August 10, 2000

Ms. Kay Drey
515 West Point Avenue
University City, MO 63130

Dear Ms. Drey:

This is my second letter in response to your letter of March 10, 2000, that you sent me, as project manager at the Nuclear Regulatory Commission (NRC) for the Callaway Plant, Unit 1. In your letter of March 10, 2000, you requested information about (1) the reactor scram event that occurred at Callaway on Sunday, February 13, 2000, and (2) the electrosleeve amendment that the NRC issued as Amendment No. 132 on May 21, 1999.

In my letter to you of April 20, 2000, I said that we requested the licensee of Callaway to address certain questions in your letter and we would address the remaining questions. In my letter to you, I provided a copy of the letter to the licensee which included a table where we listed your questions and identified those questions that we requested the licensee to address. I said that after we have received and reviewed the licensee's responses and addressed the other questions, we would respond in writing to you on all the questions. This letter is our response to you on all your questions, including Questions A.6 and B.9 that were addressed in my letter of April 20, 2000; however, the attachments referenced in the responses to these two questions were provided in my April 20, 2000, letter.

The licensee provided answers to their identified questions in its letter of June 5, 2000, to the NRC. Enclosed are the licensee's and staff's responses to your questions. I hope that the responses relieve your concerns about the Callaway reactor scram event that occurred on Sunday, February 13, 2000, and the electrosleeve amendment. If you have any questions, please contact me at 301-415-1307 or, through the internet, at jnd@nrc.gov.

Sincerely,

/RA/

Jack Donohew, Senior Project Manager, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosure: Responses to Questions

cc w/encl:
Mr. Garry L. Randolph
Vice President and Chief Nuclear Officer
Union Electric Company
Post Office Box 620
Fulton, MO 65251

Ms. Kay Drey
515 West Point Avenue
University City, MO 63130

August 10, 2000

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OFFICE	PDIV-2/PM	PDIV-2/LA	EMCB/BC	RGN IV	PDIV-2/SC
NAME	JDonohew:am	EPeyton	WBateman	WJohnson by e-mail	SDembek
DATE	07/06/00	08/07/00	07/13/00	07/12/00	08/07/00

OFFICE	EEIB/BC	SPSB/BC(A)
NAME	CHolden for JCalvo	RJBarrett w/changes
DATE	07/20/00	08/03/00

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RESPONSES TO QUESTIONS IN LETTER OF MARCH 10, 2000

The following lists the questions from the letter of March 10, 2000, from Ms. Kay Drey to Jack Donohew, project manager at the Nuclear Regulatory Commission (NRC) for the Callaway Plant, Unit 1. The following responses to the questions are either from the licensee for Callaway, taken from the licensee's letter of June 5, 2000, to the NRC, or from the staff, or in a few cases from both the licensee and the staff. The identification of which questions would be addressed by the licensee and which questions would be addressed by the staff was in the table attached to the staff's letter of April 12, 2000, to the licensee. A copy of the April 12, 2000, letter was sent to Ms. Drey in the staff's letter of April 20, 2000, to Ms. Drey.

The staff reviewed the licensee's responses to questions in the licensee's letter of June 5, 2000. If the staff had any comment on a licensee's response, it is provided in the staff's response to the same question. Where the staff added a phrase to clarify a statement in the licensee's response, the phrase is within brackets (i.e., within "[...]"). Where the licensee referred to itself as other than the licensee or AmerenUE (i.e., Ameren Union Electric), the reference was replaced by "[the licensee ...]." Most questions will only have a response from the licensee. The staff's responses to Questions A.6 and B.9 that were provided in the staff's letter of April 20, 2000, are also provided below for completeness; however, the enclosures referred to in these responses were provided in the April 20, 2000, letter.

The licensee in its letter of June 5, 2000, introduced its responses to the questions identified in the staff's letter of April 12, 2000, with the following statement concerning the characterization of steam generator tube ruptures in the beginning of the March 10, 2000, letter from Ms. Drey:

> There is no relationship whatsoever between a Steam Generator Tube Rupture and the 2/13/00 trip. There has never been a Steam Generator Tube Rupture at Callaway nor was there ever a threat of such an event during the 2/13/00 trip.

Question A1: Did any of the components fail independently, or were the components linked with one another and thus failed interactively (a form of common-mode failure)?

Or, as another form of this question: Has the NRC confirmed as yet whether or not the grid system voltage fluctuations caused only one (circuit or power supply) breaker to malfunction (which in turn caused the four reactor coolant pumps to trip and the rest of the sequence of trips, etc., to occur) - - or were other electrical controls or components of the nuclear steam supply system also directly affected by the electrical grid irregularities?

Licensee's Response: No components "failed" at Callaway. All safety-related equipment performed as designed. The transmission system disturbance was initiated by a transmission line fault within a neighboring rural electric cooperative's transmission system.

Staff's Response: On February 13, 2000, an automatic actuation of the reactor protection system was initiated due to a low reactor coolant flow condition at Callaway Plant. This condition resulted when a reactor coolant pump (RCP) motor's protective relay sensed an electrical disturbance occurring on the transmission system, subsequently tripping the pump. The cause of the disturbance was attributed to a transmission line breaker malfunction due to a defective electrical connection with a neighboring electric cooperative organization's protective relaying scheme. The licensee reviewed their transmission system and RCP protective relaying setpoints and determined that the subject relays functioned as designed. NRC inspectors reported as documented in Inspection Report No. 50-483/00-01 (dated March 7, 2000, that was attached to the staff's letter dated April 20, 2000), that the only electrical component associated with the subject event was an incorrectly adjusted safety valve position instrument. The inspection report cited no other electrical components or components of the nuclear reactor steam supply directly affected by the electrical grid irregularities.

Question A2: How frequently has NRC been informed of similar disruptions in power that have resulted in a potentially dangerous chain of events? More specifically, have fluctuating voltages frequently affected the operability of safety systems? (These questions are especially important to Missourians because of our many thunderstorms.)

Licensee's Response: Again, all safety systems performed as designed. Based on experience at Ameren, the events that led to the 2/13/00 trip would be categorized as highly irregular. The fact that the fault lasted long enough to burn down a conductor and cause a 345/161kV transformer to fail catastrophically separates this from a storm or contact-related fault. The time to clear the fault was substantially longer than is typical. The frequency of such events is rare.

