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10 CFR 50.90

2CAN121903

December 18, 2019

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: License Amendment Request
Revise Control Element Assembly Drop Time

Arkansas Nuclear One, Unit 2
NRC Docket No. 50-368
Renewed Facility Operating License No. NPF-6

As required by 10 CFR 50.90, Entergy Operations, Inc. (Entergy) hereby requests changes to the Arkansas Nuclear One, Unit 2 (ANO-2) Technical Specifications (TSs) to revise the individual and average Control Element Assembly (CEA) drop times established in TS 3.1.3.4, "CEA Drop Time." Entergy proposes to increase both the individual and average CEA drop time limits by 0.2 seconds to establish margin impacted by installation of new high temperature upper gripper coils associated with the Control Element Drive Mechanism (CEDM) for each CEA. The proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c).

Approval of the proposed amendment is requested by January 29, 2021. Once approved, the amendment shall be implemented within 60 days.

This amendment request contains no new regulatory commitments.

In accordance with 10 CFR 50.91, Entergy is notifying the State of Arkansas of this amendment request by transmitting a copy of this letter and enclosure to the designated State Official.

If there are any questions or if additional information is needed, please contact Tim Arnold, Manager, Regulatory Assurance, Arkansas Nuclear One, at 479-858-7826.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on December 18, 2019.

Respectfully,

ORIGINAL SIGNED BY RON GASTON

Ron Gaston

RWG/dbb

Enclosure: Evaluation of the Proposed Change

Attachments to Enclosure:

1. Technical Specification Page Markups
2. Retyped Technical Specification Pages

cc: NRC Region IV Regional Administrator

NRC Senior Resident Inspector – Arkansas Nuclear One

NRC Project Manager – Arkansas Nuclear One

Designated Arkansas State Official

Enclosure to

2CAN121903

Evaluation of the Proposed Change

EVALUATION OF THE PROPOSED CHANGE

1.0 SUMMARY DESCRIPTION

Entergy Operations, Inc. (Entergy) requests U.S. Nuclear Regulatory Commission (NRC) review and approval of a proposed amendment to the Arkansas Nuclear One, Unit 2 (ANO-2) Renewed Facility Operating License NPF-6, Appendix A, Technical Specifications (TSs), to revise the individual and average drop times for the Control Element Assemblies (CEAs) established in TS 3.1.3.4, "CEA Drop Time." Both drop time limits are increased by 0.2 seconds to gain margin impacted by installation of new high temperature upper grip coils associated with the Control Element Drive Mechanism (CEDM) for each CEA

2.0 DETAILED DESCRIPTION

2.1 System Design and Operation

Reactivity control at ANO-2 is provided by two independent systems, the Control Element Drive Mechanism Control System (CEDMCS) and the Chemical and Volume Control System (CVCS). The CEDMCS controls short-term reactivity changes and is used for rapid shutdown. This system employs rods referred to as CEAs which when inserted into the reactor core, absorb neutrons. When these rods are withdrawn from the core, more neutrons will be available for the chain reaction and reactor power will increase. All of the 81 CEAs in the ANO-2 reactor are full length CEAs. The full-length CEAs consist of five elements, each containing boron carbide pellets in a hollow tube. Boron carbide is a very strong or "black" absorber of neutrons, so that effectively all neutrons in the vicinity of the CEA will be absorbed. The CEA elements are clustered into groups of five "fingers" sharing a common CEA drive mechanism.

Each CEA is positioned via a magnetic jack CEDM mounted on the reactor vessel closure head. The extension shaft joins with the CEA spider and connects the CEA to the CEDM. The magnetic jack CEDM is an electromechanical device which uses induced magnetic fields to operate a mechanism for moving a control element assembly. Coils mounted in a coil stack assembly slide over the mechanism pressure housing and rests upon a locating shoulder. These coils provide the magnetic flux which operates the mechanical parts of the drive within the pressure housing. Linear motion of these parts causes operation of latching devices which translate the motion of these parts to the control element drive shaft. Driving and holding of the control element occurs when power is sequentially applied to the coils. Each gripper sub-assembly moves between upper and lower stationary stops. The upper gripper is used to perform the lift part of the step and is engaged during holding. The lower gripper assembly holds the CEA during repositioning of the upper grippers. It also performs a load transfer function to minimize latch and drive shaft wear. This function allows the grippers to close about the drive shaft without contacting the drive shaft surface. Contact is made by either lifting the grippers or by lowering the drive shaft down onto the closed grippers.

All CEAs can be inserted rapidly by gravity into the core by interrupting power to the gripper coils within the CEDMs which hold the individual CEAs. The reactor trip switchgear performs this function of interrupting CEDM power following receipt of an automatic trip signal from the Reactor Protective System (RPS) or receipt of a manual trip signal initiated by Control Room operators. When this happens, all CEAs insert fully adding negative reactivity.

2.2 Current Requirement

Currently, the ANO-2 TS 3.1.3.4 Limiting Condition for Operation states the following:

> The individual CEA drop time, from a fully withdrawn position, shall be ≤ 3.7 seconds and the arithmetic average of the CEA drop times of all CEAs, from a fully withdrawn position, shall be ≤ 3.2 seconds from when the electrical power is interrupted to the CEA drive mechanisms until the CEAs reach their 90 percent insertion positions with:
>
> a. $T_{avg} \geq 525$ °F, and
>
> b. All reactor coolant pumps operating.

2.3 Reason for the Proposed Change

Due to historical degradation of the upper gripper coils (UGCs) which holds each CEA in place, Entergy replaced all CEA UGCs during the 2R26 refueling outage (fall 2018) with high efficiency UGCs. The previously installed UGCs experienced intermittent failures which led to the need for exigent TS changes in the past in order to avoid exercising a CEA having a degraded UGC.

Post-installation testing of the new UGCs was performed following heat-up to normal Reactor Coolant System (RCS) operating pressure and temperature with all four Reactor Coolant Pumps (RCPs) in operation, as required by ANO-2 TS 3.1.3.4. Test results indicated an increased CEA drop time for each CEA which slightly exceeded the aforementioned TS drop time limits (approximately 3.212 seconds for the arithmetic average and 3.712 seconds for the worst individual CEA drop time). Because no changes to fuel design, core flow characteristics, or CEA design (other than the new UGCs) occurred in 2R26, and because all CEA drop times have increased, the cause has been determined to be associated with the new UGCs.

