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Donald F. Schnell
Vice President

January 14, 1986

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Denton: ULNRC-1242

DOCKET NUMBER 50-483
CALLAWAY PLANT
REQUEST FOR ADDITIONAL INFORMATION
REGARDING FUNCTIONAL AND TASK ANALYSIS

Reference: Youngblood to Schnell letter dated November 4, 1985

The referenced letter transmitted the subject request for additional information. The attachment provides Union Electric's response to this request.

Very truly yours,

Donald F. Schnell

Donald F. Schnell

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Attachment

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STATE OF MISSOURI)
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CITY OF ST. LOUIS)

Donald F. Schnell, of lawful age, being first duly sworn upon oath says that he is Vice President-Nuclear and an officer of 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 Donald F. Schnell
Donald F. Schnell
Vice President
Nuclear

SUBSCRIBED and sworn to before me this 14th day of January, 1986.

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NOTARY PUBLIC, STATE OF MISSOURI
MY COMMISSION EXPIRES APRIL 22, 1989
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Attachment to ULNRC-

UNION ELECTRIC RESPONSE TO THE
NRC CONCERNING FUNCTIONAL
AND TASK ANALYSIS

REQUEST 1

Demonstrate that the task analysis based on Revision 1 of the Emergency Response Guidelines (ERGs) is applicable to Callaway.

RESPONSE

The Callaway Emergency Operating Procedures (EOPs) follow the generic ERGs in format and identification. (See Response to Request 5).

REQUEST 2

Modify the Procedures Generation Package (PGP) to state that the task analysis which supported the Emergency Operating Procedures (EOP) Upgrade Program was described as part of the Detailed Control Room Design Review (DCRDR).

RESPONSE

The PGP (APA-ZZ-00102) has been modified to indicate the task analysis, which was done as part of the DCRDR, is used in the EOP Update Program.

REQUEST 3

Describe and justify the deviations from Revision 1 of the ERGs indicated in the Task Analysis Final Report, Findings 1, 6, 8, 9, and 10.

RESPONSE

Finding 1 ECA-0.0, step 23 requires BIT temperature indication in the control room. None is provided.

JUSTIFICATION

Per ERG background document, monitoring BIT temperature for solubility limitations is only a concern for systems having Boron concentrations greater than 7000 ppm. The reduction of BIT Boron concentration from greater than 20,000 ppm at the reference plant to 2000 ppm at Callaway is addressed in SLNRC 84-0070 dated April 17, 1984. It is also reflected in FSAR Sections 6.3.2.2 and 15.3.

ERG ECA-0.0, Loss of all AC Power, Step 23a which required the operator to check BIT temperature was deleted in ECA-0.0 due to the reduced Boron concentration.

CONCLUSION

BIT Boron solubility is not a concern at Callaway, therefore, the need to monitor BIT temperature and the need for control room indication is not necessary. It does not constitute a safety-significant deviation from Revision 1 of the ERGs.

Finding 6 UE procedure E-1, step 13A requires operator action at greater than 535 GPM. Control room indicators are graduated in increments of 100 GPM. Therefore, this value of 535 GPM cannot be read accurately.

JUSTIFICATION

E-1, Rev. 2 used RHR pump recirculation valve automatic closure at 535 GPM to be indication of RHR flow to RCS. The procedure was changed to use a calculated value of 550 GPM (which includes instrument errors) as read on EJ FI-618 or EJ FI-619. This change results in positive indication of minimum RHR injection flow to the RCS.

The Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERG) Procedure ERG E-1 step 13A directs operator action after verifying RHR injection into the RCS. Callaway Emergency (EOP) procedure E-1, step 13A Revision 1 requires operator action after verifying RHR pump flow to the RCS to be greater than 550 gpm, which indicates flow to RCS. The operator action is the same so there is no deviation from WOG ERG guideline actions.

CONCLUSION

This remains within the WOG ERG and is not a deviation.

Finding 8 The ERG background documentation for FR-C.1, Step C-1b, lists CCW to RHR heat exchanger flow as an instrumentation requirement. No instrument for this exists in the control room.

