October 27, 2006

Mr. R. T. Ridenoure
Vice President - Chief Nuclear Officer
Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
Post Office Box 550
Fort Calhoun, NE 68023-0550

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: CHANGES TO THE UPDATED SAFETY ANALYSIS REPORT RELATED TO THE RADIOLOGICAL CONSEQUENCES OF EVENTS AFFECTED BY THE PLANNED 2006 REPLACEMENT OF THE STEAM GENERATORS AND PRESSURIZER (TAC NO. MC8857)

Dear Mr. Ridenoure:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 243 to Renewed Facility Operating License No. DPR-40 for the Fort Calhoun Station, Unit No. 1 (FCS). The amendment consists of changes to the Updated Safety Analysis Report in response to your application, dated October 31, 2005, as supplemented by letter dated July 25, 2006.

The amendment revises the FCS Updated Safety Analysis Report Sections related to the radiological consequences of events affected by the planned 2006 replacement of the steam generators and pressurizer.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA by M. Fields for/

Alan B. Wang, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosures: 1. Amendment No. 243 to DPR-40
2. Safety Evaluation

cc w/encls: See next page

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NAME	AWang	LFeizollahi	MKotzalas	JNakoski	AHodgdon	DTerao:MFields for/
DATE	10/1/06	10/17/06	9/18/06	2/7/06	10/24/06	10/27/06

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OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

FORT CALHOUN STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 243
License No. DPR-40

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Omaha Public Power District (the licensee), dated October 31, 2005, as supplemented on July 25, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, by Amendment No. 243, changes to the Updated Safety Analysis Report Sections related to the radiological consequences of events affected by the planned 2006 replacement of the steam generators and pressurizer as set forth in the application for amendment by the licensee, dated October 31, 2005, as supplemented by letter dated July 25, 2006, are authorized.

3. The license amendment is effective as of its date of issuance and shall be implemented within 90 days of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA M. Fields for/

David Terao, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Date of Issuance: October 27, 2006

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 243 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-40

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT NO. 1

DOCKET NO. 50-285

1.0 INTRODUCTION

By application dated October 31, 2005, as supplemented on July 25, 2006 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML053130369 and ML062070048, respectively), Omaha Public Power District (OPPD, the licensee) requested changes to the Updated Safety Analysis Report (USAR) for the Fort Calhoun Station, Unit No. 1 (FCS).

The proposed amendment would revise the FCS USAR Sections related to the radiological consequences of events affected by the planned 2006 replacement of the steam generators and pressurizer. Specifically, the proposed changes would revise the USAR, Section 14.1, as well as the radiological consequences analyses for the events of Seized Rotor (SR), Section 14.6.2.8; Main Steam Line Break (MSLB), Section 14.12.6; Control Element Assembly Ejection (CEAE), Section 14.13.4; and Steam Generator Tube Rupture (SGTR), Section 14.14.3.

The supplemental letter dated July 25, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 20, 2005 (70 FR 75493).

2.0 REGULATORY EVALUATION

The U.S. Nuclear Regulatory Commission (NRC) staff evaluated the radiological consequences of postulated events against the dose criteria specified in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67(b)(2).

Implementation of an alternative source term was previously reviewed and approved by the NRC staff in License Amendment No. 201 (ADAMS Accession No. ML013030027). This safety evaluation (SE) addresses the impact of the proposed replacement steam generator (RSG) and replacement pressurizer (RPRZ) on previously analyzed design-basis accident (DBA) radiological consequences and acceptability of the revised analysis results. The revised USAR Sections are Section 14.1, “General,” as well as the radiological consequence analyses for Seized Rotor Accident (SRA), Section 14.6.2.8; Main Steamline Break (MSLB), Section 14.12.6; Control Element Assembly Ejection (CEAE), Section 14.13.4; and Steam Generator Tube Rupture (SGTR), Section 14.14.3.

The regulatory requirements upon which the NRC staff based its acceptance are: the accident dose criteria in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 of Regulatory Guide (RG) 1.183, and Standard Review Plan (SRP) 15.0.1. The licensee has not proposed any significant deviation or departure from the guidance provided in RG 1.183. The NRC staff’s evaluation is based upon the following regulatory codes, guides, and standards:

- 10 CFR Section 50.67, “Accident source term,”
- 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants”: Criterion 19, “Control room,”
- SRP, Section 6.4, “Control Room Habitability Systems,”
- SRP, Section 15.0.1, “Radiological Consequence Analysis using Alternative Source Terms,” and
- RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.”

