

Enclosure 1

SALP 10

FINAL SALP REPORT

U.S. NUCLEAR REGULATORY COMMISSION
REGION III

SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE

Inspection Report No. 50-483/92001

Union Electric Company

Callaway Plant

February 1, 1990, through January 31, 1992

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Callaway Nuclear Plant

A. Summary of Meeting with Union Electric Company on April 22, 1992

The findings and conclusions of the SALP Board are documented in Report No. 50-483/92001 and were discussed with the licensee on April 22, 1992, at Steedman, Missouri.

While the meeting was primarily a discussion between the licensee and NRC, it was open to members of the public as observers.

The following licensee and NRC personnel were in attendance, as well as the noted observers.

Union Electric Company

Don F. Schnell, Senior Vice-President Nuclear
Garry L. Randolph, Vice-President Nuclear Operations
John D. Blosser, Manager Callaway Plant
Milton A. Stiller, Manager Nuclear Safety and Emergency Preparedness
Joe V. Laux, Manager of Quality Assurance

Nuclear Regulatory Commission

A. Bert Davis, Regional Administrator, RIII
William Forney, Deputy Director, Division of Reactor Projects
John Hannon, Director, Project Directorate III-3, NRR
Richard Hague, Chief, Reactor Projects Section 3C
Bruce L. Bartlett, Senior Resident Inspector, Callaway
L. Raynard Wharton, Project Manager, NRR
Desiree R. Calhoun, Resident Inspector, Callaway
Karen R. Marcus, Region III DRS Intern

Other

Tom Lang, State of Missouri Department of Natural Resources

B. Comments Received from Licensee

Union Electric Company's response to the Callaway Initial SALP 10 Report dated May 6, 1992, included several comments that have resulted in a minor revision to the Initial SALP Report. These changes are listed in Enclosure 2 and the revised pages are included as Enclosure 3.

The affected pages of the Initial SALP Report should be replaced with the corrected pages included in Enclosure 3.

REVISION SHEET

<u>PAGE</u>	<u>LINE</u>	<u>NOW READS</u>	<u>SHOULD READ</u>
3	4	...all seven at-power...	...all six at-power...
12	36	...and seven reactor...	...and six reactor...
Basis:			
The miscount of reactor trips was caused by the inclusion of a January 23, 1992, turbine trip which did not result in a reactor trip due to the low power level.			
10	2	...weaknesses were..	...a weakness was...
10	4-7	...weaknesses found were... ...analyses, the method...	...the weakness found was... ...analyses. (Delete remainder of sentence)

Basis:

The initial SALP report refers to three weaknesses in the licensee's MOV program. This input to the SALP report was provided prior to resolution of two of the cited weaknesses and issuance of Inspection Report 50-483/91020(DRS) which identified one weakness.

C. Regional Administrator's Conclusions Based on Consideration of Licensee Comments

With regard to the comments submitted in Union Electric Company's letter of May 6, 1992, concerning the outage dose, the safety injection pump inoperability, and the core alteration issues, we may not have clearly communicated our concerns to the Union Electric Company representatives in the initial SALP report or the SALP meeting.

The initial SALP report indicates that outage scheduling priorities contributed to "...the high dose in 1990..." My staff and I continue to believe that scheduling was a factor contributing to the 1990 dose. While we can not conclude that scheduling pressure was the primary contributor to increased dose for Refuel Outage 4, we believe that scheduling was a factor. I note that a Union Electric Company radiation exposure task team report dated February 26, 1991, similarly concluded "...it is probable that with a less aggressive schedule Callaway would have been able to reduce personnel exposure considerably during the Refuel 4..."

Concerning the safety injection pump inoperability, we acknowledge the aggressiveness of Union Electric Company engineers' handling of the mispositioned safety injection throttle valve, and we concur with the assessment of the minimal safety significance of this specific event. However, it is the view of the NRC that failures to make timely and correct operability declarations constitute a safety concern.

We concur that in the case of the specific core alteration mentioned in the initial SALP Report, there was no impact on safety. The event was highlighted because it involved a cognitive personnel error by a licensed Shift Supervisor which resulted in a violation of Technical Specifications.

I have concluded that the overall ratings in the affected areas have not changed.

Enforcement history in this area continued to be very good with no significant change in performance. Two Severity Level IV violations were identified during this period.

