Duke Energy®

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January 19, 2011 10 CFR 50.90

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: Duke Energy Carolinas, LLC (Duke Energy)
Catawba Nuclear Station, Units 1 and 2
Docket Numbers 50-413 and 50-414
Proposed Technical Specifications (TS) Amendments
TS 3.6.10, "Annulus Ventilation System (AVS)"
TS 3.6.16, "Reactor Building"
License Amendment Request to Enable Opening of the Reactor Building Pressure Boundary with Administrative Controls in Place
(TAC Nos. ME3982 and ME3983)

Reference: Letter from Duke Energy to NRC, same subject, dated May 20, 2010

The reference letter requested amendments to Catawba Facility Operating Licenses NPF-35 and NPF-52 and the subject TS. These amendment requests proposed to revise the subject TS to allow the reactor building pressure boundary to be opened under administrative controls.

On December 2, 2010, a conference call was held between Duke Energy and the NRC to discuss Requests for Additional Information (RAIs). Following the conference call, the NRC formally placed the RAIs into the ADAMS database. Duke Energy indicated in the conference call that responses would be provided to all but one of the RAI questions (Question 5) by January 13, 2011 and that the response to Question 5 would be provided by February 1, 2011. It was subsequently determined that the complete response to Question 8 will be dependent on the Question 5 response. Accordingly, this letter constitutes the first phase of our response (Questions 1, 2, 3, 4, 6, 7, 9, and 10).

The responses to the RAI questions are provided in Attachment 1 to this letter. Revised marked-up TS and Bases pages resulting from these RAI responses are provided in Attachment 2 to this letter. The No Significant Hazards Consideration and the Environmental Consideration provided in the reference letter are not required to be revised as a result of this RAI response. There are no regulatory commitments contained in this letter or its attachments.

A001
NRR

In accordance with 10 CFR 50.91, Duke Energy is notifying the State of South Carolina of this RAI response by transmitting a copy of this letter and its attachments to the designated state official.

Should you have any questions concerning this information, please contact L.J. Rudy at (803) 701-3084.

Very truly yours,

James R. Morris

James R. Morris

LJR/s

Attachments

James R. Morris affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

James R. Morris, Vice President

Subscribed and sworn to me: 1-19-11
Date

Notary Public

My commission expires: 1-31-2018
Date

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ATTACHMENT 1

Response to RAI Questions

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.

[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."

Duke Energy Response:

In submitting this License Amendment Request (LAR) (Ref. 1), it was not the intent of Duke Energy "to exchange previously approved safety-related engineering controls for compensatory measures...." Rather, Duke Energy proposes to use the compensatory measure (to close an open reactor building door within 10 minutes of initiation of a Design Basis Accident (DBA) for the reactor building and AVS as an additional measure of defense-in-depth to those already provided. This may be seen with a comparison of the proposed amendments with the restrictions in the current TS (Ref. 2) and the Standard TS for Westinghouse Plants (Ref. 3).

The TS for the AVS and reactor building are TS 3.6.10 and TS 3.6.16, respectively. The corresponding Westinghouse Standard TS are WSTS 3.6.13 for the Shield Building Air Cleanup System (SBACS), which is the generic equivalent of the Catawba AVS, and WSTS 3.6.8 for the Shield Building, which is the generic equivalent of the Catawba reactor building. Currently, both Catawba TS 3.6.16 and WSTS 3.6.8 require that if inoperable, the reactor building be restored to operable status within 24 hours. Both Catawba TS 3.6.16 and WSTS 3.6.8 include a Surveillance Requirement (SR) that the door in each access opening to the reactor building is closed. From this it is seen that the reactor building is not operable if an access door is not closed (except for normal use – cf. the response to Question 10.a).

A significant difference between Catawba TS 3.6.10/TS 3.6.16 and WSTS 3.6.13/WSTS 3.6.8 relevant to this LAR is that the annulus drawdown SR is located in TS 3.6.10 for the AVS. In the WSTS, the annulus drawdown SR is located in WSTS 3.6.8 for the Shield Building rather than in WSTS 3.6.13 for the SBACS. It also is noted that the drawdown SR in Catawba TS 3.6.10 does not have a criterion for drawdown time as does the SR in WSTS 3.6.8. Removal of the time criterion from the drawdown SR and moving it from TS 3.6.16 to TS 3.6.10 (as SR 3.6.10.6) was approved on September 30, 2005 as part of Catawba License Amendments 227/222 (Ref. 11, cf. Ref. 5 Attachment 3 Part 1 and Ref. 15 Response to Question 3). Here it is noted that placement of the drawdown SR into TS 3.6.10 for the AVS allows entry into the appropriate action statement for failure to meet the drawdown requirement on account of one or two inoperable AVS trains, which correctly prompts entry into either TS 3.6.10 Condition A or TS

Limiting Condition for Operation (LCO) 3.0.3. On the other hand, any problems with reactor building inleakage requiring entry into TS 3.6.16 Condition A would be detected by completing the annulus vacuum SR of TS 3.6.16 (SR 3.6.16.2 – cf. Ref. 15 Response to Question 3). As described in the original submittal (Ref. 1), Catawba License Amendment 227/222 resulted in the unintended and unwanted effect that opening a reactor building door prompting entry into TS 3.6.16 Condition A also prompted entry into TS LCO 3.0.3 due to the inability to meet SR 3.6.10.6.

The only material effect of the proposed amendments to TS 3.6.10 and its Bases (Insert 1 and Insert 3) is to provide entry into TS 3.6.16 Condition A and only TS 3.6.16 Condition A due to the prolonged opening of a reactor building door as follows:

- **Direct entry into TS 3.6.16 Condition A (via non conformance with LCO 3.6.16) and**
- **Inability to meet SR 3.6.10.6 causing...**
- **Entry into TS 3.6.10 Condition B requiring**
- **Entry into "applicable Conditions and Required Actions of LCO 3.6.16 'Reactor Building'" meaning...**
- **Entry into TS 3.6.16 Condition A.**

Prolonged opening of a reactor building door under WSTS 3.6.8 prompts direct entry into Condition A. As seen above, prolonged opening of a reactor building door at Catawba with its TS amended as proposed causes entry into TS 3.6.16 Condition A. In this sense, the proposed amendments in this LAR are equivalent to WSTS 3.6.8. On the other hand, failure to satisfactorily perform an annulus drawdown SR at Catawba because of problems with the AVS prompts entry into either TS 3.6.10 Condition A or TS LCO 3.0.3 as appropriate, as it should. This is a path not provided by WSTS 3.6.8 and WSTS 3.6.13.

In summary, the proposed amendments to TS 3.6.10 and its Bases do not replace or circumvent any existing engineered controls. Rather, with the amendments in place, TS 3.6.10 and TS 3.6.16 will mandate entry into the appropriate Condition and Required Action given an inoperable AVS train(s) or an inoperable reactor building.