3.0 TECHNICAL EVALUATION

3.1 Core Inventory

The NRC staff evaluated the methodology used to determine the core inventory and the gap fractions used for non-loss-of-coolant accidents (non-LOCA) transients and accidents.

The FCS equilibrium core inventory for radiological consequence calculations was determined using ORIGEN-S calculations (Reference 3). The NRC staff has concluded that the use of ORIGEN-S is acceptable, in accordance with Section 3.1 of RG 1.183.

The amendment application did not request approval of the fuel/clad gap fractions contained in NUREG/CR-5009 for the fuel handling accident (FHA). The fuel gap fractions assumed for the FCS FHA continues to be the use of a 2.0 multiplier on the RG 1.183 values. Therefore, the NRC staff has noted that the fuel gap fractions for the FHA will continue to reflect previously approved FCS fuel gap fractions.

For the CEAE analysis, the licensee assumed 10 percent of the core inventory of noble gases and iodines are present in the fuel/clad gap. This assumption is consistent with RG 1.183. The NRC staff concludes that this assumption is acceptable.

For the other non-LOCA events which experience fuel damage (e.g. Locked Rotor), using the fuel/clad gap fission-product inventory from NUREG/CR-5009 (Reference 2) is conservative, since on a core-wide basis, only a small fraction of the fuel rods exceed the applicability criteria. In addition, the cycle-maximum radial peaking factor was applied to all failed fuel rods. For core-wide events, the NRC staff finds that the gap fractions from NUREG/CR-5009 are bounding for this amendment request.

Gap Fractions - Non-LOCA Events		
Nuclide Group	**Regulatory Guide 1.183**	**NUREG/CR 5009**
I-131	0.08	0.12
Kr-85	0.10	0.30
Other Noble Gases	0.05	0.10
Other Halogens	0.05	0.10
Alkali Metals	0.12	0.17

Based upon the above review, the NRC staff finds the determination of pressurized-water reactor core inventory and the use of the gap fractions from NUREG/CR-5009 for non-LOCA transients and accidents are acceptable.

3.2 Confirmatory Dose Calculations

The NRC staff performed independent confirmatory dose calculations, for DBA events affected by the RSG/RPRZ, using the NRC-sponsored radiological consequence computer code, NUREG/CR-6604, "RADTRAD: Simplified Model for RADionuclide Transport and Removal And Dose Estimation," Version 3.03. The RADTRAD code, developed by the Sandia National Laboratories for the NRC, estimates transport and removal of radionuclides and radiological consequence doses at selected receptors. The findings of this SE input are based on the description of the licensee's analysis and other supporting information docketed by OPPD.

The proposed RSG/RPRZ will change the reactor coolant system (RCS) volume and will affect some parameters used in the radiological consequence analyses on record. These parameters are: timing of steam generator tube uncovery, steam release duration, percent of failed/melted fuel, gap fractions used for non-LOCA events, and the duration of the concurrent iodine spike (CIS). The change in parameters will have an impact on some of the DBA radiological consequences analyses. A systematic review of the analyses on record revealed that the affected DBAs are SRA, CEAE, MSLB outside containment, and SGTR.

The associated FCS thermal-hydraulic analysis, including equilibrium fission products core inventory and gap fractions used in re-analysis of the affected DBAs, was reviewed in Section 3.1, "Core Inventory."

Thermal-hydraulic analysis indicates that the percentage of failed/melted fuel for the SRA, CEAE, SGTR, and MLSB events is 0 percent/0 percent. Since the SGTR and MLSB events do not challenge fuel integrity, the licensee will maintain the current licensing basis values of 0 percent/0 percent failed/melted fuel.

Although the thermal-hydraulic analyses of the SRA and CEAE events indicate no fuel failure or melting, to account for residual uncertainties regarding fuel behavior, the revised SRA dose analysis assumes 0.5 percent/0 percent, and the revised CEAE dose analysis assumes 1 percent/1 percent of failed/melted fuel. The current licensing basis contains values of 1 percent/0 percent and 10 percent/1 percent, respectively, for these accidents. The proposed fuel failure/melt fractions remain conservative and, therefore, are acceptable.

