AmerenUE
Callaway Plant

PO Box 620
Fulton, MO 65251

October 27, 2004

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Stop P1-137
Washington, DC 20555-0001

Ladies and Gentlemen: ULNRC-05070

DOCKET NUMBER 50-483
CALLAWAY PLANT
UNION ELECTRIC COMPANY
DELETION OF OPERATING LICENSE SECTION 2.F,
DELETION OR REVISION OF LICENSE CONDITIONS, AND REVISION
TO TECHNICAL SPECIFICATION TABLES 5.5.9-2 AND 5.5.9-3

Pursuant to 10 CFR 50.90, AmerenUE, requests an amendment to the Facility Operating License No. NPF-30 for Callaway Plant.

This amendment application proposes to delete Section 2.F of the Operating License which requires reporting any violations of the requirements contained in Section 2.C of the license (except for Section 2.C(2)). The proposed change to delete Section 2.F is administrative in nature and will eliminate notification and reporting requirements from the Facility Operating License which are adequately governed by the reporting requirements of 10 CFR 50.72 and 10 CFR 50.73.

Additionally, License Conditions 2.C.(1), 2.C.(3), 2.C.(4), 2.C.(5), 2.C.(6), 2.C.(7), 2.C.(8), 2.C.(9), 2.C.(10), 2.C.(11), 2.C.(12), 2.C.(13) and 2.C.(14) are being deleted or revised as these conditions are obsolete or adequately described elsewhere.

Finally, this application proposes to revise Technical Specification Table 5.5.9-2, "Steam Generator Tube Inspection," and Table 5.5.9-3, "Steam Generator Repaired Tube Inspection", to delete the requirement to notify the NRC pursuant to 10 CFR 50.72(b)(2) if the steam generator tube inspection results in a C-3 classification. On October 25, 2000, the NRC issued a final rule that amended the event reporting requirements for nuclear power reactors to reduce or eliminate unnecessary reporting burden associated with events of little or no safety significance. The current reporting requirement in Table 5.5.9-2 is incorrect based on the issuance of this final rule.

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Attachment 1 to this submittal provides the required Affidavit. Attachment 2 provides a detailed description, safety analysis of the proposed changes, and the Callaway determination that the proposed change does not involve a significant hazard consideration. Attachment 3 provides the existing marked-up Operating License pages, Attachment 4 provides the existing TS pages marked-up to show the proposed changes and Attachment 5 provides a clean copy of the proposed Technical Specification pages.

This letter identifies actions committed to by AmerenUE and Callaway Plant in this submittal. Other statements are provided for information purposes and are not considered to be commitments. A summary of the regulatory commitments included in this submittal is provided in Attachment 6.

It has been determined that this amendment application does not involve a significant hazard consideration as determined per 10 CFR 50.92. In addition, pursuant to 10 CFR 51.22(b), no environmental assessment need be prepared in connection with the issuance of this amendment.

AmerenUE requests approval of this proposed License Amendment by October 2005. The approved amendment will be implemented within 60 days of approval.

Pursuant to 10 CFR 50.91(b)(1), AmerenUE is providing the State of Missouri with a copy of this proposed amendment.

If you should have any questions on the above or attached, please contact Dave Shafer at (314) 554-3104 or Dwyla Walker at (314) 554-2126.

Very truly yours,

Keith D. Young

Keith D. Young
Manager, Regulatory Affairs

DJW/jdg

Attachments:
1) Affidavit
2) Evaluation
3) Markup of Existing Operating License pages
4) Markup of Existing Technical Specification pages
5) Clean copy of Technical Specification revised pages
6) Summary of Regulatory Commitments

cc: U.S. Nuclear Regulatory Commission (Original and 1 copy)
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ULNRC-05070
October 27, 2004

STATE OF MISSOURI)
) SS
CALLAWAY COUNTY)

Keith D. Young of lawful age, being first duly sworn upon oath says that he is Manager - Regulatory Affairs, for Union Electric Company; that he has read the foregoing document and knows the content thereof; that he has executed the same for and on behalf of said company with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By Keith D. Young
Keith D. Young
Manager, Regulatory Affairs

SUBSCRIBED and sworn to before me this 27 day of OCTOBER, 2004

LORI L. TWILLMAN
Notary Public - Notary Seal
STATE OF MISSOURI
Callaway County
My Commission Expires: Aug. 3, 2007

Lori L. Twillman

ULNRC-05070

ATTACHMENT 2

EVALUATION

EVALUATION

1.0 INTRODUCTION

This letter is a request to amend Operating License NPF-30 for Callaway Plant.

This proposed License Amendment Request (LAR) is a request pursuant to 10 CFR 50.90 to delete Section 2.F of the Callaway Plant Facility Operating License and to delete License Conditions 2.C.(3), 2.C.(4), 2.C.(6), 2.C.(7), 2.C.(8), 2.C.(9), 2.C.(10), 2.C.(11), 2.C.(12), 2.C.(13) and 2.C.(14) and to revise License Conditions 2.C.(1) and 2.C.(5).

Additionally, this application proposes to revise Technical Specification Table 5.5.9-2, "Steam Generator Tube Inspection," and Table 5.5.9-3, "Steam Generator Repaired Tube Inspection," to delete the requirement to notify the NRC pursuant to 10 CFR 50.72(b)(2) if the steam generator tube inspection results in a C-3 classification.

There are no changes to the Technical Specification Bases or the Callaway updated Final Safety Analysis Report currently anticipated as a result of this LAR.

The proposed changes do not alter design bases or technical requirements.

2.0 DESCRIPTION OF PROPOSED AMENDMENT

The proposed License Amendment would delete Section 2.F of the Callaway Plant Facility Operating License which requires reporting any violations of the requirements contained in Section 2.C of the license (except for Section 2.C.(2)). Additionally, the below License Conditions are being deleted or revised as these conditions are obsolete or adequately described elsewhere.

2.C.(1) Maximum Power Level

2.C.(3) Environmental Qualification (Section 3.11, SSER#3)

2.C.(4) Surveillance of Hafnium Control Rods (Section 4.2.3.1 (10) SER and SSER#2)

2.C.(5) Fire Protection (Section 9.5.1.7 SER and Section 9.5.1.8, SSER#3)

2.C.(6) Qualification of Personnel (Section 13.1.2, SSER#3, Section 18, SSER#1)

2.C.(7) NUREG-0737 Conditions (Section 22, SER)

2.C.(8) Post-Fuel-Loading Intial Test Program (Section 14, SER)

2.C.(9) Inservice Inspection Program (Sections 5.2.4 and 6.6, SER)

2.C.(10) Emergency Planning

2.C.(11) Steam Generator Tube Rupture (Section 15.4.4, SSER#3)

2.C.(12) Low Temperature Overpressure Protection (Section 15, SSER#3)

2.C.(13) LOCA Reanalysis (Section 15.3.7, SSER#3)

2.C.(14) Generic Letter 83-28

Finally, Technical Specification Table 5.5.9-2, "Steam Generator Tube Inspection," and Table 5.5.9-3, "Steam Generator Repaired Tube Inspection," are revised to delete the requirement to notify the NRC pursuant to 10 CFR 50.72(b)(2) if the steam generator tube inspection results in a C-3 classification.

3.0 BACKGROUND

The Callaway Plant Unit 1 Facility Operating License No. NPF-30 was issued on October 18, 1984. The license was issued with conditions containing various requirements to be completed by specified dates or prior to exceeding specified power levels. These activities have been completed and the license conditions are either obsolete or no longer needed.