Staff's Response: The NRC's search of the data reported under 10 CFR 50.72 for the years 1998-2000 did not identify any plant events that originated from a fluctuating voltage besides the Callaway event of February 13, 2000. Nuclear stations are designed to perform all the required safety functions without relying on offsite power sources that are often subject to varying levels of voltage fluctuations. The electrical power system at the nuclear stations have monitoring relays that sense the grid voltage conditions. Under unacceptable voltage conditions, the plant electrical system automatically disconnects from the power grid and the safety grade emergency diesel generators power the onsite safety related systems. The grid voltage fluctuations could result in a plant trip that disrupts the power generation from the nuclear station. The power from the grid would be used to power the safety systems only when it is capable of providing acceptable level of voltage. The voltage is continuously monitored at several critical points to ensure acceptable voltage for safety related systems needed for reactor safety. The NRC regulations require redundant onsite power sources to adequately perform safety functions even in the absence of any power from the grid. Our examination of the recent history has not revealed any condition that resulted in the absence of sufficient operable safety related systems to maintain nuclear safety.

Question A3: To what extent are surge protectors required on safety related equipment at nuclear power plants? If they are required, are they rated as safety-related, as per 10CFR50, Appendix B.

Licensee's Response: NRC Regulatory Guide 1.32, Revision 2 and IEEE Standard 308-1974 provide the basic requirements for safety-related, class 1E-power systems. All Callaway safety-related equipment complies with these requirements. Protection from lightning is also part of the basic design of the electrical power system. Lightning arrestors are applied at connections to overhead power lines in the electrical distribution system. The lightning arrestors are part of the non-safety related power distribution system and, as such, are not supplied safety-related per 10 CFR50, Appendix B. Nevertheless, lightning arrestors are surge protectors. While they are not safety related, they act to prevent unnecessary safety-related equipment actuations by prohibiting switching and lightning surges from coming into the plant.

Question A4: Although the warning sirens in the plant's emergency planning zone were fortunately not needed for the 2/13 transient, does anyone know if any of them became inoperable during the period of fluctuating voltages?

Licensee's Response: [The licensee agrees] with Ms. Drey that there was no need to activate the emergency warning sirens since there was never a threat to the public resulting from the 2/13/00 trip. In response to Ms. Drey's question, no emergency sirens lost power during the 2/13/00 trip. All remained operable. However, should a siren ever lose normal AC power for any reason, they would remain operational since each siren has an 8-hour battery backup power supply.

Question A5: According to NRC Information Notice 98-07, "Offsite Power Reliability Challenges from Industry Deregulation," the reliability of power from the transmission system grid to nuclear power plants may be adversely affected by the deregulation of the electric power industry. Aside from changes in grid loading that may affect the reliability of an off-site power source, do you anticipate other problems, such as potential decrease in funding for the maintenance of transmission systems?

Licensee's Response: AmerenUE maintains, and will continue to maintain, transmission facilities in accordance with regulations and interregional agreements to assure reliability. No change is anticipated because of electric power industry deregulation. From a business standpoint it is in AmerenUE's best interest to maintain a reliable transmission system.

Staff's Response: Although the staff does not anticipate that other potential industry changes such as a decrease in the funding for the maintenance of transmission system will directly impact the reliability of offsite power sources to a nuclear power plant, the staff is monitoring industry developments which may potentially adversely affect offsite power capability. The staff is taking follow-up action on risk-based analyses and operating experience and accident sequence precursor program evaluations in order to assure that the licensing basis for offsite power at nuclear power plants is maintained. Recently, the NRC issued Information Notice 2000-06, "Offsite Power Voltage Inadequacies" on March 22, 2000, to alert licensees to a possible concern regarding the voltage adequacy of offsite power sources. In addition, the staff met with the Nuclear Energy Institute on May 18, 2000, in order to discuss the development of a voluntary industry initiative to address offsite power concerns.

Question A6: I understand that AmerenUE officials will be attending a meeting at the NRC's Region IV office in Texas this Monday, March 13, 2000, to discuss the NRC's special inspection regarding degraded switchyard voltage conditions associated with a different reactor scram - the manual scram of August 11, 1999. Does the NRC expect to conduct a similar special inspection regarding the February 13, automatic rector scram caused by off-site voltage irregularities? The 8/11/99 incident apparently was caused in part by AmerenUE's failure to verify the operability of Callaway's off-site power sources (its failure to detect a low switchyard voltage condition) following a trip of the reactor and main generator.

Just as with the February 13 incident, the 8/11/99 incident included a sequence of environmental, economic and human error conditions involving offsite and onsite electrical sources and distribution systems. The August event included a turbine building steam-pipe rupture, a deenergized computer, an inadvertently severed fiber optic channel (that broke off communication of the switchyard voltage data between the Callaway control room and Ameren's Energy Supply Operations facility in downtown St. Louis), and the establishment of erroneous and nonconservative alarm setpoint caused by the transposition error during the inputting of voltage parameters in the plant computer.

Apparently the August 11-12, 1999, voltage problems may have been caused in part by near-peak summertime power wheeling ("excessive voltage support to the grid"), a condition perhaps related to "the potential impacts of power market deregulation on the reliability of the electrical grid relative to the design and licensing basis of your facility." (quoting from NRC Special Inspection Report No. 99-15, February 15, 2000; emphasis added).

Licensee's Response: Ms. Drey's brief description of the August 11, 1999 event is accurate except for her attempt to compare the 2/13/00 trip with the 8/11/99 event. The two events were totally different. The 8/11/99 event related to steady state voltage levels and potential inoperability associated with events in August of 1999. The 2/13/00 trip involved a transmission system disturbance initiated by a transmission line fault within a neighboring rural electric cooperative's transmission system. All safety systems performed as designed. Contrary to Ms. Drey's assertion, the 2/13/00 trip was not related to any economic factors nor did it involve any human error at Callaway Plant.