The licensee also proposes to revise the licensing basis for fuel gap fractions, as summarized in the table below, for non-LOCA events. The proposed gap inventory, based on NUREG/CR-5009, is more conservative than stated in RG 1.183 and, therefore, is acceptable. The revised gap inventories, used in this application for the updated analyses of SRA, CEAE, MSLB, and SGTR, are intended to be used in future USAR revisions.

Nuclide Group	**RG 1.183 Gap Fraction**	**Current FCS Licensing Basis**	**Proposed Revised Licensing Basis (NUREG/CR-5009)**
I-131	0.08	0.16	0.12
Kr-85	0.10	0.20	0.30
Other noble gases	0.05	0.10	0.20
Other halogens	0.05	0.10	0.10
Alkali metals	0.12	0.24	0.17

The licensee’s dose calculations conform to the NRC staff’s guidance in RG 1.183 for calculating the total effective dose equivalent (TEDE), i.e., the sum of the committed effective dose equivalent from inhalation and the deep dose equivalent from external exposure. Therefore, this is acceptable to the NRC staff.

Assumptions regarding the control room (CR) design, operation, and transport model are the same as those previously reviewed and approved by the NRC staff in License Amendment No. 201.

Major assumptions and key parameters used in the revised analyses are presented in Tables 1 and 2. The licensee’s results are presented in Table 3 for offsite receptors and Table 4 for the CR. These tables are attached to this SE.

3.2.1 <u>Seized Rotor Accident</u>

Assessment of the SRA follows the guidance provided in RG 1.183, with two exceptions. One exception is the assumed breathing rates, which are consistent with current licensing basis. These breathing rates are based on the draft guide (DG 1081) rather than on the final version of RG 1.183. The effect of this deviation is negligible. The other exception is the assumed fuel gap fractions which were discussed above.

The released gap activity is assumed to be instantaneously and homogeneously mixed within the RCS and leaked to the secondary side of the steam generators at the rate of 1 gallon per minute, or 1,440 gallons per day. The leakage is considerably greater than the Technical Specification (TS) limit of 150 gallons per day for each of the two cooling loops.

The main condenser is conservatively assumed to be unavailable after a reactor trip to maximize the environmental releases. Also, to maximize calculated dose consequences, all of the radioactivity is assumed to be released through the main steam safety valves (MSSVs) and atmospheric dump valve (ADV). A conservatism of this assumption is that a portion of the release would actually occur through the turbine exhaust of the turbine-driven auxiliary feedwater (AFW) pump, which has an associated atmospheric dispersion factor bounded by that of the MSSVs/ADV.

The assumptions regarding operation of the CR emergency ventilation system remain unchanged, which is acceptable since the effect of RSG and RPRZ on timing of the CR emergency system is negligible.

3.2.2 Control Element Assembly Ejection

Following the guidance of RG 1.183, the licensee postulated two pathways for environmental release. One is through containment leakage and the other is through primary-to-secondary steam generator (SG) tube leakage. The pathways are evaluated separately.

The containment is assumed to leak at the TS rate of 0.1 percent-volume per day, decreasing to half that value after 24 hours. The SG tubes are assumed to leak at the rate of 1 gallon per minute, or 1,440 gallons per day.

The calculated SG tube uncovery occurs at 50.69 seconds and lasts for a period of 112 minutes. The release of activity to the environment is terminated at 61.5 hours due to the initiation of shutdown cooling (SDC).

The RSG and RPRZ affect reactor trip timing, RCS mass, tube uncovery timing and duration, and steam mass release used in the CEAE accident dose analysis. All other analysis assumptions and inputs are unaffected and remain the same as in the current licensing basis.

3.2.3 Main Steamline Break Outside Containment

The MLSB event is assumed to happen simultaneously with a loss of offsite power (LOOP), with the condenser assumed to be unavailable. The MSSVs/ADV of the intact SG (unbroken main steam line) is used to cool down the reactor until the SDC system is initiated at 156.4 hours, at which point the environmental releases are terminated. However, releases from the affected SG (broken main steam line) continue until the reactor coolant temperature reaches 212 degrees Fahrenheit at 159.2 hours.

Since there is no postulated fuel damage associated with this accident, the main radiation source is the activity in the primary and secondary coolant system. Two iodine spiking cases are considered: a pre-accident iodine spike (PIS) and a CIS. Per RG 1.183, the assumed iodine composition released to the environment is 97 percent elemental and 3 percent organic.