Based on further testing, the time it takes for the magnetic flux of the new UGCs to decay when de-energized is longer than that of the previously installed UGCs due to the increased number of windings associated with the new UGCs. This caused the increase in overall CEA drop times. The operating voltage of the new UGCs was lowered subsequent to the initial failed CEA drop time test in an attempt to lower the time required for the UGC magnetic flux to decay. Each voltage adjustment was followed by a CEA drop time re-test. However, voltage can only be lowered to a certain value without affecting coil operation with respect to manual movement of a CEA. Subsequently, portions of the overall margin were reallocated in conjunction with lowering the subject voltage. The subsequent tests were successful in reducing the drop times to less than the TS 3.1.3.4 limits, but with limited margin. Therefore, a change to the CEA drop time TS limits is necessary.

2.4 Description of the Proposed Change

The current TS requirement described in Section 2.2 above is proposed to be revised as follows:

> The individual CEA drop time, from a fully withdrawn position, shall be ≤ 3.9~~7~~ seconds and the arithmetic average of the CEA drop times of all CEAs, from a fully withdrawn position, shall be ≤ 3.4~~2~~ seconds from when the electrical power is interrupted to the CEA drive mechanisms until the CEAs reach their 90 percent insertion positions with:

The proposed change does not require a change to the associated TS Bases. Attachment 1 of this enclosure provides the existing ANO-2 TS page marked-up to show the proposed changes. Attachment 2 of this enclosure provides the re-typed (revised) TS page.

3.0 TECHNICAL EVALUATION

The ANO-2 TSs place limits on both the individual CEA drop time and the average drop time of all CEAs following receipt of a reactor trip signal. Verification of CEA drop times determines that the maximum CEA drop time permitted is consistent with the assumed drop time used in the safety analysis.

During startup testing for Cycle 27, ANO-2 experienced difficulty confirming the individual and average 90% insertion CEA rod drop times in accordance with TS 3.1.3.4. Compliance with TS 3.1.3.4 drop time requirements was met only after lowering CEDMCS voltage in an attempt to reduce the time for UGC magnetic flux decay time and reallocating portions of the overall margin. To regain margin, Entergy proposes to increase both the individual and average CEA drop time requirements established in TS 3.1.3.4 by 0.2 seconds. 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.

3.1 SAR Chapter 15 Evaluations

Chapter 15 of the ANO-2 Safety Analysis Report (SAR) contains the accident analyses associated with Design Bases Accidents (DBAs) and evaluation of other non-DBA events. The table below provides a listing of accidents and events assessed to determine which are impacted by the proposed CEA drop times. Note that the ANO-2 RPS design includes Core Protection Calculators (CPCs) that continuously assess core conditions and will initiate a reactor trip if local power density (LPD) or departure from nucleate boiling ratio (DNBR) limits are exceeded. The CPCs will also generate a reactor trip when certain inputs are out of range or due to an asymmetric Steam Generator transient (ASGT) or during a variable overpower condition. The ASGT functions to protect against transients that cause power asymmetries between the two Steam Generators (SGs), such as an inadvertent closure of a Main Steam Isolation Valve (MSIV). The variable overpower trip (VOPT) functions to protect against transients having a rapid increase in power such as a low power CEA withdrawal event.

ANO-2 also has a non-vital Core Operating Limits Supervisory System (COLSS) which is capable of providing highly accurate indication of core parameters but having a slower response time than that of the CPCs. Because of its accuracy, the TSs establish additional operating restrictions when COLSS is out of service.

A DBA/Event denoted as not impacted indicates that the current analysis is bounding and/or contains sufficient margin to accommodate the proposed increase in CEA drop times.

SAR Section	DBA/Event	Results Impacted	Not Impacted	Not a DBA/Event
15.1.1	Uncontrolled CEA Withdrawal from a Subcritical Condition	X		
15.1.2	Uncontrolled CEA Withdrawal from Critical Conditions		X	
15.1.3	CEA Misoperation		X	
15.1.4	Uncontrolled Boron Dilution Incident		X	
15.1.5	Total Loss of Reactor Coolant Forced Flow	X		
15.1.5	Partial Loss of Reactor Coolant Forced Flow		X	
15.1.6	Idle Loop Startup		X	
15.1.7	Loss of External Load and/or Turbine Trip	X		
15.1.8	Loss of Normal Feedwater Flow		X	
15.1.9	Loss of All Normal and Preferred AC Power to the Station Auxiliaries		X	
15.1.10	Excess Heat Removal due to Secondary System Malfunction		X	
15.1.11	Failure of the Regulating Instrumentation		X	
15.1.12	Internal and External Events Including Major and Minor Fires, Floods, Storms, and Earthquakes		X	
15.1.13	Major Rupture of Pipes Containing Reactor Coolant up to and Including Double-ended Rupture of Largest Pipe in the RCS (Loss of Coolant Accident)		X	
15.1.14	Major Secondary System Pipe Breaks with or without a Concurrent Loss of AC Power		X	
15.1.15	Inadvertent Loading of a Fuel Assembly into the Improper Position		X	
15.1.16	Waste Gas Decay Tank Leakage or Rupture		X	
15.1.17	Failure of Air Ejector Lines (BWR)			X
15.1.18	SG Tube Rupture (SGTR) with or without a Concurrent Loss of AC Power		X	
15.1.19	Failure of Charcoal or Cryogenic System (BWR)			X
15.1.20	CEA Ejection		X	
15.1.21	The Spectrum of Rod Drop Accidents (BWR)			X
15.1.22	Break in Instrument Line or Other Lines from RCS Boundary that Penetrate Containment		X	

SAR Section	DBA/Event	Results Impacted	Not Impacted	Not a DBA/Event
15.1.23	Fuel Handling Accident		X	
15.1.24	Small Spills or Leaks of Radioactive Material Outside Containment		X	
15.1.25	Fuel Cladding Failure Combined with SG Leak		X	
15.1.26	Control Room Uninhabitability		X	
15.1.27	Failure or Overpressurization of Low Pressure Residual Heat Removal System		X	
15.1.28	Loss of Condenser Vacuum		X	
15.1.29	Turbine Trip with Coincident Failure of Turbine Bypass Valves to Open		X	
15.1.30	Loss of Service Water (SW) System		X	
15.1.31	Loss of One DC System		X	
15.1.32	Inadvertent Operation of Emergency Core Cooling System (ECCS) During Power Operation		X	
15.1.33	Turbine Trip with Failure of Generator Breaker to Open		X	
15.1.34	Loss of Instrument Air System		X	
15.1.35	Malfunction of Turbine Gland Sealing System		X	
15.1.36	Transients Resulting from the Instantaneous Closure of a Single MSIV		X	
COLSS and CPCs	COLSS and CPCs database parameters and uncertainties		X	

Each DBA/Event is discussed individually below.