Random and isolated equipment failures caused all seven at-power reactor-trips during this assessment period. None of these reactor trips were caused by errors by licensed operators. Root causes were promptly determined and effective corrective actions implemented. No pattern or common root cause was discerned.

The number of events attributed to personnel errors decreased during this assessment period. No licensed operator personnel errors resulted in reactor trips. One of the events involved a core alteration with less than the required number of source range monitors operable.

Plant operations continued to be sustained at a high level of performance. Professionalism, communications, and team work with groups, both inside and outside of the operations department were evident throughout most plant evolutions. The operating crews, with minor exceptions, maintained a close cognizance of the plant, control room boards, and operating conditions. Members of the operating crews demonstrated a well executed, disciplined response to unplanned events. Planning and scheduling of routine evolutions were implemented in such a manner that perturbations were minimized.

Operating history continued to improve. Even though there were seven reactor trips from power, the prior record of continuous days on line was exceeded. The licensee instituted a program to identify and eliminate sources of single-failure-point trip vulnerabilities in the non-safety-related portions of the plant.

Management effectiveness in ensuring quality was excellent and was evident throughout all aspects of plant operations. Standards of performance directed towards the safe, efficient operation of the facility were communicated to all members of the plant staff. Management involvement in root-cause analysis and performance of corrective actions helped to ensure that problems were identified in a timely manner and did not recur. Shutdown risk management was good in that industry experience was incorporated.

Staffing levels and qualifications were good and personnel were dedicated and knowledgeable. Use of overtime was maintained within NRC guidelines.

The effectiveness of the licensee's training and qualification programs was good. The pass rate on initial operator license examinations increased from the last rating period; excellent performance in the requalification examination program was maintained.

General housekeeping during normal operations was good but declined during refueling outages.

2. Performance Rating

Performance is rated Category 1 in this area. Performance was rated Category 1 during the previous assessment period.

room. While conservatism was exhibited in the resolution of most problems, weaknesses were found in the motor operated valve (MOV) program (Generic Letter 89-10). Although no equipment operability concerns were identified, weaknesses found were the use of nonconservative power factors for MOV degraded voltage analyses, the method used to determine differential pressure values in MOV calculations, and the failure to evaluate the effect of high ambient temperatures on the performance of MOV motors.

Staffing levels were good and overtime was well controlled. The staff of the systems, project, and design engineering organizations were knowledgeable and experienced. Turnover remained low, and a technical career advancement path was implemented, comparable to that available to managers.

The training and qualification effectiveness for licensed personnel was good as demonstrated by the knowledge and ability of the licensed operators. Weaknesses identified during an operator licensing examination were corrected before administration of the next examination. There was a significant commitment of facilities and competent instructors to train personnel working on the implementation of the MOV program.

2. Performance Rating

Performance is rated Category 1 in this area. Performance was rated Category 1 in the previous assessment period.

3. Recommendations

None.

G. Safety Assessment/Quality Verification

1. Analysis

Evaluation of this functional area was based on the results of routine inspections by resident and regional inspectors and special team inspections. In addition, licensee requests for amendments, exemptions or relief, responses to NRC generic communications, and other interactions with the NRC staff were considered.

Enforcement history in this functional area was excellent with no violations identified.

Management effectiveness in ensuring quality was mixed. Management actively supported self-assessment efforts as demonstrated by the performance of detailed safety system functional assessments (SSFAs) of the essential service water (ESW), residual heat removal (RHR), and vital electrical systems. Substantial corporate resources were committed this assessment period to the ongoing comprehensive corrective actions responding to program problems identified by the ESW SSFA conducted the previous assessment period. Plant management actively followed up on the findings of all major self-assessment efforts, including the SSFA of the RHR system. Completion of corrective actions in response to these efforts was routinely tracked. The vital electric SSFA was completed at the end of this assessment period and corrective actions were being reviewed.

During the assessment period, the licensee requested and the NRC granted several temporary waivers of compliance from TS requirements. These generally involved brief extensions to LCO action statements in complete repairs and/or confirm equipment operability. Sound technical justifications were provided. In one instance, the licensee discovered an erroneous TS surveillance requirement for diesel generator load reject testing. Although the licensee concluded that appropriate surveillances were being conducted, it properly determined that a waiver and TS change were needed.

The 10 CFR 50.59 reviews performed in conjunction with design modifications were thorough and well documented. QA oversight of the modification process was evident and effective. Evaluations of 10 CFR Part 21 reportability requirements were acceptable, with one exception of an isolated failure to make a timely notification that occurred early in the assessment period.

