UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 30, 2011

Mr. Adam C. Heflin
Senior Vice President and
Chief Nuclear Officer
Union Electric Company
P.O. Box 620
Fulton, MO 65251

SUBJECT: CALLAWAY PLANT, UNIT 1 - ISSUANCE OF AMENDMENT RE: REVISION OF TECHNICAL SPECIFICATION 3.3.8 (TAC NO. ME5173)

Dear Mr. Heflin:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 204 to Facility Operating License No. NPF-30, for the Callaway Plant, Unit 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated December 10, 2010, as supplemented by letters dated June 16, October 27, and December 13, 2011.

The amendment adds new Surveillance Requirement (SR) 3.3.8.6, to TS 3.3.8, "Emergency Exhaust System (EES) Actuation Instrumentation." The new SR requires the performance of response time testing on the portion of the EES required to isolate the normal fuel building ventilation exhaust flow path and initiate the fuel building ventilation isolation signal mode of operation. The amendment also revises TS Table 3.3.8-1, "EES Actuation Instrumentation," to state that the new SR 3.3.8.6 applies to automatic actuation Function 2, Automatic Actuation Logic and Actuation Relays (BOP ESFAS), and Function 3, Fuel Building Exhaust Radiation - Gaseous. In addition, the specified frequency of new SR 3.3.8.6 would be relocated and controlled in accordance with the licensee's Surveillance Frequency Control Program.

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,

Mohan C. Thadani

Mohan C. Thadani, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosures:
1. Amendment No. 204 to NPF-30
2. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 204
License No. NPF-30

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

   A. The application for amendment by Union Electric Company (UE, the licensee), dated December 10, 2010, as supplemented by letters dated June 16, October 27, and December 13, 2011, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's 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 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, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-30 is hereby amended to read as follows:

 (2) Technical Specifications and Environmental Protection Plan*

 The Technical Specifications contained in Appendix A, as revised through Amendment No. 204 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of its date of issuance, and shall be implemented within 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

FOR

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License No. NPF-30 and Technical Specifications

Date of Issuance: December 30, 2011

ATTACHMENT TO LICENSE AMENDMENT NO. 204

FACILITY OPERATING LICENSE NO. NPF-30

DOCKET NO. 50-483

Replace the following pages of the Facility Operating License No. NPF-30 and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Facility Operating License

REMOVE	INSERT
-3-	-3-

Technical Specifications

REMOVE	INSERT
3.3-77	3.3-77
3.3-78	3.3-78

(4) UE, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source of special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) UE, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

UE is authorized to operate the facility at reactor core power levels not in excess of 3565 megawatts thermal (100% power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan*

The Technical Specifications contained in Appendix A, as revised through Amendment No. 204 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Environmental Qualification (Section 3.11, SSER #3)**

Deleted per Amendment No. 169.

* Amendments 133, 134, & 135 were effective as of April 30, 2000 however these amendments were implemented on April 1, 2000.

** The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.8.5	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program
SR 3.3.8.6	------------------------------ NOTE ------------------------------ Radiation monitor detectors are excluded from response time testing. -- Verify Fuel Building Ventilation Exhaust ESF RESPONSE TIMES are within limits.	In accordance with the Surveillance Frequency Control Program

Table 3.3.8-1 (page 1 of 1)
EES Actuation Instrumentation

	FUNCTION	APPLICABLE MODES OR SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	NOMINAL TRIP SETPOINT
1.	Manual Initiation	(a)	2	SR 3.3.8.4	NA
2.	Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	(a)	2 trains	SR 3.3.8.3 SR 3.3.8.6	NA
3.	Fuel Building Exhaust Radiation - Gaseous	(a)	2	SR 3.3.8.1 SR 3.3.8.2 SR 3.3.8.5 SR 3.3.8.6	(b)

(a) During movement of irradiated fuel assemblies in the fuel building.
(b) Nominal Trip Setpoint concentration value ($\mu Ci/cm^3$) shall be established such that the actual submersion dose rate would not exceed 4 mR/hr in the fuel building.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 204 TO