SAR Section 15.1.1 – CEA Bank Withdrawal from a Subcritical Condition

The subcritical CEA withdrawal (CEAW) DBA analysis was previously reviewed and approved by the NRC during the 2002 ANO-2 power uprate (Reference 1) and updated in 2008 to account for the Next Generation Fuel (NGF) WSSV-T critical heat flux (CHF) correlation (Reference 2). With respect to the proposed increase in CEA drop times, the two subcritical CEA withdrawal cases from a subcritical condition contained in SAR Table 15.1.1-7 were re-analyzed:

Case 1: 2.5E-4 $\Delta\rho$/sec (maximum reactivity rate), Fq of 6.8 (total neutron flux factor)

Case 2: 2.0E-4 $\Delta\rho$/sec, Fq of 9.0

Case	SAR	Analysis of Record	Revised Evaluation	Acceptance Criteria
Minimum DNBR – Case 1	> 1.23	1.4159	1.2583	> 1.23
Minimum DNBR – Case 2	> 1.23	2.2297	1.8542	> 1.23
CTM – Case 1 (°F)	Not specified*	≤ 2900	< 3400	< 4681
CTM – Case 2 (°F)	Not specified*	≤ 2700	< 3400	< 4681

* ANO-2 TS 2.1.1.2, "Peak Fuel Centerline Temperature," specifies a limit of < 5080 °F (decreasing by 58 °F per 10,000 MWD/MTU for burnup and adjusting for burnable poisons per CENPD-275-P, Revision 1-P-A and CENPD-382-P-A)

The results demonstrate increases in peak heat flux and corresponding decreases in minimum DNBR. However, the results do not challenge the minimum DNBR specified acceptable fuel design limit (SAFDL) or the Centerline Temperature Melt (CTM) SAFDL.

SAR Section 15.1.2 – Uncontrolled CEA Withdrawal from Critical Conditions

The CEAW from critical conditions was previously reviewed and approved by the NRC during the 2002 ANO-2 power uprate (Reference 1) and updated in 2008 to account for the NGF WSSV-T CHF correlation (Reference 2). The revised evaluation started with the analysis of record (AOR) and only revised the rod drop time to determine the impact.

The CEAW events credit the CPC VOPT. The CPC VOPT modeled a response time of 0.6 seconds whereas the assumed response time is 0.4 seconds. An additional 0.2 seconds of margin was historically included in the CPC response time (equating to a total of 0.6 seconds) but is not required or necessary. The 0.4-second response time includes conservatisms to ensure the assumed CPC response time remains greater than the actual CPC response time. Therefore, the 0.2-second additional CPC response time can be applied to offset the 0.2-second increase in the CEA drop times. Thus, there is no impact on this CEAW event due to the revised CEA drop times.

SAR Section 15.1.3 – CEA Misoperation

The CEA Misoperation events evaluated in the SAR include the CEA single and subgroup drop events and the single CEAW from within the CPC dead-band event. None of the single CEA and subgroup drop events or the single CEAW from within the CPC dead-band generate a reactor trip. Thus, the revised CEA drop times do not impact these events.

SAR Section 15.1.4 – Uncontrolled Boron Dilution Incident

The boron dilution event is analyzed for all modes of plant operation. The boron dilution events during operation in Modes 1 and 2 are bounded by the hot full power (HFP) and hot zero power (HZP) CEAW events. The revised CEA drop time does not change the event characteristics; therefore, the uncontrolled boron dilution event remains bounded by existing SAR events.

For the limiting uncontrolled boron dilution event in operational Modes 3, 4, 5 and 6, all CEAs have been already inserted (i.e., CEA insertion is not credited in the boron dilution event for the subject modes of operation). Thus, the revised CEA drop time has no impact on the uncontrolled boron dilution event.

SAR Section 15.1.5 – Total and Partial Loss of Reactor Coolant Forced Flow

The loss of flow analyses in the SAR is subdivided into two events: 1) loss of forced reactor coolant flow and, 2) seized RCP rotor.

The loss of coolant flow resulting from an electrical failure event was previously reviewed and approved by the NRC during the 2002 ANO-2 power uprate (Reference 1) and updated in 2008 to account for the NGF WSSV-T CHF correlation (Reference 2).

To partially offset the reduced DNBR margin due to the revised CEA drop times, an additional 2% DNBR margin of the 7% difference in analysis required overpower margin (ROPM) and COLSS ROPM was credited. It was confirmed that the minimum DNBR remains above the SAFDL. Only the maximum subcooling scenario was evaluated since it had been previously determined that the maximum subcooling was more limiting than the minimum subcooling case.

As indicated below, the loss of coolant flow resulting from an electrical failure remains within acceptance criteria. This event is not limiting with respect to peak linear heat rate (LHR) or peak primary pressure. This event does not require changes to the current COLSS database to support the revised CEA drop time.

Case	SAR	Analysis of Record	Revised Evaluation	Acceptance Criteria
Minimum DNBR	> 1.23	1.272	1.285	≥ 1.23

The seized rotor event was previously reviewed and approved by the NRC during the 2002 ANO-2 power uprate (Reference 1), reviewed and approved by the NRC during the adoption of Alternate Source Terms (AST) (References 4 and 5), and updated in 2008 to account for the NGF WSSV-T CHF correlation (Reference 2).

The SAR seized rotor evaluation conservatively assumes an instantaneous reduction in flow to the 3-pump value. Hence, only the 3-pump flow fraction is important in the calculations of potential fuel failure. No credit is assumed for the heat flux reduction due to the revised CEA drop times. Thus, the change in CEA drop time has no impact on the calculation of potential fuel failure.

Although no transient response is required, a representative seized rotor case was performed based on a conservative response time delay of 0.5 seconds and a holding coil delay time of 0.6 seconds for a total delay time of 1.1 seconds. The revised rod drop time (holding coil delay of 0.8 seconds) and an assumed response time of 0.4 seconds, results in a total response time of 1.2 seconds. The increase in total response time of 0.1 seconds has a very small impact on the representative seized rotor case. The time to minimum DNBR would increase by approximately 0.1 seconds. The loss of load/turbine trip events (see SAR Section 15.1.7 below) are the limiting peak primary pressure events and demonstrate that for a 0.1-second increase in UGC delay time, that the increase in primary pressure is insignificant.

