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1CAN021101

February 28, 2011

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

SUBJECT: Supplemental to License Amendment Request Related to Changes to Technical Specification – Completion Times for One Inoperable RCS Cooling Loop
Arkansas Nuclear One, Unit 1
Docket No. 50-313
License No. DPR-51

REFERENCES:
1. Entergy letter dated August 24, 2010, *License Amendment Request Changes to Technical Specification Related to Completion Times for One Inoperable RCS Cooling Loop* (1CAN081004)
2. NRC email dated October 13, 2010, *Changes to Technical Specification Related to Completion Times for One Inoperable RCS Cooling Loop – Request for Additional Information* (TAC No. ME4563)
3. Entergy letter dated November 12, 2010, *Supplement to License Amendment Request Related to Changes to Technical Specification - Completion Times for One Inoperable RCS Cooling Loop* (1CAN111001)
4. NRC email dated December 29, 2010, *Follow-up RAI for the Review of TAC No. ME4563*
5. NRC email dated January 27, 2011, *Request for Additional Information on License Amendment Request (LAR) Re.: Extension of Completion Times for One inoperable RCS Cooling Loop (Question number 2 in RAI email dated December 29, 2010 is revised in this RAI) TAC No. ME4563*

Dear Sir or Madam:

In Reference 1, Entergy Operations, Inc. (Entergy) proposed a change to the Arkansas Nuclear One, Unit 1 (ANO-1) Technical Specifications (TSs). Specifically, the change would revise several TSs to permit a greater time period for one of two required Reactor Coolant System (RCS) cooling loops to be inoperable. The affected TSs are applicable in lower Modes of Operation (Modes 4, 5, and 6). In Reference 2, the NRC requested additional information (RAI) with regard to the Entergy request. Entergy provided response to the RAI in Reference 3. The NRC provided follow-up questions to the RAI in Reference 4, as revised by Reference 5.

The attachment to this letter provides the requested follow-up information. Note that the responses to NRC questions summarize various procedural and administrative controls relating to RCS cooling loops and makeup sources. Copies of actual procedures and processes can be made available upon request.

There are no new commitments in this letter.

If you have any questions or require additional information, please contact Stephenie Pyle at 479-858-4704.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on February 28, 2011.

Sincerely,

***Original signed by Christopher J. Schwarz***

CJS/dbb

Attachment: Follow-up RAI – Completion Times for One Inoperable RCS Cooling Loop

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**Attachment to**

**1CAN021101**

**Follow-up RAI**
**Completion Times for One Inoperable RCS Cooling Loop**

**Request for Additional Information**
**Completion Times for One Inoperable RCS Cooling Loop**

By letter dated August 24, 2010, Entergy Operations, Inc. (Entergy) made application to amend the Arkansas Nuclear One, Unit 1 (ANO-1) Technical Specifications (TSs).

The proposed amendment would revise several TSs: TS 3.4.6, "RCS Loops – Mode 4," TS 3.4.7, "RCS Loops – Mode 5, Loops Filled," TS 3.4.8, "RCS Loops – Mode 5, Loops Not Filled," and TS 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation – Low Water Level" to permit a greater time period for one of two required Reactor Coolant System (RCS) cooling loops to be inoperable. The affected TSs are applicable in lower Modes of Operation (Modes 4, 5, and 6).

The NRC Staff reviewed the amendment request and determined that additional information is required to complete the review. Additional information was provided to the NRC in letter dated November 12, 2010 (1CAN111001). A follow-up request for additional information (RAI) was received from the NRC via email on December 19, 2010 and appears below. The Entergy response immediately follows each question.

<u>Follow-up to RAI 1</u>

1. *Identify cases when the reactor coolant system (RCS) cooling loop or decay heat removal (DHR) loop is not operable on demand, and events of a loss of the RCS cooling loops or DHR systems in Modes 4, 5, and 6 at ANO-1. Discuss the causes of the identified cases and events to show that corrective actions taken will prevent the similar cases or events from occurring.*

<u>Response</u>

With regard to cases when a required cooling loop may not be available in Modes 4, 5, and 6, such cases are extremely limited. Short periods during transition from one cooling method to another or to another train of cooling can sometimes result in inoperability of one loop under certain conditions. The TSs currently permit sufficient time to restore cooling loops during such transitioning. Other than transitioning, however, inoperability of a required cooling loop rarely occurs, if ever. TS-required cooling loops, even if only a back-up to the in-service cooling loop, are not permitted to be removed from service in support of elective maintenance activities. Maintenance can only be performed on cooling loops that are no longer required to be operable per the TSs (generally, during RCS – Filled or defueled conditions).