Another feature in this LAR is the proposed amendment to the TS 3.6.16 Bases to cite the administrative controls. As amended by Insert 5, the TS 3.6.16 Bases will include the following statement:

"For open reactor building pressure boundary door(s) such that the AVS trains cannot establish or maintain the required negative pressure to prevent unfiltered primary containment leakage from the reactor building, action must be taken to close the door(s) within 24 hours. *During the period that the door(s) are open, appropriate compensatory measures (consistent with the intent, as applicable, of GDC 19, 41, and 10 CFR 50.67) shall be utilized to protect the plant personnel and* [offsite] *boundary from radiological releases....*" (emphasis added)

The administrative controls, denoted in Insert 5 as "appropriate compensatory measures", will be put into place if a reactor building door is to be open for longer than associated with normal use (entry and exit). The compensatory measures mandate the closure of open reactor building door(s) within 10 minutes of a DBA for the reactor building. As proposed in this submittal, the citation to the administrative controls appears only in the Bases for TS 3.6.16 Required Action A.1. The administrative controls are to be put into place during and only during entry into Condition A. Accordingly, the administrative controls do not provide a replacement for or an avenue for circumvention of the restrictions concerning the reactor building. Furthermore, neither WSTS 3.6.13 (SBACS), WSTS 3.6.8 (Shield Building), nor their Bases include any such compensatory measures. As demonstrated above, the amended Catawba TS 3.6.10 (AVS) and TS 3.6.16 (Reactor Building) are equivalent to WSTS 3.6.13 and WSTS 3.6.8 in limiting the opening of reactor building door(s) to 24 hours *without the compensatory measures*. It then becomes evident that the compensatory measures associated with Catawba TS Bases 3.6.16 as amended with Insert 5 provide a measure of defense-in-depth not provided in WSTS 3.6.8. As seen in the response to Question 2, the adequacy of this measure of defense-in-depth has been validated.

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.

Duke Energy Response:

The Catawba Nuclear Station Updated Final Safety Analysis Report (UFSAR) (Ref. 4) reports the analyses of radiological consequences of the following DBAs:

- **Steam system piping failure (denoted as the main steam line break or MSLB - UFSAR 15.1.5.3)**
- **Locked rotor accident (UFSAR 15.3.3.3)**
- **Rod ejection accident (REA - UFSAR 15.4.8.3)**
- **Break in instrument line or other lines from the reactor coolant pressure boundary that penetrate containment (UFSAR 15.6.2).**
- **Steam generator tube rupture (UFSAR 15.6.3.3)**
- **Loss of coolant accident (LOCA - UFSAR 15.6.5.3)**
- **Fuel handling accident (UFSAR 15.7.4)**
- **Weir gate drop (UFSAR 15.7.4)**

Of these DBAs, releases of fission products to containment are assumed and simulated only for the REA and LOCA. Fission products could be released from a MSLB in containment. However, in the MSLB scenarios limiting for post-accident radiation doses, fission products are assumed to be released from a broken steam line outside containment, escaping directly to the environment without the benefit of holdup in containment or the reactor building or cleanup by the AVS. The only fission product release paths assumed for the remaining DBAs bypass the containment and reactor building. For these reasons, the only DBAs for the reactor building and AVS with respect to post-accident radiation doses are the REA and LOCA. The impact of the administrative controls proposed in the LAR of May 20, 2010 on offsite and control room radiation doses following these two DBAs was evaluated in validating its adequacy.

An evaluation was completed to determine that of the REA and LOCA, the REA was the limiting DBA with respect to the impact of the proposed administrative controls on post-accident radiation doses. This conclusion was reached as follows: The proposed administrative controls mandate that an open reactor building door causing entry into TS 3.6.16 Condition A be closed within 10 minutes of accident initiation. Following a LOCA, only some of the fission products are assumed to be released. Specifically, the post-LOCA gap release phase presumably begins 30 seconds after event initiation and is not complete until 30 minutes at which time 5% of the core inventory of noble gases (krypton and xenon), iodine, and alkali metals (rubidium and cesium) are released (Ref. 5 Page A-72, cf. Ref. 6 Tables 2 and 4 and Appendix A). This equates to the release of approximately 1.7% of these fission products at 10 minutes after event initiation. In the analysis of the REA, the gap inventory of all fuel pins predicted

to experience clad failure (from departure from nucleate boiling) is assumed to be released to containment instantly with event initiation (Ref. 7 - cf. Ref. 8). Currently, it is assumed that immediately following a rod ejection, 40% (down from the 50% originally reported in Ref. 7) of the fuel pins experience clad failure with all gap inventory released to containment. The gap inventory assumed for the rod ejection accident consists of 10% of the core inventory of noble gas and iodine, and 12% of the alkali metals (Ref. 7 - cf. Ref. 8). This equates to 4% of the core inventory of noble gases and iodine and 4.8% of the core inventory of alkali metals released to containment instantly with event initiation. The radioactive source term for the REA significantly exceeds that for the LOCA over the first 10 minutes after event initiation. In addition, the guideline values (Ref. 6 Table 6) for post-LOCA radiation doses match the regulatory limits of 10 CFR 50.67 (25 Rem Total Effective Dose Equivalent (TEDE) at offsite locations and 5 Rem TEDE in the control room). However, the guideline value for post-REA offsite doses is only 6.3 Rem TEDE. For these reasons, the REA was determined to be the limiting DBA with respect to the effect of the proposed administrative controls on post-accident radiation doses.

A computer analysis of radiological consequences of a REA was completed to determine the adequacy of the administrative controls mandated in the proposed amendments. The model, method, and assumptions for the baseline AST analysis of the REA taken for this analysis are the same as those reported to the NRC (Ref. 7, cf. Ref. 5 & 11) with one exception: the design basis value of post-REA fuel clad failure now is set to 40% as noted above (cf. Ref. 4 Table 15-26). As noted (Ref. 7, cf. Ref. 5 and 11), the staff has reviewed the model, method, and assumptions for the REA. Nonetheless, a synopsis of those aspects of the model, method, and assumptions relevant to the proposed administrative controls are provided. Refer to the submittal of April 6, 2005 for additional discussion, including justification.

- The REA scenario limiting for post-accident radiation doses at offsite locations and in the control room was taken. This scenario includes a REA at Unit 2 with offsite power available and a Minimum Safeguards failure (Ref. 7).

- The Bechtel proprietary computer code LOCADOSE (Ref. 12-14) was used to validate the adequacy of the proposed administrative controls. This computer code was used to complete the baseline AST analysis of the LOCA and REA (Ref. 5 and 7).

- The following two transport and release paths are considered in the baseline AST analysis of the REA. These paths include the following:

    1) Releases to containment through the breach in the vessel left by the presumably ejected control rod assembly. Fission products may be released either to the containment atmosphere and from there to the environment with post-accident containment leakage or to the containment sump and from there released to the environment with Engineered Safety Features (ESF) leakage.

2) Releases to the reactor coolant and from it to the steam generators (SGs) with leaks assumed to be in the SG tubes. From there, fission products may be released to the environment with SG steam releases.

The method in the Catawba current licensing basis for accounting for these release paths consists of (1) calculating the constituents of each release path to the post-REA radiation doses, (2) identifying the containment transport path (containment leakage or ESF leakage) with the higher constituents to post accident radiation doses, and (3) adding the constituents for that containment transport path to the corresponding constituents for the SG steam release paths to obtain the current licensing basis values for post-REA radiation doses. Refer to the submittal of April 6, 2005 (Ref. 7) and Pages 11 and 12 for further discussion. In the baseline analysis, the containment transport path including fission product releases with post-accident containment leakage was identified as producing higher post-accident radiation doses than the containment transport path including ESF leakage. The proposed administrative controls have associated with them no interactions with post-accident ESF leak paths (or post-accident SG release paths); they have the interaction with the post-accident containment leak path in that they may extend the time span for which fission products released to the annulus presumably bypass it and escape directly to the environment. Therefore, with the compensatory measures, the post-REA containment leakage path will continue to yield higher post-accident radiation doses than the post-accident ESF leakage path (Ref. 5).