Since the non-loss of coolant accident (LOCA) seized rotor transient analyses are not impacted, the seized rotor event AST doses are not impacted. Thus, the revised CEA drop times have no impact on the seized rotor potential fuel failures and AST doses.

SAR Section 15.1.6 – Idle Loop Startup

This SAR section discusses reactor operations with two RCPs in service on one of the two RCS loops. This mode of operation is not allowed per TS 3.4.1.1, "Reactor Coolant Loops and Coolant Circulation – Startup and Power Operation." ANO-2 cannot operate with less than all four RCPs in operation when the reactor is critical.

SAR Section 15.1.7 – Loss of External Load and/or Turbine Trip

The analysis presented in SAR Section 15.1.7 models a loss of external load (LOL) concurrent with a loss of feedwater in order to bound the LOL and turbine trip event as well as the loss of condenser vacuum event. The LOL event was previously reviewed and approved by the NRC during the 2002 ANO-2 power uprate (Reference 1) and updated in 2008 to account for the NGF WSSV-T CHF correlation (Reference 2). For the CEA drop time analysis, the AOR was first baselined to the most recent code version of the current NRC-approved CENTS model listed in the Core Operating Limits Report (COLR), and then updated with the revised CEA drop times to determine the impact of the drop time change.

For the main steam safety valves (MSSVs) out-of-service cases where a reactor trip is credited with mitigating the event, the low SG level (LSGL) trip provides reactor protection. The LSGL trip was modeled in the analysis with a response time of 1.3 seconds. The LSGL trip response time has been reduced from 1.3 to 1.1 seconds. The 1.1-second response time continues to ensure the assumed LSGL trip response time remains greater than the actual LSGL trip response time. Therefore, the 0.2-second reduction in the reactor protection response time for the LSGL trip can be applied to offset the 0.2-second increase in the UGC decay time. Note that for a subset of the MSSVs out-of-service cases, the limiting condition occurs well before reactor trip occurs. For these cases, the increase in the UGC decay time has no impact on the limiting conditions. Thus, there is no impact on the MSSVs out-of-service cases due to the revised CEA drop time.

Case	SAR	AOR Baseline	Revised Evaluation	Acceptance Criterion
Peak Primary Pressure	2688 psia	2694 psia	2695 psia	< 2750 psia
Peak Secondary Pressure	1208 psia*	1207 psia	1207 psia	< 1210 psia
Peak Pressurizer Water Volume	Not Reported	1122 ft^3	1130 ft^3	< 1200 ft^3

* Note that the SAR reported value rounds the peak secondary side pressure of 1208.4 psia to 1209 psia. All values reported herein are rounded up or rounded down as appropriate.

SAR Section 15.1.8 – Loss of Normal Feedwater Flow

The loss of normal feedwater flow event (LONF) was previously reviewed and approved by the NRC during the replacement of the SGs in 2000 (Reference 3).

The current AOR was evaluated by only revising the CEA drop times and the assumed LSGL trip response time to determine impacts. The LONF event credits the LSGL trip. The LSGL trip assumed response time of 1.3 seconds was reduced to 1.1 seconds. The 1.1-second response time continues to ensure the assumed LSGL trip response time remains greater than the actual LSGL trip response time. Therefore, the 0.2-second reduction in the LSGL trip assumed response time can be applied to offset the 0.2-second increase in the UGC decay time. Thus, there is no impact on the LONF event due to the revised CEA drop time.

SAR Section 15.1.9 – Loss of All Normal and Preferred AC Power to the Station Auxiliaries

This event is bounded by the loss of total and partial loss of reactor coolant forced flow event (SAR Section 15.1.5), loss of external load and/or turbine trip event (SAR Section 15.1.7), and the loss of normal feedwater flow event (SAR Section 15.1.8). The revised CEA drop times does not change the event characteristics; therefore, the event continues to be bounded by these three SAR events.

SAR Section 15.1.10 – Excess Heat Removal due to Secondary System Malfunction

The excess heat removal due to secondary system malfunction event was previously reviewed and approved by the NRC during the replacement of the SGs in 2000 (Reference 3) and was updated to account for the NGF WSSV-T CHF correlation (Reference 2).

The current AOR was evaluated by only revising the CEA drop times to determine impacts. This event credits the CPCs VOPT. The CPCs VOPT modeled a response time of 0.6 seconds whereas the assumed response time is 0.4 seconds (see SAR Section 15.1.2 discussion above). Therefore, the 0.2-second additional CPC response time can be applied to offset the 0.2-second increase in the UGC decay time. Thus, there is no impact on the excess heat removal due to secondary system malfunction event due to the revised CEA drop time.

SAR Section 15.1.11 – Failure of the Regulating Instrumentation

SAR Section 15.1.11 assesses malfunctions or failures of regulating/control systems that could result in deviations of plant process parameters from the prescribed values. Such deviations would initiate a reactor trip in the event a core safety limit was approached. The event characteristics and conclusions would not be impacted by the revised CEA drop times. Thus, this event is not impacted by the revised CEA drop times.

SAR Section 15.1.12 – Internal and External Events Including Major and Minor Fires, Floods, Storms, and Earthquakes

The event characteristics of the subject phenomena are not impacted by the revised CEA drop time, i.e., safe shutdown of the plant would not be prevented. Thus, there is no impact to the evaluation of these phenomena due to the revised CEA drop times.

SAR Section 15.1.13 – Major Rupture of Pipes Containing Reactor Coolant up to and including Double-Ended Rupture of Largest Pipe in the Reactor Coolant System (Loss of Coolant Accident)

The large break loss of coolant accident (LOCA) and post-LOCA long term cooling analyses do not explicitly model or credit core reactivity due to rod insertion; thus, the revised CEA drop time has no impact on these analyses.

The small break LOCA analysis does credit the CEA drop time insertion. The small break LOCA AOR reactivity-versus-time curve was compared to the reactivity insertion using the revised CEA drop time curve. The current small break LOCA analysis reactivity curve bounds that of the revised CEA drop time curve; thus, there is no impact to the SAR results and conclusions for this event.

The LOCA AST dose analyses (References 4, 5, and 6) do not explicitly model the CEA drop time insertion. Hence, the LOCA AST dose analyses are not impacted by the revised CEA drop time curve. Thus, the SAR results and conclusions for the LOCA AST dose evaluations are not impacted.

