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October 31, 1994

U.S. Nuclear Regulatory Commission
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Gentlemen:

ULNRC- 3090
TAC NO. M90154

DOCKET NUMBER 50-483
CALLAWAY PLANT
BORON DILUTION REANALYSIS

References: 1. ULNRC-3051 dated 8-4-94
2. NRC Request for Additional Information (RAI) Letter dated 10-4-94

Reference 1 submitted an amendment application to reflect the reanalysis of the Inadvertent Boron Dilution event in FSAR Section 15.4.6. NRC Staff requested additional information to complete their review via Reference 2. The attachment to this letter provides the requested information.

If you have any questions on the attachment, please contact us.

Very truly yours,

Al Passwater
for Donald F. Schnell

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Attachment: Response to NRC RAI Letter

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Alan C. Passwater, of lawful age, being first duly sworn upon oath say~ that he is Manager, Licensing and Fuels (Nuclear) for Union Electric Company; that he has read the foregoing document and knows the content thereof; that he has executed the same for and on behalf of said company with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By Alan C Passwater
Alan C. Passwater
Manager, Licensing and Fuels
Nuclear

SUBSCRIBED and sworn to before me this 31st day of October, 1994.

Barbara J. Pfaff

BARBARA J. PFAFF
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NRC Question:

1. Please describe how the Callaway plant-specific ICRR versus boron curve was obtained and how is it verified that this curve is bounding for the entire cycle, as well as for future cycles.

Response:

The Callaway plant-specific ICRR versus boron concentration curve was obtained during routine 1/M plots for the last startup performed by diluting to critical (Cycle 5). The startups for Cycles 6 and 7 were performed by withdrawing rods to critical. It was prudent to use the Cycle 5 ICRR data for a mature core most similar to current loading patterns that was obtained by diluting to critical.

The ICRR curve is based on startup data taken with the highest boron concentrations present during the cycle. While it is true that as boron concentration decreases to zero at the end of the cycle, thereby removing the neutron capture reaction and changing the relative proportion of reactivity variant vs. secondary source neutrons seen by the source range detectors, (the importance of which is discussed at the top of page 2 and in response to NRC Question #2), it is also true that BOL conditions represent the worst case for the effects of an Inadvertent Boron Dilution event. As such, it is felt that the ICRR curve used in the analysis should reflect the time in life where the accident consequences are greatest.

The following discussion addresses the assurance, by experience, that the Cycle 5 ICRR data will be applicable for future cycles.

As a part of the various communications between TU Electric and the NRC (meetings and letters of correspondence), issues related to the analysis of the Inadvertent Boron Dilution event were discussed. From the viewpoint of the Comanche Peak Plant, obtaining acceptable solutions to these issues was quite bleak in that nothing attempted yielded acceptable analysis results. The "fixes" considered by TU Electric included a relocation of the secondary sources, a reduction in the instrumentation setpoint to an acceptable level, and the use of a cycle-specific source range detector response. Each of these individually and all of them collectively did not produce acceptable analysis results for TU Electric. This, however, is not true for Callaway. Despite the various analysis concerns during the past years, acceptable licensing-basis Inadvertent Boron Dilution analysis results have always been obtained for Callaway.

The validity of ICRR data used in the analysis of the licensing-basis Inadvertent Boron Dilution event in comparison to the plant-specific data taken during plant startup is critical to confirmation of the acceptability of the safety analysis

results. The response of the neutron count rate to changing boron concentration in the moderator directly affects the response time of the Boron Dilution Mitigation System (BDMS). A very large change in the count rate with little change in boron concentration causes a steep slope in the ICRR data and little time for mitigation action following the flux-multiplication signal. A gradual slope with a linear relationship between the ICRR data and boron concentration provides for acceptable analysis results with respect to the acceptance criterion; such a relationship is based primarily on the secondary source vs. source range (SR) detector geometry and the core configuration being such that the source range detector response is dominated by reactivity variant neutrons during the approach to criticality.