Except for Section 2.C(2), Operating License Section 2.F, provides for initial notification of any violations of the requirements contained in Section 2.C of the license to be made within 24 hours. Initial notification is made in accordance with the provisions of 10 CFR 50.72 with written follow-up in accordance with the procedures described in 10 CFR 50.73 (b), (c), (d), and (e). The Nuclear Regulatory Commission's requirements for immediate notification with written follow-up requirements (Licensee Event Reports) of events at operating nuclear power reactors are stated in 10 CFR 50.72 (Reference 8.1) and 10 CFR 50.73 (Reference 8.2). Thus, the requirements of Operating License Section 2.F are adequately governed by the requirements of 10 CFR 50.72 and 10 CFR 50.73.

License Conditions 2.C.(3), 2.C.(6), 2.C.(7), 2.C.(8), 2.C.(9), 2.C.(10), 2.C.(11), 2.C.(12), 2.C.(13), and 2.C.(14) are conditions that have been completed and are considered obsolete. License Conditions 2.C.(1) and 2.C.(5) are revised to remove sub-actions that are completed and considered obsolete.

Technical Specification Table 5.5.9-2, "Steam Generator Tube Inspection," and Table 5.5.9-3, "Steam Generator Repaired Tube Inspection", require notification to the NRC pursuant to 10 CFR 50.72(b)(2) if the results of the steam generator tube inspection identify more than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective (Category C-3). On October 25, 2000, the NRC issued a final rule (Reference 8.3) that amended the event reporting requirements for nuclear power reactors to reduce or eliminate the unnecessary reporting burden associated with events of little or no safety significance. Prior to the final rule, 10 CFR 50.72(b)(2)(i) required a four hour report for any event found while the reactor is shutdown, that , had it been found while the reactor was in operation, would have resulted in the nuclear power plant, including its principle safety barriers, being seriously degraded or being in an unanalyzed condition that significantly compromises plant safety. The final rule revised section (b)(2) of the regulation to only apply to initiation of a plant shutdown required by the Technical Specifications. Under the final rule, 10 CFR 50.72(b)(3)(ii) specifies an eight hour reporting requirement for a principle safety barrier being significantly degraded or the plant being in an unanalyzed condition. NUREG-1022, Revision 2, "Event Reporting Guidelines 10 CFR 50.72 and 50.73," Section 3.2.4 identifies serious steam generator tube degradation as an example of a reportable event or condition under 10 CFR 50.72(b)(3)(ii).

4.0 TECHNICAL ANALYSIS

<u>Deletion of Operating License Section 2.F</u>

This amendment application proposes to delete Section 2.F of the Operating License which requires reporting any violations of the requirements contained in Section 2.C of the license (except for Section 2.C(2)). Operating License Section 2.C lists conditions regarding Maximum Power Level, Technical Specifications and Environmental Protection Plan, Fire Protection and other license conditions. Except for Section 2.C.(2), Operating License Section 2.F requires that AmerenUE (Union Electric (UE)) notify the NRC of violations of the requirements in Section 2.C within 24 hours in accordance with the provisions of 10 CFR 50.72. The initial 24 hour notification is followed with written notification in accordance with the procedures described in 10 CFR 50.73 (b), (c), (d), and (e). These requirements of Section 2.F of the license are adequately addressed by the reporting requirements identified in 10 CFR 50.72 and 10 CFR 50.73. As such, Operating License Section 2.F is not required.

Section 2.F of the license can be deleted because any deviations from the conditions regarding Maximum Power Level, Technical Specifications, Fire Protection, and Additional Conditions are adequately addressed by the requirements of 10 CFR 50.72 and 10 CFR 50.73.

Revision or Deletion of Specific License Conditions, Section 2.C

The following provides the justification for the deletion or revision of specific License Conditions in Section 2.C of the Operating License.

Current 2.C.(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 and in Attachment 1 to this license. The preoperational tests, startup tests and other items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license."

Proposed Revision 2.C.(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."

Justification for Revision to 2.C.(1)

2.C.(1) is revised to delete reference to Attachment 1 and Attachment 1 is deleted from the Operating License. Attachment 1 required (A) the implementation of Radiation/Chemical Technical refresher training within six months following fuel load and (B) the installation of a permanent area monitor on the manipulator crane prior to entering Mode 6 (refueling mode).

Technical Specification 5.3 and FSAR Chapter 13 provide staffing and qualification requirements for plant personnel. Since Callaway has been in operation since 1984, the requirements of Attachment 1, Item A, have been completed and are, therefore, proposed to be deleted.

By letter, ULNRC-01254, dated February 7, 1986, Callaway informed the NRC of its plan to install a permanent area radiation monitor on the manipulator crane to satisfy Attachment 1, Item B of the Callaway Plant Operating License. Item B of Attachment 1 was completed and is, therefore, proposed to be deleted.

2.C.(3) Environmental Qualification (Section 3.11, SSER#3)

" (a) Prior to November 30, 1985, UE shall environmentally qualify all electrical equipment according to the provisions of 10 CFR 50.49.

(b) Prior to restart following the first refueling outage, UE shall have qualified the reactor vessel level instrumentation system high volume sensor."

Justification for Deletion of 2.C.(3)

Letter SLNRC 85-24, dated November 29, 1985, notified the NRC that the electrical equipment required to be qualified under 10 CFR 50.49 has been evaluated and determined to be qualified to the provisions of 10 CFR 50.49. Letter SLNRC 86-002, dated January 17, 1986 provided a final report on the independent review of environmental qualification programs. The NRC responded by letter dated March 17, 1986, indicating that the staff finds that License Condition 2.C.(3)(a) has been fulfilled.

FSAR Section 3.11(B), "Environmental Design of Mechanical and Electrical Equipment," provides information on the environmental conditions and design bases for which the mechanical, instrumentation, and electrical portions of the engineered safety features, the reactor protection systems, and other safety-related systems are designed to ensure acceptable performance during normal and design basis accident environmental conditions. As such, changes to the environmental qualification of the equipment specified in the FSAR would be reviewed in accordance with 10 CFR 50.59.

License Condition 2.C.(3)(b) is complete based on AmerenUE's conformance with Regulatory Guide 1.97, Revision 2 and based on a letter to the NRC, ULNRC-01287, dated April 4, 1986, providing notification that the Reactor Vessel Level Instrumentation System (RVLIS) High Volume Sensor, as well as, all other RVLIS components were evaluated and determined qualified, therefore fulfilling License Condition 2.C.(3)(b). In a letter dated April 10, 1985 the NRC staff found AmerenUE's conformance to Regulatory Guide 1.97, Revision 2 acceptable.

The requirements of this license condition have been met and are, therefore, proposed to be deleted.

2.C.(4) Surveillance of Hafnium Control Rods (Section 4.2.3.1(10), SER an SSER#2)

"UE shall perform a visual inspection of a sample of hafnium control rods during one of the first five refueling outages. A summary of the results of these inspections shall be submitted to the NRC."

Justification for Deletion

Letter ULNRC-02116, dated November 30, 1989 informed the NRC that an inspection of all hafnium rod cluster control assemblies (RCCAs) was performed during Refuel-2 in the Fall of 1987. However, during Callaway Cycle-3, swelling of hafnium RCCAs was identified at other plants with Westinghouse supplied nuclear steam supply systems. Although swelling was not identified at Callaway during the Refuel-2 eddy current inspection, Union Electric made a decision to replace all hafnium RCCAs with Silver-Indium-Cadmium RCCAs during Callaway Refuel-3. Replacement was completed in the Spring of 1989. This information closed Callaway License Condition 2.C.(4). The requirement of this license condition was closed and is, therefore, proposed to be deleted.