If the requested TS change were approved, the idle (or standby) cooling loop could be removed from service for a limited period of time (no more than 8 hours in Modes 5 and 6) in support of testing activities. While outage testing activities cannot always be predicted, the major routine activity that the TS change would accommodate is response time testing. Response time test can cause the idle cooling pump to automatically start, resulting in system perturbations and could challenge the operating cooling loop. Therefore, the breaker for the idle pump is racked-down to prevent its start during response time testing. No reconfigurations occur on the idle train that could not be promptly recovered to support placing the idle pump in service if needed. On occasion, other short duration testing, such as that associated with the 4160V switchgear bus and system flow testing may require temporary inoperability of the idle cooling loop. In any case, recovery of the pump can be accomplished in a very short period of time. Approval of the proposed TS change will not result in substantial increased unavailability of a TS-required RCS cooling loop.

With regard to unintended loss of one or more required DHR loops in Modes 4, 5, and 6, a review of Licensee Event Reports (LERs) and site Condition Reports (CRs) was performed for the last five years. A list of each event, followed by a short summary follows:

**LER 50-368/2009-004-00**

Note that this was an event on ANO, Unit 2 (ANO-2), and had no impact on ANO-1. The following is an excerpt from the LER:

> *During the performance of planned surveillance testing, the 2K-4A Emergency Diesel Generator (EDG) automatically started. An Offsite Power Transfer Test was being performed to test automatic transfer from the Startup 3 Offsite Transformer (SU3) to the Startup 2 Offsite Transformer (SU2). This transfer test is accomplished by momentarily using a handheld jumper to simulate the loss of voltage on SU3. This causes the SU3 feeder breaker 2A-113 to the 4160 Volt Bus 2A1 (2A1) to open, which in turn closes a permissive contact in the SU2 feeder breaker 2A-111 to Bus 2A1. If SU2 and the 2A1 Bus are synchronized, the SU2 to 2A1 feeder breaker will then close immediately, thus completing a fast bus transfer. If they are not synchronized, the SU2 to 2A1 feeder breaker will not close immediately; instead, 2A1 will momentarily de-energize, then the 2A1 undervoltage relay will close the SU2 to 2A1 feeder breaker, completing a slow bus transfer. When the Offsite Power Transfer Test was initiated, a slow transfer of the 2A1 Bus occurred instead of the expected fast transfer. The slow transfer of 2A1 resulted in a momentary loss of power, for approximately two seconds, to the 4160 Volt Safety Electrical Bus 2A3 (2A3) which is powered from 2A1. The momentary undervoltage condition on 2A3 caused the 2K-4A EDG to auto start as designed. The EDG did not power 2A3, since 2A3 was successfully powered from 2A1 after the slow transfer completed. During the momentary loss of power, 2A3 automatically shed all loads as designed. This caused the running Shutdown Cooling Pump (Low Pressure Safety Injection Pump 2P-60A) to secure, which resulted in a loss of Shutdown Cooling flow to the Reactor Coolant System for approximately three and one half minutes. Shutdown Cooling flow was successfully restored using the applicable procedure.*
>
> *The cause of this event was attributed to a high resistance contact on a contact block located in the SU2 to 2A1 Bus Feeder Breaker. This high resistance contact interfered with the voltage supply to the 2A1 Bus Synchronizing Relay, resulting in the slow transfer of the 2A1 Bus.*

To resolve the immediate issue, a different contact block was selected. Testing was completed satisfactorily. However, it was noted that some routine testing coordinated largely by the Operations department was not scheduled appropriately during outages. Some Operations-related tests were scheduled with significant “float” (i.e., test could be performed anytime during a several day period) and were not tied directly to a specified plant configuration. In the above event, the test was performed within the “float” period, but on a bus supplying power to the operating DHR train (referred to as Shutdown Cooling on ANO-2). A Category A (highest level) root cause evaluation (RCA) was completed and numerous actions assigned to prevent recurrence of such an event, including the requirement to link testing with the train of equipment not being relied upon for DHR or RCS inventory control. The RCA contains substantial insight and additional information, and is available upon request.

**CR-ANO-1-2007-01279**

This report documents the lifting of a DHR train relief valve on May 10, 2007, during Mode 5, RCS – Filled conditions. The idle train was placed in service and the operating train secured in order to seat the relief valve. Steam Generators (SGs) were available. While the affected DHR train was conservatively declared inoperable, TSs continued to be met by the redundant DHR train and the RCS cooling loops via respective SGs. No loss of DHR occurred during this event.