The ICRR data typical of a first core (which has no active secondary sources), or a core with the secondary sources on the outer row, or a core with fresh fuel within the outboard assemblies will present itself as being very non-linear with increasing slope as criticality is approached. Such data is not conducive towards obtaining acceptable boron dilution safety analysis results. However, ICRR data typical of a mature core with the secondary sources located in assemblies in core locations H-3 and H-13 such as at Callaway (assemblies in the third row deep) and burned fuel in the outboard assemblies exhibits a linear relationship with changing boron concentration. Such ICRR data is not dominated by the secondary source neutrons and yields acceptable boron dilution safety analysis results with respect to the available mitigation time.

Prediction of the plant-specific ICRR data is based solely on the experience gained in plant operations from cycle to cycle and is not derived from any calculational basis. The designed core loading patterns for Callaway have been consistent from cycle to cycle with the locations of the secondary sources being constant and the outboard assemblies between the source and the detector being previously burned. Based on this experience, the observed linear relationship of the Callaway ICRR data as a function of boron concentration, and the loading pattern design constraints delineated below, it is expected that the safety analysis results of the Inadvertent Boron Dilution event will remain valid as long as future loading patterns remain consistent. Based on this observation, there is no need to reproduce the ICRR data for future cycles to validate the assumptions in the boron dilution safety analysis.

The current safety analysis for an Inadvertent Boron Dilution event at Callaway assumes that the ICRR curve measured during the Callaway Cycle 5 startup remains applicable. The ICRR curve relates the source range detector response to a decrease in soluble boron concentration in order to determine how much time will elapse between the flux-multiplication signal and an inadvertent criticality. Currently, Westinghouse performs a qualitative evaluation of each loading pattern to determine if this analysis assumption is still valid for any given cycle. The use of Cycle 5 ICRR data for future cycles is acceptable

provided that each cycle-specific loading pattern continues to meet these criteria as identified below:

- Secondary source components should be in core locations H-3 and H-13. The reason for this is to be consistent with the Cycle 5 loading pattern, and to ensure that the source range detector signal is an indicator of core reactivity and not secondary source strength.

- There should be no feeds in rows 1 and 15. The reason for this is again to be consistent with Cycle 5 and to ensure that the feeds do not provide a strong source to the detector, such that the detector response is driven by spontaneous fission sources in the adjacent burnt fuel rather than by core reactivity. The placement of low reactivity once or twice burnt fuel in these locations is recommended, and is consistent with good loading pattern economics.

- There should be no feeds under the source components in core locations H-3 and H-13. Again, this is consistent with Cycle 5, and will help to ensure that the detector response is not dominated by one localized source of neutrons. This requirement does not exist for new secondary sources or outage lengths greater than 180 days.

Loading patterns that meet the above criteria will provide assurance that the cycle-specific ICRR data will be sufficiently similar to the Cycle 5 ICRR data, and therefore that the conclusions reached for the current licensing-basis boron dilution event analysis in Modes 3, 4 and 5 will remain valid for future cycles.

Generic data had always been used for prior Callaway analyses of the Inadvertent Boron Dilution event in Modes 3, 4, and 5. In November 1993, documentation was completed which confirmed that the Callaway Cycle 5 ICRR data was less limiting than the generic (reference) data used in the safety analyses (Figure 1). The conclusion was drawn that the Cycle 5 data was valid for Cycle 7 and future cycles. As stated above, the basis for this conclusion is engineering judgment, founded on the following tenets:

- Even though only the Cycle 5 ICRR data for Callaway was compared to the reference data, a large amount of data from other plants (3 loops & 4 loops) was also reviewed and compared to the reference analysis ICRR data. In at least half of the instances, the plant-specific data was more limiting than the reference data. For Callaway, though, the ICRR data is more close to linear than is the reference data, as shown on Figure 1.

- The core loading patterns for Cycle 7 and prior cycles indicate consistent locations for the secondary sources as well as the type of assemblies positioned between the secondary source assemblies and the source range detectors. Conversations with the Callaway core design engineer indicate that this was the typical pattern for Callaway.

- Subsequently, written verification was provided in April 1994, as discussed above, confirming the design of the Callaway loading patterns such that the validity of the ICRR data taken during Cycle 5 would be preserved with some degree of certainty.

Therefore, the most recent analysis of the Callaway Inadvertent Boron Dilution event in Modes 3, 4, and 5 remains valid for future reload cycles unless there is a reload-related change that discredits it. The Cycle 5 ICRR data also remains valid for future cycles unless there is some change (in the loading pattern, at the plant, etc.) which would lead to a change in the data.