<u>2.C.(5) Fire Protection (Section 9.5.1.7 SER and Section 9.5.1.8, SSER #3)</u>

"(a) *Within 60 days of acquisition on the 100% power data for thermal and dynamic testing, UE shall have operable the Halon systems in the north electrical penetration room (fire area A-18).*

(b) *Prior to restart following the first extended outage of known duration greater than two weeks occurring after February 15, 1985 or prior to restart following the first refueling outage which ever occurs first, UE shall have completed the installation of the five new isolation switches and modification to the four existing isolation switches identified in the August 23, 1984 SNUPPS letter."*

<u>Proposed Revision 2.C.(5) Fire Protection (Section 9.5.1.7 SER and Section 9.5.1.8, SSER #3)</u>

"(a) *Deleted per Amendment No. xxx*

(b) *Deleted per Amendment No. xxx"*

<u>Justification for Deletion of Items (a) and (b)</u>

(a) Letter ULNRC-01045, dated February 21, 1985 to the NRC addressed the License Condition 2.C.(5)(a) concern. The license condition required the Halon system be operable within 60 days of acquisition of the 100% power data for thermal and dynamic testing. ULNRC-01045 informed the NRC that the Halon concentration test for the north electrical penetration room was completed and the results approved in February 1985. Callaway startup testing was completed in December 1984. The requirements of the license condition were completed and are, therefore, proposed to be deleted.

(b) Letter ULNRC-01259, dated February 20, 1986, addressed License Condition 2.C.(5)(b). The license condition concerned the installation of five new isolation switches and the modification of four existing isolation switches. The switches enhanced the ability to shut the plant down with a postulated fire in the Main Control Room. ULNRC-01259 informed the NRC that the switch installations and modifications were completed in compliance with the license condition. Because the requirements were completed, the license condition is proposed to be deleted.

2.C.(6) Qualification of Personnel (Section 13.1.2, SSER#3, Section 18, SSER#1)

"(a) *UE shall have on each shift operators who meet the requirements described in Attachment 2.*

(b) *UE shall have a senior individual with previous operating experience on a commercial PWR assigned to assist the Plant Manager as an advisor during the startup test program and for one year following full power operation."*

Justification for Deletion

(a) Attachment 2 to the Operating License states, in part:

"The Operating Corporation shall have a licensed senior operator on each shift who has had at least six months of hot operating experience on a same type plant, including at least six weeks at power levels greater than 20% of full power, and who has had startup and shutdown experience. .... The NRC shall be notified at least 30 days prior to the date the Operating Corporation proposes to release the advisors from further service."

Letter ULNRC-01081, dated April 15, 1985, notified the NRC of the intent to release the shift advisors from further service on May 31, 1985. This license condition concerned having additional shift advisors on shift during initial plant criticality and operation. Technical Specification 5.3 and FSAR Chapter 13 provides the shift staffing and qualification requirements for operations personnel. Since Callaway has been in operation since 1984, the requirements of this license condition have been completed and are, therefore, proposed to be deleted.

(b) Letter ULNRC-0823, dated May 16, 1984 transmitted to the NRC the resume for the Advisor to the Plant Manager. The Callaway Startup Test Program was completed on December 19, 1984 and Callaway plant has been in operation since 1984. The requirement for an advisor to assist the Plant Manager was met, is now obsolete and is, therefore proposed to be deleted.

2.C.(7) NUREG-0737 Conditions (Section 22, SER)

"UE shall complete the following conditions to the satisfaction of the NRC. These conditions reference the appropriate items in Section 22.2, "TMI Action Plan Requirements for Applicants for Operating Licenses," in the Safety Evaluation Report and Supplements 1, 2, 3 and 4 NUREG-0830.

(a) *Detailed Control Room Design Review (I.D.1, SSER #4)*

Prior to May 1, 1985, UE shall submit for review and approval by the

NRC staff, the results of the function and task analysis. For those Human Engineering Discrepancies (HEDs) identified by this analysis that require correction, the submittal shall include the proposed correction and implementation schedule: and for those HEDs for which no planned correction is proposed, a basis for that determination shall be documented.

(b) *<u>Emergency Response Capabilities (Generic Letter 82-33, Supplement 1 to NUREG-0737)</u>*

Prior to restart following the first refueling outage, UE shall have a fully functional Technical Support Center and Emergency Operations Facility and a fully operable Emergency Response Facilities Information System (ERFIS).

(c) *<u>Regulatory Guide 1.97 (Section 7.5.2.3, SSER #3)</u>*

Prior to restart following the first refueling outage, UE shall have installed and operable the following instrumentation.

1) *Source range instrumentation qualified to post-accident conditions*

2) *Reactor vessel water level instrumentation*

3) *Subcooling monitors*

4) *Radiation monitors for releases from steam generator safety/relief valves or atmospheric dump valves, and*

5) *Auxiliary feedwater pump turbine exhaust monitor"*

<u>Justification for Deletion</u>

(a) <u>Detailed Control Room Design Review (I.D.1, SSER #4)</u>

Letter SLNRC 85-12, dated April 26, 1985, transmitted the Final Report for the Task Analysis for SNUPPS Detailed Control Room Design Review (DCRDR) to the NRC. With this transmittal, all human factors issues addressed for Callaway License Condition 2.C.(7)(a) were considered closed.

NRC letter dated August 27, 1985, provided the results of the NRC review of Callaway submittals regarding the detailed control room design review and the related function and task analysis. The letter stated that the NRC staff found the requirements of License Condition 2.C.(7)(a) were satisfied with respect to the

DCRDR, but the task analysis results remained under review as relating to the upgrade of emergency operating procedures.

NRC letter dated March 4, 1987 provided the completion of NRC reviews relating to Callaway submittals of the DCRDR and task analysis and its application in upgrading Callaway Emergency Operating Procedures. Based on the review, the NRC concluded that procedural changes adopted at Callaway provide adequate guidance and information to the operator to cope with emergencies and achieve the pertinent objectives of the Westinghouse Emergency Response Guidelines. The letter concluded that the staff had completed its review and that License Condition 2.C.(7)(a) had been satisfied.

(b) <u>Emergency Response Capabilities (Generic Letter 82-33, Supplement 1 to NUREG-0737)</u>

Letter ULNRC-01288, dated April 8, 1986 confirmed to the NRC the Callaway commitment to complete the requirements of License Condition 2.C.(7)(b). In order to assess the operability of the emergency response facilities for Callaway, the letter describes the Emergency Response Facility Assessment Program that was established. Based on the program, the Technical Support Center, the Emergency Operations Facility, and information systems were determined to be fully functional.

(c) <u>Regulatory Guide 1.97 (Section 7.5.2.3, SSER #3)</u>

Letter ULNRC-01289, dated April 7, 1986 confirmed to the NRC that the Requirements of License Condition 2.C (7)(c) are completed. The letter confirmed that certain Regulatory Guide 1.97, Post Accident Monitoring (PAM), instruments were ready to be declared operable at the time of startup for Callaway Cycle 2.

Based on the above information, the requirements of this license condition are complete and are, therefore, proposed to be deleted.

<u>2.C.(8) Post-Fuel-Loading Initial Test Program (Section 14, SER)</u>

"UE shall conduct the post-fuel-loading initial test program described in Chapter 14 of the FSAR, as amended, without making any major modifications unless such modifications have prior NRC approval. Major modifications are defined as:

*(a) elimination of any safety-related test**

(b) modification of objectives, test method, or acceptance criteria for any safety-related test

(c) *performance of any safety-related test at a power level different from that stated in the FSAR by more than 5 percent of rated power*

(d) *failure to satisfactorily complete the entire initial start-up test program by the time core burnup equals 120 effective full power days*

(e) *deviation from initial test program administrative procedures or quality assurance controls described in the FSAR*

(f) *delays in test program in excess of 30 days (14 days if power level exceeds 50 percent), concurrent with power operation. If continued power operation is desired during a delay, the licensee shall provide justification that adequate testing has been performed and evaluated to demonstrate that the facility can be operated at the planned power level with reasonable assurance that the health and safety of the public will not be endangered."*

Justification for Deletion

This license condition is obsolete since the Initial Test Program is complete and the unit is currently in operation in Cycle 14.