Regarding [the licensee's] delayed detection of a low switchyard voltage condition, several corrective actions have been taken to address the 8/11/99 event. These include real-time computer contingency analysis, formal agreements between Energy Supply Operations and Callaway Plant, procedure revisions, annunciator modifications, plant computer alarms and increased training. In addition, modifications are ongoing and planned to add voltage regulation equipment at Callaway. These modifications will provide for operation under a wider range of grid voltages. Finally, projected grid conditions are evaluated prior to each summer and winter peak season to assure plant equipment will operate properly.

Staff's Response: The question was if NRC expected to conduct a special inspection, similar to that conducted for the August 11, 1999, event, for the February 13, 2000, automatic reactor scram event at Callaway. The staff's response was the following:

For Question A.6, upon review of the event, we decided that the February 13, 2000, event did not warrant a special inspection, such as was conducted for the August 11, 1999, manual scram event. The licensee's description of the second event is documented in its licensee event report (LER) 2000-002-00, "Automatic Reactor Trip Initiated by Reactor Coolant Pump Trip Caused by Motor Current Imbalance Due to Transmission System Disturbance," dated March 13, 2000 (Enclosure 2 to the staff's letter of April 20, 2000) and Revision 1 to this LER dated May 1, 2000 (Attachment 1 to this enclosure). The licensee stated in the LER that all safety-related and non-safety-related equipment functioned as designed, and there was no release of radioactivity and added in the revised LER that no significant radioactivity was released during the event.

Thus, although both events were electrical grid-plant interactions, the August 11, 1999, event was an <u>unexpected</u> impact on the grid from a plant scram, and the February 13, 2000, event was an <u>expected</u> plant response to a grid perturbation. We, therefore, concluded that the February 13, 2000, event did not warrant a special inspection; however, a routine inspection was conducted of the event by the Callaway resident inspectors and documented in Inspection Report 50-483/00-01 dated March 7, 2000 (Enclosure 3 to the staff's letter of April 20, 2000). The meeting summary that the staff issued for the meeting on March 13, 2000, was issued on March 14, 2000 (Enclosure 4 to the staff's letter of April 20, 2000). The focus of the meeting was the transmission system perspective and corrective actions for the August 11, 1999, event because the impact on the grid had been unexpected. These were addressed by the licensee and none of the slides in the licensee's handout addressed the February 13, 2000, event. Although the February 13, 2000, event was briefly discussed in the meeting, neither the staff or the licensee believed the event needed to be extensively addressed in this meeting because the plant response to the event that occurred outside the Callaway system was as expected from the plant design.

<u>Question B1(a):</u> What was the concentration level, per liter, of radioactive contaminants in the secondary coolant <u>prior to</u> the February 13 electrical grid fluctuations? (I would appreciate it if you would include as part of your answer the levels of dissolved and entrained radioactive noble gases, and tritium.)

<u>Licensee's Response:</u> Secondary system activity in the last samples taken prior to the event, including Xe and Kr dissolved and entrained noble gases, is shown in the tabulation below. Although sample analysis includes a full spectrum of isotopes, only those listed were detected. In cases where all activity in the sample is found to be below the analysis lower limit of detection it is reported in the tabulation as **<u>N</u>**ot **<u>D</u>**etected. These activity concentrations are well below levels that would cause any health impact and do not significantly contribute to radiation dose to the public (i.e., a fraction of 1% of the regulatory limit). Although the activity listed below was detected in samples taken from the secondary system, the Atmospheric Steam Dump (ASD) radiation effluent monitor detected no activity in the steam released to the atmosphere through the ASD. (i.e., activity was below the detection limit of the ASD effluent monitor).

Sample	Isotope	Concentration uCi/ml	Concentration uCi/liter	Sample Date
Secondary System Tritium (condensate pump discharge)	H-3	3.8E-5	3.8E-2	02/01/00
Condenser Offgass	Xe-133	4.2E-8	4.2E-5	02/09/00
	Xe-135	2.0E-8	2.0E-5	02/09/00
Steam Generator - A	ND	ND	ND	02/11/00
B	I-133	1.23E-8	1.23E-5	02/11/00
C	Kr-88	2.08E-8	2.08E-5	02/11/00
D	ND	ND	ND	02/11/00

Question B1(b): How much in advance of the February 13 incident had the secondary coolant samples been collected and analyzed, and reported to the NRC?

Licensee's Response: Sample dates are provided in the response to Question B1(a) above. Technical Specifications require steam generators to be sampled every 72 hours. This information is available to NRC but not reported unless found to be above that allowed by plant Technical Specifications.

Question B2: How many pounds of steam did the licensee estimate were released during the "steam generator atmospheric dumps (that) were used for no more than 20 minutes?" If it is correct that each generator loop has two atmospheric steam dump valves, how many were full or partially open? Was it noisy as I'm told such dumping is?

Licensee's Response: Only one ASD opened on the 'D' Steam Generator for approximately 40 minutes following this event. Approximately 397,000 pounds of steam was released from this ASD. There is a substantial amount of noise associated with steam relief through this valve.

Question B3: Were radioactive gases, iodine, or other fission/corrosion/activation products released during the February 13 event from any of the following paths, other than the atmospheric steam dump?: (a) air ejector discharge, (b) turbine-driven auxiliary feed pump exhaust, or (c) gland steam exhaust? If so, please describe.

Licensee's Response: Since a small amount of activity was detected in samples taken of the secondary side, very small releases of radioactive gases and other fission products occurred from the paths mentioned above. These releases were well below levels that would cause any health impact and did not significantly contribute to radiation dose to the public (i.e., a small fraction of 1% of the regulatory limit). Any releases from the secondary side are reported as part of [the licensee's] Annual Effluent Release Report.