JUSTIFICATION

The WOG ERG FR-C.1 cautions the operator to verify that the RHR pumps are not operated longer than a specified time without CCW flow to RHR heat exchanger to prevent pump damage. Callaway EOP FR-C.1 contains the same caution. Acceptable alternatives exist for the indication of CCW flow. Control room annunciators 51A and 53A alert the operator to HI/LO CCW flow conditions. RHR inlet/outlet temperature indication across the heat exchangers is an acceptable indication of CCW flow to the heat exchangers, and is available in the control room. In addition, CCW to RHR heat exchanger flow indication is available locally and on the BOP CRT located in the control room. Since the reactor operator has

adequate CCW flow information avai ble, the actions of the FR-C.1 caution remain within the WOG ERGs.

CCW to RHR heat exchanger flow indication is classified as backup plant instrumentation per the Instrumentation Section of the Generic Issues portion of the Executive Volume. Backup plant instrumentation, as defined in Generic Issues, is not required to meet the stringent design, qualification and display requirements of key plant instrumentation. For example, backup instrumentation is not required to be redundant, powered from a highly reliable source, and is not needed to be either accessible on demand or recorded. Therefore, the instrumentation used in the EOP to verify CCW flow to the RHR heat exchanger meets the ERG criteria and is not an instrument and control deviation.

CONCLUSION

Since CCW flow information is available to the operator and the actions and instrumentation remain within the WOG ERGs, this is not a deviation.

Finding 9 ERG background documents for eight of the ERGs list CCW Flow to Seal Water Heat Exchanger as an information requirement. No instrumentation for this information is provided in the control room.

JUSTIFICATION

The eight EOPs that were referenced in the finding are listed below with the WOG equivalent procedure and step cross-referenced.

1. EOP FR-I.1, Response to Pressurizer Flooding, Step 4 (ERG FR-I.1, Step 2)
2. EOP E-3, Steam Generator Tube Rupture, Step 34 (ERG E-3, Step 34)
3. EOP ES-11, Post-LOCA Cooldown and Depressurization, Step 26 (ERG ES-1.2, Step 26)
4. EOP ES-03, SI Termination, Step 16 (ERG ES-1.1, Step 16)
5. EOP C-21 Uncontrolled Depressurization of ALL Steam Generators, Step 27 (ERG ECA-2.1, Step 27)
6. EOP C-31, SGTR with Loss of Reactor Coolant-Subcooled Recovery, Desired, Step 31 (ERG ECA-3.1, Step 31)
7. EOP C-32, SGTR with Loss of Reactor Coolant-Saturated Recovery Desired, Step 25 (ERG ECA-3.2, Step 25)
8. EOP C-33, SGTR Without Pressurizer Pressure Control, Step 19 (ERG ECA-3.3, Step 18)

These steps reference CCW flow to the Seal Water Heat Exchanger. This indication is also classified as backup plant instrumentation. Acceptable alternatives exist in the Control Room for indication of proper CCW flow. A check of service loop flow on EC-FI-55A (of which the seal return heat exchanger is a part), CCW to and from service loop valve positions, and the absence of annunciator 54F "CCW Seal HX Flow HILO" are sufficient to assure that proper CCW flow thru the seal return heat exchanger exists.

CONCLUSION

The ERG criteria has been met and this finding is not a deviation.

Finding 10 The background document for ERG E-3, Step 2, lists steamline radiation monitors as an instrument requirement. None is provided in the SNUPPS control room.

JUSTIFICATION

The intent of the Emergency Response Guidelines is to utilize steamline monitors as one possible means to identify which steam generator(s) have ruptured tubes (this is the "purpose" for E-3, step 2). The Executive Volume and Background Documents allow for alternate instruments, such as the steam generator level indication (for larger leaks) or the sampling system (effective for smaller leaks). Per the Generic Instrumentation Section of Generic Issues portion of Executive Volume, only two channels of secondary radiation detection are necessary. Callaway procedure EOP E-3, step 2, utilizes SG blowdown monitors and SG sample monitor, and high radiation from any steamline.

Therefore, the two channel criteria is met, and one backup method of determining SG radiation is provided.

CONCLUSION

Because the ERG lists the steamline radiation monitors as one of several options, and because we meet the two channel criteria, steamline radiation monitors are not an instrument requirement. Therefore, a safety-significant deviation from WOG guidelines does not exist.

