Enclosure 2

2CAN032001

Response to Request for Additional Information Related to Proposed Increase in Control Element Assembly Drop Times

(NON-PROPRIETARY)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION RELATED TO PROPOSED INCREASE IN CONTROL ELEMENT ASSEMBLY DROP TIMES

(NON-PROPRIETARY)

By letter dated December 18, 2019 (Reference 1), Entergy Operations, Inc. (Entergy), requested NRC approval of a proposed change to the Arkansas Nuclear One, Unit 2 (ANO-2) Technical Specifications (TSs) that would increase the existing individual and average Control Element Assembly (CEA) drop times established in TS 3.1.3.4, "CEA Drop Time," by 0.2 seconds.

The NRC issued a request for additional information (RAI) on February 24, 2020 (Reference 2), providing a 30-day response period. Each question associated with the subject RAI is repeated below followed immediately by Entergy's response to the specific question.

Background

The CEA insertion on a reactor trip signal provides negative reactivity to the core to shut down the unit for numerous design-basis events discussed in the Chapter 15 of the ANO-2 safety analysis report (SAR). The licensee assessed all the SAR Chapter 15 analyses to determine the impact of the proposed CEA drop times changes and identified that the following events would be affected by the changes: (1) SAR Section 15.1.1 – CEA Bank Withdrawal from a Subcritical Condition; (2) SAR Section 15.1.5 – Total Loss of Reactor Coolant Forced Flow; and (3) SAR Section 15.1.7 – Loss of External Load and/or Turbine Trip. The licensee reanalyzed the events to demonstrate their compliance with the applicable SAR Chapter 15 acceptance criteria and provided the results in the LAR for the NRC staff to review and approve.

RAI-1 – Computer Codes, Critical Heat Flux (CHF) Correlation, and Initial Conditions

(a) Confirm that the methods, computer codes and CHF correlations used in the reanalysis are consistent with those used in the analysis of record for each event reanalyzed. If changes were made, identify and justify the changes.

(b) Confirm that the initial conditions (i.e., RCS core inlet temperature, RCS pressure, core flow rate, power level, axial power shape, and total neutron heat flux factor) used in the reanalysis are consistent with those used in the analysis of record for each event reanalyzed. If the initial conditions were different, identify and justify the differences.

Entergy Response (Parts (a) and (b))

The reanalysis methods, computer codes, and CHF correlations used in the reanalysis are consistent with the descriptions in the SAR sections listed in the following table. In addition, the initial conditions in the reanalysis remain unchanged from those contained in the SAR. The applicable SAR locations respective to Parts (a) and (b) of RAI 1 above are included in the following table.

	Methods, Computer Codes, and CHF Correlations	Initial Conditions
CEA Bank Withdrawal from a Subcritical Condition	SAR Section 5.1.1.4.5	SAR Table 15.1.1-7
Total Loss of Reactor Coolant Forced Flow	SAR Sections 15.1.5.2.3.7 and 15.1.5.2.3.9.	SAR Tables 15.1.5-12 and 15.1.5-13
Loss of External Load and/or Turbine Trip	SAR Section 15.1.7.4.2	SAR Table 15.1.7-6

RAI-2 – CEA Insertion Reactivity Curve

ANO-2 SAR Section 15.1.0.2.3, Shutdown CEA Reactivity, indicates that the CEA insertion reactivity curves in SAR Figure 15.1.1.0-1D (representing CEA reactivity vs. insertion time) and Figure 15.1.1.0-1E (representing CEA insertion position vs. insertion time) were used in the transients and accidents analyses applicable to Cycle 13 and subsequent cycles. The curves were developed based on the assumption that the arithmetic average drop time to 90% inserted was 3.2 seconds, of which 0.6 seconds is attributed to holding coil delay time.

(a) Discuss whether the same CEA insertion curves in SAR Figure 15.1.1.0-1D and Figure 15.1.1.0-1E were used in the reanalysis or not.

(b) If the same curves were used, address the acceptability of their use considering that the curves were based on the arithmetic average drop time to 90% inserted of 3.2 seconds vs. 3.4 seconds for the application to the reanalysis.

(c) If different curves were used, provide the curves and discuss their derivation.

(d) Clarify how the CEA insertion curves used in the reanalysis would be bounding throughout the upcoming and subsequent operating cycles.