- Data pertaining to the radioactive source term assumed to be released to containment is taken from the submittal of April 6, 2005 (Ref. 7) except for the decrease in assumed fraction of post-accident fuel pins with clad failures and shown below:

Table Q2-1: Data Pertaining to the Radioactive Source Term Released to Containment Following a REA

Parameter	Value
Fraction of fuel pins with clad failure	40%
Post REA melted fuel pins?	No
Radial peaking included?	Yes
Number of fuel pins in a fuel assembly	264
Number of fuel assemblies in the core	193
Fuel Assembly Isotopic Radioactivity Level	See Table Q2-2
Design Basis REA gap fractions	
Alkali metals	0.12
All other isotopes	0.10
Equilibrium reactor coolant gross gamma activity	100/E-Bar uCi/gm
Equilibrium DEI reactor coolant specific activity	1 uCi/gm
Reactor coolant mass	
Unit 1	537,793 lbm
Unit 2	481,637 lbm
Concurrent iodine spike	No

Table Q2-2: - Limiting Radioactivity Levels in a LEU Fuel Assembly (Ref. 7 - Accounts for Radial Peaking)

Radio Isotope	Activity (Ci)	Radio Isotope	Activity (Ci)	Radio Isotope	Activity (Ci)
Noble Gases		Halogens		Alkali Metals	
Kr83m	1.27E+05	Br83	1.27E+05	Rb86	1.68E+03
Kr85m	2.85E+05	Br85	2.85E+05	Rb88	8.48E+05
Kr85	7.31E+03	Br87	4.72E+05	Rb89	1.13E+06
Kr87	5.86E+05	I130	2.52E+04	Rb90	1.07E+06
Kr88	8.29E+05	I131	7.52E+05	Cs134	1.91E+05
Kr89	1.07E+06	I132	1.11E+06	Cs136	4.16E+04
Xe131m	9.63E+03	I133	1.60E+06	Cs137	9.15E+04
Xe133m	4.88E+04	I134	1.86E+06	Cs138	1.59E+06
Xe133	1.57E+06	I135	1.52E+06	Cs139	1.51E+06
Xe135m	3.20E+05				
Xe135	4.14E+05				
Xe137	1.48E+06				
Xe138	1.52E+06				

- **The entire REA radioactive source term presumably is released to containment immediately with event initiation. It mixes homogeneously and instantly with the air in the lower compartment (Ref. 7).**

- **Iodine released to the containment presumably takes the form of 97% elemental iodine and 3% organic iodine compounds. Only the alkali metals take the form of particulates (Ref. 7).**

- **Fission products released from the REA into the lower compartment presumably are transported through the ice condenser to the upper compartment beginning with the start of the Containment Air Return System (ARS) fans. No credit is taken for scrubbing of the fission products from the flow through the ice condenser. The fans also blow air from the upper compartment to the lower compartment, establishing a path for recirculation of air through the containment (Ref. 5).**

- **One ARS fan presumably has failed to start and run. This is a consequence of the Minimum Safeguards failure.**

- **Full containment leakage begins immediately with event initiation. The leakage is reduced to half its initial values at 24 hours after event initiation (Ref. 5).**

- **All containment leakage presumably bypasses the annulus until the AVS draws the annulus pressure to –0.25 in. w.g. everywhere in the annulus. Afterwards, the containment leakage is partitioned into leakage into the annulus and leakage bypassing the annulus (Ref. 5).**

- **No credit is taken for washout of fission products from the containment atmosphere with operation of the Containment Spray System (Ref. 7).**

- **Credit is taken for deposition of alkali metals on internal structures in containment. Of the assumptions pertaining to transport of fission products in containment, this one was made for the AST analysis of the REA only (not the LOCA). The deposition time constants and decontamination factors are listed below (Ref. 7).**

Table Q2-3: Time Constants and DFs for Natural Deposition
Following a CNS Design Basis REA

Time Step End Point (Hours)		Time Constant	Decontamination
Beginning	End	hour^{-1}	Factor
0.00	0.50	0.02801	1.0134
0.50	1.80	0.05713	1.0944
1.80	3.80	0.06502	1.3220
3.80	11.80	0.09151	1.3220
11.80	13.80	0.09146	3.9270
13.80	22.22	0.09146	3.9270
22.22	27.78	0.03770	8.2920
27.78	33.33	0.02770	8.2920
33.33	720.00	0.00000	1.0000

- **One AVS fan presumably has failed to start and run. This also is a consequence of the Minimum Safeguards failure (Ref. 5).**

- **All fission products escaping from containment presumably do so from the unit vent stack, whether they bypass the annulus or not (Ref. 5).**

- **Additional data pertaining to transport of fission products in containment following a REA and releases with post-accident containment leakage is presented as follows (Ref. 5, cf. Ref. 7):**

Table Q2-4: Data Pertaining to Post-REA
Fission Product Transport in Containmnent and Annulus

Parameter	Value
Containment volumes	
Lower Compartment	346,895 cu.ft.
Upper Compartment	669,559 cu.ft.
Containment leak rate (L_a)	
0- 24 hours	0.3% volume per day
24-720 hours	0.15% volume per day
Containment bypass leak rate	7% of L_a
ARS fan response time	600 seconds
ARS airflow rate (one fan)	40,000 cfm
Annulus volume	484,090 cu.ft.
AVS response time	23 seconds
AVS airflow rate (one fan)	8,100 cfm
AVS filter data (Ref. 16 - cf. the note below)	
Efficiency for absorption of diatomic iodine	92%
Efficiency for absorption of organic iodine	92%
Efficiency for absorption of particulates	95%
Bypass airflow fraction	1%

Note: In a safety evaluation report (Ref. 11), the staff cites the AVS system efficiencies as 95% for diatomic iodine and particulates and 80% for organic iodine compounds. These values were used in the AST analysis of the LOCA reported in the LAR of

November 25, 2002 (Ref. 5). They were revised to the values reported immediately above in response to a question from the NRC (Ref. 17).

- The dispersion data is set as follows (Ref. 5 and 18):

Table Q2-5: Dispersion Data for the AST Analysis of Radiological Consequences of Post-REA Containment Leakage

Parameter	Value
Exclusion Area Boundary χ/Q (0-2 hours)	4.78×10^{-4} sec/m^3
Low Population Zone boundary χ/Q	
0 - 8 hours	6.85×10^{-5} sec/m^3
8 - 24 hours	4.00×10^{-5} sec/m^3
24 - 96 hours	2.00×10^{-5} sec/m^3
96 – 720 hours	7.35×10^{-6} sec/m^3
Control room χ/Q (releases from the unit vent stack and both control room intakes open)	
0 - 2 hours	1.04×10^{-3} sec/m^3
2 - 8 hours	8.82×10^{-4} sec/m^3
8 - 10 hours	4.14×10^{-4} sec/m^3
10 - 24 hours	3.68×10^{-4} sec/m^3
24 - 96 hours	2.67×10^{-4} sec/m^3
96 – 720 hours	1.87×10^{-4} sec/m^3

- The maximum 2 hour releases, as determined by the maximum 2 hour TEDE at the Exclusion Area Boundary (EAB) falls within the interval 0-2 hours. This is taken into account in the value taken for the control room atmospheric dispersion factor (χ/Q) for 0-2 hours.

- Both control room outside air intakes are open. This is taken into account in the values taken for the control room χ/Q values for all intervals (Ref. 7, cf. Ref. 5).