REQUEST 4

Review the method used for identifying deviations and describe and justify all potentially safety-significant deviations not identified in Items 1 and 3 above.

RESPONSE

The method for identifying deviations from generic instrumentation and control characteristics was submitted via SLNRC 85-11

dated April 1, 1985. The actions and information requirements were developed independent of existing control room instrumentation and utilized WOG ERGs Revision 1. The results of this review are documented in SLNRC 85-012 dated April 26, 1985 and clarified by SLNRC 85-016 dated May 24, 1985. The findings that resulted consisted of human factor findings which address instrumentation and control characteristics but are not necessarily deviations from the generic guidelines.

A review of the EOP's and background material was conducted to identify potentially safety-significant plant-specific technical deviations from the WOG ERGs. The following criteria was utilized during the review:

1. Plant-specific steps which differ from the WOG HP Rev. 1 reference plant procedure steps were not considered to be deviations if they agreed with step conversion guidance of the background document or generic issues section of the administrative volume.

2. Control and instrumentation criteria was reviewed only for those cases where differences existed. This control and instrumentation criteria was then reviewed to ensure that the ERG criteria was still adhered to.

3. The following were not considered as deviations because APA-ZZ-00102 specifically exempts them:

   a) Level of detail.

   b) Rewording to conform to standard Callaway Procedure Terminology.

4. Setpoints were not reviewed because they underwent independent review during the plant specific procedure generation process.

The potential safety-significant deviations that resulted from this review are described below. After reviewing these, it was determined that no safety-significant deviations exist.

ITEM 1

DESCRIPTION

The option to use the procedure without RVLIS was added to ten (10) EOP's. The affected procedures are listed below.

EOP ES-0.4, Natural Circulation Cooldown with Steam Void in Vessel (without RVLIS), Symptom (ERG ES-0.4)
EOP ECA-1.1, Loss of Emergency Coolant Recirculation, Step 18a (ERG ECA-1.1)
EOP ECA-3.1, SGTR With Loss of Reactor Coolant-Subcooled Recovery Desired, Step 2b of Foldout (ERG ECA-3.1)

EOP ECA-3.2, SGTR With Loss of Reactor Coolant-Saturated Recovery Desired, Step 20a and RNO, Step 1 & 2b of Foldout (ERG ECA-3.2)
EOP ECA-3.3, SGTR Without Pressurizer Pressure Control, Step 7c, Step 11, Steps 1 & 2b of Foldout (ERG ECA-3.3)
EOP FR-C.1, Response to Inadequate Core Cooling, Step 6 (ERG FR-C.1)
EOP FR-C.2, Response to Degraded Core Cooling, Steps 6 (ERG FR-C.2)
EOP FR-P.1, Response to Imminent Pressurized Thermal Shock Condition, Steps 5 & 12 (ERG FR-P.1)
EOP FR-I.3, Response to Void in Reactor Vessel, Steps 8a & RNO, Steps 10a & RNO (ERG FR-I.3)

Each procedure was modified to allow the reactor operator to perform the procedure step with or without RVLIS.

JUSTIFICATION

The ERG's provide guidance for developing procedures for using RVLIS or procedures if RVLIS is not installed. Since the ERG's contained guidance for plants without RVLIS, Plant Management made the decision to develop Contingency Actions for the case when RVLIS may not be operable. These Contingency Actions were developed using the ERG's.

This effort provides additional information to the operator and does not detract from the plant specific EOP's.

CONCLUSION

Providing guidance for the condition when RVLIS is not operable does not constitute a safety-significant deviation.

ITEM 2

DESCRIPTION

The reset of SI has been added to EMG ES-0.4, Natural Circulation With Steam Void In Vessel (Without RVLIS), Step 1. This is a potentially safety significant deviation from the generic guidelines due to a plant design difference.

JUSTIFICATION

The step was added to ensure undervoltage relays do not actuate when the reactor coolant pump(s) are started. This reflects a commitment regarding Confirmatory Issue #18 in SLNRC 83-006 dated February 2, 1983. Resetting the SI signal prior to an attempt to start RCPs C or D will reset the SI output relays and the immediate undervoltage trip is removed from the offsite power breaker control circuits.

CONCLUSION

This is not a safety-significant deviation.