This was a newly installed relief valve. Subsequent investigation revealed the valve had been satisfactorily pretested and no deficiencies identified related to why the valve opened at a pressure approximately 85 psig below its setting. The apparent cause evaluation concluded the most likely cause of early lift was due to a very small piece of debris lodged in the spring/washer area of the valve, which was flushed out when the valve opened. Due to ANO programs and controls with regard to maintenance and installation of such equipment, it was noted that such occurrence was extremely rare. A second pre-tested relief valve was installed in the system without event.

2. *A discussion of OP 1015.002 indicated that the steam generator nozzle dams are used during reduced inventory operations. The nozzle dams will fail if the RCS pressure (resulting from a loss of DHR) exceeds the nozzle dam design pressure without a pressure vent/release path, thus creating a LOCA event. Provide a discussion of plant procedures to show that appropriate instruction is available to prevent RCS pressurization from exceeding the nozzle dam design pressure, thus insure that reactor water is not lost as a result of DHR.*

This question pertains only to Mode 5 conditions. The RCS cannot be drained or nozzle dams installed in Mode 4. In Mode 6, the RCS is open or the reactor vessel head is removed, precluding any pressurization of the RCS upon a loss of DHR removal. In accordance with OP 1103.011, Draining and N2 Blanketing the RCS, vent paths are established via the control rod drive (CRD) top closure assemblies, SG hand-holes, and/or high point vents from the hot leg and Pressurizer when initially draining the RCS. As stated in OP 1015.048, Shutdown Operations Protection Plan, having all CRD top closure assemblies removed or the SG hand-hole cover removed places the RCS in an “open” condition. These paths, among others, are sufficient to preclude RCS pressurization upon a loss of DHR event. OP 1103.018, Maintenance of RCS Water Level, provides Operator guidance for RCS level adjustments, including adequate venting, during Modes 5 and 6.

By design, the lower SG manway must be removed to install the nozzle dam and to provide a path for the nozzle dam air supply lines in the SG bowl area. Because the ANO-1 SGs are once-through SGs, this creates a vent path up through the SG tubes into the hot leg, which penetrates the top of the SG.

As stated in the ANO-1 Nozzle Dams Safety Evaluation Report 51-1176759-00, the nozzle dam design pressure is 30 psig. With the refueling canal full, a static pressure of 25.6 psig is present on the dam. The nozzle dams are designed such that they will remain in place upon failure of the primary seal and will limit leakage to 2 gpm each at 30 psig, even if all supply air to the inflatable seals is lost. During original design, the dams were tested at 30 psig / 225 °F and shown not to exhibit permanent deformation or failed components at 45 psig. Therefore, the design leak rate if all primary seals failed or all supply air was lost is 8 gpm at 30 psig. While this more directly relates to Mode 6 conditions (reactor vessel head removed), it does provide information regarding the pressure necessary in Mode 5 for leakage to be initiated through a nozzle dam, considering that the RCS is open while SG nozzle dams are installed. This leakage can easily be accommodated by available makeup sources. Although not credible, the complete failure of a nozzle dam alone does not result in a loss of the DHR system. However, any reduction in RCS inventory will reduce the estimated time-to-boil.

Based on the nozzle dam design and the numerous RCS vent methods available, sufficient leakage through one or all nozzle dams will not readily lead to a loss of the DHR cooling loops.

<u>Follow-up to RAI 2</u>

*Specify the 'mission times' and the associated bases to restore the RCS cooling loop, or decay heat removal (DHR) loop for each of the worst cases in Modes 4, 5, and 6 for ANO-1 in the case of a total loss of the DHR systems and RCS cooling loops (hereinafter referred to as the "subject systems"). Provide a discussion of plant procedures and analysis to show that operators can restore the decay heat removal capability of the "subject systems" or alternate core cooling systems within the specified mission times.*

<u>Response</u>

Mission times are generally not applicable to emergency operating procedures (EOPs). Only the Alternate Shutdown abnormal operating procedure (a condition where the control room must be evacuated) contains mission times. However, as discussed in the original submittal, such emergencies are practiced by the Operators via the Training Facility Simulator. Historical training in this regard has not revealed a challenge to core uncovery during loss of core cooling simulations. Nevertheless, a discussion of the subject modes of operation and recovery procedures is provided below.