2.C.(9) Inservice Inspection Program (Sections 5.2.4 and 6.6, SER)

"Within nine months of the date of this license, UE shall submit for staff review and approval, the inservice inspection program which conforms to the ASME Code in effect 12 months prior to the date of issuance of this license."

Justification for Deletion

Letter ULNRC-01143, dated July 16, 1985, as supplemented by letters ULNRC-01399, dated November 7, 1986, ULNRC-01457, dated March 3, 1987, ULNRC-01481, dated April 2, 1987, and ULNRC-01753, dated April 11, 1988 transmitted the Callaway Inservice Inspection Program Plan (including relief requests) for NRC review and approval. In a letter from the NRC dated December 14, 1988, the staff concluded that the Callaway Inservice Inspection Program Plan, with additional information provided and the specific written reliefs, met the requirements of 10 CFR 50.55a and Technical Specifications and was acceptable. The requirements of this license condition are considered complete and are, therefore, proposed to be deleted.

2.C.(10) Emergency Planning

"In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR

Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply."

Justification for Deletion

Facility Operating License No. NPF-30 for Callaway Plant, Section 2.C, states in part:

"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 1 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; ....."

10 CFR Chapter 1 includes Part 50, "Domestic licensing of production and utilization facilities," and as such, indicates that 10 CFR Section 50.54(s) is currently applicable to Callaway Plant. This license condition is duplicative of requirements that are currently applicable and enforceable. Deletion of this license condition would not result in a reduction of requirements and is, therefore, proposed to be deleted.

2.C.(11) Steam Generator Tube Rupture (Section 15.4.4, SSER #3)

"Prior to restart following the first refueling outage, UE shall submit for NRC review and approval an analysis which demonstrates that the steam generator single-tube rupture (SGTR) analysis presented in the FSAR is the most severe case with respect to the release of fission products and calculated doses. Consistent with the analytical assumptions, the licensee shall propose all necessary changes to Appendix A to this license."

Justification for Deletion

Letter SLNRC 86-001, dated January 8, 1986, transmitted a report which demonstrated that the SGTR analysis presented in the FSAR is the most severe case with respect to release of fission products and calculated doses. Letter ULNRC-01238, dated January 14, 1986, submitted a license amendment request to incorporate a limiting condition for operation and surveillance requirements into the technical specifications for the steam generator atmospheric relief valves to assure the availability of mitigating equipment assumed in the SGTR analysis. Amendment No. 45, dated May 16, 1989, approved the changes to the technical specifications. NRC letter dated August 6, 1990 provided a Safety Evaluation associated with the SGTR analysis. Subsequently in March 2004, Amendment No. 159, dated March 11, 2004, approved an update to the SGTR analysis. The results of the SGTR analysis are described in Chapter 15 of the FSAR. As such, changes to the analysis described in the FSAR are reviewed in accordance with 10 CFR 50.59. The requirements of this license condition have been met and are, therefore, proposed to be deleted.

2.C.(12) Low Temperature Overpressure Protection (Section 15, SSER #3)

"By January 1, 1985, UE shall submit for NRC review and approval a description of equipment modifications to the residual heat removal system (RHRS) suction isolation valves and to closure circuitry which conform to the applicable staff requirements (SRP 5.2.2). Within one year of receiving NRC approval of the modifications, UE shall have the approved modifications installed. Alternately, by January 1, 1985, UE shall provide acceptable justification for reliance on administrative means alone to meet the staff's RHRS isolation requirements, or otherwise, propose changes to Appendix A to this license which remove reliance on the RHRS as a means of low temperature overpressure protection."

Justification for Deletion

Letter ULNRC-01003, dated December 28, 1984 provided AmerenUE's response to License Condition C(12) regarding low temperature overpressure protection. The license condition concerned the use of Residual Heat Removal (RHR) suction relief valves for low temperature overpressure protection (LTOP). The letter to NRC described plant modifications to be completed within one year of receiving NRC approval. The modifications included adding an alarm circuit to reactor coolant system (RCS) and RHR system valves to provide assurance that the RHR system properly isolates from the RCS when the plant returns to operating pressure following use of the RHR relief valves for LTOP. In a letter dated July 30, 1985, the NRC approved the modification and stated that the submittal requirement of License Condition 2.C.(12) had been met. After supplemental transmittals to NRC, due to minor design changes, Letter ULNRC-01306, dated May 7, 1986, informed the NRC that the modification was completed. The requirements of this license condition were met and are, therefore, proposed to be deleted.

2.C.(13) LOCA Reanalysis (Section 15, SSER #3)

"Prior to restart following the first refueling outage, UE shall submit for NRC review and approval a reanalysis for the worst large break LOCA using an approved ECCS evaluation model. At this time that model is the 1981 Westinghouse model. A modified version of the 1981 model which includes the BART computer code may be used."

Justification for Deletion

Letter ULNRC-01207, dated November 15, 1985 transmitted to NRC an application for a reload license amendment for Callaway Cycle 2, which included a large break Loss of Coolant Accident (LOCA) analysis based on the Westinghouse BASH model. In a subsequent letter to NRC, ULNRC-01247, dated January 28, 1986, AmerenUE transmitted a new large break LOCA analysis based on the Westinghouse BART model, which was the most recent Westinghouse, large break LOCA model approved by the

NRC. Amendment No. 15, dated April 8, 1986, approved the changes to the Technical Specifications and provided a Safety Evaluation Report (SER). As part of the SER, the NRC staff concluded that the license condition requiring reanalysis of the worst large break LOCA was met. The requirements of this license condition were met and are, therefore, proposed to be deleted.

2.C.(14) Generic Letter 83-28

"UE shall submit responses to and implement the requirements of Generic Letter 83-28 on a schedule which is consistent with that given in its May 21, 1984 letter."

Justification for Deletion

The table below provides a summary of the responses and NRC review of the requirements of Generic Letter 83-28.

ITEM	UNION ELECTRIC/NRC Response
Item 1.1 – Post Trip Review Program Description and Procedure	NRC letter dated May 7, 1985, indicates that the Post-Trip Review Program and Procedures for Callaway Plant are acceptable.
Item 1.2 – Post Trip Review Data and Information Capability	NRC letter dated July 24, 1986, indicates that the Post-Trip Review Data and Information Capability for Callaway Plant are acceptable.
Item 2.1.1 – Equipment Classification (Reactor Trip System Components) Item 2.1.2 – Vendor Interface (Reactor Trip System Components)	NRC letter dated July 21, 1986, indicates that the NRC had completed its review for Callaway Plant and found the equipment classification program at Callaway acceptable. NRC letter dated December 22, 1986, indicates that the NRC staff completed its review of the equipment classification and vendor interface for the Callaway reactor trip system and that these items are acceptable for Callaway. This letter completed NRC review for Item 2.1 of Generic Letter 83-28.
Item 2.2.1 – Equipment Classification (Programs for all Safety-Related Components)	NRC letter dated April 10, 1987, indicates that the NRC staff completed its review of licensee submittals ULNRC-00687, dated November 18, 1983; ULNRC-00763, dated March 12, 1984; ULNRC-00829, dated May 21, 1984; ULNRC-01002, dated December 27, 1984; ULNRC-01098, dated May 17, 1985, and ULNRC-01435, dated