FACILITY OPERATING LICENSE NO. NPF-30

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

1.0 INTRODUCTION

By application dated December 10, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML103470204), as supplemented by letters dated June 16, October 27, and December 13, 2011 (ADAMS Accession Nos. ML111680233, ML113010383, and ML113480060, respectively), Union Electric Company (the licensee), requested U.S. Nuclear Regulatory Commission (NRC) approval of a license amendment request (LAR) of the Callaway Plant, Unit 1 (Callaway) Facility Operating License. The amendment would add new Surveillance Requirement (SR) 3.3.8.6 to Technical Specification (TS) 3.3.8, "Emergency Exhaust System (EES) Actuation Instrumentation." The new SR would require the performance of response time testing on the portion of the EES required to isolate the normal fuel building ventilation exhaust flow path and initiate the fuel building ventilation isolation signal (FBVIS) mode of operation. The amendment also would revise TS Table 3.3.8-1, "EES Actuation Instrumentation," to indicate that new SR 3.3.8.6 applies to automatic actuation Function 2, Automatic Actuation Logic and Actuation Relays (BOP ESFAS), and Function 3, Fuel Building Exhaust Radiation - Gaseous. In addition, the specified frequency of new SR 3.3.8.6 would be relocated and controlled in accordance with the licensee's Surveillance Frequency Control Program.

The supplemental letter dated June 16, 2011, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 8, 2011 (76 FR 12766).

By supplemental letter dated October 27, 2011, the licensee requested that the specified frequency of new SR 3.3.8.6 be relocated and controlled in accordance with the licensee's Surveillance Frequency Control Program. The no significant hazards consideration determination of the additional change was not evaluated in the initial notice (76 FR 12766; March 8, 2011); therefore, on November 29, 2011, the NRC staff republished the notice of proposed no significant hazards consideration incorporating the new change (76 FR 73733).

The supplemental letter dated December 13, 2011, provided additional information that clarified the application, as supplemented, did not expand the scope of the application, as supplemented, and did not change the NRC staff's revised proposed no significant hazards consideration determination, as published in the *Federal Register* on November 29, 2011 (76 FR 73733).

2.0 REGULATORY EVALUATION

This safety evaluation describes the NRC staff's evaluation of the impact of the proposed changes on analyzed design-basis accident (DBA) radiological consequences as a result of a postulated fuel handling accident (FHA) event. General Design Criterion (GDC) 61, "Fuel storage and handling and radioactivity control," in Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 states that

> The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal , and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

A postulated FHA in the fuel building as discussed in Callaway's Final Safety Analysis Report (FSAR), Section 15.7.4, "Fuel Handling Accidents," assumes credit for automatic actuation of the EES following receipt of an FBVIS or a safety injection signal. Actuation of the FBVIS mode of the EES isolates the outside air intake system, trips the supply air handling units, and closes the corresponding dampers in the normal exhaust ductwork to the auxiliary building in order to isolate the fuel building and mitigate the radiological consequences of an FHA.

This evaluation has been conducted to verify that the radiological dose consequence analyses continue to meet the dose acceptance criteria given in 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance," for offsite doses, GDC 19, "Control room habitability," in Appendix A of 10 CFR Part 50, for control room dose, and the regulatory guidance provided in NRC Regulatory Guide (RG) 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," May 2003 (ADAMS Accession No. ML031490640). The applicable acceptance criterion is 5 roentgen equivalent man (rem) whole body in the control room, 300 rem thyroid and 25 rem whole body at the exclusion area boundary (EAB), and 300 rem thyroid and 25 rem whole body at the outer boundary of the low population zone (LPZ). Furthermore, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR [Light-Water Reactor] Edition" (SRP), Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents," provides a more specific acceptance criterion of 25 percent or less of the 10 CFR Part 100 limits (i.e., 75 rem thyroid and 6.25 rem whole body at both the EAB and LPZ boundaries, respectively) for an FHA. In addition, the

NRC staff's technical evaluation is based upon the licensee's previously approved DBA analysis of record, and the descriptions and results presented in the proposed LAR.