- Data pertaining to the control room and Control Room Area Ventilation System (CRAVS) is presented below (Ref. 5):

Table Q2-6: Data Pertaining to the Control Room and CRAVS

Parameter	Value
Control room volume	117,920 cu.ft.
Control room intake airflow split	60%/40%
CRAVS airflow rates to the control room	
Total	3,500 cfm
Recirculation	1,500 cfm
Rate of control room unfiltered inleakage	100 cfm
CRAVS filter data (Ref. 16 - cf. the note below)	
Efficiency for diatomic iodine	98.1%
Efficiency for organic iodine compounds	98.1%
Efficiency for particulates	99%
Bypass airflow fraction	0.05%
Control room occupancy factors	
0 - 24 hours	1.0
24 - 96 hours	0.6
96 – 720 houors	0.4

Note: In a safety evaluation report (Ref. 11), the staff cites the CRAVS system efficiencies as 99% for diatomic iodine and particulates and 95% for organic iodine compounds. These values were used in the AST analysis of the LOCA reported in the LAR of November 25, 2002 (Ref. 5). They were revised to the values reported immediately above in response to a question from the NRC (Ref. 17).

- Radioactivity levels at the offsite locations and in the control room were converted to radiation doses using the following data (Ref. 5):

Table Q2-7: Data for Conversion from Activity to Dose

Parameter	Value
Method	R.G. 1.183
References for dose coefficients	
External dose	FGR-12
Inhaled dose	FGR-11
Offsite breathing rates	
0 - 8 hours	3.5×10^{-4} sec/m^3
8 - 24 hours	1.8×10^{-4} sec/m^3
24 – 720 hours	2.3×10^{-4} sec/m^3
Control room breathing rate	3.5×10^{-4} sec/m^3

The compensatory measure in the administrative controls is to close a reactor building door held open within 10 minutes of accident initiation. To simulate this, the time at which the AVS lowers the pressure in the annulus to at least –0.25 in. w.g. everywhere in it was increased by 10 minutes. The design basis analysis of post-accident annulus conditions and AVS response predicts annulus pressure decreasing to –0.25 in. w.g. everywhere in it at 41.4 sec after initiation of an event with a Minimum Safeguards failure (REA - Ref. 7). To simulate the effect of having a reactor building door open under the administrative controls, this value was reset to 10 minutes and 41 seconds after event initiation. The time dependent AVS recirculation and exhaust airflow rates (Ref. 5 Attachment 3 Appendix A Page A-52) were reset by adding 10 minutes to the beginning and end of the time step over which each airflow rate originally was assigned. The resulting values of AVS exhaust and recirculation airflow rates and revised time interval start and end are presented (compared to the baseline values) in the response to Question 4.

For this scenario, the following post-REA radiation doses were calculated:

- TEDE at the Exclusion Area Boundary (EAB)
- TEDE at the boundary of the Low Population Zone (LPZ)

- TEDE in the control room

The post-REA radiation doses associated with the proposed administrative controls were compared the design basis values (Ref. 4 Table 15-26). The comparison is shown below:

Table Q2-8: Radiation Doses Following a Rod Ejection Accident at Catawba Nuclear Station

Type of Post-REA Radiation Dose	Baseline	Post-REA Radiation Dose (Rem) With Administrative Controls	Increase
EAB TEDE			
Cont leakage	2.30	3.71	1.40
SG release	2.44	2.44	0.00
Total	4.75	6.15	1.40
LPZ TEDE			
Cont leakage	2.43	2.63	0.20
SG release	0.39	0.39	0.00
Total	2.82	3.03	0.20
Control Room TEDE			
Cont leakage	1.39	1.49	0.10
SG release	1.31	1.31	0.00
Total	2.70	2.80	0.10

The guideline limits for the post-REA radiation doses at offsite locations and in the control room are, respectively, 6.3 Rem and 5.0 Rem (Ref. 6). It is seen that all post-REA radiation doses remain within the NRC guideline values (Ref. 6) with a reactor building door initially open then closed under the proposed administrative controls.

The other DBA for the reactor building and AVS is the LOCA. As noted above, the proposed compensatory action is seen to have a smaller impact on post-LOCA radiation doses than on radiation doses following a REA. Upper bounds to radiation doses following a LOCA with the proposed compensatory action taken are estimated by adding the increases in the post-REA radiation doses to the values for the baseline post-LOCA radiation doses. The values taken for the baseline post-LOCA radiation doses correspond to the implementation of the "Water Management" initiatives (Ref. 9 and 10). Adding the post-REA increases provides the following upper bounds for radiation doses following a LOCA with a reactor building door initially open then closed under the proposed administrative controls:

Table Q2-9: Upper Bounds to Radiation Doses Following a LOCA at Catawba Nuclear Station

Type of Post-LOCA Radiation Dose	Post-LOCA Radiation Dose (Rem) Baseline	With Administrative Controls
EAB TEDE	8.79	10.19
LPZ TEDE	3.78	3.99
Control Room TEDE	3.31	3.41

The guideline limits (Ref. 6) for the LOCA are the same as the regulatory dose limits in 10 CFR 50.67: 25 Rem TEDE at offsite locations and 5 Rem in the control room. It is seen that the radiation doses following a LOCA with a reactor building door initially open then closed under the proposed administrative controls will remain under the regulatory limits of 10 CFR 50.67.

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)).

Duke Energy Response:

The response is given in two parts as follows:

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.).

The compensatory measures will be put into place whenever a reactor building door is open for any reason other than normal use (ingress/egress), causing the reactor building to be inoperable and TS 3.6.16 Condition A to be entered. The compensatory measures would be in place only for the time allowed for an inoperable reactor building per TS 3.6.16 Condition A. The compensatory measures would not be in effect at any other time. That is, the reactor building will not be considered operable during the time the compensatory measures are in place. In summary, the compensatory measures are appropriate for and only for entry into TS 3.6.16 Condition A for reasons of one or more access doors not closed for reasons other than normal use.

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)).

As noted above, the compensatory measures will be put into place during and only during entry into TS 3.6.16 Condition A. They will have been lifted and the associated reactor building access door(s) closed within the 24 hours allowed per TS 3.6.16 Condition A. For this reason, the proposed administrative controls and associated compensatory measures are considered not to be part of a "proposed design basis" (referring to the last sentence of Question 4). Accordingly, the post-accident radiation doses calculated for REA and LOCA scenarios with a reactor building door open under the administrative controls are considered not to provide the baseline for evaluating "changes, tests, and experiments" pursuant to 10 CFR 50.59 and with use of the guidance of NEI 96-07. The UFSAR will not be updated to report them.

Any proposed revision to the compensatory measure will be evaluated to ensure that with the revised compensatory measure in place the radiation doses for the reactor building and AVS DBAs (LOCA and REA) remain within their guideline values (Ref. 6). Also, any modification, procedure revision, test, or experiment with the potential to affect the radiation doses following a postulated LOCA or REA will be evaluated to validate the adequacy of the compensatory measure.

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.

Duke Energy Response:

The compensatory measure is written to mandate that the integrity of the annulus be restored within 10 minutes of indication of initiation of any DBA for it and the AVS. The simulation of the compensatory action was discussed in the response to Question 2. Additional details of this simulation are provided here.

The analyses of radiological consequences of the LOCA and REA make use of a calculation of post-accident conditions in the annulus and response of the AVS. This baseline calculation has been presented to the NRC (Ref. 5 Attachment 3 Appendix A § 1.4 and Enclosure 4 – cf. Ref. 7). Its features germane to the response to this question are discussed here. As noted in the response to Question 2, the DBA scenario limiting with respect to the proposed administrative controls is a REA with offsite power available and with a Minimum Safeguards failure. For this scenario, one AVS fan starts and comes to full speed within 23 seconds after event initiation. It first draws the annulus pressure to –0.25 in. w.g. everywhere in the annulus at 41.4 seconds after event initiation. At that time, most (93%) of the containment leakage is retained in it with the AVS removing most of the fission products entering the annulus with containment leakage. At 54 seconds after initiation of this scenario, the AVS draws the annulus pressure to a setpoint it maintains by recirculating some of the airflow in it. These and other salient characteristics of post-accident AVS operation were calculated with the computer code CANVENT (Ref. 5 Attachment 3 Appendix A § 1.1) and with assumptions already presented to the staff (Ref. 5 Attachment 3 Appendix A § 1.4). The output includes AVS exhaust and recirculation airflow rates over a number of time intervals (Ref. 5 Attachment 3 Appendix A Enclosure 4).