SAR Section 15.1.14 – Major Secondary System Pipe Breaks with or without a Concurrent Loss of AC Power

This SAR analysis is subdivided into two sections: 1) steam line break accident and, 2) feedwater line break accident. The steam line break (SLB) event was previously reviewed and approved by the NRC during the replacement of the SGs in 2000 (Reference 3) and during adoption of AST doses (References 4 and 5), and was updated to account for the NGF WSSV-T CHF correlation (Reference 2).

The SLB event evaluation includes different break locations for inside and outside containment (IC and OC), with and without a loss of offsite power (LOOP), combined with different single failures.

The steam system piping failures at hot zero power (HZP) and hot full power (HFP) with and without LOOP for IC and OC events are dominated by the total scram reactivity added and the rate of the primary-side cooldown accounting for the associated moderator and fuel feedback effects. The total reactivity added has not changed due to the revised CEA drop times. The primary-side cooldown is directly dependent on the secondary-side cooldown. Since the SG blowdown is not impacted by the revised CEA drop time, there is no change in the secondary-side cooldown. As there is no change to the secondary-side cooldown, there is no change in the primary side-cooldown. Hence, there is no significant, impact on the steam system piping failures events due to the revised CEA drop times.

The steam system piping failures result in an insignificant impact on the maximum post-trip fission power, maximum post-trip reactivity, post-trip minimum DNBR, and maximum peak LHR. Thus, the revised CEA drop times do not appreciably impact the SLB potential fuel failures and AST dose analyses.

The feedwater line break (FWLB) event was previously reviewed and approved by the NRC during the 2002 ANO-2 power uprate (Reference 1).

The current FWLB AOR was evaluated by only revising the CEA drop times and the assumed LSGL trip response time to determine impacts.

The FWLB event credits the LSGL and the high pressurizer pressure (HPP) trips. The LSGL trip modeled a response time of 1.3 seconds. The LSGL trip assumed response time has been reduced from 1.3 to 1.1 seconds. The 1.1-second response time continues to ensure the assumed LSGL trip response time remains greater than the actual LSGL trip response time. Therefore, the 0.2-second reduction in the assumed analysis response time for the LSGL trip can be applied to offset the 0.2 second increase in the UGC decay time.

The HPP trip modeled a response time of 0.9 seconds whereas the assumed response time is 0.65 second. The 0.65-second response time continues to ensure the assumed HPP trip response time remains greater than the actual HPP trip response time. Therefore, 0.25 seconds of the additional HPP trip response time can be applied to offset the 0.2-second increase in the UGC decay time. Thus, there is no impact on the FWLB event due to the revised CEA drop times.

SAR Section 15.1.15 – Inadvertent Loading of a Fuel Assembly into the Improper Position

The inadvertent loading of a fuel assembly into the improper position could cause a difference in the local power peaking for the misloaded core. This event is analyzed with respect to thermal margin for power operations. Since, the event does not require a reactor trip; the revised CEA drop times have no impact on this event.

SAR Section 15.1.16 – Waste Gas Decay Tank Leakage or Rupture

SAR Section 15.1.16 describes the waste gas decay tank leakage or rupture as an unexpected and uncontrolled release to the atmosphere of radioactive fission gases that are stored in one waste gas decay tank event. Since, the event does not require a reactor trip; the revised CEA drop times do no impact this event.

SAR Section 15.1.17 – Failure of Air Ejector Lines (BWR)

This event is applicable only to boiling water reactors (BWRs). Because ANO-2 is a pressurized water reactor (PWR), this event is not applicable.

SAR Section 15.1.18 – Steam Generator Tube Rupture with or without a Concurrent Loss of AC Power

The SGTR event was previously reviewed and approved by the NRC during the 2002 ANO-2 power uprate (Reference 1) and updated to account for the AST doses (References 4 and 5).

The SGTR analysis credits the CPCs RCP shaft speed – Low trip for the without AC power cases and the CPCs Hot Leg (saturation) temperature (Tsat) – High trip for the with AC power cases.

The CPC RCP shaft speed trip modeled a response time of 1.0 second whereas the assumed response time is 0.4 seconds. The 0.4-second response time continues to ensure the assumed CPC RCP shaft speed trip response time remains greater than the actual CPC RCP shaft speed trip response time. Therefore, 0.2 seconds of the 0.6-second additional CPC RCP shaft speed trip response time can be applied to offset the 0.2-second increase in the UGC delay time. Thus, there is no impact on the SGTR without AC power available cases.

The CPC Tsat trip modeled a response time of 3.0 seconds whereas the assumed response time is 2.45 seconds. Therefore, 0.2 seconds of the 0.55-second additional CPC Tsat trip response time can be applied to offset the 0.2-second increase in the UGC decay time. Since the non-LOCA SGTR transient analyses are not impacted, the SGTR event AST doses are not impacted. Thus, there is no impact on the SGTR with AC power available cases.

SAR Section 15.1.19 – Failure of Charcoal or Cryogenic System (BWR)

This event is only applicable to BWRs. Because ANO-2 is a PWR, this event is not applicable.

SAR Section 15.1.20 – CEA Ejection

The CEA ejection event was previously reviewed and approved by the NRC during the 2002 ANO-2 power uprate (Reference 1) and updated to account for the AST doses (References 4 and 5).

The current CEA ejection AOR was evaluated by only revising the CEA drop times to determine impacts. The CEA ejection event credits the CPCs VOPT. The CPCs VOPT trip modeled a response time of 0.6 seconds whereas the assumed response time is 0.4 seconds (see SAR Section 15.1.2 discussion above). Therefore, the 0.2-second additional CPC response time can be applied to offset the 0.2-second increase in the UGC decay time. Since the non-LOCA CEA ejection transient analysis is not impacted, the CEA ejection AST doses are not impacted. Thus, there is no impact on the CEA ejection event due to the revised CEA drop times.

SAR Section 15.1.21 – The Spectrum of Rod Drop Accidents (BWR)

This event is only applicable to BWRs. Because ANO-2 is a PWR, this event is not applicable.

SAR Section 15.1.22 – Break in Instrument Line or Other Lines from Reactor Coolant Pressure Boundary that Penetrate Containment

Since there are no instrument lines from the RCS that penetrate containment, this event is not applicable and there is no impact to this event due to the revised CEA drop times.

SAR Section 15.1.23 – Fuel Handling Accident

The Fuel Handling Accident analysis has been updated to account for the AST doses (References 4 and 5). The CEA drop time is not modelled for this evaluation. Thus, the SAR results and conclusions for this event are not impacted by the revised CEA drop times.