In 10 CFR 50.36, "Technical specifications," the NRC established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. The regulations in 10 CFR 50.36(c)(3), "Surveillance requirements," state that,

> Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

The licensee's request for relocation of the frequency of new SR 3.3.8.6 to the licensee's Surveillance Frequency Control Program is based on a previously approved methodology for Callaway in Amendment No. 202 dated July 29, 2011 (ADAMS Accession No. ML111661877). That methodology was based on the Commission's policy on performance-based/risk-informed considerations specified in the following regulatory guides.

- NRC Regulatory Guide (RG) 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," November 2002 (ADAMS Accession No. ML023240437), describes a risk-informed approach, acceptable to the NRC, for assessing the nature and impact of proposed permanent licensing-basis changes by considering engineering issues and applying risk insights. This regulatory guide also provides risk acceptance guidelines for evaluating the results of such evaluations.

- NRC RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," August 1998 (ADAMS Accession No. ML003740176), describes an acceptable risk-informed approach specifically for assessing proposed permanent TS changes.

- NRC RG 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," January 2007 (ADAMS Accession No. ML070240001), describes an acceptable approach for determining whether the quality of the probabilistic risk assessment (PRA), in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision making for light-water reactors.

3.0 TECHNICAL EVALUATION

Radiological consequence analyses show that an FHA is postulated to occur during movement of irradiated fuel assemblies within the fuel building and/or reactor building. During fuel handling operations within the fuel building, a fuel assembly could be dropped in the transfer canal, in the fuel storage pool, or in the cask loading pool. In addition to the area radiation monitors located on the wall around the fuel storage pool, portable radiation monitors capable of emitting audible alarms are also located in this area. Should a fuel assembly be dropped in the canal, in the cask loading pit, or in the pool, and release radioactivity above a prescribed level, the radiation monitors will sound an audible alarm and automatic actuation of the EES will commence following receipt of an FBVIS or a safety injection signal.

The fuel building at Callaway is served by an outside air supply system which provides fresh outside air, either heated or cooled as required, to all areas of the fuel building. Within the fuel building, the auxiliary/fuel building normal exhaust system takes suction from the area above the spent fuel pool and mixes that air with the air from the auxiliary building prior to processing it through the auxiliary/fuel building filter absorber train and discharging it to the unit vent. The EES ensures that radioactive materials in the fuel building atmosphere following an FHA are filtered and adsorbed prior to being exhausted to the environment. Accordingly, a control room ventilation isolation signal is also initiated to ensure control room habitability. These actions reduce the radioactive content in the fuel building exhaust following an FHA so that offsite doses remain within the guidelines specified in 10 CFR 100.11, and that control room habitability is maintained within dose limits specified by GDC 19.

In the analysis of the radiological consequences of an FHA in the fuel building, it is assumed that fuel building ventilation is switched from the auxiliary/fuel building normal exhaust system to the EES within an assumed time interval. This time interval begins from the time when radioactivity in the exhaust reaches the exhaust duct and produces a count rate from the detector that corresponds to the actuation setpoint for the radiation monitor. Additionally, the activity released before completion of the fuel building ventilation switchover is assumed to be discharged directly to the environment with no credit for filtration or dilution. 32.1 seconds was the previously approved parameter to be assumed for the fuel building isolation time in the previous FHA in the fuel building analysis of record. However, the licensee proposed to use a 90-second response time as an appropriate limit for the fuel building ventilation exhaust engineered safety feature (ESF) response.

The NRC staff reviewed the licensee's regulatory and technical assessments, as they relate to the radiological consequences of DBA analyses, in support of the proposed TS changes and change in the analysis assumptions. In Attachment 1 of the licensee's letter dated December 10, 2010, the licensee stated that the only DBA analysis affected by the proposed change is the design-basis FHA accident. Only the postulated FHA in the fuel building, as described in Callaway's FSAR Section 15.7.4, "Fuel Handling Accidents," relies on automatic actuation of the EES and realignment to the FBVIS mode of operation to mitigate the radiological consequences. This review evaluates the specific effect of the proposed TS changes on Callaway's current FHA analysis of record. The LAR proposes the following changes for Callaway, which are described in Sections 3.1 through 3.4 of this safety evaluation:

- Addition of new SR 3.3.8.6 to TS 3.3.8;
- Use of a 90-second response time as an appropriate limit for the fuel building ventilation exhaust ESF (fuel building isolation time);
- Revision to TS Table 3.3.8-1; and
- Relocation of the frequency of new SR 3.3.8.6 to licensee's Surveillance Frequency Control Program.