Mode 4

As discussed in the original and supplemental submittals related to this TS change request, the RCS remains pressurized and capable of being cooled via the SGs or via the DHR loops, depending on RCS pressure, and Pressurizer level is maintained in the normal operating band. With RCS pressure above DHR system design pressure, the RCS can easily be depressurized to place a DHR loop in service should SG cooling capability be lost. Conversely, RCS pressure may be easily raised if DHR loops were lost to accommodate cooling via the SGs. Because four means of cooling can be made available in Mode 4 (cooling via either SG or either DHR loop), it is not credible to assume an unrecoverable loss of cooling in this mode during non-accident conditions that would lead to core uncovery. Procedures are available providing Operator direction should a loss of cooling occur in Mode 4.

Defining a worse-case condition is not possible due to various equipment configurations that could exist. However, if an upper temperature limit for Mode 4 (280 °F) were assumed at four hours post-shutdown, the Time-to-Boil (TTB) would be approximately 35 minutes and Time-to-Core-Uncovery (TTCU) approximately three hours, assuming no make up to the RCS and normal post-trip Pressurizer level of 100 inches (normal operating RCS level is approximately 220 inches). Because a loss of the operating core cooling method can easily be accommodated by placing the standby train in service or via one of the methods described above, sufficient time is reasonably available to complete recovery procedure steps and restore core cooling. Note that TS 3.5.3, ECCS – Shutdown, requires makeup sources to be operable in this mode. Therefore, any challenge to core inventory is further minimized by the ability to makeup to the RCS.

When SG cooling becomes unavailable in Mode 4 (DHR system not in service), the EOPs govern corrective action. These procedures will diagnose the cause of the loss and refer the Operator to the best recovery procedure. For example, if a loss of all feedwater has occurred, the Operator will transition to the Overheating EOP. This EOP contains recovery actions for Emergency Feedwater (EFW), Main Feedwater (MFW), and Auxiliary Feedwater (AFW). Other EOPs that may be applicable depending on the event include Overcooling, Degraded Power, and Station Blackout. Abnormal Operating Procedures (AOP) that may provide additional guidance include Natural Circulation Cooldown, Reactor Coolant Pump Trip, and Loss of Steam Generator Feed, among others. These procedures work in conjunction to restore cooling sources and to transition the plant, as necessary, to place the DHR system in service.

When operating with the DHR system in service and DHR is lost, the Loss of Decay Heat Removal AOP provides guidance to the Operator for recovery. This procedure envelopes several events including a loss of RCS inventory, loss of Service Water to DHR coolers, DHR pump trips, DHR pump vortexing, etc. The procedure provides guidance for increasing pressure and starting reactor coolant pumps, if necessary, to establish cooling via the SGs when operating with a steam bubble in the Pressurizer. The significant guidance available to the Operators provides high confidence that core uncovery will be precluded in all events. In addition and as discussed in previous correspondence associated with this proposed TS change, Operators receive classroom and simulator training with specific emphasis on shutdown events such as loss of DHR removal and loss of RCS inventory.

Modes 5

Determining the “worst-case” condition for this mode is not practical due to the various activities, decay heat load, RCS inventory, and plant configurations that could exist. Therefore, an assumption of lowered inventory conditions to SG nozzle dam level (~ 371 feet) in Mode 5 at 140 °F (maximum temperature for draining the RCS) and 12 hours after shutdown (to account for approximate minimum time to stabilize and cooldown the RCS following reactor shutdown) would result in a TTB of 11 minutes and TTCU of just under 2 hours. For this reason, the proposed TS change provides an additional restriction for testing a TS-required idle DHR loop such that the idle loop must be capable of being restored and placed in service prior to the TTB calculated with consideration of the existing or projected worst-case plant configuration for the testing period. The time required for idle loop recovery can vary depending on the test to be performed. However, a 30-minute recovery time is estimated for the purposes of discussion (this is considered a conservative assumption for known routine outage-related testing). Assuming 30 minutes is required to recover the idle DHR loop and assuming full core of irradiated fuel, testing could not be performed on a TS-required idle DHR loop until approximately 21 days after initial shutdown. This is why procedures prohibit any challenge to a DHR loop when the RCS is in a lowered inventory condition (≤ 376’ 5”).