SAR Section 15.1.24 – Small Spills or Leaks of Radioactive Material Outside Containment

The subject event characteristics and conclusions are not impacted by the revised CEA drop times. Thus, there is no impact on this event due to the revised CEA drop times.

SAR Section 15.1.25 – Fuel Cladding Failure Combined with Steam Generator Leak

Since this is a SAR Chapter 11 (Radioactive Waste Management) event and the CEA drop time is not of significance with respect to event mitigation, the SAR results and conclusions are not impacted by the revised CEA drop time curve.

SAR Section 15.1.26 – Control Room Uninhabitability

The rod insertion characteristics are not an input for the subject evaluation. Thus, the SAR results and conclusions are not impacted by the revised rod drop time curve.

SAR Section 15.1.27 – Failure or Overpressurization of Low Pressure Residual Heat Removal System

The rod insertion characteristics are not an input for the subject evaluation. Thus, the SAR results and conclusions are not impacted by the revised rod drop time curve.

SAR Section 15.1.28 – Loss of Condenser Vacuum

SAR Section 15.1.28 describes the loss of condenser vacuum (LOCV) event, which is similar to the loss of load and/or turbine trip event in SAR Section 15.1.7. The LOCV event is comparable to or less adverse (bounded) than the loss of load and/or turbine trip event. The revised CEA drop times do not change the event characteristics, therefore, the loss of load and/or turbine trip event remains the bounding event.

SAR Section 15.1.29 – Turbine Trip with Coincident Failure of Turbine Bypass Valves to Open

This event is encompassed in the SAR Section 15.1.7 evaluation of loss of load and/or turbine trip event. The revised CEA drop times do not change the event characteristics, therefore, the loss of load and/or turbine trip event remains the bounding event.

SAR Section 15.1.30 – Loss of Service Water System

The CEA drop time is independent of the loss of the SW system and is not of significance with respect to event mitigation. The event characteristics and conclusions are not impacted by the revised CEA drop times. Thus, the SAR results and conclusions for this event are not impacted by the revised CEA drop time curve.

SAR Section 15.1.31 – Loss of One DC System

The CEA drop time is independent of the loss of one DC system and is not of significance with respect to event mitigation. The event characteristics and conclusions are not impacted by the revised CEA drop times. Thus, the SAR results and conclusions for this event are not impacted by the revised CEA drop time curve.

SAR Section 15.1.32 – Inadvertent Operation of ECCS during Power Operation

During power operation, the primary pressure range is above the shutoff head of the High Pressure Safety Injection (HPSI) pumps and above the pressure of the Safety Injection Tanks (SITs). The CEA drop time is independent of the HPSI pumps and SITs and is not of significance with respect to event mitigation. Thus, the SAR results and conclusions for this event are not impacted by the revised CEA drop time curve.

SAR Section 15.1.33 – Turbine Trip with Failure of Generator Breaker to Open

The CEA drop time is independent of the turbine trip with failure of the generator output breaker to open and is not of significance with respect to event mitigation. Thus, the SAR results and conclusions for this event are not impacted by the revised CEA drop time curve.

SAR Section 15.1.34 – Loss of Instrument Air System

The CEA drop time is independent of the loss of Instrument Air system and is not of significance with respect to event mitigation. The event characteristics and conclusions are not impacted by the revised CEA drop times. Thus, the SAR results and conclusions for this event are not impacted by the revised CEA drop time curve.

SAR Section 15.1.35 – Malfunction of Turbine Gland Sealing System

The CEA drop time is independent of the malfunction of the gland sealing system and is not of significance with respect to event mitigation. The event characteristics and conclusions are not impacted by the revised CEA drop times. Thus, the SAR results and conclusions for this event are not impacted by the revised CEA drop time curve.

SAR Section 15.1.36 – Transients Resulting from the Instantaneous Closure of a Single MSIV

SAR Section 15.1.36 describes the transients resulting from the instantaneous closure of a single MSIV or loss of load to one SG event. The LOL to one SG event was previously reviewed and approved by the NRC during the replacement of the SGs in 2000 (Reference 3) and was updated to account for the NGF WSSV-T CHF correlation (Reference 2).

The current AOR was evaluated by only revising the CEA drop times to determine impacts. The LOL to one SG event credits the CPCs ΔTcold trip, which modeled a response time of 0.6 seconds whereas the assumed response time is 0.4 seconds (see SAR Section 15.1.2 discussion above). Therefore, the 0.2 seconds additional analysis response time can be applied to offset the 0.2-second increase in the UGC decay time. Thus, there is no impact on the LOL to one SG event due to the revised CEA drop times.

COLSS and CPC Impact

The setpoints analyses calculate COLSS and CPC database parameters and uncertainties. The COLSS and CPC functionality is used at ANO-2 to measure plant parameters (e.g., temperature, pressure, flow, incore/excore detector signals). When COLSS-monitored parameters are outside of established limits, the COLSS computer will provide alarms to the operator. When critical CPC-related parameters are outside of the limits, the CPCs will initiate a reactor trip. Neither the COLSS or the CPCs provide any further action upon the onset of a reactor trip. Therefore, the COLSS and CPCs are not directly affected by a change in the rod drop times.

The COLSS and CPC database parameters and addressable constants are updated as part of the cycle-specific reload process. The initial assessment of the next reload analysis is that the revised CEA drop time insertion curve will not impact any COLSS or CPC parameters. Therefore, the setpoints analyses are not expected to be affected by the revised CEA drop times.

3.2 Physics

The revised CEA drop time has no impact on the core design, thermal-hydraulics, and fuel rod design analyses; however, physics do support the margin between the average CEA drop time modeled in the safety analysis and the maximum individual CEA drop time. The reactivity worth of a CEA is a function of the power or neutron flux environment surrounding the CEA. During a reactor trip, the faster CEAs will be in higher flux regions sooner and will subsequently have a greater contribution to the net negative reactivity insertion than the slower CEAs. Therefore, the negative reactivity insertion for any reasonable distribution of CEAs is more directly correlated to, and can be represented by, the average CEA insertion.

The proposed change to TS 3.1.3.4 increases the individual CEA drop time from 3.7 to 3.9 seconds. This 0.2-second increase simply follows from the 0.2-second shift in the entire insertion curve which increases the average drop time from 3.2 to 3.4 seconds. Since the 0.2-second increase is due to an increase in the UGC decay time, there is no impact on the physics assessment of the individual CEA drop time limit.