<u>NRC Question</u>:

2. Is the temperature at which the ICRR curve was obtained consistent with the temperature(s) used in the dilution reanalysis? If not, how are temperature differences accounted for?

<u>Response</u>

The ICRR data was taken at hot zero power conditions, approximately 557°F, just prior to criticality whereas the same analysis curve is used for the lower Modes 3-5. The effects of reduced temperature on the ICRR curve should be negligible or to slightly reduce the contribution of the secondary source (SS) on the overall count rate. The reduction in the SS contribution should make the measurements at Mode 2 directly applicable or slightly more conservative for the lower modes. There are two contributing terms to the SR detector signal, the fixed source and the reactivity variant neutrons. The fixed source neutrons consist of those generated by photo-neutron (γ, n) reactions from the Sb-Be SS. The reactivity variant neutrons include those neutrons from spontaneous fission decay and from subcritical multiplication fissions. Using the knowledge that the SS neutrons are of moderate energy (~20 Kev), where the fission spectrum has neutrons up to the 6 MeV range, temperature is expected to change the relative contributions. The mean free path an epithermal neutron travels prior to undergoing an interaction is strongly impacted by the water density. A positive MTC exists at BOL core conditions during which ICRR data is taken and for which the effects of an Inadvertent Boron Dilution event are greatest. A reduction in moderator temperature at BOL leads to a reduction in reactivity given the high boron concentration. This reduction in reactivity is due to the

increased boron density. The increased density results in a greater probability that an epithermal SS neutron would be captured rather than be leaked or thermalized to cause additional subcritical multiplication fissions. Therefore, the fixed source term contributed by the SS neutrons would be reduced. This should make the ICRR curve less concave at reduced temperature, rendering the hot measurements more conservative for the analysis. However, there are no measurements or design tools to prove this premise.

NRC Question:

3. Describe how the source range detector uncertainties for count rates less than 10 cps (55.5% uncertainty) and greater than 10 cps (25.5% uncertainty) were obtained. Was the Callaway source-detector geometry used?

Response:

The uncertainty calculation performed for the BDMS was generic in scope and not based on any plant-specific characteristics. The uncertainty calculation envelopes the uncertainties associated with the electronics of the source range channel, which supplies the BDMS flux-multiplication monitor, and the electronics of the monitor itself. The neutron leakage fluence characteristics (which the source-detector geometry and characteristics are a part of) are effectively modeled in the ICRR curve characteristics used in the analyses.

The various error components evaluated or considered in the uncertainty calculation are:

Source range channel

- noise
- power supply
- pulse shaper and driver
  - count rate variation
  - temperature sensitivity

log pulse integrator and level amplifier

accuracy

reproducibility

isolation amplifier

BDMS flux-multiplication monitor

setpoint setting error

reproducibility

temperature sensitivity

drift

The evaluation concluded that the error magnitudes, particularly the BDMS flux-multiplication monitor reproducibility, were generally a function of the count rate input. The larger the count rate, the more stable the input and the circuit characteristics improved. This is due to the basic design of the source range channel. It is a six decade logarithmic circuit in which the best response characteristics are in the range of 10^2 to 10^5 cps. The original intent of this circuit was to provide only a monitoring function in the range of 1 to 10^4 cps, where plants are expected to achieve criticality in the 10^2 to 10^4 cps region. A second function was to provide a reactor trip at 10^5 cps, thus it should come as no surprise that the major effort in the design was put into the high end where a potential safety function was located. However, for an Inadvertent Boron Dilution event, this system must also perform in the region of 2 to 100 cps. Count rates this low may be experienced during the refueling interval and in this region the response characteristics of the circuit change markedly. For high count rates (10^3 cps or greater) the total error for the system can be as little as ~10%, for intermediate count rates (10 to 10^3 cps) the total error is ~25%, and at very low count rates (less than 10 cps) the total error has been determined to be ~55% of the setpoint. Without significant circuit redesign these errors are not expected to improve. Since the uncertainty calculation is conservatively based on the circuit design, which is common to all plants utilizing the BDMS, and the ICRR curve characteristics can be bounded based on experience and secondary source location requirements as discussed above, a generic uncertainty calculation is acceptable.