The method to simulate the effects of a reactor building door open then closed under the proposed administrative controls was as follows: All times in the CANVENT output from the baseline calculation were increased by 10 minutes or 600 seconds (or 0.166666667 hours). As noted above, this is the time assumed to close a reactor building door open under the compensatory action and restore reactor building integrity. A comparison of the assumptions pertaining to post-accident annulus conditions and AVS operation is presented in the two tables below:

Table Q4-1: Sequence of Events for Post-Accident Annulus Conditions and AVS Operation

	Time (seconds after event initiation)	
Event	Baseline	With Administrative Controls
Event initiation	0	0
AVS reaches full operation (one fan)	23.0	23.0
One AVS fan draws the annulus pressure to -0.25 in. w.g. everywhere in it	41.4	641.4
One AVS fan draws the annulus pressure to its modulation setpoint	54.0	654.0

Table Q4-2: Post-Accident AVS Airflow Rates (Minimum Safeguards)

Time (seconds after event initiation) Baseline		With Administrative Controls in Place		AVS Airflow Rate (cfm)	
Start	End	Start	End	Exhaust	Recirc
0	23	0	23	0.0	0.0
23	41.4	23	41.4	8100.0	0.0
41.4	54	641.4	654	8100.0	0.0
54	60	654	660	5613.0	2487.0
60	75	660	675	6150.0	1950.0
75	90	675	690	6604.8	1495.2
90	105	690	705	7099.3	1000.7
105	120	705	720	7416.4	683.6
120	135	720	735	7675.9	424.1
135	150	735	750	7884.1	215.9
150	172	750	772	8100.0	0.0
172	347	772	947	8100.0	0.0
347	400	947	1000	7747.1	352.9
400	500	1000	1100	7382.8	717.2
500	600	1100	1200	6414.3	1685.7
600	700	1200	1300	5447.1	2652.9
700	800	1300	1400	4602.9	3497.1
800	900	1400	1500	3929.0	4171.0
900	1000	1500	1600	3430.5	4669.5
1000	3000	1600	3600	3859.8	4240.2
3000	7200	3600	7800	3361.9	4738.1
7200	9000	7800	9600	3361.9	4738.1
9000	28800	9600	29400	3190.9	4909.1
28800	2592000	29400	2592000	3136.2	4963.8

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?

Duke Energy Response:

Duke Energy plans to provide a response to this question by February 1, 2011.

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.

Duke Energy Response:

As indicated in the December 2, 2010 conference call with the NRC, Catawba never intended that the proposed Insert 2 Note allowing the reactor building pressure boundary to be open under administrative control would circumvent the requirement to declare the reactor building inoperable and enter the 24 hour Condition. However, Catawba concurs that the proposed Insert 2 Note (and/or its proposed Insert 4 Bases discussion) could be subject to future misinterpretation by persons unfamiliar with the details of this amendment request. Therefore, Catawba is withdrawing proposed Inserts 2 and 4. In addition, Catawba is revising proposed Inserts 1 and 3 to clarify that they only apply to situations where reactor building pressure boundary door(s) are open. Attachment 2 contains the revised marked-up TS and Bases pages reflecting these changes. These pages should replace those pages contained in Attachments 2 and 3 of the original LAR in their entirety.

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.

Duke Energy Response:

Following the withdrawal of proposed Inserts 2 and 4 as indicated in the response to Question 6, there are only two substantive changes remaining in this amendment application. They are as follows: 1) the change to allow the 24 hour Completion Time of TS 3.6.16 Required Action A.1 to be utilized when both AVS trains are inoperable SOLELY due to open reactor building pressure boundary door(s), and 2) the change describing the proposed compensatory actions to be utilized while this situation is in effect. The remainder of the proposed changes are editorial in nature (i.e., they reflect the re-lettering of Conditions and Required Actions). For the two remaining substantive changes, the responses to the other questions provide the appropriate technical and regulatory evaluation.

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.

Duke Energy Response:

As indicated in the response to Question 6, proposed Inserts 2 and 4 are being withdrawn. For Insert 5, the details regarding the proposed compensatory measures cannot be fully formulated until the response to Question 5 has been developed. However, in general terms, the compensatory measures will involve stationing a dedicated individual at the door who can close the door within 10 minutes of the occurrence of the relevant DBA. Duke Energy plans to provide a response to this question by February 1, 2011.

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.

Duke Energy Response:

As indicated in the response to Question 6, proposed Inserts 2 and 4 are being withdrawn. These Inserts, along with Insert 5, were the ones that were based on TSTF-287-A. The remaining proposed changes, which are described in the response to Question 7, are justified (or will be justified) by the analytical basis supporting the responses to Questions 1 through 5 and 8.

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?

Duke Energy Response:

a. **TS 3.6.16, SR 3.6.16.1 states, "Verify the door in each access opening is closed, except when the access opening is being used for normal transit entry and exit." This SR has a 31 day Frequency. Therefore, when a door is opened for normal routine passage of personnel, LCO 3.6.16 is still considered to be met and Condition A for an inoperable reactor building is not entered. At present, Catawba has no procedural guidance for opening these doors for reasons other than normal routine passage of personnel in Modes 1 through 4, since with an open door, the affected unit would be in LCO 3.0.3 due to the inability to meet SR 3.6.10.6. SR 3.6.10.6 states, "Verify each AVS train produces a pressure equal to or more negative than -0.88 inch water gauge when corrected to elevation 564 feet." This SR has an 18 month Frequency. Should the NRC approve this amendment request to allow the 24 hour Completion Time of TS 3.6.16 to be utilized for open door(s), then Catawba's practice would be to enter TS 3.6.16, Condition A whenever a door is propped open for a time period longer than it takes to generate a security alarm. This time period is presently 60 seconds. When a door is open for less than 60 seconds, it is Catawba's position that this constitutes normal transit entry and exit and that the allowance of SR 3.6.16.1 applies. It is also Catawba's position that a door can be propped open to allow for the passage of tools, material, or equipment as long as the 60 second security alarm limit is not exceeded.**

b. **As indicated in the response to Question 6, proposed Inserts 2 and 4 are being withdrawn. Therefore, this question is no longer applicable.**

c. **Insert 5 has been re-worded and re-located (i.e., it is being placed after the existing Required Action A.1 Bases discussion) in order to make it clear that taken in its entirety, the revised Bases discussion will apply to all causes of reactor building inoperability, not merely to those causes involving the reactor building pressure boundary.**

References:

1) James R. Morris (Duke Energy) to U.S. Nuclear Regulatory Commission, "Duke Energy Carolinas, LLC (Duke Energy) Catawba Nuclear Station, Units 1 and 2 Docket Numbers 50-413 and 50-414 Proposed Technical Specification (TS) Amendments 3.6.10, 'Annulus Ventilation System (AVS)' TS 3.6.16 'Reactor Building' License Amendment Request to Enable Opening of the Reactor Building Pressure Boundary with Administrative Controls in Place," May 20, 2010.

2) Catawba Nuclear Station Technical Specifications, with Amendments Through 261/257.

3) USNRC, Standard Technical Specifications Westinghouse Plants, NUREG-1431 Volume 1 Rev 3.0, June 2004.

4) Catawba Nuclear Station Updated Final Safety Analysis Report, Rev 14 (effective April 18 2009).