ITEM	UNION ELECTRIC/NRC Response
	January 27, 1987, and found them to be acceptable.
Item 2.2.2 – Vendor Interface (Programs for all Safety-Related Components)	NRC letter dated April 10, 1987, indicates that the NRC staff completed its review of licensee submittals ULNRC-00687, dated November 18, 1983; ULNRC-00763, dated March 12, 1984; ULNRC-00829, dated May 21, 1984; ULNRC-01002, dated December 27, 1984; ULNRC-01098, dated May 17, 1985, and ULNRC-01435, dated January 27, 1987, and found them to be acceptable. Generic Letter 90-03 was issued on March 20, 1990, to clarify the staff position in Part 2 of Item 2.2 of Generic Letter 83-28. Letter ULNRC-02294, dated September 21, 1990, responded to Generic Letter 90-03. NRC letter dated December 3, 1990, indicated that the staff reviewed the response and found it acceptable. Letter ULNRC-02354, dated January 10, 1991, confirmed to NRC that the enhanced vendor interface program commitments for Generic Letter 90-03 had been implemented.
Items 3.1.1 and 3.1.2 – Post Maintenance Testing (Reactor Trip System Components)	Letters ULNRC-00687, dated November 18, 1983 and ULNRC-00763, dated March 12, 1984 addressed Generic Letter 83-28 items concerning post-maintenance testing and reactor trip system reliability. NRC letters dated June 25, 1985 and July 3, 1985 determined that the Callaway programs outlined were acceptable.
Items 3.1.3 – Post Maintenance Testing – Changes to Test Requirements (Reactor Trip System Components)	Letters ULNRC-00687, dated November 18, 1983 and ULNRC-01356, dated August 12, 1986 addressed Generic Letter 83-28 items concerning post-maintenance testing and reactor trip system reliability. NRC letter dated October 7, 1986, determined that the Callaway program for post maintenance testing was acceptable.

ITEM	UNION ELECTRIC/NRC Response
Items 3.2.1 and 3.2.2 – Post Maintenance Testing (all other safety-related components)	Letters ULNRC-00687, dated November 18, 1983 and ULNRC-00763, dated March 12, 1984 addressed Generic Letter 83-28 items concerning post-maintenance testing and reactor trip system reliability. NRC letters dated June 25, 1985 and July 3, 1985 determined that the Callaway programs outlined were acceptable.
Item 3.2.3 – Post Maintenance Testing – Changes to Test Requirements (all other safety-related components)	Letters ULNRC-00687, dated November 18, 1983 and ULNRC-01356, dated August 12, 1986 addressed Generic Letter 83-28 items concerning post-maintenance testing and reactor trip system reliability. NRC letter dated October 7, 1986, determined that the Callaway program for post maintenance testing was acceptable.
Item 4.1 – Reactor Trip System Reliability (Vendor-Related Modifications)	Letters ULNRC-00687, dated November 18, 1983 and ULNRC-00763, dated March 12, 1984 addressed Generic Letter 83-28 items concerning post-maintenance testing and reactor trip system reliability. NRC letters dated June 25, 1985 and July 3, 1985 determined that the Callaway programs outlined were acceptable.
Items 4.2.1 and 4.2.2 – Reactor Trip System Reliability – Maintenance and Testing	Letters ULNRC-00687, dated November 18, 1983; ULNRC-00763, dated March 12, 1984; ULNRC-0829, dated May 21, 1984; and ULNRC-01002, dated December 27, 1984 addressed Generic Letter 83-28 items concerning maintenance and trending programs for reactor trip breakers. NRC letter dated October 28, 1985, determined that the Callaway programs outlined were acceptable.
Items 4.2.3 and 4.2.4 – Reactor Trip System Reliability – Life Cycle Testing of Reactor Trip Breakers	Letters ULNRC-01678, dated November 13, 1987 and ULNRC-02670, dated July 23, 1992 addressed Generic Letter 83-28 items concerning life cycle testing of reactor trip breakers. NRC letter dated October 7, 1992 determined that the required actions associated with items 4.2.3 and 4.2.4 were no longer needed and licensee actions in response to items 4.2.3 and 4.2.4 were no longer necessary.

ITEM	UNION ELECTRIC/NRC Response
Item 4.3 – Reactor Trip System Reliability (Automatic Actuation of Shunt Trip Attachment for Westinghouse Plants)	Letters ULNRC-00763, dated March 12, 1984 and ULNRC-01002, dated December 27, 1984 initially responded to this Generic Letter 83-28 item. NRC letter dated July 18, 1984 provided NRC's review and positions concerning the Callaway design for automatic shunt trip for scram breakers. In response to the staff positions, Union Electric submitted ULNRC-01031, dated January 29, 1985 to revise the technical specifications. Subsequently, NRC Generic Letter 85-09 was issued on May 23, 1985. Letter ULNRC-01240, dated January 9, 1986 superceded the submittal of ULNRC-01031. Letter ULNRC-01240 submitted proposed technical specifications per the guidance in Generic Letter 85-09, "Technical Specifications for Generic Letter 83-28, Item 4.3." NRC approved the proposed technical specifications as Amendment 19, dated March 3, 1987. Also in accordance with the Generic Letter 85-09 guidance, ULNRC-01602, dated September 10, 1987, submitted proposed technical specifications addressing the reactor trip bypass breakers. NRC approved the license amendment request as Amendment 34, dated February 17, 1988.
Item 4.4 – Reactor Trip System Reliability (Improvements in Maintenance and Test Procedures for B&W Plants)	Not applicable to Callaway.
Item 4.5.1 – Reactor Trip System Reliability (System Functional Testing)	Letters ULNRC-00687, dated November 18, 1983 and ULNRC-00763, dated March 12, 1984 addressed Generic Letter 83-28 items concerning post-maintenance testing and reactor trip system reliability. NRC letters dated June 25, 1985 and July 3, 1985 determined that the Callaway programs outlined were acceptable.

ITEM	UNION ELECTRIC/NRC Response
Item 4.5.2 and 4.5.3 – Reactor Trip System Reliability (On-line System Functional Testing)	Letter ULNRC-00687, dated November 18, 1983, originally responded these Generic Letter 83-28 items. Union Electric endorsed the Westinghouse Owners Group Report WCAP-10271 as being applicable to Callaway. In addition, letter ULNRC-01174, dated October 16, 1985, submitted proposed technical specifications based upon implementing changes approved generically from NRC's review of WCAP-10271 and Supplement 1. NRC approved the proposed technical specifications as Amendment 17 dated September 8, 1986. NRC letter dated June 12, 1989, indicated that the staff completed the Callaway response and the Westinghouse Owners Group Report and concluded the Item 4.5.3 to be complete for Callaway. In addition, the staff noted that Callaway Plant is designed to permit on-line functional testing of the reactor trip system, including testing of the diverse trip features of the reactor trip breakers (the undervoltage and shunt trip attachments) and Generic Letter 83-28, Item 4.5.2 is not applicable.

Revision to Technical Specification Table 5.5.9-2 and Table 5.5.9-3

Technical Specification Table 5.5.9-2, "Steam Generator Tube Inspection," and Table 5.5.9-3, "Steam Generator Repaired Tube Inspection", require notification to the NRC pursuant to 10 CFR 50.72(b)(2) if the results of the steam generator tube inspection identify more than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective (Category C-3). On October 25, 2000, the NRC issued a final rule (Reference 8.3) that amended the event reporting requirements for nuclear power reactors to reduce or eliminate the unnecessary reporting burden associated with events of little or no safety significance. Prior to the final rule, 10 CFR 50.72(b)(2)(i) required a four hour report for any event found while the reactor is shutdown, that , had it been found while the reactor was in operation, would have resulted in the nuclear power plant, including its principle safety barriers, being seriously degraded or being in an unanalyzed condition that significantly compromises plant safety. The final rule revised section (b)(2) of the regulation to only apply to initiation of a plant shutdown required by the Technical Specifications. Under the final rule, 10 CFR 50.72(b)(3)(ii) specifies an eight hour reporting requirement for a principle safety barrier being significantly degraded or the plant being in an unanalyzed condition. NUREG-1022, Revision 2, "Event Reporting Guidelines 10 CFR 50.72 and 50.73," Section 3.2.4

identifies serious steam generator tube degradation as an example of a reportable event or condition under 10 CFR 50.72(b)(3)(ii). The current reporting requirement in Table 5.5.9-2 and Table 5.5.9-3 is incorrect based on the issuance of the final rule. Table 5.5.9-2 and Table 5.5.9-3 are revised to delete the reporting requirement. Deletion of this requirement from the Technical Specifications does not change the requirement to report results that satisfy the criteria of 10 CFR 50.72(b)(3). Additionally, Technical Specification 5.6.10c still requires reporting the results of steam generator tube inspections, which fall into Category C-3, in a report within 30 days and prior to resumption of plant operation.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

This amendment application proposes to delete Section 2.F of the Operating License which requires reporting any violations of the requirements contained in Section 2.C of the license (except for Section 2.C(2)). The proposed change to delete Section 2.F is administrative in nature and will eliminate notification and reporting requirements from the Facility Operating License which are adequately governed by the reporting requirements of 10 CFR 50.72 and 10 CFR 50.73.