Reportable events remained at a low level throughout the assessment period. The quality of LERs was good and root-cause evaluations and corrective actions were thorough. In general, timeliness requirements were met with the one exception of the LER of a design error in a modification to the steam generator low-low-level trip time delay circuitry, the licensee took an excessively long time to determine that the event was reportable. This was considered to be an isolated occurrence.

Staffing levels in the QA and QC organizations were appropriate and sufficient for the existing workload. QA personnel were technically competent. QA and QC supervisors and staff were very knowledgeable.

2. Performance Rating

Performance is rated Category 1 declining in this area. Performance was rated Category 1 in the previous assessment period.

3. Recommendations

None.

IV. SUPPORTING DATA AND SUMMARIES

A. Licensee Activities

The Callaway Plant operated routinely throughout the majority of the SALP assessment period (except for short duration power reductions and outages for maintenance/surveillance activities and equipment repairs). The plant was shutdown from August through October 1990 for its scheduled Cycle 4 refueling outage.

The Callaway Plant experienced nine engineered safety feature actuations and seven reactor trips. All reactor trips occurred above 15 percent power.

Enforcement history in this area continued to be very good with no significant change in performance. Two Severity Level IV violations were identified during this period.

Random and isolated equipment failures caused all six at-power reactor-trips during this assessment period. None of these reactor trips were caused by errors by licensed operators. Root causes were promptly determined and effective corrective actions implemented. No pattern or common root cause was discerned.

The number of events attributed to personnel errors decreased during this assessment period. No licensed operator personnel errors resulted in reactor trips. One of the events involved a core alteration with less than the required number of source range monitors operable.

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Management effectiveness in ensuring quality was excellent and was evident throughout all aspects of plant operations. Standards of performance directed towards the safe, efficient operation of the facility were communicated to all members of the plant staff. Management involvement in root-cause analysis and performance of corrective actions helped to ensure that problems were identified in a timely manner and did not recur. Shutdown risk management was good in that industry experience was incorporated.

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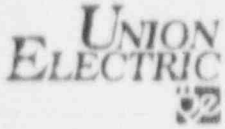
None.

IV. SUPPORTING DATA AND SUMMARIES

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May 6, 1992

Donald F. Schnell
Senior Vice President
Nuclear

Mr. A. Bert Davis
Regional Administrator
U. S. Nuclear Regulatory Commission
Region III
799 Roosevelt Road
Glen Ellyn, IL 60137

ULNRC-2634

Dear Mr. Davis:

INITIAL SALP 10 BOARD REPORT NO. 50-483/92001
CALLAWAY PLANT

This responds to your letter dated March 30, 1992, which transmitted the Initial SALP 10 Board Report for Callaway Plant covering the period February 1, 1990 through January 31, 1992.

We have reviewed the report and have the following comments and clarifications:

A. Page 3, paragraph 2:

We experienced six at-power reactor trips during this assessment period. The error in this paragraph may be due to counting the 6/12/90 manual trip in this group. It actually occurred when subcritical.

B. Page 3, paragraph 3:

The core alteration described in this paragraph was an attempt to remove a reactor vessel specimen. These specimens are located outside the reactor core barrel. The potential reactivity change during this operation is too small to measure. Literally speaking, moving a vessel specimen can be interpreted as a core alteration, but the report should characterize the operation as having no impact on safety.

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Mr. A. Bert Davis

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C. Page 4, paragraph 5:

We acknowledge that total person-rem dose during Refuel IV was higher than expected. However, it is inaccurate to say that outage scheduling pressure was a primary contributor to increased dose. As pointed out in our April 22 meeting, we terminated RTD bypass piping decontamination efforts simply because the hydrolazer didn't work as expected. The greater-than-expected dose accumulated during ISI might have been reduced through better work planning as job problems were encountered. We have since instituted a programmatic requirement to conduct in-progress dose assessments during the performance of jobs with high dose potential. Again, we do not believe schedule pressure was a primary contributor.

Our experience during Refuel IV prompted a number of initiatives to improve ALARA performance. As we discussed with you, we have:

1. initiated design changes to reduce dose fields;
2. reduced source through an improved methodology for RCS chemical shock;
3. improved planning for work in containment through the use of a revised containment grid location system;
4. increased accountability for ALARA through the establishment of an outage review board; and
5. increased work productivity through the use of special tools and specialized vendors.