5) G.R. Peterson (Duke Energy) to U.S. Nuclear Regulatory Commission, "Duke Energy Corporation Catawba Nuclear Station, Units 1 and 2 Docket Numbers 50-413 and 50-414 Proposed Technical Specifications and Bases Amendment Technical Specification and Bases 3.6.10 Annulus Ventilation System (AVS) Technical Specification and Basis 3.6.16 Reactor Building Technical Specification Bases 3.7.10 Control Room Area Ventilation System (CRAVS) Technical Specification Bases 3.7.12 Auxiliary Building Filtered Ventilation Exhaust System (ABFVES) Technical Specification Bases 3.7.13 Fuel Handling Ventilation Exhaust System (FHVES) Technical Specification and Bases 3.9.3 Containment Penetrations Technical Specification 5.5.11 Ventilation Filter Testing Program (VFTP)," November 25, 2002.

6) USNRC, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, Regulatory Guide 1.183 (Rev 0), July 2000.

7) D.M. Jamil (Duke Energy) to U.S. Nuclear Regulatory Commission, "Duke Energy Corporation Catawba Nuclear Station, Units 1 and 2 Docket Numbers 50-413 and 50-414 Proposed Technical Specifications and Bases Amendment Technical Specification and Bases 3.6.10 Annulus Ventilation System (AVS) Technical Specification and Basis 3.6.16 Reactor Building Technical Specification Bases 3.7.10 Control Room Area Ventilation System (CRAVS) Technical Specification Bases 3.7.12 Auxiliary Building Filtered Ventilation Exhaust System (ABFVES) Technical Specification Bases 3.7.13 Fuel Handling Ventilation Exhaust System (FHVES) Technical Specification and Bases 3.9.3 Containment Penetrations Technical Specification 5.5.11 Ventilation Filter Testing Program (VFTP) TAC Numbers MB7014 and MB7015," April 6, 2005.

8) Regulatory Guide 1.183 Rev 0 (Ref. 6) and Appendix H.

9) Jon Thompson (USNRC) to J.R. Morris, "Catawba Nuclear Station, Units 1 and 2, Issuance of Amendments Regarding Technical Specification Changes to Allow Manual Operation of the CSS (TAC Nos. MD9752 and MD9753)," June 28, 2010.

10) James R. Morris to U.S. Nuclear Regulatory Commission, "Duke Energy Carolinas, LLC (Duke) Catawba Nuclear Station, Units 1 and 2 Docket Numbers 50-413 and 50-414 Technical Specifications (TS) and/or Bases Sections 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation 3.3.3, Post Accident Monitoring (PAM) Instrumentation, 3.5.4, Refueling Water Storage Tank (RWST) 3.6.6 Containment Spray System License Amendment Request for Emergency Core Cooling System (ECCS) Water Management Initiative," September 2, 2008.

11) Sean E. Peters (USNRC) to D.M. Jamil (Duke Energy), Catawba Nuclear Station, Units 1 and 2 Re: Issuance of Amendments (TAC Nos. MB7014 and MB7015)," September 30, 2005.

12) Bechtel Corporation, <u>LOCADOSE NE-319 User's Manual</u> (Rev 8), May 1999.

13) Bechtel Corporation, <u>LOCADOSE NE-319 Theoretical Manual</u> (Rev 8), May 1999.

14) Bechtel Corporation, <u>LOCADOSE NE-319 Validation Manual</u> (Rev 9), May 1999.

15) D.M. Jamil (Duke Energy) to U.S. Nuclear Regulatory Commission, "Duke Energy Corporation Catawba Nuclear Station, Units 1 and 2 Docket Numbers 50-413 and 50-414 Proposed Technical Specifications and Bases Amendment Technical Specification and Bases 3.6.10 Annulus Ventilation System (AVS) Technical Specification and Basis 3.6.16 Reactor Building Technical Specification Bases 3.7.10 Control Room Area Ventilation System (CRAVS) Technical Specification Bases 3.7.12 Auxiliary Building Filtered Ventilation Exhaust System (ABFVES) Technical Specification Bases 3.7.13 Fuel Handling Ventilation Exhaust System (FHVES) Technical Specification and Bases 3.9.3 Containment Penetrations Technical Specification 5.5.11 Ventilation Filter Testing Program (VFTP) TAC Numbers MB7014 and MB7015," November 13, 2003.

16) D.M. Jamil to U.S. Nuclear Regulatory Commission, "Duke Energy Corporation Catawba Nuclear Station, Units 1 and 2 Docket Numbers 50-413 and 50-414 Proposed Technical Specifications and Bases Amendment Technical Specification and Bases 3.6.10 Annulus Ventilation System (AVS) Technical Specification and Basis 3.6.16 Reactor Building Technical

Specification Bases 3.7.10 Control Room Area Ventilation System (CRAVS) Technical Specification Bases 3.7.12 Auxiliary Building Filtered Ventilation Exhaust System (ABFVES) Technical Specification Bases 3.7.13 Fuel Handling Ventilation Exhaust System (FHVES) Technical Specification and Bases 3.9.3 Containment Penetrations Technical Specification 5.5.11 Ventilation Filter Testing Program (VFTP) TAC Numbers MB7014 and MB7015," August 17, 2005.

17) James R. Morris to U.S. Nuclear Regulatory Commission, "Duke Energy Corporation Catawba Nuclear Station, Units 1 and 2 Docket Numbers 50-413 and 50-414 Proposed Technical Specifications and Bases Amendment Technical Specification and Bases 3.6.10 Annulus Ventilation System (AVS) Technical Specification and Basis 3.6.16 Reactor Building Technical Specification Bases 3.7.10 Control Room Area Ventilation System (CRAVS) Technical Specification Bases 3.7.12 Auxiliary Building Filtered Ventilation Exhaust System (ABFVES) Technical Specification Bases 3.7.13 Fuel Handling Ventilation Exhaust System (FHVES) Technical Specification and Bases 3.9.3 Containment Penetrations Technical Specification 5.5.11 Ventilation Filter Testing Program (VFTP) TAC Numbers MB7014 and MB7015," July 8, 2005.

18) D.M. Jamil to U.S. Nuclear Regulatory Commission, "Duke Energy Corporation Catawba Nuclear Station, Units 1 and 2 Docket Numbers 50-413 and 50-414 Proposed Technical Specifications and Bases Amendment Technical Specification and Bases 3.6.10 Annulus Ventilation System (AVS) Technical Specification and Basis 3.6.16 Reactor Building Technical Specification Bases 3.7.10 Control Room Area Ventilation System (CRAVS) Technical Specification Bases 3.7.12 Auxiliary Building Filtered Ventilation Exhaust System (ABFVES) Technical Specification Bases 3.7.13 Fuel Handling Ventilation Exhaust System (FHVES) Technical Specification and Bases 3.9.3 Containment Penetrations Technical Specification 5.5.11 Ventilation Filter Testing Program (VFTP) TAC Numbers MB7014 and MB7015," September 8, 2005.

ATTACHMENT 2

Revised Marked-Up TS and Bases Pages

Revised TS and TS Bases Inserts

INSERT 1

B. Two AVS trains inoperable due to open reactor building pressure boundary door(s).	B.1 Enter applicable Conditions and Required Actions of LCO 3.6.16, "Reactor Building."	Immediately

INSERT 2

This proposed insert has been withdrawn.

INSERT 3

B.1

With two AVS trains inoperable due to open reactor building pressure boundary door(s), the applicable Conditions and Required Actions of LCO 3.6.16, "Reactor Building," must be immediately entered. In this situation, an LCO 3.0.3 entry is not required, since the AVS trains would otherwise be OPERABLE if the door(s) were in their normal closed position.