Entergy Response

Please note that the actual SAR figure numbers associated with this RAI are Figures 15.1.0-1D and 15.1.0-1E. Figures 1 and 2 provide Reactivity Insertion versus CEA Insertion time (Figure 15.1.0-1D) and CEA Insertion versus Time (Figure 15.1.0-1E) curves, respectively, for the current SAR figures based on 0.6 second holding coil delay (HCD) time, along with revised curves for the 0.8 second HCD time.

(a) The SAR Figure 15.1.0-1D and Figure 15.1.0-1E CEA insertion curves were used in the reanalysis, unchanged except that the time of the upper gripper coil (UGC) release of the CEA was delayed by 0.2 seconds. This was intended to be captured in the original license amendment request (Reference 1) which states: "This change impacts the reactivity versus time data that is used in the accident analyses. The evaluation of impacts on affected accident analyses applies the additional drop time to the time at which the UGC releases the CEA. The reactivity insertion timing after the UGC releases the CEA is unaffected." This is also reflected in Figure 1 and Figure 2 below as the dashed line.

(b) With respect to the use of the current SAR figures noted above, should the actual motion of the CEAs be slower than that indicated in SAR Figures 15.1.0-1D and 15.1.0-1E, but the UGC release of the CEA be sooner than the 0.2 second delay from that indicated in the modified CEA insertion curves (Figures 1 and 2 below), then the insertion of negative reactivity would be faster (reducing reactor power more quickly) than that modeled in the analysis. Therefore, the use of the CEA insertion curves in SAR Figures 15.1.0-1D and 15.1.0-1E is conservative provided the results of the CEA drop time testing meets the revised TS acceptance criteria in the same manner as it is with the current TS acceptance criteria.

(c) Because the same SAR curves (delayed by 0.2 seconds) were used, no response to RAI 2, Part (c) is necessary.

(d) Prior to each fuel cycle, Entergy reviews the key input parameters relative to the SAR Chapter 15 analyses. Pre-startup testing of the CEA drop times ensures that the revised TS acceptance criteria are met. As indicated in the response to RAI-5, Part (b), on approval of the revised TS acceptance criteria, station procedures and licensing basis documents (including the SAR) are revised to reflect the Reactor Protective System (RPS) response time changes upon which the revised CEA drop time acceptance criteria rely. Changes to the RPS response time acceptance criteria in the associated surveillance procedures are also revised accordingly.

Figure 1

Reactivity versus CEA Insertion Time

(SAR Figure 15.1.0-1D)