To provide a comparison relating to the affect of initial RCS level, with RCS level just above the 376.5' lowered inventory condition (approximately 377') and maintaining the other assumptions above, a 30-minute TTB is achieved approximately 6 days following shutdown. By this time in the outage, notwithstanding unforeseen circumstances, the reactor vessel head has been removed and the plant has entered Mode 6 conditions (discussed later). Based on these physical restrictions and that testing is normally performed near the end of a refueling outage (after all associated maintenance is complete), the proposed TS change with TTB restrictions will prevent the idle DHR loop from being removed from service for testing prior to core offload. Also note that when an idle loop is removed from service to support testing, it is generally placed in a configuration with the main pump breaker racked down and possibly a patch cord installed. This enables response time testing to be performed without actually starting the pump. The breaker location is less than 2 minutes from the Control Room and Operator stations. With 5 minutes estimated for an Operator to don protective clothing for working around high voltage equipment, the patch cord itself can be removed within 5 minutes (possibly up to 15 minutes depending on exact location of electrician when event occurs) and the breaker racked up and ready for pump start in an additional 3 minutes. While these local actions are taking place, the Control Room staff prepares for pump start and placing the idle DHR loop in service in accordance with procedure. Upon notification from the local Operator that the breaker has been restored, the pump will immediately be placed in service and DHR flow stabilized within the following 5 minutes. Note these estimations are conservative (account for slight delays and personnel briefings). Because Operators have been well trained in such recovery efforts and fully comprehend the seriousness of a loss of core cooling, actual restoration will likely be performed in less than the 20 minutes indicated in this example.

As discussed above, the Loss of Decay Heat Removal AOP provides guidance to the Operator for recovery. With the Pressurizer steam bubble collapsed, this procedure will guide Operators in maintaining RCS inventory while recovering a DHR loop. Options are also available such as establishing cooling using the SGs (if available) via natural circulation and forming a Pressurizer bubble (if possible) and establishing forced circulation cooling using SGs. The various causes and recovery actions for a loss of DHR are complex and cannot be adequately captured by discussion in this letter. However, procedures can be made available for NRC review upon request.

Mode 6

With fuel in the core, required DHR loops and RCS makeup sources are protected at all times when in lowered inventory conditions. The examples provided in the above Mode 5 discussion remain valid for Mode 6 since the selected TTB and TTCU calculations conservatively considered RCS open conditions. Reactor vessel head removal level is approximately equal to the transition point to lowered inventory conditions. When the RCS is filled to refueling water level following head removal, the TTB will be approximately 5.6 hours (about 4 days following shutdown) with a TTCU of more than 3 days. As refueling progresses, the TTB and TTCU calculations may account for fuel transferred to the Spent Fuel Pool (reduced core decay heat load).

Guidance and recovery actions are provided in the same procedures discussed above for Mode 5 conditions. The significant time for recovery afforded in Mode 6 provides reasonable assurance of core cooling recovery following a loss of the operating DHR system.

Follow-up to RAI 3

*RAI response stated that "not all available RCS volume is credited." The license should expand this statement by including the specific values of RCS water inventory used for applicable conditions in each of Modes 4 through 6 operations and show that the TTB calculation has considered the effect of incomplete mixing of RCS water and the values of RCS water used are conservative, resulting in minimum TTBs.*

*RAI 3 also requested the licensee to discuss the initial RCS water inventory used in the TTB calculation for TS 3.9.5 - Mode 6 conditions. The response does not provide adequate information to this part of the RAI. Since the refueling cavity water above the reactor vessel may not be completely mixed, the licensee should address the incomplete mixing effect of the RCS water on the TTB calculations.*

Response

In accordance with ANO calculation CALC-89-E-0017-01, the TTB with the reactor vessel head on credits the water in both the reactor pressure vessel (in and above the core only) and the hot legs, but not other RCS volumes (e.g. cold legs, pressurizer or SG tubes). This is one of several conservatisms used in the TTB and TTCU calculations for this configuration. With the reactor vessel head removed, credit is taken for the refueling canal inventory that exists above the reactor vessel flange. While natural convection currents will form above the core, it is recognized that this convection is not adequate to bring the bulk water temperature to the boiling point prior to some localized boiling at the surface directly above the core. The TTB calculations do not consider this phenomenon because the heat associated with such boiling will quickly dissipate. Therefore, calculation of the time to bulk-boiling was considered acceptable and appropriate for this application. DHR loops and RCS makeup sources are protected at and below lowered inventory conditions. With no canal volume available, the TTCU early in the outage (see estimates provided in response to RAI 2 above), is approximately 2 hours. This increases significantly as canal level is raised. Because the TTCU is sufficiently large due to available inventory at levels above the reactor vessel flange area, the mixing phenomenon is neglected.