3.1 Addition of SR 3.3.8.6 to TS 3.3.8, "Emergency Exhaust System (EES) Actuation Instrumentation"

New SR 3.3.8.6 would state:

> -- NOTE --
> Radiation monitor detectors are excluded from response time testing.
> --
>
> Verify Fuel Building Ventilation Exhaust ESF RESPONSE TIMES are within the limits.

New SR 3.3.8.6 would require the performance of response time testing on a portion of the EES required to isolate the normal fuel building ventilation exhaust flow path and initiate the FBVIS mode of operation.

The fuel building FHA analysis was performed by the licensee with an assumed fuel building ventilation exhaust ESF response time of 90 seconds. Regular response time testing will enhance the confidence that, if the fuel building ventilation exhaust redundant radiation monitors indicate that as the radiation level exceeds the allowed limit, the response will occur within 90 seconds as assumed in the analysis of record. Therefore, the NRC staff concludes that the proposed TS change is acceptable.

3.2 Use of 90 Seconds as an Appropriate Fuel Building Ventilation Exhaust ESF Response Time

The proposed use of the 90-second ESF response time directly affects the radiological dose model and consequence analyses of the design-basis FHA. In its letter dated June 16, 2011, in response to an NRC staff request for additional information (RAI) dated May 19, 2011 (ADAMS Accession No. ML111390388), the licensee stated that no other model, inputs, parameters, assumptions, and/or methodologies within the FHA consequence analyses at Callaway are affected by the proposed changes.

The current radiological analysis of record credits an FBVIS for actuating a Control Room Ventilation Isolation Signal, within 32.1 seconds following a FHA in the fuel building at Callaway. In addition, the current response time assumption of 32.1 seconds is based on historical EES damper stroke times measured under Callaway's safety-related preventive maintenance program, as well as response-time testing performed per SR 3.3.7.6 for similar design elements associated with the control room emergency ventilation system. As presented in Attachment 5 of the licensee's letter dated December 10, 2010, the ESF response time (or building isolation time) is a direct input into the radiological model to estimate both control room and offsite radiological consequences. To support the proposed change, the licensee reanalyzed the FHA in the fuel building at Callaway to determine if the proposed ESF response time of 90 seconds would result in acceptable radiological consequences.

In Table 15.7-7, "Parameters Used in Evaluating the Radiological Consequence of a Fuel-Handling Accident," and Table 15.7-8, "Radiological Consequences of a Fuel Handling Accident," in Attachment 5 to the LAR, and its letter dated June 16, 2011, the licensee provided the inputs and/or parameters used for its radiological model and the resulting offsite and control room radiological dose consequence results. The NRC staff evaluated the proposed modifications and resulting dose consequences by comparing the proposed modifications against the current analysis of record for Callaway. The NRC staff conducted a confirmatory analysis using the proposed ESF response time of 90 seconds. The comparison of the current analysis of record against the proposed ESF response time of 90 seconds is outlined below in Table 1 for the FHA analysis in the fuel building. The NRC staff's comparison validates that no other inputs, parameters, and/or assumptions have been changed and/or credited in the proposed LAR.

Table 1

Comparison of Parameters (Inputs) Used in Evaluating the Radiological Consequences of a Fuel Handling Accident in the Fuel Building at Callaway Plant (Proposed Modifications Versus Current Analysis of Record)

	Current Analysis of Record	Proposed Modifications
Source Data		
Core Power Level (MWt)	3,636	3,636
Radial Peaking Factor	1.65	1.65
Decay Time	72	72
Number of Fuel Assemblies	1	1
Fraction of fission product gases contained in the gap region of the fuel assembly	Per RG 1.25[1]	Per RG 1.25
Atmospheric Dispersion Factors	See FSAR Table 15A-2	See FSAR Table 15A-2
Activity Release Data		
Percent of affected fuel assemblies gap activity release	100	100
Pool decontamination factors		
1. Iodine	100	100
2. Noble Gas	1	1
Filter efficiency (%)	0 until isolation, 90 thereafter	0 until isolation, 90 thereafter
Building mixing volumes assumed (% of total volume)	0	0
HVAC exhaust rate (cfm)	20,000 until isolation, 9,000 thereafter	20,000 until isolation, 9,000 thereafter
Building isolation time (ESF response time)	32.1 sec	90 sec
Activity release period (hours)	2	2

[1] NRC Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," dated March 1972 (ADAMS Accession No. ML083300022).