Experience in the current outage indicates significant savings in dose have occurred, and such benefits will continue to accrue in the future.

D. Page 10, paragraph 1:

The only weakness identified in the MOV program documented in inspection report 50-483/91020, was our use of a non-conservative power factor in degraded voltage analyses. Based on discussions in our April 22 meeting, we understand the SALP report will be revised to correct this paragraph.

Mr. A. Bert Davis

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E. Page 11, paragraph 1:

We are troubled by the examples used to demonstrate that management effectiveness in insuring quality was mixed. As pointed out in our meeting, estimated critical position errors were first recognized in cycle 4. A task team evaluated these errors, initiated improvements and verified that they did not represent a nuclear safety concern. Estimated critical positions calculated during startup for cycle 5 and after the December 30, 1990 trip showed good agreement between actual and predicted ECPs. The next indication of a problem in ECP prediction did not occur until recovery from the November 1991 trip. Prior to this event, we had no indication that cycle 4 improvements had not resolved the ECP prediction error. Some months earlier, we had formed a cycle 5 task team to evaluate the cause of axial flux responses which did not match those predicted. The ECP error was linked with the axial flux response and addressed by that team. Our cycle 5 team systematically investigated both problems, providing input to revised operating limits and assuring these problems were not a nuclear safety concern.

Regarding our reactor trip reduction efforts, we agree that results over the two-year SALP period are less satisfactory than expected. As you point out on page 3, however, no pattern or common root cause for the trips was discerned and none were caused by licensed operator error. Our program has initiated many improvements that have reduced the probability of reactor trips, and we acknowledge the need to continue efforts in this area.

We have already discussed the assertion that schedule pressure resulted in increased dose during the 1990 refueling outage.

F. Page 11, paragraph 2:

Your belief that sufficient evidence of a mispositioned throttle valve existed to make an earlier declaration of safety injection pump inoperability is a matter of opinion. We disagree with your characterization of this situation as a "less than conservative approach to safety" due to the timing of a declaration of inoperability. We believe this overlooks the depth, scope and aggressiveness of the review effort performed to identify the actual facts in this situation. The concern involving safety injection flows was identified by our engineering personnel during a related procedure development effort. Resolution of the concern was aggressively pursued

Mr. A. Bert Davis

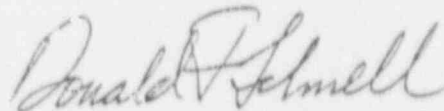
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over a 72-hour period. The investigation involved reviews of records over several refueling outages and development and implementation of a comprehensive action plan which ultimately identified a mispositioned valve. A parallel effort evaluated the nuclear safety impact of the safety injection flow unbalance and concluded that the difference in flow would have no impact on the ability of the system to carry out its safety function. An earlier declaration of inoperability would have shortened our period of noncompliance with Technical Specifications, but it would not have hastened resolution of the issue from a nuclear safety perspective.

We appreciate the opportunity to discuss the items noted here with members of the Region III staff and representatives of NRR during our SALP meeting on April 22, 1992. We appreciate your recognition of our performance and acknowledge those sections of the report which identify areas for increased attention and improvement. Your critique of our operation and support of Callaway and our discussion of your observations will help us continue to improve the safety and performance of the plant.

Very truly yours,



Donald F. Schnell

DFS/JCG/lkr

cc: R. L. Hague - Chief, Reactor Projects Section 3C, USNRC Region III
L. R. Wharton - USNRC Licensing Project Manager (2 copies)
USNRC Document Control Desk
Manager - Electric Department, Missouri Public Service Commission
B. L. Bartlett - USNRC Senior Resident Inspector
Shaw, Pittman, Potts & Trowbridge

Mr. A. Bert Davis

Page 5

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bcc: A160.0761 (QA Record) (CA-460)
A160.0411 (92001) Commercial Record (CA-460)
A160.0411 (92001 - File/Logging) (QAQS CA-460)
Nuclear Date File (S. L. Dale) (470)
E210.0001
D. F. Schnell (Chrono)
G. L. Randolph
J. V. Laux
Licensing and Fuels (A. C. Passwater/D. E. Shafer) (470)
T. P. Sharkey
NSRB (S. L. Dale) (470)
M. P. Barrett (100)
K. K. Chernoff (WCNOC)
C. S. Petzel