From: Thompson, Jon
Sent: Thursday, December 02, 2010 3:48 PM
To: Rudy, Lawrence J
Cc: 'Hart, Randy'; Blumberg, Mark; Martin, Kamishan; Bettle, Jerome; Hamm, Matthew; Kulesa, Gloria; OBrien, Margaret; Blechman, Paula
Subject: Request for Information for Amendment dated May 20, 2010

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 (CATAWBA 1 AND 2), REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING LICENSE AMENDMENT REQUEST RELATED TO ENABLING THE OPENING OF THE REACTOR BUILDING PRESSURE BOUNDARY WITH ADMINISTRATIVE CONTROLS IN PLACE (TAC NOS. ME3982 AND ME3983)

By letter dated May 20, 2010, Duke Energy Carolinas, LLC (the licensee), submitted a proposed license amendment to change the Catawba 1 and 2 Technical Specifications (TSs). The proposed change would change TSs 3.6.10 and 3.6.16 to enable the opening of the reactor building pressure boundary with administrative controls in place.

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the licensee's submittal and determined that additional information is needed in order to complete our review. The enclosed document describes this RAI. A written response should be provided to the NRC staff for all these RAI questions, with the exception of number five, by January 13, 2011, in order to support our timely review of this application. The response for RAI question number 5 is expected by February 3, 2011. Please inform me as soon as possible if you are unable to support this response timeframe.

If you have any questions, please call me at 301-415-1345.

Sincerely,

Jon Thompson, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-413, 50-414

Enclosure:
As stated

REQUEST FOR ADDITIONAL INFORMATION (RAI)

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REGARDING LICENSE AMENDMENT REQUEST (LAR) RELATED TO

ENABLING THE OPENING OF THE REACTOR BUILDING PRESSURE BOUNDARY

WITH ADMINISTRATIVE CONTROLS IN PLACE

CATAWBA NUCLEAR STATION, UNITS 1 AND 2 (CATAWBA 1 and 2)

The following RAI from the Nuclear Regulatory Commission (NRC) staff pertains to the LAR to Enable Opening of the Reactor Building Pressure Boundary with Administrative Controls in Place for Catawba 1 and 2 as described in the LAR submitted by letter dated May 20, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML101470066), submitted by Duke Energy Carolinas, LLC (Duke, the licensee):

1. Inserts 1 and 3 of Attachment 2 of the LAR, link the operability of the Annulus Ventilation System (AVS) to the Catawba 1 and 2 Technical Specification (TS) 3.6.16, "Reactor Building," which allows 24 hours to restore operability of the reactor building pressure boundary before an orderly shutdown of the reactor is required. Insert 5 of Attachment 2 of the LAR states in part that "if the reactor building pressure boundary is inoperable such that the AVS trains cannot establish or maintain the required negative pressure to prevent unfiltered primary containment leakage from the reactor building, … appropriate compensatory measures (consistent with the intent, as applicable, of GDC [General Design Criteria] 19, GDC 41 [of Appendix A, "General Design Criteria" to Title 10 of the *Code of Federal Regulations* Part 50 (10 CFR Part 50)] and 10 CFR 50.67) *should* [emphasis added] be utilized to protect the plant personnel and boundary from radiological releases." The condition can be entered for planned and unplanned entries into the condition.

   For the changes described above (Inserts 1, 3, and 5), the licensee proposes to exchange previously approved safety-related engineered controls for compensatory measures which are undefined (in the TSs or TS bases). Currently, these safety-related engineering controls are required to be operable, but the new proposed compensatory actions are optional. Previous methods, used by the licensee to determine the adequacy of the engineered controls, were provided by the licensee in an application required by 10 CFR 50.90[1] "Application for amendment of license, construction permit, or early site permit." These methods and controls were reviewed and approved by the NRC staff. The proposed change does not define any specific engineered controls or methods (i.e. Regulatory Guidance (RG)) used to determine the adequacy of the proposed controls. The proposed change, therefore, in total, shifts the control for determining the adequacy and presence of the reactor building boundary (and associated compensatory actions) for protecting the health and the safety of the plant personnel and the public to the licensee's control.