ITEM 3

DESCRIPTION

A foldout in the generic guidelines which addressed RWST switchover criteria has been deleted in EOP ES-1.3, Transfer to Cold Leg Recirculation Following Loss of Reactor Coolant.

JUSTIFICATION

The foldout was deleted to avoid confusion since RWST switchover to cold leg recirculation must occur prior to initiating hot leg recirculation.

CONCLUSION

The deletion of the foldout step that addressed RWST switchover criteria is not a safety-significant deviation.

ITEM 4

DESCRIPTION

The phrases "Rod bottom light-lit" and "Rod position indicators-at zero" in ERG ECA-0.0, Loss of all AC Power, Step 1 were deleted in EOP ECA-0.0.

JUSTIFICATION

The WOG Guidelines instruct the operator in ECA-0.0, Step 1 to verify reactor trip by the following:

- Rod bottom lights-lit
- Reactor trip and bypass breakers open
- Rod position indicators-at zero
- Neutron flux-decreasing

In the SNUPPS design, the rod bottom lights indicator and rod position indicator is the same indication. Also, upon loss of AC power, this indication is deenergized. Reactor trip is adequately verified by in EOP ECA-0.0, Step 1 by verifying the reactor trip and bypass breakers are open and decreasing neutron flux.

CONCLUSION

This is not a safety-significant deviation.

ITEM 5

This item is discussed in the response to Request 4 under Finding 1.

REQUEST 5

Provide a cross-reference of the Callaway EOPs to Revision 0 and Revision 1 of the ERGs and identify each step of the eight EOPs given in Enclosure 2 that lists the Component Coolant Water Flow to the Seal Water Heat Exchanger as an instrumentation requirement.

RESPONSE

Comparison of Emergency Procedures

GENERIC ERG		Wolf Creek	Callaway
a.	FR-I.1	FR-I.1	FR-I.1
b.	E-3	E-3	E-3
c.	ES-1.2	ES-11	ES-1.2
d.	ES-1.1	ES-03	ES-1.1
e.	ECA-2.1	C-21	ECA-2.1
f.	ECA-3.1	C-31	ECA-3.1
g.	ECA-3.2	C-32	ECA-3.2
h.	ECA-3.3	C-33	ECA-3.3

All the referenced procedural steps have been reviewed in both the Callaway EOP's and generic ERG's. All steps that are questioned by the NRC involve plant specific steps for system restoration. The generic ERG considers these items to be plant specific, thus the instrumentation requirement for CCW flow to the seal water heat exchanger is not identified in the generic ERG step.

REQUEST 6

Describe the indications, other than steam generator water level, that the operator will use to identify the steam generator with a ruptured tube.

RESPONSE

Operators at Callaway, in addition to observing SG water level, will utilize the following to identify the faulted steam generator(s):

1. Check for abnormal radiation from any of the following:

   a. High Steamline Radiation

   b. Inline Steam Generator Blowdown Monitor for abnormal radiation levels by utilizing one generator at a time.

   c. Inline Steam Generator Sample Radiation Monitor for abnormal radiation levels by utilizing one generator at a time.

2. If actions and indications of Item 1 do not positively identify the ruptured steam generator, then the operator is directed to re-establish SG sample and to request chemistry to obtain a grab sample of the most suspect steam generator, followed by grab samples of all other steam generators.

Operator action times have previously been provided in SLNRC 84-044 dated March 16, 1984 and SLNRC 84-129 dated December 3, 1984.

Main Steam Rad Monitors

AB	RE-114	'A' SG PORV plume monitor (M-12AB01/1)
AB	RE-113	'B' SG PORV plume monitor (M-12AB01/1)
AB	RE-112	'C' SG PORV plume monitor (M-12AB01/1)
AB	RE-111	'D' SG PORV plume monitor (M-12AB01/1)

SG Blowdown

BM RE-25	SG Blowdown non-regenerative heat exchanger outlet (M02BM02/11)
BM RE-52	SG Blowdown surge TK outlet to liquid radwaste discharge header (M02BM04/5)

SG Sample

SJ RE-2	SG sample downstream of the sample isolation valves (solenoid operated) and sample flow indicators. Note: Sample can be individually restored to determine ruptured SG. Also, grab sample may be drawn for analysis.

A revision to the Task Analysis Final Report is not required.