INSERT 4

This proposed insert has been withdrawn.

INSERT 5 (to be added following existing Required Action A.1 Bases discussion)

For open reactor building pressure boundary door(s) such that the AVS trains cannot establish or maintain the required negative pressure to prevent unfiltered primary containment leakage from the reactor building, action must be taken to close the door(s) within 24 hours. During the period that the door(s) are open, appropriate compensatory measures (consistent with the intent, as applicable, of GDC 19, 41, and 10 CFR 50.67) shall be utilized to protect the plant personnel and boundary from radiological releases. Preplanned measures shall be available to address these concerns for intentional and unintentional entry into the Condition. The 24 hour Completion Time is reasonable considering the limited leakage design of the containment, low probability of a Design Basis Accident during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair most problems with the reactor building pressure boundary door(s).

3.6 CONTAINMENT SYSTEMS

3.6.10 Annulus Ventilation System (AVS)

LCO 3.6.10 Two AVS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One AVS train inoperable.	A.1 Restore AVS train to OPERABLE status.	7 days
INSERT 1		
~~B~~ C. One or more Annulus Ventilation System (AVS) train(s) heater inoperable.	~~B~~ C.1 Restore AVS train(s) heater to OPERABLE status. OR ~~B~~ C.2 Initiate action in accordance with Specification 5.6.6.	7 days 7 days
~~C~~ D. Required Action and associated Completion Time not met.	~~C~~ D.1 Be in MODE 3. AND ~~C~~ D.2 Be in MODE 5.	6 hours 36 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.10.1	Operate each AVS train for ≥ 10 continuous hours with heaters operating.	31 days
SR 3.6.10.2	Perform required AVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.10.3	Verify each AVS train actuates on an actual or simulated actuation signal.	18 months
SR 3.6.10.4	Verify each AVS filter cooling bypass valve can be opened.	18 months
SR 3.6.10.5	Verify each AVS train flow rate is ≥ 8100 cfm and ≤ 9900 cfm.	18 months
SR 3.6.10.6	Verify each AVS train produces a pressure equal to or more negative than -0.88 inch water gauge when corrected to elevation 564 feet.	18 months

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3.6 CONTAINMENT SYSTEMS

3.6.16 Reactor Building

LCO 3.6.16 The reactor building shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor building inoperable.	A.1 Restore reactor building to OPERABLE status.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.16.1	Verify the door in each access opening is closed, except when the access opening is being used for normal transit entry and exit.	31 days

(continued)

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.16.2	Verify that during the annulus vacuum decay test, the vacuum decay time is $\geq$ 87 seconds.	18 months
SR 3.6.16.3	Verify reactor building structural integrity by performing a visual inspection of the exposed interior and exterior surfaces of the reactor building.	3 times every 10 years, coinciding with containment visual examinations required by SR 3.6.1.1

B 3.6 CONTAINMENT SYSTEMS

B 3.6.10 Annulus Ventilation System (AVS)

BASES

BACKGROUND

The AVS is required by 10 CFR 50, Appendix A, GDC 41, "Containment Atmosphere Cleanup" (Ref. 1), to ensure that radioactive materials that leak from the primary containment into the reactor building (secondary containment) following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment.

The containment has a secondary containment called the reactor building, which is a concrete structure that surrounds the steel primary containment vessel. Between the containment vessel and the reactor building inner wall is an annulus that collects any containment leakage that may occur following a loss of coolant accident (LOCA) or rod ejection accident. This space also allows for periodic inspection of the outer surface of the steel containment vessel.

under post-accident conditions

The AVS establishes a negative pressure in the annulus between the reactor building and the steel containment vessel. Filters in the system then control the release of radioactive contaminants to the environment.

The AVS consists of two separate and redundant trains. Each train includes a heater, prefilter/moisture separators, upstream and downstream high efficiency particulate air (HEPA) filters, an activated carbon adsorber section for removal of radioiodines, and a fan. Ductwork, valves and/or dampers, and instrumentation also form part of the system. The prefilters/moisture separators function to remove large particles and entrained water droplets from the airstream, which reduces the moisture content. A HEPA filter bank upstream of the carbon adsorber filter bank functions to remove particulates and a second bank of HEPA filters follow the adsorber section to collect carbon fines. Only the upstream HEPA filter and the carbon adsorber section are credited in the analysis.

The reactor building is required to be OPERABLE to ensure retention of containment leakage and proper operation of the AVS.

BACKGROUND (continued)

A heater is included within each filter train to reduce the relative humidity of the airstream, although no credit is taken in the safety analysis. The heaters are not required for OPERABILITY since the carbon laboratory tests are performed at 95% relative humidity, but have been maintained in the system to provide additional margin (Ref. 6). Continuous operation of each train, for at least 10 hours per month, with heaters on, reduces moisture buildup on their HEPA filters and adsorbers.

The system initiates and maintains a negative air pressure in the reactor building annulus by means of filtered exhaust ventilation of the reactor building annulus following receipt of a safety injection (SI) signal. The system is described in Reference 2. The AVS reduces the radioactive content in the annulus atmosphere following a DBA. Loss of the AVS could cause site boundary doses, in the event of a DBA, to exceed the values given in the licensing basis.

APPLICABLE SAFETY ANALYSES

The AVS design basis is established by the consequences of the limiting DBA, which is a LOCA. The accident analysis (Ref. 3) assumes that only one train of the AVS is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the remaining one train of this filtration system. The amount of fission products available for release from containment is determined for a LOCA.

The modeled AVS actuation in the safety analyses is based upon a worst case response time following an SI initiated at the limiting setpoint. The CANVENT computer code is used to determine the total time required to achieve a negative pressure in the annulus under accident conditions. The response time considers signal delay, diesel generator startup and sequencing time, system startup time, and the time for the system to attain the required pressure.

The AVS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 4).

LCO

In the event of a DBA, one AVS train is required to provide the minimum iodine removal assumed in the safety analysis. Two trains of the AVS must be OPERABLE to ensure that at least one train will operate, assuming that the other train is disabled by a single active failure.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could lead to fission product release to containment that leaks to the reactor building. The large break LOCA, on which this system's design is based, is a full power event. Less severe LOCAs and leakage still require the system to be OPERABLE throughout these MODES. The probability and severity of a LOCA decrease as core power and Reactor Coolant System pressure decrease. With the reactor shut down, the probability of release of radioactivity resulting from such an accident is low.

In MODES 5 and 6, the probability and consequences of a DBA are low due to the pressure and temperature limitations in these MODES. Under these conditions, the AVS is not required to be OPERABLE.

ACTIONS

A.1

With one AVS train inoperable, the inoperable train must be restored to OPERABLE status within 7 days. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant AVS train and the low probability of a DBA occurring during this period. The Completion Time is adequate to make most repairs.

INSERT 3

~~B~~.1 and ~~B~~.2 C C

With one or more AVS heaters inoperable, the heater must be restored to OPERABLE status within 7 days. Alternatively, a report must be initiated within 7 days per Specification 5.6.6, which details the reason for the heater's inoperability and the corrective action required to return the heater to OPERABLE status.

The heaters do not affect OPERABILITY of the AVS filter trains because carbon adsorber efficiency testing is performed at 30°C and 95% relative humidity. The accident analysis shows that site boundary radiation doses are within 10 CFR 50.67 limits during a DBA LOCA under these conditions.