NRC Question:

4. Your safety analysis states that the Technical Specification (TS) limits on shutdown margin in Modes 3-5 will be met for boron dilution events from these Modes. Although total shutdown margin may not be lost, aren't the TS shutdown margin limits degraded due to an inadvertent deboration event and, therefore, not met?

Response:

The Safety Evaluation statements on pages 5 and 6 of Attachment 1 to Reference 1 have been taken out of context. Those statements say that the "Technical Specification limits on shutdown margin in Modes 3-5 will be met," not that they will be met at the conclusion of an Inadvertent Boron Dilution event occurring during those modes.

The SRP acceptance criteria refer to terminating the dilution transient "before the shutdown margin is eliminated" and to having sufficient operator action time between an alarm and "the time of loss of shutdown margin." This has never been interpreted by Union Electric to mean that the analysis must maintain the 1.3% (Modes 3, 4) and 1.0% (Mode 5) SDMs at the end of an inadvertent boron dilution event. Our boron dilution analyses have always been based on precluding an inadvertent criticality, not on preventing a reduction below the Tech Spec LCO. The Tech Specs reflect initial conditions for accident analyses, not acceptance criteria for the endpoints of those analyses. The Safety Evaluation statements could have been clearer by saying that the Modes 3-5 SDMs will continue to be met during normal operation to ensure the validity of the analysis initial conditions.

NRC Question:

5. How were the delay times for opening and closing Chemical Volume Control System (CVCS) isolation valves and purge of CVCS piping determined?

Response:

For each of the three shutdown modes (Modes 3, 4 and 5) for which the automatic Boron Dilution Mitigation System (BDMS) is utilized. the analyses assume 10 seconds for the signal delay from the BDMS microprocessor to mitigation actuation. A switchover of suction for the centrifugal charging pumps from the Volume Control Tank (VCT) to the Refueling Water Storage Tank (RWST) then occurs. The delivery of borated water from the RWST is accomplished by a sequential opening of the RWST isolation valves and closing of the VCT isolation valves.

The RWST isolation valves, BN-LCV-0112D and -0112E, are 8-inch valves. The VCT isolation valves, BG-LCV-0112B and -0112C, are 4-inch valves. All four valves have a 10 second maximum stroke time limit in the Callaway Inservice Testing program for pumps and valves. The extra 5 seconds in stroke time for BN-LCV-0112D and -0112E represent calculational margin. These stroke times were previously the subject of OL Amendment Number 22 dated May 4, 1987 in which the sequential opening of the RWST isolation valves in 15 seconds followed by the closing of the VCT isolation valves in 10 seconds was reflected in the ESFAS response times of Technical Specification Table 3.3-5 (amendment requested via letter ULNRC-1493 dated April 16, 1987).

The CVCS purge volume is composed of two segments. The first segment consists of the piping volume from the RWST suction isolation valves for the CCPs to the tee on the CCP discharge header for the RCP seals and normal charging. The second segment involves the piping volume from that tee to the RCS.

The CVCS purge delay time is calculated by dividing the above purge volume by the appropriate dilution flow rate, 150 gpm for Mode 5 (limited by closing BG-V-0178 causing the flow path to include orifice BG-FO-0010) or 260 gpm for Modes 3 and 4. That is why the purge delay time is 131 seconds for Modes 3 and 4 vs. 212 seconds for Mode 5. Another conservative aspect of the boron dilution reanalysis is that these flow rates, 150 gpm and 260 gpm, are based on the dilution source conditions for reactor makeup water at 37°F and 14.7 psia. However, the analysis results presented in Table 1 of Attachment 1 to Reference 1, i.e. the time from the flux-multiplication alarm to criticality, are based on density-compensated dilution flow rates which reduce the time to criticality. Therefore, the purge delay time is maximized by using the lower non-compensated flow rates while the time to criticality is minimized by using the higher, density-compensated flow rates.

The dilution flow rates, 150 gpm and 260 gpm, have also been verified to be conservative based on plotting the calculated reactor makeup water system resistances (Mode 5 includes the flow orifice resistance) against the reactor makeup water pump performance curve (pump head vs. flow characteristic curve).

FSAR Section 15.4.6 was updated in Revision OL 7 (May 1994) to include the results of this reanalysis.

FIGURE 1

ICRR DATA COMPARISON