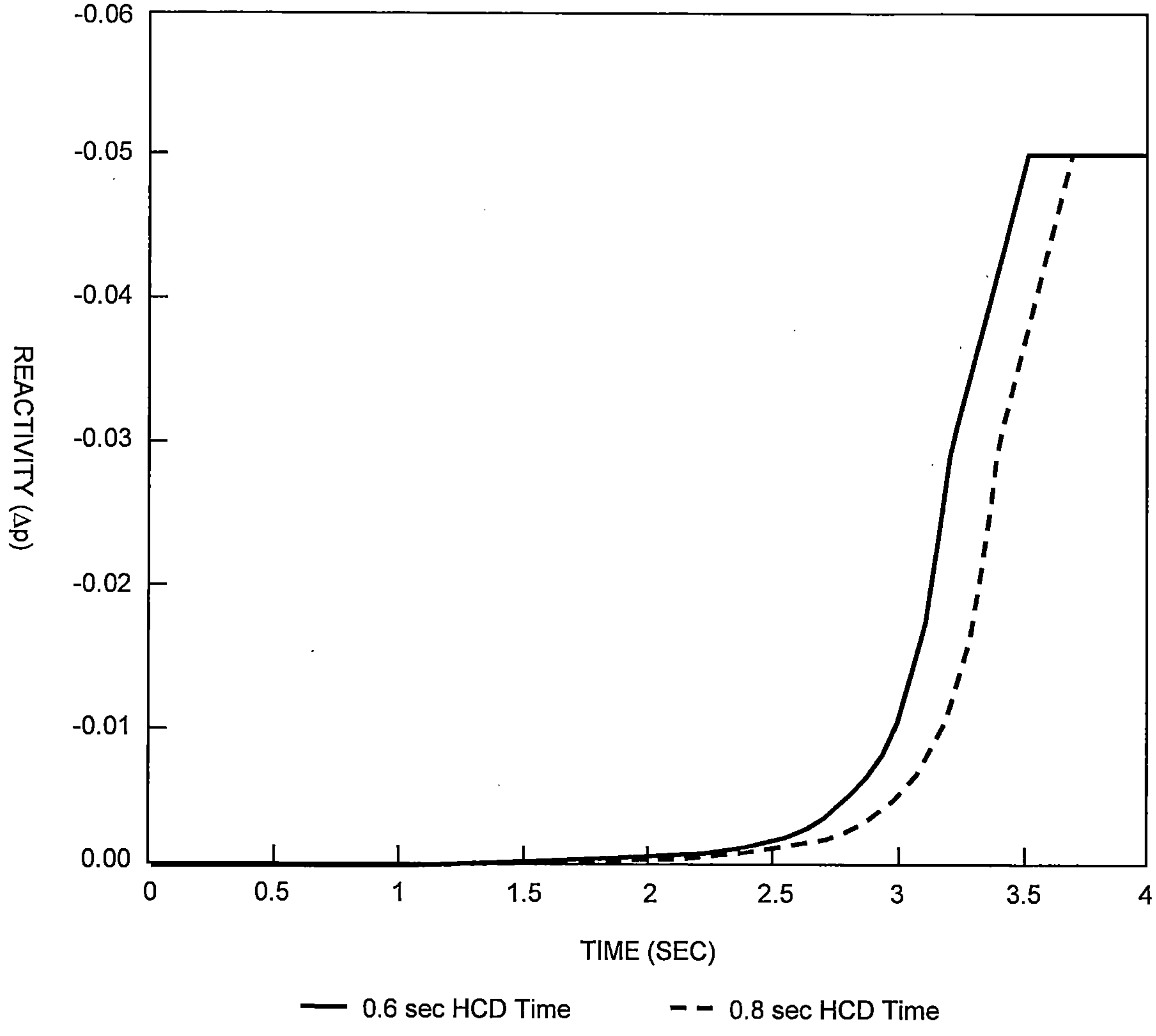
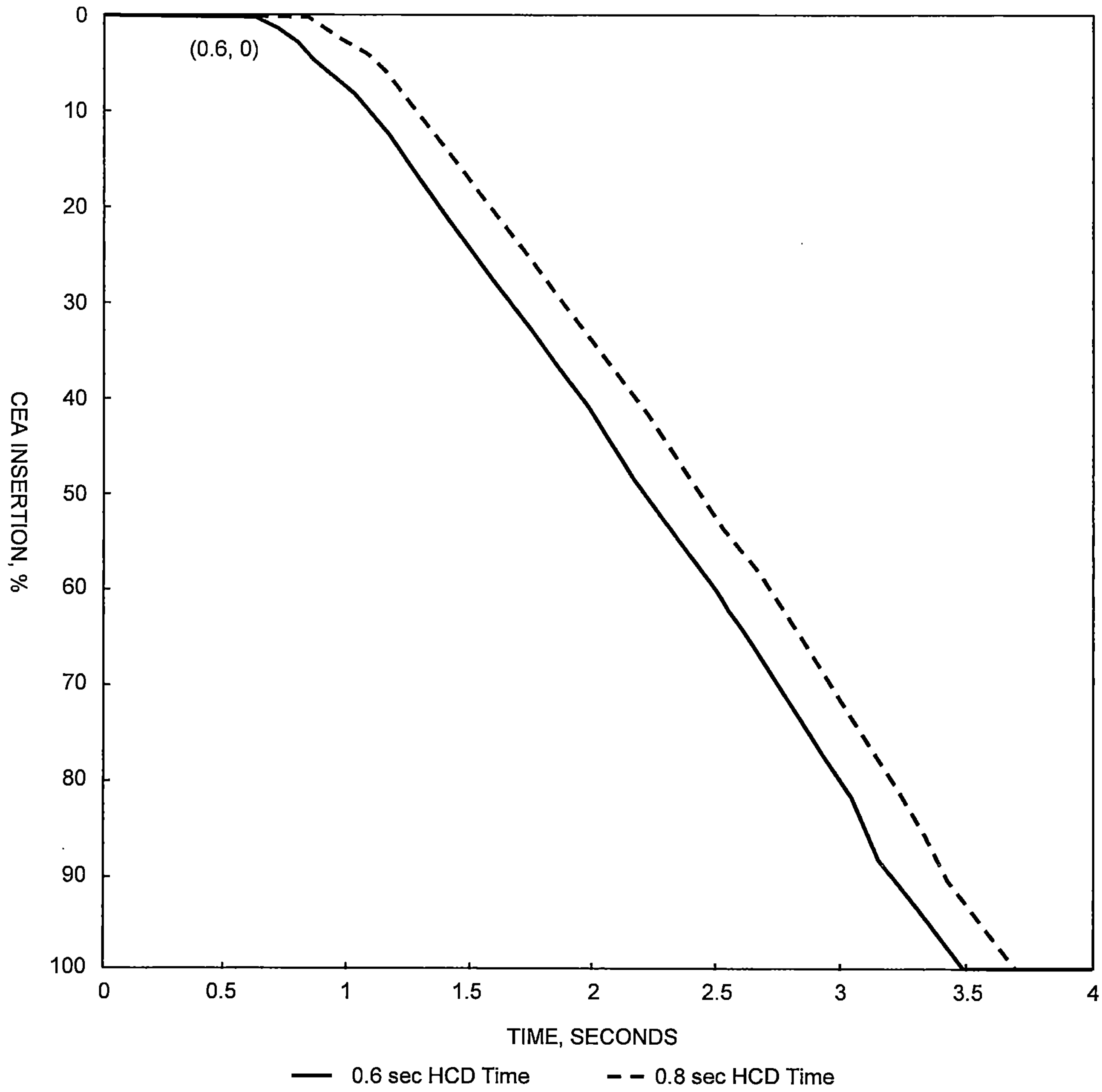