The duration of CIS is assumed to be 4 hours, which is a change of current licensing basis duration time of 8 hours. The updated duration of the spike is based on depletion of all activity available in the gap of defective fuel. Such an approach is consistent with RG 1.183 and, therefore, is acceptable.

The iodine activity gap fraction is a combination of that stated in RG 1.183 and the one proposed for the revised licensing basis (i.e., iodine gap fraction for once and twice-irradiated fuel (two-thirds of the core) is based on RG 1.183; iodine gap fraction for fuel irradiated three times (one-third of the core) is based on NUREG/CR-5009). The resultant iodine gap fraction (0.093) is still more conservative than stated in RG 1.183 (0.08) and, therefore, is acceptable.

Two release paths are considered: one from the breakpoint (Room 81 Pressure Relief Domes) and one from the MSSVs/ADV. As in the case of SRA, a release through the turbine exhaust of the turbine-driven AFW pump is conservatively modeled as occurring through the MSSVs/ADV.

The RSG and RPRZ affect reactor trip timing, RCS mass, tube uncovery timing and duration, and steam mass release used in the MSLB accident dose analysis. All other analysis assumptions and inputs are unaffected and remain the same as in the current licensing basis.

The revised analysis reduces the margin to the regulatory limit for the CR dose following an MSLB with a concurrent iodine spike, due to the significant delay in the reactor coolant reaching 212 degrees Fahrenheit (i.e., 159.2 hours versus 10.94 hours previously used). The NRC staff recognizes, however, that there is substantial conservatism in the assumption of 1 percent defective fuel versus 0.28 percent, as implied by the coolant activity concentration limit specified in the TS.

3.2.4 Steam Generator Tube Rupture

The worst case SGTR scenario with RSG/RPRZ is a tube rupture at the top of the tubesheet on the hot-leg side of the SG. Based on revised thermal hydraulic analysis, the reactor trip will occur at 629.5 seconds after the postulated event. To maximize the activity releases, the main condenser is assumed to be unavailable. The environmental releases are terminated after the SDC initiation at 144.6 hours.

Since there is no postulated fuel damage associated with this accident, the main radiation source is the activity in the primary and secondary coolant systems. Two iodine spiking cases are considered in accordance with RG 1.183: PIS and CIS.

The releases to the environment from the faulted SG are from the main condenser air ejector until the reactor trip. After that, the releases continue through the MSSVs/ADV until the SG is isolated 2 hours after the accident.

The releases to the environment from the intact SG are through the MSSVs/ADV until SDC initiation.

The CR emergency ventilation initiation occurs as a result of a safety injection actuation signal at 678.5 seconds. An additional delay of 44 seconds is assumed to account for a coincident LOOP.

The RSG and RPRZ affect reactor trip timing, RCS mass, tube uncovery timing and duration, and steam mass release used in the SGTR accident dose analysis. All other analysis assumptions and inputs are unaffected and remain the same as in the current licensing basis.

3.2.5 Atmospheric Dispersion

The licensee used current licensing basis CR, exclusion area boundary (EAB) and low-population zone (LPZ) atmospheric dispersion factors (X/Q values) to perform dose assessments to evaluate the impact of the planned 2006 RSG and RPRZ. The X/Q values were previously generated by the licensee in support of FCS Amendment No. 201, dated December 5, 2001. In the July 25, 2006, response to the NRC request for additional information, the licensee confirmed that the release paths for the analysis of the current license amendment request are the same as those of Amendment No. 201. Based on the review described in the SE associated with the Amendment No. 201, the NRC staff has concluded that the FCS X/Q values on record are acceptable for use in the DBA CR, EAB and LPZ dose assessments performed in support for this license amendment request.

3.3 Summary of Revised Analyses

The licensee evaluated the radiological consequences resulting from the postulated non-LOCA events, affected by RSG/RPRZ, and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose criteria provided in 10 CFR 50.67 and accident dose guidelines specified in SRP 15.0.1. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the staff are discussed in Section 3.0. The staff performed independent calculations using the licensee's assumptions. The NRC staff analyses confirmed the licensee's calculated results. The NRC staff finds that the EAB, LPZ, and CR doses, estimated by the licensee for the SRA, CEAE, MSLB, and SGTR, presented in Tables 3 and 4, meet the applicable dose criteria and are, therefore, acceptable.