The licensee provided the following additional information in an email dated June 26, 2000, to the project manager for Callaway (ADAMS Accession No. ML003728228). The two dates referenced in the email were confirmed by the licensee to be February 11 and 13, 2000:

> These are the quantitative answers to question B3. The partitioning factor used for all these calculations was assumed to be 1. [The quantities below] are based on the pre-event samples taken on February 11, 2000 and [the post-event] samples taken on February 13, 2000.
>
> Air Ejector Discharge:
>
> The effluent of the Air Ejector Discharge is one of several flow paths directed to the Unit Vent for release to the environment. The flow going out the Unit Vent is sampled and the release to the environment is quantified. The Unit Vent results for the period of February 10 through February 17 is as follows:
>
> Tritium: 8.06 E-01 curies
> Xe-133: 8.51 E-01 curies
>
> Turbine-Driven Auxiliary Feed Pump:
>
> The turbine-driven auxiliary feed pump is supplied with steam from B & C steam generators. After the steam passes through the turbine, the exhaust is discharged to the atmosphere. The release to the environment is quantified using sample results from B & C steam generator and the steam flow through the turbine. The turbine-driven auxiliary feed pump was run for four hours on February 13. The results are as follows:
>
> Using February 11 steam generator results:
>
> I-133: 1.66E-06 curies
> Kr-88: 2.75E-06 curies
> H-3: 5.04E-03 curies
>
> Using February 13 steam generator results:
>
> I-131: 1.54E-04 curies
> I-132: 2.98E-05 curies
> I-133: 2.15E-04 curies
> I-135: 9.26E-05 curies
> Cs-137: 5.59E-06 curies
> Xe-133: 2.94E-05 curies
> Xe-135: 9.30E-06 curies
> Xe-135m: 9.37E-05 curies
> H-3: 5.04E-03 curies

Gland Steam Exhaust:

When the reactor is shutdown, there is no steam being supplied to the main turbine, therefore, no steam release from the gland steam exhaust into the turbine building. To quantify what was released to environment from gland steam exhaust for this event, the time from startup at 0 percent power to 100 percent power was used. This time period was 55 hours. The amount of activity discharged was:

Using February 11 steam generator results:

I-133:	5.25E-07 curies
Kr-88:	8.72E-07 curies
H-3:	1.58E-03 curies

Using February 13 steam generator results:

I-131:	4.87E-05 curies
I-132:	9.45E-06 curies
I-133:	6.83E-05 curies
I-135:	2.93E-05 curies
Cs-137:	1.77E-06 curies
Xe-133:	9.32E-06 curies
Xe-135:	2.95E-06 curies
Xe-135m:	2.97E-05 curies
H-3:	1.58E-03 curies

Question B4: Did the fluctuating voltages affect the steam monitoring equipment or any other electronic radiation detectors at the plant?

Licensee's Response: There was no loss of power or any other effect on the electronic radiation detectors or on steam monitoring equipment due to this event. As mentioned in our response to question B1(a), no activity was detected in the steam released to the atmosphere by the ASD radiation effluent monitor which was properly calibrated and functioning normally.

Question B5: Has any condenser (tertiary) cooling water, with its aggressive chemicals, leaked into the secondary coolant of the generators over the years, causing damage to the tubes or other steam generator internals?

Licensee's Response: Callaway has experienced 2 significant main condenser tube leaks over the life of the plant. The first one was in 1985; the second was in February 1997. The 1997 leak resulted in steam generator sulfates exceeding approximately 600 ppb and resulted in shutting down the plant per the Electric Power Research Institute (EPRI) Secondary Chemistry Guidelines and plant procedures. The Operations and Chemistry departments followed their procedures and took actions that mitigated the potential effects of the tube leak on the steam generators. Based on hideout return studies performed during the shutdown following the 1997 leak, it was evident that a fairly insignificant amount of sulfates were "hiding out" in the sludge and steam generator crevices, which suggests that the steam generator

crevice environment was not adversely affected by the main condenser tube leaks. The 1997 condenser tube leak had no perceptible effect on steam generator tube reliability.

The 1985 leak was significantly smaller (7 gpm compared to 145 gpm in 1997). The 1985 leak resulted in steam generator sulfates of ~100 ppb. The plant was not shut down as a result of the 1985 leak. Condensate Polishers were used to clean up the secondary side, minimizing any effect on the Steam Generators. Therefore, like the 1997 leak, the 1985 condenser tube leak had no perceptible effect on steam generator tube reliability.

Question B6: According to the NRC February 13 event report: "During the period of time before the reactor coolant pumps were restarted, the reactor coolant system pressure increased, causing one pressurizer PORV (one of the two power operated relief valves) to lift and reseat." (a) What is the normal operating pressure? (b) Does the fact that a PORV lifted mean that the reactor coolant system pressure exceeded the 2335 and 2485 psig setpoints? (c) Does anyone know for how long a period the PORV remained open before it reseated, and if any primary coolant escaped during that time? These questions are, of course, inspired by memories of the Three Mile Island-Unit 2 accident.

Licensee's Response: Normal operating reactor coolant system pressure is 2235 psig. Without pressurizer spray, the reactor coolant pressure rose to 2330 psig before both pressurizer PORVs lifted for about 1 second to relieve pressure. Taking into consideration the tolerance of the PORV instrumentation and the tolerance of the pressure instrumentation the operator was observing, the PORVs operated consistent with the intended setpoint of 2335 psig. Reactor coolant system pressure never approached the Safety Valve setpoint of 2485 psig. All primary coolant remained in a closed system, and none escaped to the containment.

Question B7: Regarding the steam generators: What is the current permissible primary-to-secondary coolant system leak rate limit at the Callaway Plant - - that is, the Technical Specification leak rate that the NRC is confident will not result in a sudden tube rupture?

Licensee's Response: Callaway Plant Technical Specifications require a plant shutdown when reaching primary-to-secondary leakage of 150 gpd [through any steam generator]. However, current plant procedures and programs contain engineering considerations and administrative controls which would result in a plant shutdown well before reaching the 150 gpd limit.

Staff's Response: There is no steam generator tube leak rate such that a tube rupture could not occur. The primary-to-secondary leak rate limit in the plant technical specifications, along with other elements in the licensee's steam generator program, is intended to ensure that structural and leakage integrity of the steam generators will be maintained within acceptable margins of safety consistent with the general design criteria of Appendix A to 10 CFR Part 50 and the accident dose guidelines of 10 CFR Part 100. The other program elements to ensure tube integrity include steam generator tube inspections (required by the technical specifications with sample size, frequency, and acceptance criteria requirements) during outages to detect and diagnose flaws in the tubes, evaluations to determine if tube integrity requirements were met at plant shutdown for the tube inspections and will be met through the next operating cycle until the next plant shutdown for the next inspections, monitoring and trending of tube leakage,

primary and secondary water chemistry controls, and training and qualification of operators on procedures for responding to tube leaks. As noted above, operating a plant within the primary-to-secondary leak rate limit will not always prevent the occurrence of a sudden tube failure. Leakage can arise from several types of tube defects. While some tube ruptures have been averted by rapid operator response, operators have not always been able to shut down a plant before leakage from certain types of tube defects became large.