Figure 2

CEA Insertion versus Time

(SAR Figure 15.1.0-1E)



RAI-3 – Initial Conditions in SAR Table 15.1.1-7 for CEAW Reanalysis

On page 5 of the enclosure to the LAR, the licensee discussed the reanalysis of the SAR Section 15.1.1 Event – CEA Bank Withdrawal (CEAW) from a Subcritical Condition. The licensee reanalyzed two cases using the initial conditions in SAR Table 15.1.1-7, "Assumptions for the Cycle 16 Uncontrolled CEA Withdrawal from a Subcritical Condition."

As the title of the table suggests, SAR Table 15.1.1-7 was applicable to the CEAW analysis for the Cycle 16 core, which was the first cycle based on the power uprate conditions. Provide the rationale for selecting the two cases and the associated initial conditions in SAR Table 15.1.1-7 for performing the reanalysis of this CEAW event.

Entergy Response

The Cycle 16 core was the first core at the power uprate conditions. This analysis is discussed in SAR Section 15.1.1.4.3. Select analysis assumptions are listed in SAR Table 15.1.1-7. Subsequent to the power uprate was the fuel transition to Next Generation Fuel (NGF) over Cycles 20 (SAR Section 15.1.1.4.4) and 21 (SAR Section 15.1.1.4.5). The transition to NGF did not affect the analysis assumptions listed in SAR Table 15.1.1-7. The CEAW reanalysis to support the revised CEA drop time started from the NGF analyses and revised only the CEA drop time. None of the assumptions listed in SAR Table 15.1.1-7 were affected by the revised CEA drop time.

The differences between the two cases listed in SAR Table 15.1.1-7 are the CEA reactivity insertion rates (RIRs) and the total (maximum) nuclear heat flux factors (Fq). Case 1 models a RIR of 2.5E-4 $\Delta\rho$/sec and a Fq of 6.8. Case 2 models a RIR of 2.0E-4 $\Delta\rho$/sec and a Fq of 9.0. These combinations of RIR and Fq define limits on the allowable core design, and the reload process confirms that these limits are met each cycle. These limits are not affected by the change in CEA drop time.

The remainder of the assumptions listed in SAR Table 15.1.1-7 are consistent between Cases 1 and 2. The inputs were chosen for their effect on maximum power (and resultant peak temperature) and minimum departure from nucleate boiling ratio (DNBR). The rationale for the remaining listed assumptions is discussed below:

- The initial core power level of 9.63E-7 Megawatt Thermal (MWt) corresponds to 6% subcritical. The CEAW analysis core is assumed to be initially subcritical by an amount which, while meeting the shutdown margin requirements, allows the core to reach criticality through the inadvertent withdrawal of a single bank of CEA's. The initial subcriticality is chosen such that it covers the worth of the bank withdrawing. For ANO-2, the maximum worth of Bank P or of the regulating banks is less than 6% $\Delta\rho$. Thus, initial subcriticality for the subcritical bank withdrawal is chosen equal to 6% $\Delta\rho$.