3.3 Non-Chapter 15 SAR Evaluations

SAR Section 5.2.2 – Overpressure Protection

This SAR section describes the overpressure protection of the RCS and SGs. The peak RCS and SG pressures used in the overpressure protection report are based on the loss of load / turbine trip analysis documented in SAR Section 15.1.7. The conclusions of SAR Section 15.1.7 are that the 0.2-second increase in the revised CEA drop time has insignificant effects on the RCS peak pressure, SG peak pressure, and the pressurizer surge following the reactor trip. Therefore, the current overpressure protection report described in SAR Section 5.2.2 remains applicable.

SAR Section 6.2.1 – Containment Functional Design

This SAR section describes the containment functional design including the mass and energy releases following a LOCA or Main Steam Line Break (MSLB). The LOCA and MSLB mass and energy releases are not impacted by the CEA drop time curve change. The LOCA mass and energy release analysis inputs negative reactivity based on voiding in the core and the CEA drop time curve is not an input to this analysis. The MSLB mass and energy release analysis was confirmed to be negligibly affected by the increase in CEA drop times.

DNB Propagation

A DNB propagation evaluation was performed for the extended power uprate (Reference 1). The evaluation was subsequently updated to account for NGF (Reference 2). The limiting event for DNB propagation concerns is the seized rotor event. The transient response for the seized rotor event is used to demonstrate that DNB propagation is precluded if the maximum clad strain is less than 29.3% and assuming the fuel rod time in DNB is less than 50 seconds (Reference 7, Appendix A, Table 3-3).

Since the seized rotor event was not impacted by the revised CEA drop time insertion, maximum strain and time in DNB results have not changed from the evaluation that was performed for NGF. Thus, there is no impact due to the revised CEA drop time insertion for DNB propagation.

Fuel Cladding Burst

Previous analysis concluded that the fuel cladding burst acceptance criteria were met for all Combustion Engineering (CE) fleet Zircalloy Diboride (ZrB_2) fuel types. This analysis determined that cladding burst is precluded if peak cladding temperature and engineering hoop stress remain below the values for cladding rupture depicted in Section 4.4.2.1 of Reference 8. A fuel cladding burst evaluation was performed for the extended power uprate (Reference 1) and was subsequently updated to account for NGF (Reference 2).

The transient for the pre-trip MSLB and rod ejection events provide the inputs to the cladding burst evaluation. As described above, due to the allocation of response time margin, the transient response that is provided as an input to the clad strain evaluation is not impacted. As a result, the fuel cladding strain evaluation performed for NGF remains valid.

Fuel Cladding Strain

Fuel cladding strain results were found to be acceptable for the extended power uprate (Reference 1) and were subsequently updated to account for NGF (Reference 2). The transient results for the HZP CEA withdrawal and the rod ejection events provide the inputs to the clad strain evaluation. As described above, due to the allocation of response time margin, the transient response that is provided as an input to the clad strain evaluation is not impacted. As a result, the fuel cladding strain evaluation performed for NGF remains valid.

Conclusion

The evaluations performed in support of increasing the both the individual and average CEA drop time TS limits by 0.2 seconds demonstrate that the ANO-2 design and licensing basis continue to be met. The conclusions documented demonstrate that all Chapter 15 accident/event analyses remain acceptable and meet requirements with the inclusion of the proposed revised CEA drop times. The evaluation also confirmed that no changes are required to the COLSS and CPC databases or constants.

In addition, results of reviews and assessments of other analyses not contained in SAR Chapter 15 (or other areas of interest) demonstrate that revising TS 3.1.3.4 to increase the 90% average CEA drop time from 3.2 to 3.4 seconds and to increase the individual CEA drop time from 3.7 to 3.9 seconds is acceptable. Therefore, Entergy has concluded that the subject changes are appropriately justified and that an acceptable margin of safety is maintained.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

Pursuant to 10 CFR 50.36(c), TS are required to include items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings, (2) limiting conditions for operation (LCOs), (3) surveillance requirements (SRs), (4) design features, and (5) administrative controls. 10 CFR 50, Appendix A, General Design Criteria (GDC) 10 states that "the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits [SAFDLs] are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences." GDC 26 states, in part:

> Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under

conditions of normal operation, including anticipated operation occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded.

The proposed change to the CEA drop time limits of ANO-2 TS 3.1.3.4 continues to support the intent of the rule and does not invalidate the assumption of the associated accident analyses or maintenance of SAFDLs with respect to compliance with GDCs 13 and 26.

In addition, 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," defines the limits to be satisfied regarding accident analyses requirements. The ANO-2 Emergency Core Cooling System acceptance criteria continues to be maintained given the proposed change to the CEA drop time limits,

4.2 Precedent

Entergy requested a TS change to increase the individual CEA drop time for ANO-2 via letter dated August 30, 2007 (Reference 9), as supplemented by letter dated December 5, 2007 (Reference 10). The NRC approved this request via ANO-2 TS Amendment 275 (Reference 11). This change was necessitated by the transition to Next Generation Fuel.

The previous full evaluation of the current ANO-2 TS CEA drop time limits (notwithstanding the aforementioned Amendment 275) was completed via review of Entergy letter dated June 15, 1989 (Reference 12) as approved via ANO-2 TS Amendment 100 (Reference 13).

St. Lucie Unit 2 requested a TS change to increase the CEA drop time in letter L-2009-127 (Reference 14). The NRC approved this request in TS Amendment 158 (Reference 15).

4.3 No Significant Hazards Consideration Analysis

Entergy Operations, Inc. (Entergy) has evaluated the proposed changes to the Arkansas Nuclear One, Unit 2 (ANO-2) Technical Specifications (TSs) using the criteria in 10 CFR 50.92 and has determined that the proposed changes do not involve a significant hazards consideration.

The proposed amendment increases the Control Element Assembly (CEA) individual and average drop time limits stated in ANO-2 Technical Specification (TS) 3.1.3.4, "CEA Drop Time," by 0.2 seconds.

Basis for no significant hazards consideration determination: As required by 10 CFR 50.91(a), Entergy analysis of the issue of no significant hazards consideration (NSHC) is presented below.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed change increases the TS 3.1.3.4 average and individual CEA drop time limits. The CEA drop time is required to be verified prior to Modes 1 or 2 of plant operations. The probability of an accident previously evaluated remains unchanged since the CEAs will continue to insert into the core as a result of an accident or transient condition, and CEA drop time does not in itself initiate an accident.