BASES

ACTIONS (continued)

~~C~~D.1 and ~~C~~D.2

If the AVS train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.10.1

Operating each AVS train from the control room with flow through the HEPA filters and carbon adsorbers ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation with the heaters on for ≥ 10 continuous hours eliminates moisture on the adsorbers and HEPA filters. Experience from filter testing at operating units indicates that the 10 hour period is adequate for moisture elimination on the adsorbers and HEPA filters. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls, the two train redundancy available, and the iodine removal capability of the Containment Spray System and Ice Condenser.

SR 3.6.10.2

This SR verifies that the required AVS filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The AVS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 5). The VFTP includes testing HEPA filter performance, carbon adsorber efficiency, system flow rate, and the physical properties of the activated carbon (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.10.3

The automatic startup on a safety injection signal ensures that each AVS train responds properly. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore the Frequency was concluded to be acceptable from a reliability standpoint. Furthermore, the SR interval was developed considering that the AVS equipment OPERABILITY is demonstrated at a 31 day Frequency by SR 3.6.10.1.

SR 3.6.10.4

The AVS filter cooling electric motor-operated bypass valves are tested to verify OPERABILITY. The valves are normally closed and may need to be opened from the control room to initiate miniflow cooling through a filter unit that has been shutdown following a DBA LOCA. Miniflow cooling may be necessary to limit temperature increases in the idle filter train due to decay heat from captured fission products. The 18 month Frequency is considered to be acceptable based on valve reliability and design, and the fact that operating experience has shown that the valves usually pass the Surveillance when performed at the 18 month Frequency.

SR 3.6.10.5

The proper functioning of the fans, dampers, filters, adsorbers, etc., as a system is verified by the ability of each train to produce the required system flow rate. The 18 month Frequency is consistent with Regulatory Guide 1.52 (Ref. 5) guidance for functional testing.

SR 3.6.10.6

The ability of the AVS train to produce the required negative pressure of at least -0.88 inch water gauge when corrected to elevation 564 feet ensures that the annulus negative pressure is at least -0.25 inch water gauge everywhere in the annulus. The -0.88 inch water gauge annulus pressure includes a correction for an outside air temperature induced hydrostatic pressure gradient of -0.63 inch water gauge. The negative

SURVEILLANCE REQUIREMENTS (continued)

pressure prevents unfiltered leakage from the reactor building, since outside air will be drawn into the annulus by the negative pressure differential.

The CANVENT computer code is used to model the thermal effects of a LOCA on the annulus and the ability of the AVS to develop and maintain a negative pressure in the annulus after a design basis accident. The annulus pressure drawdown time during normal plant conditions is not an input to any dose analyses. Therefore, the annulus pressure drawdown time during normal plant conditions is insignificant.

The AVS trains are tested every 18 months to ensure each train will function as required. Operating experience has shown that each train usually passes the surveillance when performed at the 18 month Frequency. Furthermore, the SR interval was developed considering that the AVS equipment OPERABILITY is demonstrated at a 31 day Frequency by SR 3.6.10.1. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 41.
2. UFSAR, Sections 6.2.3 and 9.4.9.
3. UFSAR, Chapter 15.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
5. Regulatory Guide 1.52, Revision 2.
6. Catawba Nuclear Station License Amendments 90/84 for Units 1/2, August 23, 1991.
7. NUREG-0800, Sections 6.2.3 and 6.5.3, Rev. 2, July 1981.

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.16 Reactor Building

BASES

BACKGROUND

The reactor building is a concrete structure that surrounds the steel containment vessel. Between the containment vessel and the reactor building inner wall is an annular space that collects containment leakage that may occur following a loss of coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the steel containment vessel.

The Annulus Ventilation System (AVS) establishes a negative pressure in the annulus between the reactor building and the steel containment vessel under post-accident conditions. Filters in the system then control the release of radioactive contaminants to the environment. The reactor building is required to be OPERABLE to ensure retention of containment leakage and proper operation of the AVS. To ensure the retention of containment leakage within the reactor building:

a. The door in each access opening is closed except when the access opening is being used for normal transit entry and exit, and

b. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

APPLICABLE SAFETY ANALYSES

The design basis for reactor building OPERABILITY is a LOCA. Maintaining reactor building OPERABILITY ensures that the release of radioactive material from the containment atmosphere is restricted to those leakage paths and associated leakage rates assumed in the accident analyses.

The reactor building satisfies Criterion 3 of 10 CFR 50.36 (Ref. 1).

LCO

Reactor building OPERABILITY must be maintained to ensure proper operation of the AVS and to limit radioactive leakage from the containment to those paths and leakage rates assumed in the accident analyses.

BASES

APPLICABILITY

Maintaining reactor building OPERABILITY prevents leakage of radioactive material from the reactor building. Radioactive material may enter the reactor building from the containment following a LOCA. Therefore, reactor building OPERABILITY is required in MODES 1, 2, 3, and 4 when a LOCA or rod ejection accident could release radioactive material to the containment atmosphere.

In MODES 5 and 6, the probability and consequences of these events are low due to the Reactor Coolant System temperature and pressure limitations in these MODES. Therefore, reactor building OPERABILITY is not required in MODE 5 or 6.

ACTIONS

A.1

In the event reactor building OPERABILITY is not maintained, reactor building OPERABILITY must be restored within 24 hours. Twenty-four hours is a reasonable Completion Time considering the limited leakage design of containment and the low probability of a Design Basis Accident occurring during this time period.

INSERT 5

B.1 and B.2

If the reactor building cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.16.1

Maintaining reactor building OPERABILITY requires maintaining the door in the access opening closed, except when the access opening is being used for normal transit entry and exit. The 31 day Frequency of this SR is based on engineering judgment and is considered adequate in view of the other indications of door status that are available.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.16.2

The annulus vacuum decay test is performed to verify the reactor building is OPERABLE. A minimum annulus vacuum decay time of 87 seconds ensures that the reactor building design outside air inleakage rate is ≤ 2000 cfm at an annulus differential pressure of -1.0 inch water gauge. Higher reactor building annulus outside air inleakage rates correlate to less holdup, mixing, and filtration of radiological effluents which increase offsite and operator doses.

The vacuum decay test is performed by isolating the pressure transmitter and starting the AVS fan to draw down the annulus pressure to a significant vacuum. Isolating the transmitter enables the fan to reduce the annulus pressure below the normal setpoint. The fan is then secured and the time it takes for the annulus pressure to decay or increase from -3.5 inches water gauge to -0.5 inch water gauge is measured. The time required for the pressure in the annulus to increase from -3.5 inches water gauge to -0.5 inch water gauge is known as the vacuum decay time.

The reactor building annulus outside air inleakage is an input to the CANVENT computer code, which provides input to the dose analyses. The CANVENT computer code is used to model the thermal effects of a LOCA on the annulus and the ability of the AVS to develop and maintain a negative pressure in the annulus after a design basis accident. The code also determines AVS exhaust and recirculation airflow rates following a LOCA. The results of the CANVENT analysis for annulus conditions and AVS response to the LOCA also are used for the rod ejection accident.

The 2000 cfm at -1.0 inch water gauge reactor building annulus outside air inleakage rate is conservatively corrected for ambient temperature and pressure as well as annulus differential pressure conditions prior to use as an input to the CANVENT computer code. The CANVENT results are then used as an input to the dose analyses.

The reactor building pressure boundary is tested every 18 months. The 18 month Frequency is consistent with the guidance provided in NUREG-0800.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.16.3

This SR would give advance indication of gross deterioration of the concrete structural integrity of the reactor building. The Frequency is based on engineering judgment, and is the same as that for containment visual inspections performed in accordance with SR 3.6.1.1.

REFERENCES

1. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
2. UFSAR, Sections 6.2.3 and 6.2.6.5.
3. NUREG-0800, Sections 6.2.3 and 6.5.3, Rev. 2, July 1981.