- The CEAW maximum power is insensitive to the magnitude of the Reactor Coolant Pump (RCP) heat. A maximum value of 18 MWt is used.

- The maximum initial core inlet temperature yields a lower minimum DNBR. The CEAW maximum power is insensitive to the value of initial temperature.

- A minimum pressurizer pressure yields a lower minimum DNBR. A minimum Reactor Coolant System (RCS) pressure also delays the possible time of a high Pressurizer pressure trip (HPPT), yielding a higher maximum power.

- The initial Steam Generator (SG) pressure is a calculated result of the defined initial conditions and is not specifically chosen.

- A minimum RCS flow yields higher peak temperatures and lower minimum DNBRs. The minimum flow results in the largest fluid heatup in the core during the transient particularly at the time of the power spike, which when coupled to the positive moderator temperature coefficient (MTC) yields the largest power spike during the event.

- The most positive MTC value of +0.5E-4 $\Delta\rho/°F$ is used to maximize the power increase.

- Use of the least negative beginning-of-core fuel temperature coefficients minimize the Doppler reactivity feedback in a power (temperature) increasing event, resulting in a higher peak power.

- The steam bypass system is modeled as being in manual (and off) to prevent its actuation as the event progresses, thus maximizing the RCS heatup during the CEAW event and minimizing the DNBR.

- The feedwater regulating system is set to manual to maintain constant feedwater flow during the event.

RAI-4 – Reanalysis of SAR Section 15.1.5 – Total Loss of Reactor Coolant Force Flow (LOF)

For the LOF reanalysis discussed on page 7 of the enclosure to the LAR, the licensee indicated that "an additional 2% DNBR margin of the 7% difference in analysis required overpower margin (ROPM) and COLSS [Core Operating Limits Supervisory System] ROPM was credited" to partially offset the reduced departure from nuclear boiling (DNBR) margin due to the revised CEA drop times.

Discuss how the additional "2% DNBR margin" was modeled in the LOF reanalysis and provide justification.

Entergy Response

The Overpower Margin (OPM) is the [ ]^{a,c}. The ROPM is the quantity of OPM set aide to ensure that the departure from nucleate boiling (DNB) SAFDL is not violated by an anticipated operational occurrence.

The limiting transient analyses define ROPM requirements, and the reload process confirms that the ROPM requirements are met each cycle. The limiting ROPM event initiated from 100% power is the Seized Rotor (SR) event, which has a ROPM of [ ]^{a,c}, whereas the LOF is a non-limiting event with respect to ROPM.

The power uprate LOF analysis was based on [ ]^{a,c} ROPM, and [ ]^{a,c} for the LOF was confirmed for the transition to NGF. The difference in ROPMs between the SR and the LOF is the "7%" referenced above.

The increase in CEA drop time would increase the thermal margin degradation resulting from the LOF. To provide assurance that the DNB SAFDL was not violated, the ROPM for the LOF was increased from [ ]^{a,c}. This is the "2% DNBR margin" referenced above. Although the difference between the LOF and the limiting SR event is reduced, the LOF continues to be a non-limiting event with respect to setting the ROPM.

RAI-5 – Trip Response Times

In support of the revised CEA drop times, Section 3.1 of the LAR discusses the analysis of the ANO-2 SAR Chapter 15 events and identifies the events that credited various reactor trips. The reactor trips and the associated trip response times discussed in the LAR are summarized as follows:

(1) the core protection calculator (CPC) variable overpower trip (VOPT) with the assumed response of 0.4 seconds for the analysis of the CEAW from critical conditions and the excess heat removal due to secondary system malfunction

(2) the low steam generator level (LSGL) trip with the assumed response time of 1.1 seconds for the analysis of the loss of normal feedwater flow event (LONF) event, the feedwater line break (FLB), and main steam safety valves (MSSVs) out-of-service cases

(3) the high pressurizer pressure trip with the assumed response time of 0.65 seconds for the analysis of the FLB

(4) The CPC RCP shaft speed trip with the assumed response time of 0.4 seconds for the analysis of the steam generator tube rupture (SGTR) event.

(5) The CPC Tsat trip with the assumed response time of 2.45 second for the analysis if the SGTR event

(6) The CPC ΔTcold trip with the assumed response time of 0.4 second for the analysis of the events resulting from the instantaneous closure of a single main steam isolation valve or loss of load to a single steam generator.