A steam generator tube rupture (SGTR) can occur if the primary-to-secondary differential pressure across a tube exceeds the structural strength of the tube. Because a tube is leaking, it does not mean that the tube strength has decreased to the point where the tube would rupture under the primary-to-secondary operating conditions. This is the reason that technical specifications allow plants to operate with primary-to-secondary tube leakage. For the past 20 years, there have been only seven SGTRs among the U.S. commercial nuclear powers and all of the SGTRs have been the rupture of one tube. It should be pointed out that the pressurized water reactor plants that use steam generators, such as Callaway, are designed for the SGTR event.

Question B8: After three or four years of negotiations between the NRC staff and Union Electric/Ameren, the NRC staff finally issued a license amendment on May 24, 1999, which permitted Ameren to repair defective steam generator tubes at Callaway by using the Framatome Electrosleeving method.

According to NRC-SECY-99-199 (August 3, 1999): “The Electrosleeve is a nano-crystalline nickel sleeve that is electrochemically deposited on the inner surface of a steam generator tube. The Electrosleeve is a proprietary process designed to span a known flaw in the steam generator tube and to function as the pressure boundary.”

Does the NRC have a report that describes the predominant tube wall deformations and defects that were detected in the Callaway steam generator tubes – such as intergranular stress corrosion cracking, pitting, denting and circumferential cracking – that led to the decision to install electrosleeves in the tubes in an effort to try to reduce leakage or imminent leakage? If so, would you please tell me the title and document number? I am particularly interested in defects in the U-bends of the tubes and in the parts of the tubes that pass through the holes in the tube support plates (the series of plates in the steam generators designed to keep the tubes properly spaced).

Licensee’s Response: [The licensee’s] Technical Specifications require that [it] submit a report to NRC within 12 months of our last steam generator inspection that describes all defects found in the tubes. [The licensee’s] last report was dated April 28, 1999 (ULNRC-4019, for the Refuel 9 inspection).

Staff’s Response: The licensee is required by Callaway Technical Specification 5.6.10 to submit a steam generator tube inspection report to NRC within 12 months of the inspection. As stated in the staff response to Question B7 above, these inspections are normally conducted in the plant refueling outages. In many cases, the licensees will have a conference call with the staff if there is anything of interest about the tubes from the inspections. Unless a problem is identified by the licensee or another party (NRC project manager or the plant resident

inspector), or there have been previous problems identified in past inspections, the staff does not normally review these reports. The routine reports are kept for the staff to be able to refer back to previous inspections if a problem should be identified in the current inspection. The staff would then act on the basis of the information contained in the report, and, if the report did not warrant any action by the staff on the licensee's operation of the steam generators, the NRC would take no action, and vice versa if the information would warrant staff action. A copy of the licensee's last three such reports, the reports dated October 9, 1997, and April 28, 1999, and June 1, 2000, for the 9th, 10th, and 11th refueling outage inspections, respectively, are attached as Attachments 2, 3, and 4, respectively, to this enclosure.

However, Callaway Technical Specification 5.6.10 also requires that if more than 10 percent of the tubes inspected are degraded or more than 1 percent, are defective, a special report is sent to the Commission within 30 days and prior to the resumption of plant operation. If this situation reflected serious tube degradation, the inspection results and the appropriate corrective actions would be discussed with the licensee.

Question B9: How did the NRC decide to allow Callaway to be the first and perhaps only plant in the U.S. licensed to experiment with the Framatome Electrosleeves.

Staff's Response: The staff's justification for its decision to approve Amendment No. 132 dated May 21, 1999, is given in the safety evaluation dated May 21, 1999, attached to the letter approving the amendment (Enclosure 5 to the staff's letter of April 20, 2000).

Question B10: In the absence of any NRC or a NRC-licensee experience with Electrosleeving at a U.S. nuclear power plant, prior to Callaway, would you please tell me what test results you required AmerenUE to provide for your evaluation of this tube repair method? For example, what manufacturer's qualification and/or Ameren acceptance testing results were submitted to the NRC that demonstrated accelerated life testing – including elevated temperature and pressure regimes, and cycling up and down – and that you deemed were sufficient to give you confidence in the integrity and performance characteristics of the Electrosleeve method and materials for the desired lifetime (of two operating cycles, or beyond)?

Licensee's Response: Framatome Technologies, Inc. "Electrosleeving Qualification for PWR Recirculating Steam Generator Tube Repair," BAW-10219P, Revision 3 is a Framatome proprietary document that contains qualification information for electrosleeving. This document formed the technical basis for our submittal to NRC. While AmerenUE provided the NRC with some additional information, particularly concerning severe accident impact and information related to ultrasonic testing ("UT"), after the above report was submitted, the report represents > 95% of all the pertinent information.

Staff's Response: The staff reviewed the document listed in the licensee's response and used the information in the document as part of its basis to approve the amendment issued to Callaway to use electrosleeved tubes. Electrosleeving, because of its metallurgical structure and because there is no cold working of the tube, has been shown in laboratory tests to be highly resistant to tube degradation, compared to the conventional sleeving process. The BAW-10219P report was submitted with the licensee's application for the amendment and is referenced in the staff's safety evaluation dated May 21, 1999, issued with the amendment

(Enclosure 5 to the staff's letter of April 20, 2000). The basis for the staff's conclusion that the licensee could operate the Callaway steam generators safely for the two operating cycles with electrosleeved tube is given in the safety evaluation issued with the amendment. Section 3.10 of the safety evaluation entitled "Future Considerations" lists what the licensee must address in order for the staff to approve electrosleeving without the two-cycle limitation.

Because BAW-10219P is proprietary, in accordance with 10 CFR 2.790, the staff cannot send a copy of the document to members of the public.

