1):

Note: For U.S. licensing, the maximum pressure in MSLB with an additional 10% allowance shall be within the containment design pressure.

Does this mean:

(1) Calculate maximum MSLB pressure add 10% and then compare to design pressure, for example design pressure is 50 psia then MSLB calculation would have to be about 45 psia or less (10% below design value 45 + 5 = 50)

or

(2) Calculated MSLB pressure is not greater than design pressure + 10%, for example would allow 55 psia if 50 psia is design pressure (55 - 5 = 50).