Additionally, License Conditions 2.C.(1), 2.C.(3), 2.C.(4), 2.C.(5), 2.C.(6), 2.C.(7), 2.C.(8), 2.C.(9), 2.C.(10), 2.C.(11), 2.C.(12), 2.C.(13) and 2.C.(14) are being deleted or revised as these conditions are obsolete or adequately described elsewhere.

Finally, this application proposes to revise Technical Specification Table 5.5.9-2, "Steam Generator Tube Inspection," and Table 5.5.9-3, "Steam Generator Repaired Tube Inspection", to delete the requirement to notify the NRC pursuant to 10 CFR 50.72(b)(2) if the steam generator tube inspection results in a C-3 classification. On October 25, 2000, the NRC issued a final rule that amended the event reporting requirements for nuclear power reactors to reduce or eliminate unnecessary reporting burden associated with events of little or no safety significance. The current reporting requirement in Table 5.5.9-2 is incorrect based on the issuance of this final rule.

AmerenUE has evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10 CFR 50.92(c) as discussed below:

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

   Response: No

This request involves administrative changes only. The changes consist of duplicates or overly burdensome reporting requirements or the deletion of completed items required by conditions from the original issuance of Operating License NPF-30. No actual plant equipment or accident analyses will be affected by the proposed changes. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No

   This request involves administrative changes only. The changes consist of duplicates or overly burdensome reporting requirements or the deletion of completed items required by conditions from the original issuance of Operating License NPF-30. No actual plant equipment or accident analyses will be affected by the proposed change and no failure modes not bounded by previously evaluated accidents will be created. Therefore, the proposed changes do not create a new or different kind of accident from any accident previously evaluated.

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

   Response: No

   Margin of safety is associated with confidence in the ability of the fission product barriers (i.e., fuel and fuel cladding, Reactor Coolant System pressure boundary, and containment structure) to limit the level of radiation dose to the public. This request involves administrative changes only.

   No actual plant equipment or accident analyses will be affected by the proposed change. The changes consist of duplicates or overly burdensome reporting requirements or the deletion of completed items required by conditions from the original issuance of Operating License NPF-30. Additionally, the proposed changes will not relax any criteria used to establish safety limits, will not relax any safety systems settings, or will not relax the bases for any limiting conditions of operation. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Based on the above evaluations, AmerenUE concludes that the activities associated with the above described changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92 and accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

10 CFR 50.72: "Immediate notification requirements for operating nuclear power reactors." This regulatory requirement contains general requirements (section (a)), requirements for reporting non-emergency events (section (b)), and requirements for providing followup notification (section c)).

10 CFR 50.73(a) Reportable events. (1) The holder of an operating license for a nuclear power plant (licensee) shall submit a Licensee Event Report (LER) for any event of the type described in this paragraph within 60 days after the discovery of the event. In the case of an invalid actuation reported under § 50.73(a)(2)(iv), other than actuation of the reactor protection system (RPS) when the reactor is critical, the licensee may, at its option, provide a telephone notification to the NRC Operations Center within 60 days after discovery of the event instead of submitting a written LER. Unless otherwise specified in this section, the licensee shall report an event if it occurred within three years of the date of discovery regardless of the plant mode or power level, and regardless of the significance of the structure, system, or component that initiated the event.

Analysis

This amendment application proposes to delete Section 2.F of the Operating License which requires reporting any violations of the requirements contained in Section 2.C of the license (except for Section 2.C(2)). Operating License Section 2.C lists conditions regarding Maximum Power Level, Technical Specifications and Environmental Protection Plan, Fire Protection and other license conditions. Operating License Section 2.F requires that AmerenUE notify the NRC of violations of the requirements in Section 2.C within 24 hours in accordance with the provisions of 10 CFR 50.72 and with written followup in accordance with the procedures described in 10 CFR 50.73 (b), (c), (d) and (e). The requirements of Section 2.F of the license are adequately addressed by the reporting requirements identified in 10 CFR 50.72 and 10 CFR 50.73.

Deviations from the conditions regarding Maximum Power Level, Technical Specifications and Environmental Protection Plan and Fire Protection are adequately governed by the requirements of 10 CFR 50.72 and 10 CFR 50.73.

The proposed changes will reduce redundant regulatory burden and will allow AmerenUE to maintain compliance with 10 CFR 50.72 and 10 CFR 50.73, without the requirement of the current Section 2.F.

Portions of License Condition 2.C.(1) are conditions that have been completed and are considered obsolete. License Conditions 2.C.(3), 2.C.(4), 2.C.(5)(a), 2.C.(5)(b), 2.C.(6), 2.C.(7), 2.C.(8), 2.C.(9), 2.C.(10), 2.C.(11), 2.C.(12), 2.C.(13) and 2.C.(14) are conditions that have been completed and are considered obsolete. The proposed changes either delete or modify existing license conditions which have been completed or are otherwise no longer in effect.

The current reporting requirement in Technical Specification Tables 5.5.9-2 and 5.5.9-3 are incorrect based on the issuance of the final rule (Reference 8.3). Tables 5.5.9-2 and 5.5.9-3 are revised to delete the reporting requirement. Deletion of this requirement from the Technical Specifications does not change the requirement to report results that satisfy the criteria of 10 CFR 50.72(b)(3). Additionally, Technical Specification 5.6.10c still requires the results of steam generator tube inspections, which fall into Category C-3, in a report within 30 days and prior to resumption of plant operation.

Conclusion

The proposed changes either delete or modify existing license conditions which have been completed or are otherwise no longer in effect. The deletion of Section 2.F of the Operating License and the changes to Technical Specification Tables 5.5.9-2 and 5.5.9-3 are consistent with the changes recently implemented in 10 CFR 50.72 and 10 CFR 50.73.

Based on the considerations discussed above, 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.

6.0 ENVIRONMENTAL CONSIDERATION

AmerenUE has determined that the proposed amendment is a change to reporting and administrative requirements as described in 10 CFR 51.22(c)(10). Accordingly, the proposed amendment meets eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(10). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed changes is not required.

7.0 PRECEDENTS

There is precedent for allowing deletion of Section 2.F and the specified conditions in Section 2.C from the Facility Operating License. The Duquesne Light Company operating licenses for Beaver Valley Power Station, Units 1 and 2 (Facility Operating License Numbers DPR-66 and NPF-73) have been amended to delete the Condition on reporting of violation of license conditions and other specified conditions via Amendments 220 and 97 respectively. The Wolf Creek Nuclear Operating Corporation operating license for Wolf Creek Generating Station (Facility Operating License Number NPF-42) has been amended to delete the Condition on reporting of violation of license conditions and other specified conditions via Amendment Number 141. TXU Power operating licenses for Comanche Peak Steam Electric Station (CPSES), Units 1 and 2 (Facility Operating License Numbers NPF-87 and NPF-89) have been amended to delete the Condition on reporting of violation of license conditions and other specified conditions via Amendment 103 respectively.