Question B11: Was the U.S. Department of Energy's Argonne National Laboratory able to resolve the NRC staff's concerns about the potential failure of the electrosleeved tubes under severe accident conditions? That is, was ANL able to determine whether or not the electrochemically deposited sleeve would survive high temperatures and "high primary side pressure and depressurized and dry secondary side" conditions of a beyond-design-basis accident? (SECY-99-199, page 2) (The NRC withheld from public disclosure the June 1999 ANL technical letter report to the NRC, because of alleged proprietary commercial information, and I was therefore not able to read their answers to that inquiry.)

Staff's Response: Argonne National Laboratory was not able to resolve the staff's concerns. The laboratory provided the results of tests to quantify the potential for electrosleeved tubes to fail under severe accident conditions. The Argonne letter report is Attachment 5 to this enclosure. It should be noted that the conditions used by the laboratory for these tests were intended to be representative of one plant, rather than bounding for all plants. The staff's electrosleeve risk assessment (Attachment 6 to this enclosure) was not used in the Callaway amendment issued May 21, 1999, because the assessment did not show that the risk would be so great that the public was not adequately protected. At the time that the Callaway amendment was issued, the staff did not have procedures for incorporating risk concerns when the application is based solely on meeting all design basis requirements. Since then, SECY-99-246, "Proposed Guidelines for Applying Risk-Informed Decisionmaking in License Amendment Reviews," dated October 12, 1999 (Attachment 7 to this enclosure), provided guidelines for applying risk-informed decisionmaking and specifically referred to the electrosleeve amendment issued to Callaway as an example of the difficulty in completing a review of a proposed modification that meets existing regulations, but may introduce new potential risks from severe accidents. Using that guidance for future electrosleeve applications, in order for the staff to require a licensee to address risk issues, the staff would have to bear the burden of showing that the proposal could raise substantial issues with regard to the risk associated with conditions that are beyond the design basis requirements for the plant. The level of risk that is considered substantial is approximately the level that would not be approved for an amendment request that was submitted under a risk-informed process. Guidance for risk-informed applications is provided in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-informed Decisions on Plant-Specific Changes to the Licensing Basis," dated July 1998 (Attachment 8 to this enclosure). Based on the staff's experience with the Callaway amendment, the staff expects to consider severe accident risks in making future decisions on the application of electrosleeved tubes. However, it should be noted that there are other plant designs that present different levels of physical challenge in the event of a severe accident, both more and less difficult than presented by Callaway.

Additionally, different plants, even those with the same general design type, will have different estimated frequencies for those challenges. Therefore, each electrolseeve application will have to be considered on a plant-specific basis in order to properly assess the risk.

Question B12: Has Ameren examined the integrity of its electrosleeved tubes as of yet? (I believe the electrosleeves were to have been installed during the fall 1999 plant refueling outage.) (a) If no, when is the first examination of the parent tubes and sleeving to be required, and has Ameren or the NRC decided if ultrasonic testing will work? (b) If yes, have any flaws been identified that are longer than 1 inch? (c) Have any of the electrosleeved tubes already required removal from service, for example, by installing plugs in the tube ends?

Licensee's Response: Fifty-seven tubes were Electrosleeved (31 in A Steam Generator, 26 in C Steam Generator) at the hot leg top of the tubesheet in Refuel [outage] 10. Following installation, an approved Ultrasonic Testing technique was used to verify proper installation parameters. In addition, an eddy current ("+Point") examination was performed to provide baseline data. Because Cycle 11 is the first operating cycle the sleeves have been in service, no in-service examination has yet been performed. The EPRI Examination Guidelines require that sleeves be inspected in the same manner as tubes. This means that we must inspect a minimum of 20% of the installed Electrosleeves in Refuel 11 (spring 2001). We currently plan to inspect about 50% of them, exceeding the minimum requirement.

Question B13: The Callaway license amendment that permits the electrosleeving method apparently limits the installation of the sleeves to a maximum of two operating cycles (approximately three years). Is it correct that existing sleeves are to be removed after that duration, with no others to be permitted subsequently, as outlined in an NRC Request for Additional Information (December 16, 1998)? If so, has Electrosleeve removal experience been accrued at any other nuclear power plant(s) in Canada or Europe as yet?

Licensee's Response: The current license requires all steam generator tubes containing an Electrosleeve to be removed from service within two cycles following installation of the first Electrosleeve. The reason this restriction is in place is the NRC's concern with our ability to inspect a tube crack propagating into the electrosleeve. AmerenUE and Framatome expect this concern to be resolved prior to the completion of the two-cycle deadline, at which time AmerenUE will apply for the restriction to be removed. If removal from service is necessary, the tube would be plugged on both ends rather than removing the elecrosleeve itself.

Question B14: I am interested in learning how many (or the percent) of the 5,625 tubes in each of the four Callaway steam generators are no longer operable or available? (a) Would you please tell me what percent of the tubes in each of the generators has been Electrosleeved? (b) If other, non-Framatome sleeves had been installed earlier, would you also please tell me how many of those sleeves (or what percent) remain in each of the generators?

Licensee's Response: Tubes that are no longer operable or available are those that are plugged. Steam Generator A has 94 tubes plugged, B has 56 plugged, C has 104 plugged and D has 92 plugged. This amounts to 346 actual plugged tubes out of 22,504 tubes. Sleeved tubes result in some loss in heat transfer. Because the amount of heat transferred is a key

parameter, when the Electrosleeved tubes are taken into consideration, the equivalent tubes plugged total is 349. Electrosleeves and Westinghouse Laser Welded Sleeves are the two types of sleeves used at the Callaway Plant. During Refuel [outage] 10, 31 electrosleeves were installed in Steam Generator A and 26 in Steam Generator C. Previously, 44 Westinghouse Laser Welded sleeves were installed in Steam Generator A and 33 in Steam Generator C. [The licensee has] experienced no in-service problems with any installed plugs or sleeves.

Question B15: In what percent of the tubes in each generator has the licensee installed welded plugs at the inlet and outlet of the tubes, following the detection of tube wall degradation? Have any plugs been removed or been dislodged?

Licensee's Response: There are a number of plugs installed in the steam generators. Some are welded, others mechanical. Steam Generator A has 1.67% of the tubes plugged, Steam Generator B has 1.00%, Steam Generator C has 1.85%, and Steam Generator D has 1.64%. [The licensee has] experienced no in-service problems with any plugs or sleeves installed to date.