The NRC staff reviewed the licensee's assumptions, methodology, and conclusions regarding the radiological dose consequence results of the reanalysis of the design-basis FHA at Callaway. Based on the confirmatory analyses performed, the NRC staff confirmed that the total radiological dose consequence of the proposed changes increased minimally for the EAB, LPZ, and control room at Callaway. Specifically, the licensee determined that EAB doses increased by 0.81 rem to 6.40 rem thyroid, and by 0.001 rem to 0.235 rem whole-body; that the LPZ doses increased by 0.081 rem to 0.640 rem thyroid, and by 0.0001 rem to 0.0235 rem whole body; and that all of the control room doses were less than 1.0 rem, all of which remain below the applicable dose acceptance guidelines and criteria described in 10 CFR 100.11 and GDC 19 and, accordingly, are acceptable. The NRC staff's confirmatory calculations validated the licensee's radiological dose consequence results for the EAB, LPZ, and control room at Callaway. Specifically, the NRC staff determined that EAB doses increased to 6.24 rem thyroid, and to 0.235 rem whole body; that the LPZ doses increased to 0.627 rem thyroid, and to 0.0235 rem whole body; and that all of the control room doses were less than 1.0 rem for each the thyroid, whole body, and beta skin doses. The radiological dose consequence results are presented below in Table 2.

Table 2
Comparison of the Radiological Dose Consequence Results of a Fuel Handling Accident at Callaway Plant (Proposed Modifications Versus Regulatory Acceptance Criteria)

Fuel Building Consequences	Current Analysis of Record(1)	Regulatory Acceptance Criteria(2)	Proposed Modifications(3)
Exclusion Area Boundary (0-2 hr)			
Thyroid (rem)	5.59	75.00	6.40
Whole body (rem)	0.234	6.25	0.235
Low Population Zone Outer Boundary (duration)			
Thyroid (rem)	0.559	75.00	0.640
Whole body (rem)	0.0234	6.25	0.0235

(1) Current analysis of record doses are from Callaway's FSAR Table 15.6-8 for the radiological consequences for the control room operator does for a large break loss-of-coolant accident, which bounds the FHA in the fuel building. These control room doses were also reported by the NRC in the safety evaluation in support of Amendment No. 152 dated September 9, 2002 (ADAMS Accession No. ML022620599).

(2) Regulatory acceptance criterion are from NRC RG 1.195, Table 4, "EAB and LPZ Accident Dose Criteria," and Section 4.5, "Control Room Acceptance Criteria."

(3) Proposed doses are from Table 15.7-8 of the LAR for the radiological consequences at the EAB and outer boundary of the LPZ for an FHA in the fuel building. The reported dose for the radiological consequences for the control room operator for an FHA in the fuel building are from the licensee's response to NRC staff RAIs.

3.3 Revision to TS Table 3.3.8-1

The Callaway LAR also proposes a revision to TS Table 3.3.8-1, "EES Actuation Instrumentation," to indicate that proposed SR 3.3.8.6 is applicable to both the automatic actuation Function 2, "Automatic Actuation Logic and Actuation Relays (BOP ESF AS)," and Function 3, "Fuel Building Exhaust Radiation – Gaseous," of the EES actuation instrumentation.

As described in Section 3.2 of this safety evaluation, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the impact of the proposed licensing basis changes on the postulated design basis FHA. The staff concludes that the licensee's revised design-basis FHA analysis conservatively accounts for the impact of the proposed changes and, following implementation of these changes, the licensee will continue to meet the applicable dose acceptance guidelines and criteria, as identified in Section 2.0 of this evaluation. The application of new SR 3.3.8.6 to the specified EES actuation instrumentation will further ensure that the assumptions in the revised FHA analysis will be met. Therefore, the NRC staff concludes that the proposed TS change is acceptable.