The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of postulated SRA, CEAE, MSLB and SGTR events, affected by the RSG/RPRZ. The NRC staff finds that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0. The NRC staff compared the doses estimated by the licensee to the applicable criteria identified in Section 2.0. The NRC staff also finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and CR doses will comply with these criteria. Therefore, the NRC staff finds reasonable assurance that FCS, as modified by the RSG/RPRZ, will continue to provide sufficient safety margins, with adequate defense-in-depth, to address unanticipated events, and to compensate for uncertainties in accident progression and analysis assumptions and parameters. Therefore, the proposed license amendment is acceptable with respect to the radiological consequences of the affected DBA events.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Nebraska State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (70 FR 75493; published on December 20, 2005). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Letter from R. T. Ridenoure (Omaha Public Power District) to the NRC, “Updated Safety Analysis Report Revision for Radiological Consequences Analysis for Replacement NSSS Components,” dated October 31, 2005.

2. NUREG/CR-5009, PNL-6258, “Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors,” February 1988, Pacific Northwest Laboratory, Richland, Washington.

3. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000, U.S. Nuclear Regulatory Commission.

Principal Contributors: A. Drozd
L. Brown
C. Brown

Date: October 27, 2006

Table 1
Selected parameter values and assumptions for SRA, CEAE, MSLB and SGTR

Containment pathway

Power level	1530 MWt
Containment free volume	1.05E+6 ft^3
Reactor coolant mass	250,000 lbm
Reactor trip	0.69 sec (629.5 sec for SGTR)
Primary to secondary tube leakage	1 gpm at STP (two SGs)
Failed /melted fuel percentage	0%/0.5% (SRA) 1%/1% (CEAE) 0%/0% (MSLB) 0%/0% (SGTR)
Termination of releases	30 days (CEAE only)

Secondary side pathway

Minimum liquid mass for intact SG	45,708 lbm per SG
Minimum liquid mass for faulted SG	70,261 lbm
Maximum liquid mass for MSLB	116,683 lbm
Released iodine composition	97% I_2, 3% organic
PIC (for MSLB and SGTR only)	60 µCi/gm DE I-131
CIS (for MSLB and SGTR only)	500 times equilibrium (MSLB) 335 times equilibrium (SGTR)
Duration of CIS	4 hours (MSLB) 8 hours (SGTR)
Environmental release point	MSSVs/ADV
Termination of releases	61.5 hours (SRA and CEAE) 159.2 hours (MSLB) 144.6 hours (SGTR)

Table 2
Selected parameter values and assumptions for FCS Control Room

Free volume	45,100 ft^3
Unfiltered normal operation intake	1000 cfm +/- 10%
Emergency intake rate	1000 cfm +/- 10%
Emergency recirculation rate	1000 cfm +/- 10%
Emergency intake filter efficiency	99% (iodine and particulates)
Emergency recirculation filter efficiency	99% (iodine and particulates)
Unfiltered inleakage	38 cfm
Occupancy factors	0-24 hr (1.), 1-4 d (0.6), 4-30 d (0.4),
Operator breathing rate	0-30 days (3.47E-4 m^3/sec)
Operator action to repair recirc damper	2 hours after accident
Emergency intake rate during repair	1000 cfm +/- 10% to 2000 cfm +/- 10%
Diesel generator start up	14 seconds
CR damper realignment	15 seconds
CR emergency fan ramp up	15 seconds
Total	44 seconds

Table 3
Revised calculated EAB and LPZ doses (TEDE)
(rem)

Accident	EAB		LPZ		Regulatory limit
	current	revised	current	revised	
SRA	0.50	0.50	0.50	0.50	2.50
CEAE	2.0	1.50	0.50	0.50	6.30
MSLB (PIS)	0.50	0.50	0.50	0.50	25.0
MSLB (CIS)	1.50	1.0	0.50	0.50	2.50
SGTR (PIS)	1.50	1.0	0.50	0.50	25.0
SGTR (CIS)	1.50	1.0	0.50	0.50	2.50

Table 4
30 day integrated CR doses (TEDE)
(rem)

Accident	Current	Revised	Regulatory limit
SRA	4.70	2.50	5.0
CREA	3.0	3.0	5.0
MSLB (PIS)	1.0	1.0	5.0
MSLB (CIS)	2.50	4.50	5.0
SGTR (PIS)	1.50	1.0	5.0
SGTR (CIS)	1.50	1.50	5.0

Ft. Calhoun Station, Unit 1

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