Question B16: Has the sleeving and plugging of the Callaway tubes caused any reduction in the Callaway steam generators' heat removal capability or the plant's power-generating capacity? If not, could you please explain why not?

Licensee's Response: The plant was designed to allow for a certain level of steam generator tube plugging. [The licensee is] currently licensed to operate with up to 10% of the tubes plugged in any steam generator. Each time a tube is removed from service, there is, of course, a very small effect on the total heat transfer ability of the steam generators and, subsequently, on plant output. However, the total effect does not become significant until a relatively large number of tubes are plugged. Sleeves typically have a much smaller effect. To date, [the licensee has] not seen any significant degradation of heat transfer capability.

Question B17: In addition to the tubes, another pathway exists for the primary coolant to reach the secondary coolant – that is, the thick "tube plate" (inside, at the bottom of each generator) through which the steam generator tubes penetrate. Already in the 1970s cracks were found in these plates in France, at Framatome reactors which are similar in design to the Callaway Westinghouse reactor.

Would you please tell me if the tube plate in each of the four Callaway steam generators is periodically inspected and tested to assess its structural and leak tight integrity? If so, would you please tell me: (a) when the most recent inspection occurred; (b) how it was performed – for example, how access was achieved for the destructive or nondestructive testing; and (c) if cracking or other signs of deterioration were detected?

Licensee's Response: The tubesheet in each steam generator is 21.23" thick. Each tube is hydraulically expanded into the tubesheet for the full length of the tubesheet. While we do not specifically inspect the tubesheet, a portion of the tubesheet is inspected during the steam generator eddy current inspection. Each outage, 100% of the tubes in two steam generators are inspected through the tubesheet with eddy current technology, which has the capability to see beyond the tube wall into the tubesheet. No tubesheet degradation has been identified.

Question B18: Recognizing the fact that the radiation fields within which workers have to make steam generator repairs rank among the highest at pressurized water reactor plants, would you please tell me whether replacing the steam generators rather than retrofitting them would perhaps have resulted in lower radiation exposure for the workers? And would that not be especially true if the electrosleeving will have to be removed (requiring more retrofitting and more worker exposure) after a maximum of only two operating cycles? What is to happen to the defective parent tubes within the generators after the sleeving is somehow removed?

Licensee's Response: On April 25, 2000, the Ameren Board of Directors approved the replacement of steam generators for Callaway Plant. This is scheduled for the fall of 2005. Although it is not considered likely, should the tubes with Electrosleeves installed require removal at the end of the two operating cycles, they will be removed from service by installing plugs at each end rather than removal of the actual electrosleeve. Plugging is a routine process and does not incur a great deal of worker radiation dose.

Staff's Response: Part 20 of Title 10 of the Code of Federal Regulations (10 CFR) requires licensees to plan work in radiation areas such that occupational exposure to workers is as low as is reasonably achievable. Therefore, the licensee would plan any work in radiation areas to reduce the occupational exposure.

Question B19: Has the NRC or AmerenUE estimated if and when the Callaway steam generators may have to be replaced?

Licensee's Response: On April 25, 2000, the Ameren Board of Directors approved the replacement of steam generators for Callaway Plant. This is scheduled for the fall of 2005.

Staff's Response: The NRC does not estimate if or when the steam generators at a plant may have to be replaced. This is a decision made by the licensee. The NRC enforces the requirements in Title 10 of the Code of Federal Regulations and in the plant Technical Specifications on the operation of the steam generators. There could come a time in the operation of a plant when the steam generators may soon not meet these requirements. Then, if the licensee wants to continue operating the plant, the steam generators would have to be replaced before the requirements were not met; however, as long as the steam generators can meet the applicable requirements, the licensee may operate the plant with the steam generators.

Question B20: In his April 7, 1999, presentation to the ACRS, the NRC's Steven Long reported on the Nuclear Energy Institute's proposal to replace the tube integrity criterion that limits a maximum permissible crack to 40% through the tube wall with a criterion that would allow through-wall cracks to remain in service. Has the NRC indeed decided to relax this criterion? If so: (a) Did you limit the permissible length of a through wall crack? (b) Did you limit the number of through-wall cracks in a given steam generator? (c) Did you agree to permit such cracks in the free-span portion of a tube, or only in areas not accessible for repair?

Staff's Response: The current requirements on permissible cracks in the Callaway steam generator tubes are in Section 5.5.9, "Steam Generator (SG) Tube Surveillance Program," of the Technical Specifications. In this section, the required imperfection depth at or beyond

which the tube shall be removed from service by plugging or repaired by sleeving is 40 percent. The required plugging limit for laser welded sleeves is 39 percent. The NRC is not considering any proposal from the licensee to relax the current criteria for Callaway.

Question B21: A non-steam generator question: knowing of concerns the NRC staff has raised intermittently about the possibility that waterhammer could exceed allowable piping stress at the Callaway plant, and remembering the continuing discovery during the plant's construction of defective stud welds on many embedded pipe supports, I would like to ask the following: was any of the series of February 13 sudden shutdowns that were caused by the voltage fluctuations sufficient to result in mechanical damage to any of the cooling water systems or other components?

Licensee's Response: The Callaway Plant Corrective Action Program requires that significant plant events be documented. The required documentation was written on the 2/13/00 trip. If the Occurrence involves an unexpected, potentially damaging Transient Event (e.g., a water hammer) to a system, our procedures require that engineering review the event and walk down the piping systems subject to the transient and examine accessible hanger and snubber structures for signs of damage. If the event is determined to be potentially damaging, then Transient Event Inspections are performed.

This plant trip was not classed as a Transient Event because the plant responded as designed, and there were no indications of abnormal waterhammer. Therefore, no piping and hanger walkdowns were conducted as a result of this plant trip.

Question B22: And one final question that I do not believe is related to the February 13 incident: according to the American Nuclear society's October 1999 Nuclear News, the Callaway plant workforce fabricates complex parts in its machine shop. "When the Callaway plant needs a spare part that is complex in design, we go to our machine shop and fabricate it ourselves. I don't know of other nuclear plants that do that as much as Callaway." (p. 24) I was surprised to read that. Reflecting back on the years when the plant was under construction, I remember that the NRC required vendors of safety-related equipment to follow elaborate quality assurance/quality control procedures and to keep thorough records. The NRC staff even occasionally made inspections at the vendors' factories. To what extent does the NRC oversee Callaway's in-house machine shop? And its documentation of purchases – such as of safety-related parts and metals? Or do you rely on oversight by the nuclear industry's Institute of Nuclear Power Operations?