The licensee stated that the existing analysis of the above events remained valid for the proposed increase in the CEA drop time of 0.2 seconds (from 3.2 second to 3.4 seconds), because: (1) the response time modeled in the analysis for each trip was greater than the "assumed" response time by at least 0.2 seconds, and (2) the "assumed" response time was greater than the "actual" response time. For example, on page 12 of the enclosure to the LAR the licensee discusses the effects of the revised CEA drop times on the existing analyses of the SGTR accidents, which credited the CPCs RCP shaft speed – Low trip for the SGTRs without alternating current (AC) power available.

The licensee stated that the CPC RCP shaft speed trip modeled a response time of 1.0 seconds as compared to 0.4 seconds for the "assumed" response time. The licensee also stated that the 0.4-second response time continued to ensure the assumed CPC RCP shaft speed trip response time remained greater than the "actual" CPC RCP shaft speed trip response time. The licensee determined that, because the 0.2 seconds of the 0.6-second additional CPC RCP shaft speed trip response time could be applied to offset the 0.2-second increase in the CEA drop time, the existing analysis of the SGTR without AC power available cases remained bounding for the revised CEA drop times conditions.

(a) Provide the "actual" response times for each of the above reactor trips and show that the "actual" response times were equal to or less than the "assumed" response times.

(b) Clarify what was done to ensure that the "actual" response times would not exceed the "assumed" response times throughout the operation of upcoming and future cycles.

<u>Entergy Response</u>

(a) For the six trips listed as credited for the events that are listed above, the following table provides a comparison of the current safety analysis modeled response times from reaching the trip parameter setpoint to opening the reactor trip breakers, to the highest measured response time in the 2003 – 2018 time-frame. The response times provided in the table below are those used in the individual accidents/events described in Items 1 through 6 as listed in the RAI above.

	Current Safety Analysis Modeled Response Time (sec)	Worst Measured Response Time 2003-2018 (sec)	Difference (sec)
CPC VOPT	0.6	0.14	0.46
Low SG Level Trip	1.3	0.78	0.52
HPPT	0.9	0.37	0.53
CPC RCP Shaft Speed Trip	1.0	0.29	0.71
CPC T_{sat} Trip	3.0	0.17	2.83
CPC ΔT_{cold} Trip	0.6	0.16	0.44

(b) On approval of the requested change to the TS CEA drop time acceptance criteria, appropriate SAR changes are required in accordance with 10 CFR 50.71(e) to reflect new response time acceptance criteria. The ANO-2 SAR serves as the source of the response time test surveillance procedure acceptance criteria. Changes to the response time acceptance criteria in the associated surveillance procedures are also implemented to reflect the changes to the SAR and TS. These updates are a normal part of the proceduralized TS change implementation process.

RAI-6 – MSLB Mass and Energy Releases for Containment Analysis

Page 16 of the enclosure to the LAR states, in part, that "the MSLB mass and energy release analysis was confirmed to be negligibly affected by the increase in CEA drop times."

Discuss what was done to confirm that the increase in CEA drop times had a negligible effect on the MSLB mass and energy release analysis.

Entergy Response

Small changes in the reactivity vs. time would have a negligible impact on the Main Steam Line Break (MSLB) analysis since the primary driver for the event is the blowdown of the SG. The effects of the RCS and core transients are less important, and the small change in the reactivity vs. time affects only the first few seconds of the event.

To confirm that the effect of the increased CEA drop time has a negligible effect on the analysis, SAR Table 6.2-9C was reviewed. The results show that for the MSLB cases, the peak temperature occurs after 48 seconds, and the peak pressure after 140 seconds. Though not reported in the SAR, the time of reactor trip in the mass and energy release analysis varies by case from approximately 2 to 5 seconds. The peaks are thereby confirmed to occur well after the reactor trip, and therefore, the increased CEA drop time has a negligible effect on the analysis.

REFERENCES

1. Entergy Operations, Inc. (Entergy) letter to U. S. Nuclear Regulatory Commission (NRC), *License Amendment Request to Revise Control Element Assembly Drop Time*, Arkansas Nuclear One, Unit 2 (2CAN121903) (ML19352F266), dated December 18, 2019.

2. NRC email to Entergy, *ANO-2 Final RAI RE: License Amendment Request to Revise Control Element Assembly Drop Time (EPID L-2019-LLA-0285),* (2CNA022002) dated February 24, 2020.