3.4 Relocation of the Frequency of New SR 3.3.8.6 to the Licensee's Surveillance Frequency Control Program

The NRC's "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," published in the *Federal Register* on July 22, 1993 (58 FR 39132), addressed the use of Probabilistic Safety Analysis (PSA, currently referred to as Probabilistic Risk Assessment (PRA)) and supports, in part, the increased use of PRA technology in "all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy." The NRC staff approved relocation of SRs to licensees' Surveillance Frequency Control Program based on Technical Specification Task Force (TSTF) change traveler TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control - RITSTF Initiative 5b," and Nuclear Energy Institute (NEI) guidance NEI 04-10, Revision 1, "Risk Informed Technical Specification Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies." TSTF-425 and NEI 04-10 justify the relocation of all periodic surveillance frequencies from the TSs to licensee control in accordance with new licensee-controlled programs called the Surveillance Frequency Control Program.

New SR 3.3.8.6 is of a recurring class of frequencies, and belongs to a class of SRs whose relocation to the licensee-controlled Surveillance Frequency Control Program was approved in Amendment No. 202 dated July 29, 2011. Therefore, the NRC staff concludes that the relocation of the frequency of SR 3.3.8.6 to the licensee's Surveillance Frequency Control Program is acceptable.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration which is presented below.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   There are no design changes associated with the proposed change. All design, material, and construction standards that were applicable prior to this amendment request will continue to be applicable.

   The proposed change will not affect accident initiators or precursors nor adversely alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained with respect to such initiators or precursors. There will be no change to fuel handling methods and procedures.

   Therefore, there will be no changes that would serve to increase the likelihood of occurrence of a fuel handling accident.

   The proposed change changes a performance requirement, but it does not physically alter safety-related systems nor affect the way in which safety-related systems perform their functions.

   The proposed TS change will serve to assure that the fuel building ventilation exhaust ESF [emergency safety feature] response time is tested and confirmed to be in accordance with the system design and consistent with the assumptions of the fuel building FHA [fuel handling accident] analysis (as revised). As such, the proposed change will not alter or prevent the capability of structures, systems, and components (SSCs) to perform their intended functions for mitigating the consequences of an accident and meeting applicable acceptance limits.

   The proposed change will not affect the source term used in evaluating the radiological consequences of a fuel handling accident in the fuel building. However, the Fuel Building Ventilation Exhaust ESF response time has been increased to 90 seconds in recognition of the total delay times involved in the generation of a fuel building ventilation isolation signal (FBVIS) and the times required for actuated components to change state to their required safety configurations. Consequently, the fuel handling accident radiological consequences as reported in FSAR [Final Safety Analysis Report] Table 15.7-8 have increased. However, the increases are much less than the upper limit of "minimal" as defined pursuant to 10 CFR 50.59(c)(2)(iii) and NEI [Nuclear Energy Institute]

96-07 Revision 1 ["Guidelines for 10 CFR 50.59 Implementation," November 2000]. Therefore, there is no significant increase in the calculated consequences of a postulated design basis fuel handling accident in the fuel building. The applicable radiological dose criteria of 10 CFR 100.11, 10 CFR 50 Appendix A General Design Criterion 19, and SRP [NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water-Reactor] Edition"] 15.7.4 will continue to be met. New SR 3.3.8.6 is added to ensure system performance consistent with the accident analyses and associated dose calculations (as revised).

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   With respect to any new or different kind of accident, there are no proposed design changes nor are there any changes in the method by which any safety-related plant SSC performs its specified safety function. The proposed change will not affect the normal method of plant operation or change any operating parameters. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures will be introduced as a result of this amendment.

   The proposed amendment will not alter the design or performance of the 7300 Process Protection System, Nuclear Instrumentation System, Solid State Protection System, BOP ESFAS, MSFIS [Main Steam and Feed Isolation System], or LSELS [Load Shedding and Emergency Load Sequencing] used in the plant protection systems.