Instrumentation and Control during Modes 4 through 6 Operations

*The NRC staff stated in NUREG-1449 that inadequate instrumentation and incomplete operating procedures, especially during periods of reduced inventory operations, have contributed to several loss-of-DHR cooling events at operating plants. In addressing the NUREG concerns, the licensee is requested to discuss its instrumentation capabilities during Modes 4 through 6 operations to show that the instrumentation will be available to provide visible and audible indications of abnormal conditions in (a) reactor vessel water level, (b) temperature, and (c) DHR heat-removal performance, and that it will enable operators to (1) continuously monitor key parameters during reduced inventory operations, and (2) detect the onset of a loss of DHR early enough that mitigating actions can be taken to restore DHR capability.*

<u>Response</u>

Requirements for entering lowered inventory conditions are found throughout procedures. Previous correspondence related to this TS change dated November 12, 2010 (1CAN111001), Attachment 1, Pages 6 and 7 focuses on the lowered inventory checklist (Attachment M of procedure OP-1015.002, Decay Heat Removal and LTOP System Control). This correspondence noted requirements for two separate level indicators, two independent Core Exit Thermocouple (CET) temperature indications, and various computer trends for monitoring the DHR system (among other parameters) including temperature, pressure, and flow rate. Selected instrumentation is powered from onsite instrument buses and is not reliant solely on offsite power. Operators are well trained with regard to monitoring and responding to indications that could represent a challenge to the DHR system such as lowering level, pressure/flow oscillations, and rising temperatures. As discussed previously, significant procedural guidance is provided to prevent or recover from any challenge to DHR systems. In addition, the aforementioned attachment also “flags” several critical alarms associated with DHR loop performance, CETs, valve position, RCS level, etc., prior to entering lowered inventory conditions. When an Operator receives any of these alarms and notes the alarm window as “flagged”, the importance of the indication is immediately recognized and prompt action taken accordingly.

Computer systems with several monitoring stations permit continuous monitoring and trending of desired parameters. These computer trends may also have setpoints established by the Operators to permit alarm function should the parameter exceed its expected range. Proceduralized checklists are performed to ensure sufficient instrumentation remains available as outage activities progress. Selected instrumentation addresses considerations of operating experience, such as the above NUREG and GL 88-17, and has been previously assessed through NRC inspections. Based on the above, ANO-1 meets the necessary instrument and procedural requirements intended to prevent a loss of DHR and minimize the consequences of a loss of DHR (see November 12, 2010, correspondence for additional detail).

<u>Effects of PWR Upper Internals</u>

*In NUREG/CR-5820, the NRC staff analyzed the assumed loss of DHR with the vessel upper internals in place to examine the possibility of core uncovery from a lack of coolant circulation flow. Such conditions could occur during the reflooding of the refueling pool cavity while preparing for fuel shuffling operations. Under these conditions, the vessel upper resistance to natural circulation flow between the refueling pool and the reactor may prevent the refueling water from cooling the core if the DHR cooling is lost.*

*Provide a discussion of the operating strategy or analysis addressing the NUREG-5820 concerns relating to a loss of DHR during Mode 6 with the upper internal in place for ANO-1.*

<u>Response</u>

Except during the 10-year core barrel inspection, the upper internals (plenum) is removed prior to initial flooding of the canal. The RCS is drained to the vessel flange level, the vessel head removed and placed on a stand, followed by removal of the plenum, which is normally placed in the deep end of the canal. Because incores penetrate through the bottom of the ANO-1 reactor vessel, it is not necessary to establish water shielding for the plenum during its initial removal from the vessel. If DHR were lost during the relatively short period of time between head removal and plenum removal, CETs would be continuously monitored and core makeup initiated as needed (forced flow through the down comber and out the top of the vessel) to support core cooling. When refueling is complete, the canal is drained to support reinstallation of the plenum and head.

Because the canal deep end is needed to store the core barrel during the aforementioned 10-year inspection, the canal is partially filled during these inspection outages to permit the plenum to be removed and stored in the shallow end of the canal. However, this does not prevent the initiation of forced cooling via core makeup as described previously.

While resistance to core convection cooling (hot water exiting the top of the vessel and cooler water descending back into the vessel) may be present during the limited periods in which the plenum is installed, procedures and administrative controls ensure core conditions are monitored and make up sources readily available during lowered inventory conditions, providing sufficient cooling until a DHR loop can be returned to service. Addition details are included in correspondence related to the proposed TS change dated November 12, 2010 (1CAN111001).