8.0 REFERENCES

8.1 10 CFR 50.72, "Immediate notification requirements for operating nuclear power reactors."

8.2 10 CFR 50.73, "Licensee event report system."

8.3 Federal Register, Vol. 65, No. 207, pg. 63769, "Reporting Requirements for Nuclear Power Reactors and Independent Spent Fuel Storage Installations at Power Reactor Sites."

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ATTACHMENT 3

MARKUP OF OPERATING LICENSE PAGES

H. This license is effective as of the date of issuance and shall expire at Midnight on October 18, 2024.

FOR THE NUCLEAR REGULATORY COMMISSION

ORIGINAL SIGNED BY H. R. DENTON

Harold R. Denton, Director
~~Office of Nuclear Reactor Regulation~~

Attachments/Appendices:

(Deleted per Amendment No. xxx)

1. Attachment 1
2. Attachment 2
3. Appendix A - Technical Specifications (NUREG-1058, Revision 1)
4. Appendix B - Environmental Protection Plan
5. Appendix C – Additional Conditions

Date of Issuance: October 18, 1984

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(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 ~~and in Attachment 1 to this license. The preoperational tests, startup tests and other items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.~~

(2) Technical Specifications and Environmental Protection Plan*

The Technical Specifications contained in Appendix A, as revised through Amendment No. 161 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. xxx

~~(a) Prior to November 30, 1985, UE shall environmentally qualify all electrical equipment according to the provisions of 10 CFR 50.49.~~

Amendment 161

* 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.

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~~(b) Prior to restart following the first refueling outage, UE shall have qualified the reactor vessel level instrumentation system high volume sensor.~~

(4) Surveillance of Hafnium Control Rods (Section 4.2.3.1(10), SER and SSER #2)

~~UE shall perform a visual inspection of a sample of hafnium control rods during one of the first five refueling outages. A summary of the results of these inspections shall be submitted to the NRC.~~

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Deleted per Amendment No. XXX

(5) Fire Protection (Section 9.5.1.7 SER and Section 9.5.1.8, SSER #3)

(a) ~~Within 60 days of acquisition of the 100% power data for thermal and dynamic testing, UE shall have operable the Halon systems in the north electrical penetration room (fire area A-18).~~

(b) ~~Prior to restart following the first extended outage of known duration greater than two weeks occurring after February 15, 1985 or prior to restart following the first refueling outage which ever occurs first, UE shall have completed the installation of the five new isolation switches and modification to the four existing isolation switches identified in the August 23, 1984 SNUPPS letter.~~

(c) The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the SNUPPS Final Safety Analysis Report for the facility through Revision 15, the Callaway site addendum through Revision 8, and as approved in the SER through Supplement 4, subject to provision d below.

(d) The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(e) Deleted (see Amendment No. 30, January 13, 1988)

(6) Qualification of Personnel (Section 13.1.2, SSER #3, Section 18, SSER #1)

~~(a) UE shall have on each shift operators who meet the requirements described in Attachment 2.~~

~~(b) UE shall have a senior individual with previous operating experience on a commercial PWR assigned to assist the Plant Manager as an advisor during the startup test program and for one year following full power operation.~~

(7) NUREG-0737 Conditions (Section 22, SER)

Deleted per Amendment No. xxx

~~UE shall complete the following conditions to the satisfaction of the NRC. These conditions reference the appropriate items in Section 22.2, "TMI Action Plan Requirements for Applicants for Operating Licenses," in the Safety Evaluation Report and Supplements 1, 2, 3 and 4 NUREG-0830.~~

~~(a) Detailed Control Room Design Review (I.D.1, SSER #4)~~

~~Prior to May 1, 1985, UE shall submit for review and approval by the NRC staff, the results of the function and task analysis. For those Human Engineering Discrepancies (HEDs) identified by this analysis that require correction, the submittal shall include the proposed correction and implementation schedule; and for those HEDs for which no planned correction is proposed, a basis for that determination shall be documented.~~

~~(b) Emergency Response Capabilities (Generic Letter 82-33, Supplement 1 to NUREG-0737)~~

~~Prior to restart following the first refueling outage, UE shall have a fully functional Technical Support Center and Emergency Operations Facility and a fully operable Emergency Response Facilities Information System (ERFIS).~~

~~(c) Regulatory Guide 1.97 (Section 7.5.2.3, SSER #3)~~

~~Prior to restart following the first refueling outage, UE shall have installed and operable the following instrumentation.~~

~~1) Source range instrumentation qualified to post-accident conditions~~

~~2) Reactor vessel water level instrumentation~~

~~3) Subcooling monitors~~

~~4) Radiation monitors for releases from steam generator safety/relief valves or atmospheric dump valves, and~~

~~(5) Auxiliary feedwater pump turbine exhaust monitor.~~

(8) Post-Fuel-Loading Initial Test Program (Section 14, SER)

~~UE shall conduct the post-fuel-loading initial test program described in Chapter 14 of the FSAR, as amended, without making any major modifications unless such modifications have prior NRC approval. Major modifications are defined as:~~

~~(a) elimination of any safety-related test*~~

~~(b) modification of objectives, test method, or acceptance criteria for any safety-related test~~

~~(c) performance of any safety-related test at a power level different from that stated in the FSAR by more than 5 percent of rated power~~

~~(d) failure to satisfactorily complete the entire initial start-up test program by the time core burnup equals 120 effective full power days~~

~~(e) deviation from initial test program administrative procedures or quality assurance controls described in the FSAR~~

~~(f) delays in test program in excess of 30 days (14 days if power level exceeds 50 percent), concurrent with power operation. If continued power operation is desired during a delay, the licensee shall provide justification that adequate testing has been performed and evaluated to demonstrate that the facility can be operated at the planned power level with reasonable assurance that the health and safety of the public will not be endangered.~~

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Deleted per Amendment No. xxx

(9) Inservice Inspection Program (Sections 5.2.4 and 6.6, SER)

~~Within nine months of the date of this license, UE shall submit for staff review and approval, the inservice inspection program which conforms to the ASME Code in effect 12 months prior to the date of issuance of this license.~~

~~* Safety-related tests are those tests which verify the design, construction, and operation of safety-related structures, and equipment.~~

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(10) Emergency Planning

~~In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provisions of 10 CFR Section 50.54(s) (2) will apply.~~

(11) Steam Generator Tube Rupture (Section 15.4.4, SSER #3)

~~Prior to restart following the first refueling outage, UE shall submit for NRC review and approval an analysis which demonstrates that the steam generator single-tube rupture (SGTR) analysis presented in the FSAR is the most severe case with respect to the release of fission products and calculated doses. Consistent with the analytical assumptions, the licensee shall propose all necessary changes to Appendix A to this license.~~

Deleted per Amendment No. xxx

(12) Low Temperature Overpressure Protection (Section 15, SSER #3)

~~By January 1, 1985, UE shall submit for NRC review and approval a description of equipment modifications to the residual heat removal system (RHRS) suction isolation valves and to closure circuitry which conform to the applicable staff requirements (SRP 5.2.2). Within one year of receiving NRC approval of the modifications, UE shall have the approved modifications installed. Alternately, by January 1, 1985, UE shall provide acceptable justification for reliance on administrative means alone to meet the staff's RHRS isolation requirements, or otherwise, propose changes to Appendix A to this license which remove reliance on the RHRS as a means of low temperature overpressure protection.~~

(13) LOCA Reanalysis (Section 15, SSER #3)

~~Prior to restart following the first refueling outage, UE shall submit for NRC review and approval a reanalysis for the worst large break LOCA using an approved ECCS evaluation model. At this time that model is the 1981 Westinghouse model. A modified version of the 1981 model which includes the BART computer code may be used.~~

(14) Generic Letter 83-28

~~UE shall submit responses to and implement the requirements of Generic Letter 83-28 on a schedule which is consistent with that given in its May 21, 1984 letter.~~

(15) Additional Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No. 133, are hereby incorporated into this license. UE shall operate the facility in accordance with the Additional Conditions.