Licensee's Response: It is the policy of AmerenUE to maintain an Operating Quality Assurance Program (OQAP) for Callaway Plant, as required by provisions of the Nuclear Regulatory Commission operating license and amendments thereto. AmerenUE has established an organization to implement the OQAP as documented in policies, manuals, and procedures. Specific OQAP requirements and corresponding organizational responsibilities are specified in the Operating Quality Assurance Manual (OQAM).

Fabrication of safety-related parts to support maintenance work, or provide spare parts to ensure reliability of safety-related plant equipment, is an activity controlled by the Operating Quality Assurance Program described above. Appropriate controls are documented in

Section 3.0 of the OQAM to address spare part design considerations and drawings. These controls include reviews for suitability of application, design verification, and technical reviews. Records of the foregoing activities, including procurements of raw materials and inspections, are maintained in compliance Section 17 of the OQAM.

In accordance with Section 18 of the OQAM, the Callaway Plant Quality Assurance Department implements a comprehensive audit program to ensure compliance with, and effectiveness of, the Operating Quality Assurance Program.

<u>Staff's Response:</u> The response has the following three parts:

Part 1 — To what extent does the NRC oversee Callaway's in-house machine shop?

Without a specific regulatory concern or licensee performance issue, the NRC does not conduct oversight reviews or inspections of the Callaway in-house machine shop. As an internal unit of the licensee's organizational structure, safety-related activities performed in the machine shop are subject to the requirements of Appendix B to 10 CFR Part 50 as implemented through the licensee's NRC-approved quality assurance (QA) program.

Part 2 — To what extent does the NRC oversee documentation of purchases (e.g., of safety-related parts and metals) by Callaway's machine shop?

The NRC does not routinely review or inspect procurement documentation for items fabricated in, or purchased by, the Callaway in-house machine shop. The licensee's QA organization is responsible for ensuring that machine shop safety-related activities, as well as all other safety-related activities performed by any organization on-site, are conducted in accordance with the requirements of Appendix B to 10 CFR Part 50.

Part 3 — Does the NRC rely on oversight by the nuclear industry's Institute of Nuclear Power Operations for Callaway's in-house machine shop safety-related activities?

The answer is no.

Attachments:

1. License Event Report 2000-02-01 (Revision 1), “Automatic Reactor Trip Initiated by Reactor Coolant Pump Trip Caused by Motor Current Imbalance Due to Transmission System Disturbance,” dated May 1, 2000.
2. Licensee’s 9th Steam Generator Tube In-service Inspection Report dated October 9, 1997
3. Licensee’s 10th Steam Generator Tube In-service Inspection Report dated April 28, 1999
4. Licensee’s 11th Steam Generator Tube In-service Inspection Report dated June 1, 2000
5. Argonne National Laboratory, “Technical Letter Report on Failure Prediction of Electrosleeved Tubes Under Severe Accident Transients,” dated October 4, 1999
6. Memorandum dated May 21, 1999, on “Revised Probablistic Safety Assessment Branch Input to Safety Evaluation Report on the Change to Technical Specifications at Callaway Plant to Allow Use of Framatone Electrosleeve Steam Generator Tube Repair Method”
7. SECY-99-246, "Proposed Guidelines for Applying Risk-Informed Decisionmaking in License Amendment Reviews," dated October 12, 1999
8. Regulatory Guide 1.174, “An Approach for Using Probablistic Risk Assessment in Risk-informed Decisions on Plant-Specific Changes to the Licensing Basis,” dated July 1998.

ATTACHMENTS

The following attachments were sent only to Ms. Kay Drey with the staff's second letter addressing the concerns in her letter of March 10, 2000. These attachments were not sent to any other party as part of the internal or external distribution of the letter to Ms. Drey. The following attachments can be found at their NUDOCS or ADAMS accession numbers given below.

1. License Event Report 2000-02-01 (Revision 1), "Automatic Reactor Trip Initiated by Reactor Coolant Pump Trip Caused by Motor Current Imbalance Due to Transmission System Disturbance," dated May 1, 2000. (Response to Question A6) (ADAMS Accession No. ML003713247)

2. Licensee's 9th Steam Generator Tube In-service Inspection Report dated October 9, 1997. (Response to Question B8) (NUDOCS Accession No. 9710170069)

3. Licensee's 10th Steam Generator Tube In-service Inspection Report dated April 28, 1999. (Response to Question B8) (NUDOCS Accession No. 9905050080)

4. Licensee's 11th Steam Generator Tube In-service Inspection Report dated June 1, 2000. (Response to Question B8) (ADAMS Accession No. ML003721791)

5. Argonne National Laboratory, "Technical Letter Report on Failure Prediction of Electrosleeved Tubes Under Severe Accident Transients," dated October 4, 1999. (Response to Question B11) (Attached to memorandum dated November 29, 1999, from Ashok C. Thadani to Samuel J. Collins)

6. Memorandum dated May 21, 1999, on "Revised Probabilistic Safety Assessment Branch Input to Safety Evaluation Report on the Change to Technical Specifications at Callaway Plant to Allow Use of Framatone Electrosleeve Steam Generator Tube Repair Method." (Response to Question B11) (NUDOCS Accession No. 9905260098)

7. SECY-99-246, "Proposed Guidelines for Applying Risk-Informed Decisionmaking in License Amendment Reviews," dated October 12, 1999. (Response to Question B11) (NUDOCS Accession No. 9910290235)

8. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-informed Decisions on Plant-Specific Changes to the Licensing Basis," dated July 1998. (Response to Question B11)

This page is to document where the above attachments can be found within NRC records. The page was only included in the ADAMS record of the staff's second letter to Ms. Drey and in the internal distribution of the staff's letter to show where these attachments could be found, but was not sent it to Ms. Drey or to Mr. Garry Randolph of Callaway plant, who received a copy of the letter sent to Ms. Drey. Mr. Randolph has access to copies of the above attachments separate from the NRC records, and does not need a copy.