   Currently, to provide reasonable assurance that limits in 10 CFR 50.67 are met, safety-related engineered controls are required and were approved using methods found acceptable to the NRC staff. Please justify how the proposed changes provide the same level of reasonable assurance that the limits in 10 CFR 50.67 are met or justify why this is not necessary.

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[1] 10 CFR 50.90, " states: "Whenever a holder of a license or construction permit desires to amend the license (including the Technical Specifications incorporated into the license) or permit, application for an amendment must be filed with the Commission, as specified in § 50.4, fully describing the changes desired, and following as far as applicable, the form prescribed for the original applications."

2. RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000 (ADAMS Accession No. ML003716792), Regulatory Position C.1.3.2, "Reanalysis Guideline," states in part:

   > The NRC staff does not expect a complete recalculation of all facility radiological analyses, but does expect licensees to evaluate all impacts of the proposed changes and to update the affected analyses and the design bases appropriately. An analysis is considered to be affected if the proposed modification changes one or more assumptions or inputs used in that analysis such that the results, or the conclusions drawn on those results, are no longer valid.

   Also, RG 1.183, Section B, "Discussion," states that with respect to evaluating the response of a facility's engineered safety features:

   > Although the LOCA is typically the maximum credible accident, NRC staff experience in reviewing license applications has indicated the need to consider other accident sequences of lesser consequence but higher probability of occurrence."

   NUREG-800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP) Chapter 15.0, "Introduction – Transient and Accident Analysis," Rev. 3, dated March 2007 (ADAMS Accession No. ML070710376) states in part that:

   > The reviewer considers the possible case variations of AOOs [anticipated operational occurrences] and postulated accidents presented to verify that the licensee has identified the limiting cases.

   Please provide an evaluation of the impact of the proposed change on all accidents and AOO's in the design bases or include a justification why an evaluation of the impact is not needed. If an evaluation of other design bases accidents is provided, please provide the regulatory bases for the acceptance criteria (i.e. 10 CFR Part 100, "Reactor Site Criteria," 10 CFR Part 50.67) and any regulatory guidance or SRPs used to make this determination.

3. Attachment 1, page 6, of the LAR states that:

   > "The proposed administrative controls will establish appropriate compensatory measures to ensure that the consequences of a Design Basis Accident which may occur during this time do not exceed the *acceptance criteria* [emphasis added] for that accident."

   How would the licensee determine whether the compensatory measures (consistent with the intent of GDC 19, GDC 41 and 10 CFR 50.67) are appropriate?

   Please specify and justify the methods used to make these determinations (i.e. RG 1.183, SRP Section 18.0, Rev. 2 "Human Factors Engineering" (ADAMS Accession No. ML070670253), etc.).

   Please confirm that, for the purposes of future evaluations conducted in accordance with the provisions of 10 CFR 50.59, "Changes, tests and experiments," the "acceptance criteria" (i.e. 25 rem total effective dose equivalent (TEDE) for the exclusion area

boundary and low population zone and 5 rem TEDE for the control room) will be used as the "calculated dose" (as discussed Section 4.3.3 of the guidance in NEI 96-07, Rev. 1, "Guidelines for 10 CFR 50.59 Implementation," (ADAMS Accession No. ML003771157) which is endorsed by RG 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments." (ADAMS Accession No. ML003759710)).

4. The secondary containment or annulus is provided with the AVS to filter primary containment leakage before release to the environment. The AVS is activated by a safety injection (SI) signal, and draws the annulus to a negative pressure.