   The proposed change does not, therefore, create the possibility of a new or different accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   There will be no effect on those plant systems necessary to assure the accomplishment of protection functions associated with reactor operation or the reactor coolant system. There will be no impact on the overpower limit, departure from nucleate boiling ratio (DNBR) limits, heat flux hot channel factor (F_Q), nuclear enthalpy rise hot channel factor ($F\Delta H$), loss of coolant accident peak cladding temperature (LOCA PCT), peak local power density, or any other limit and associated margin of safety.

> Required shutdown margins in the COLR [Core Operating Limits Report] will not be changed.
>
> The proposed change does not eliminate any surveillances or alter the frequency of surveillances required by the Technical Specifications. The proposed change would add a new Technical Specification Surveillance Requirement for assuring the satisfactory performance of the fuel building ventilation exhaust ESF function in response to a[n] FBVIS. The accident analysis for a fuel handling accident in the fuel building was re-performed to support the proposed Fuel Building Ventilation Exhaust ESF response time, and this reanalysis demonstrated that the acceptance criteria continue to be met with only a slight increase in radiological consequences (i.e., less than one percent).
>
> Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The licensee's request in its letter dated October 27, 2011, for relocating the specified frequency of new SR 3.3.8.6 to the licensee-controlled Surveillance Frequency Control Program is the type of SR whose relocation to the licensee's Surveillance Frequency Control Program was previously approved in Amendment No. 202, dated July 29, 2011 (ADAMS Accession No. ML111661877). That amendment was approved based on the determination that the proposed change did not involve a significant increase in probability or consequences of accidents previously evaluated, did not create the possibility of creating a new or different kind of accident, and would not result in a significant reduction of margin of safety. The no significant hazards consideration was published in the *Federal Register* on January 11, 2011 (76 FR 1649).

In reviewing the licensee's proposed addition of the specified frequency to the licensee's Surveillance Frequency Control Program, the NRC staff determined that the surveillance frequencies are not an initiator of any accidents previously evaluated. Consequently, they do not result in any increase of probability of previously evaluated accidents. The systems and components required to be operable by Technical Specifications will continue to assure that the relocation of the specified frequency in the licensee's Surveillance Frequency Control Program will not significantly increase the consequences of previously evaluated accidents. Therefore, the NRC staff concludes that the proposed change does not involve a significant increase in probability or consequences of an accident previously evaluated.

The change does not involve any alteration of the plant, or change in the methods of operation. The change does not impose any new or different kind of requirements. The change is consistent with the safety analysis assumptions. Therefore, the NRC staff concludes that the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The design, operation, testing methods, and acceptance criteria for systems, structures, and components specified in the applicable requirements will continue to be met as described in the plant's licensing basis. In addition, to evaluate the change in the relocated surveillance frequencies, the licensee performed a probabilistic risk evaluation using the guidance in NRC approved NEI 04-10 Revision 1, that was consistent with the NRC staff's guidance in Regulatory Guide 1.177. The licensee found that the proposed change would not significantly reduce the margin of safety. Based on the evaluation, the NRC staff concludes that the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis, as discussed above, and, based on its review, the staff concludes that the amendment meets the three criteria of 10 CFR 50.92(c). Therefore, the NRC staff has made a final determination that the amendment involves no significant hazards consideration.

5.0 STATE CONSULTATION

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

6.0 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 made a final finding that the amendment involves no significant hazards consideration. 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.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) the amendment does not (a) involve a significant increase in the probability or consequences of an accident previously evaluated; or (b) create the possibility of a new or different kind of accident from any accident previously evaluated; or (c) involve a significant reduction in a margin of safety; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (3) such activities will be conducted in compliance with the Commission's regulations; and (4) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: L. Benton
M. Thadani

Date: December 30, 2011

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/

Mohan C. Thadani, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosures:
1. Amendment No. 204 to NPF-30
2. Safety Evaluation

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OFFICE	NRR/LPL4/PM	NRR/LPL4/LA	NRR/DRA/AADB/BC	NRR/DSS/STSB/BC
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DATE	12/15/11	12/14/11	10/7/11	12/22/11
OFFICE	NRR/DE/EICB/BC	OGC NLO	NRR/LPL4/BC	NRR/LPL4/PM
NAME	GWilson RStattel for	STurk	MMarkley (LWilkins for)	MThadani
DATE	12/16/11	12/30/11	12/30/11	12/30/11

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