   The proposed change to the CEA drop time requirements have been evaluated for impact on the accident analyses. The accident analyses assumptions remain valid and, therefore, accident analysis results remain within applicable regulatory acceptance criteria.

   Based on the above, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or involve a change in the methods governing plant operation. The ANO-2 Safety Analysis Report (SAR) has previously evaluated conditions where a reactor trip fails to occur upon a valid signal or where CEAs fail to insert following a reactor trip. The proposed change to CEA drop times does not affect the SAR assumptions respective to these failure modes. In addition, CEA drop times are not associated with accident initiators.

   Therefore, this change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The increase in CEA drop times as proposed has been determined to have no significant impact on the accident analyses described in the SAR and, therefore, the proposed change does not result a significant reduction on the existing margins of safety for the fuel, the fuel cladding, the reactor coolant system boundary, or the containment building. The change in CEA drop time does not impact the fuel rod design or mechanical design analysis. The slightly slower drop time would produce a smaller impact on the fuel assembly and lower stresses on the CEAs. The accident analysis consequences are slightly more adverse, but all remain within the regulatory acceptance limits.

   Therefore, this change does not involve a significant reduction in a margin of safety.

4.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

The proposed change would change or relocate a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20, and would change or relocate an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

6.0 REFERENCES

1. U.S. Nuclear Regulatory Commission (NRC) to Entergy Operations, Inc. (Entergy), "Arkansas Nuclear One, Unit No. 2 – Issuance of Amendment RE: Increase in Licensed Power Level (TAC No. MB0789)," (2CNA040207) (ML021130826), dated April 24, 2002.

2. NRC letter to Entergy, "Arkansas Nuclear One, Unit No. 2 – Issuance of Amendment RE: Technical Specification 6.6.5, "Core Operating Limits Report (COLR)" (TAC Nos. MD6220 and MD6268)," (2CNA030807) (ML080840015), dated March 26, 2008.

3. NRC letter to Entergy, "Arkansas Nuclear One, Unit No. 2 – Issuance of Amendment RE: Technical Specification Changes and Unreviewed Safety Question Resolution related to Applicable Limits and Setpoints for Steam Generator Replacement (TAC No. MA7299)," (2CNA090002) (ML003758482), dated September 29, 2000.

4. NRC letter to Entergy, "Arkansas Nuclear One, Unit No.2 – Issuance of Amendment RE: Use of Alternate Source Term (TAC No. ME3678)," (2CNA041102) (ML110980197), dated April 26, 2011.

5. NRC letter to Entergy, "Arkansas Nuclear One, Unit No.2 – Correction to Safety Evaluation for Amendment No. 293 related to Use of Alternate Source Term (TAC No. ME3678)," (2CNA081101) (ML111511001), dated August 9, 2011.

6. NRC letter to Entergy, "Arkansas Nuclear One, Units 1 And 2 – Safety Evaluation related to Revised Dose Consequences based on Alternate Source Term (TAC Nos. MF0524 and MF0525)," (0CNA121305) (ML13326A502), dated December 24, 2013.

7. CEN-372-P-A, Revision 0, "Fuel Rod Maximum Allowable Gas Pressure," May 1990.

8. CENPD-404-P-A, Revision 0, "Implementation of ZIRLO^{TM} Cladding Material in CE Nuclear Power Fuel Assembly Designs," November 2001.

9. Entergy letter to NRC, "License Amendment Request to Revise Technical Specification 3.1.3.4, CEA Drop Time," (2CAN080701) (ML072610630), dated August 30, 2007.

10. Entergy letter to NRC, "Supplement to Amendment Request to Revise Technical Specification 3.1.3.4, CEA Drop Time," (2CAN120701) (ML073510407), dated December 5, 2007.

11. NRC letter to Entergy, "Arkansas Nuclear One, Unit No. 2 – Issuance of Amendment RE: Control Element Assembly (CEA) Drop Time (TAC No. MD6627," (2CNA030801) (ML080520223), dated March 5, 2008.

12. Entergy letter to NRC, "Control Element Assembly Drop Time Technical Specification Change Request," (2CAN068902) (Accession No. 8906260087), dated June 15, 1989.

13. NRC letter to Entergy, "Issuance of Amendment No. 100 to Facility Operating License No. NPF-6 – Arkansas Nuclear One, Unit No. 2 (TAC No. 73440," (2CNA108903) (ML021500377), dated October 12, 1989.

14. St. Lucie Unit 2 letter to NRC, "Technical Specification Modification Regarding Control Rod Assembly Drop Time (L-2009-127)," (ML091460050), dated May 21, 2009.

15. NRC letter to St. Lucie Unit 2, "St. Lucie Unit 2 Technical Specification Amendment 158 – Control Element Assembly Drop Time," (ML100990275), dated May 31, 2010.

<u>ATTACHMENTS</u>

1. Technical Specification Page Markups

2. Retyped Technical Specification Pages

Enclosure Attachment 1 to

2CAN121903

Technical Specification Page Markups
(1 page)

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual CEA drop time, from a fully withdrawn position, shall be ≤ 3.9~~7~~ seconds and the arithmetic average of the CEA drop times of all CEAs, from a fully withdrawn position, shall be ≤ 3.4~~2~~ seconds from when the electrical power is interrupted to the CEA drive mechanisms until the CEAs reach their 90 percent insertion positions with:

a. $T_{avg} \geq 525$ °F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the CEA drop times determined to exceed either of the above limits, restore the CEA drop times to within the above limits prior to proceeding to MODE 1 or 2.

b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of all CEAs shall be demonstrated through measurement prior to reactor criticality:

a. For all CEAs following each removal of the reactor vessel head,

b. For specifically affected individuals CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. In accordance with the Surveillance Frequency Control Program.

Enclosure Attachment 2 to

2CAN121903

Retyped Technical Specification Pages
(1 page)

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual CEA drop time, from a fully withdrawn position, shall be ≤ 3.9 seconds and the arithmetic average of the CEA drop times of all CEAs, from a fully withdrawn position, shall be ≤ 3.4 seconds from when the electrical power is interrupted to the CEA drive mechanisms until the CEAs reach their 90 percent insertion positions with:

a. T_{avg} ≥ 525 °F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the CEA drop times determined to exceed either of the above limits, restore the CEA drop times to within the above limits prior to proceeding to MODE 1 or 2.

b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

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