   Per the letter dated September 30, 2005, (ADAMS Accession No ML052730312) the NRC staff stated:

   > The licensee calculated and reported doses for each of the above failure scenarios for both the containment leakage and ECCS leakage pathways. Based on its analysis, Duke assumed that the AVS brings the annulus to the required negative pressure and begins filtration within 41.1 seconds for the DBA LOCA with minimum safeguards, and within 30.5 seconds for the other LOCA scenarios with two AVS fans in operation. Until the time the AVS is assumed operational, all of the primary containment leakage of 0.3 volume percent per day is assumed to be released directly to the environment. After that time, 93 percent of the primary containment leakage is assumed to be mixed in 50 percent of the annulus volume, then filtered for release by the AVS. These assumptions on the annulus mixing and percentage filtered by the AVS are consistent with the current licensing basis.

   For all design basis accidents in the Catawba 1 and 2 licensing basis affected by the proposed change, what is the minimum time for establishing annulus integrity so that the "required negative pressure" can be obtained and the licensing basis dose consequences required by 10 CFR 50.67 can be met? Please provide the methods, acceptance criterion, inputs and assumptions used to make this determination. If the inputs, methods, acceptance criterion, assumptions and different from the current licensing basis it would be helpful to provide a table of the current design basis, the proposed design basis and a justification for each change.

5. For proposed compensatory actions that involve human actions, please justify how these actions can be assured to be completed with the potential for harsh environments (radiation, temperature, humidity, failure of high energy pipes) which could impede or prevent human actions?

6. Inserts 2 and 4 of Attachment 2 of the LAR appear to allow the reactor building pressure boundary to be open under administrative control for an undefined amount of time. This appears to override the intent of the TS 3.6.16, "Reactor Building," which provides for the reactor building to be inoperable for only 24 hours before Required Actions to be in Mode 3 are started. Please justify why it is acceptable to allow the reactor building to be opened for an indefinite amount of time.

7. The LAR requests many changes to the Catawba 1 and 2 TSs. For each applicable change, provide a summary of the change and a justification including a regulatory and technical evaluation against the applicable regulatory criteria in 10 CFR 50.67, 10 CFR 100, and GDC 19.

8. Please provide specific information regarding what “administrative controls” (as stated in Insert 2), “method[s] to rapidly close the opening,” (Insert 4) and “compensatory measures” (Insert 5) will be used so that the NRC staff can make a determination of whether there is reasonable assurance that the design basis accidents for Catawba 1 and 2 will remain within the limits of 10 CFR 50.67, 10 CFR 100, and GDC 19, as applicable.

9. This LAR cites consistency with technical specifications task force (TSTF) traveler TSTF-287-A as a significant supporting basis for the acceptability of the requested changes. The NRC staff does not concur that TSTF-287-A is applicable. TSTF-287-A and the justifications it contains were approved for changes to specific sections in Standard Technical Specifications section 3.7 “PLANT SYSTEMS” and not for section 3.6 “CONTAINMENT SYSTEMS.” Although there may be some similarities, the differences were not evaluated. Please provide the justification for the subject requested changes independent of TSTF-287-A.

10. If the TS Bases changes described in the LAR are to reflect a potentially acceptable justification, the following clarifications are needed for an evaluation of the TS changes.

    a. When the doors are opened for normal routine passage of personnel, would the respective Limiting Condition for Operation (LCO) be entered? If not, what specific guidance is provided in plant procedures for how long the doors could be open before the LCO would be entered?

    b. INSERT 4 for the TS mark-up pages in the LAR states “For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room.” If a door were to be held/propped open longer than the time for normal/routine personnel passage, would the same compensatory actions be taken, along with LCO entry, as for the other openings?

    c. INSERT 5 appears to be intended to replace the current TS Bases Action A.1 discussion. However, INSERT 5 discusses only reactor building pressure boundary inoperability, dropping the applicability to inoperability due to conditions other than pressure boundary issues that could impact the effectiveness of the annulus ventilation system. Was this an intended reduction in the scope of the Action A.1 bases statement?

E-mail Properties

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