D. An Exemption from certain requirements of Appendix J to 10 CFR Part 50, are described in the October 9, 1984 staff letter. This exemption is authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, this exemption is hereby granted pursuant to 10 CFR 50.12. With the granting of this exemption the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. UE shall fully implement and maintain in effect all provisions of the ~~Commission-approved physical~~ security, guard training and qualification, and safeguards contingency plans ~~including~~ amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) ~~and to~~ the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Callaway Security Plan," with revisions submitted through November 17, 1987; "Callaway Security Force Training and Qualification Plan," with revision submitted through November 21, 1986; and "Callaway Safeguards Contingency Plan," with revisions submitted through November 21, 1986. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

Deleted per Amendment No.xxx

F. ~~With the exception of 2.C(2) UE shall report any violations of the requirements contained in Section 2.C, of this license within 24 hours. Initial notification shall be made in accordance with the provisions of 10 CFR 50.72 with written follow-up in accordance with the procedures described in 10 CFR 50.73(b), (c), (d), and (e).~~

G. UE shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

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ATTACHMENT 1

~~This attachment identifies items which must be completed to the Commission's satisfaction in accordance with the operational modes as identified below.~~

~~A The licensee shall implement Radiation/Chemical Technician refresher training within six months following fuel load.~~

~~B The licensee shall install a permanent area monitor on the manipulator crane prior to the entering Mode 6 (refueling mode).~~

Deleted per Amendment No. xxx

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ATTACHMENT 2

~~Operating Staff Experience Requirements~~

~~UE shall have a licensed senior operator on each shift who has had at least six months of hot operating experience on a same type plant, including at least six weeks at power levels greater than 20% of full power, and who has had startup and shutdown experience. For those shifts where such an individual is not available on the plant staff, an advisor shall be provided who has had at least four years of power plant experience, including two years of nuclear plant experience, and who has had at least one year of experience on shift as a licensed senior operator at a similar type facility. Use of advisors who were licensed only at the RO level will be evaluated on a case-by-case basis. Advisors shall be trained on plant procedures, technical specifications and plant systems, and shall be examined on these topics at a level sufficient to assure familiarity with the plant. For each shift, the remainder of the shift crew shall be trained in the role of the advisors. The training of the advisors and remainder of the shift crew shall be completed prior to exceeding 5% power. Prior to exceeding 5% power, UE shall certify to the NRC the names of the advisors who have been examined and have been determined to be competent to provide advice to the operating shifts. These advisors shall be retained until the experience levels identified in the first sentence above have been achieved. The NRC shall be notified at least 30 days prior to the date UE proposes to release the advisors from further service.~~

Deleted per Amendment No. XXX

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ATTACHMENT 4

MARKUP OF TECHNICAL SPECIFICATION PAGES

5.5 Programs and Manuals (continued)

TABLE 5.5.9-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N.A.	N.A.	N.A.	N.A.
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N.A.	N.A.
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug or repair defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N.A	N.A.
	C-3	Inspect all tubes in this S.G., plug or repair defective tubes and inspect 2S tubes in each other S.G. ~~Notification to NRC pursuant to §50.72(b)(2) of 10 CFR Part 50~~	All other S.G.s are C-1	None	N.A.	N.A.
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N.A.	N.A.

(continued)

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5.5 Programs and Manuals

TABLE 5.5.9-2

STEAM GENERATOR TUBE INSPECTION
(continued)

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug or repair defective tubes. ~~Notification to NRC pursuant to §50.72(b)(2) of 10 CFR Part 50~~	N.A.	N.A.

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$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

(continued)

5.5 Programs and Manuals (continued)

TABLE 5.5.9-3

STEAM GENERATOR REPAIRED TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required
A minimum of 20% of repaired tubes (1) (2)	C-1	None	N.A.	N.A.
	C-2	Plug defective repaired tubes and inspect 100% of the repaired tubes in this S.G.	C-1	None
			C-2	Plug defective repaired tubes
			C-3	Perform action for C-3 result of first sample
	C-3	Inspect all repaired tubes in this S.G., plug defective tubes and inspect 20% of the repaired tubes in each other S.G. ~~Notification to NRC pursuant to §50.72(b)(2) of 10 CFR Part 50~~	All other S.G.s are C-1	None
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of first sample

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(continued)

5.5 Programs and Manuals

TABLE 5.5.9-3

STEAM GENERATOR REPAIRED TUBE INSPECTION
(continued)

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required
			Additional S.G. is C-3	Inspect all repaired tubes in each S.G. and plug defective tubes. ~~Notification to NRC pursuant to §50.72(b)(2) of 10 CFR Part 50~~

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(1) Each repair method is considered a separate population for determination of initial inservice inspection and scope expansion.

(2) The inspection of repaired tubes may be performed on tubes from 1 to 4 steam generators based on outage plans.

(continued)

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ATTACHMENT 5

RETYPED MARKUP OF TECHNICAL SPECIFICATION PAGES

5.5 Programs and Manuals (continued)

TABLE 5.5.9-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N.A.	N.A.	N.A.	N.A.
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this S.G	C-1	None	N.A.	N.A.
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug or repair defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N.A	N.A.
	C-3	Inspect all tubes in this S.G., *plug or repair* defective tubes and inspect 2S tubes in each other S.G.	All other S.G.s are C-1	None	N.A.	N.A.
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N.A.	N.A.

(continued)

5.5 Programs and Manuals

TABLE 5.5.9-2

STEAM GENERATOR TUBE INSPECTION
(continued)

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
			Additional S.G. is C-3	Inspect all tubes in each S.G and plug or repair defective tubes.	N.A.	N.A.

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

(continued)

5.5 Programs and Manuals (continued)

TABLE 5.5.9-3

STEAM GENERATOR REPAIRED TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required
A minimum of 20% of repaired tubes (1) (2)	C-1	None	N.A.	N.A.
	C-2	Plug defective repaired tubes and inspect 100% of the repaired tubes in this S.G	C-1	None
			C-2	Plug defective repaired tubes
			C-3	Perform action for C-3 result of first sample
	C-3	Inspect all repaired tubes in this S.G., plug defective tubes and inspect 20% of the repaired tubes in each other S.G	All other S.Gs are C-1	None
			Some S.Gs C-2 but no additional S.G are C-3	Perform action for C-2 result of first sample

(continued)

5.5 Programs and Manuals

TABLE 5.5.9-3

STEAM GENERATOR REPAIRED TUBE INSPECTION
(continued)

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required
			Additional S.G. is C-3	Inspect all repaired tubes in each S.G. and plug defective tubes.

(1) Each repair method is considered a separate population for determination of initial inservice inspection and scope expansion.
(2) The inspection of repaired tubes may be performed on tubes from 1 to 4 steam generators based on outage plans.

(continued)

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ATTACHMENT 6

SUMMARY OF REGULATORY COMMITMENTS

SUMMARY OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by AmerenUE, Callaway Plant in this document. Any other statements in this submittal are provided for information purposes and are not considered to be commitments. Please direct questions regarding these commitments to Dave E. Shafer, Superintendent, Licensing at AmerenUE, Callaway Plant, (314) 554-3104.

COMMITMENT	Due Date/Event
The proposed amendment will be implemented within 90 days after approval	90 days following NRC approval