Entergy

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March 31, 2010

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

SUBJECT: License Amendment Request: Control Rod Drive Exercise Surveillance

Palisades Nuclear Plant
Docket 50-255
License No. DPR-20

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Nuclear Operations, Inc. (ENO) requests Nuclear Regulatory Commission (NRC) review and approval of a proposed license amendment for the Palisades Nuclear Plant. The proposed amendment requests NRC approval to add new license condition 2.C(4) stating that performance of Technical Specification (TS) surveillance requirement (SR) 3.1.4.3, which verifies control rod freedom of movement, is not required for control rod drive (CRD) 22 during cycle 21 until the next entry into Mode 3 in a maintenance or refueling outage, whichever is earlier.

SR 3.1.4.3 is required to be performed every 92 days, providing increased confidence that all full-length control rods continue to be trippable, even if they are not regularly tripped. However, exercising a control rod in accordance with this SR may aggravate existing seal degradation on CRD-22, causing excessive seal leakage, which could result in a forced shutdown. ENO has determined that the risk incurred to shutdown and repair CRD-22 is considered to be greater than the risk of continued full power operation.

Attachment 1 provides a detailed description of the proposed change, background and technical analysis, No Significant Hazards Consideration Determination, and Environmental Review Consideration. Attachment 2 provides the annotated operating license page showing the proposed change. Attachment 3 provides the revised operating license page reflecting the proposed change. Attachment 4 provides a qualitative assessment of risk due to a forced outage to repair CRD-22.

ENO requests approval of this proposed license amendment prior to May 14, 2010, with the amendment being implemented within 15 days. The approval date was selected to avoid exercising CRD-22 at the next scheduled surveillance due date of May 18, 2010.

A001
NRR

A copy of this request has been provided to the designated representative of the State of Michigan.

Summary of Commitments

This letter contains two new commitments that would apply if the proposed amendment request is approved:

- ENO will make repairs to correct the existing seal leakage on CRD-22 prior to entering Mode 2, following the next Mode 3 entry.

- ENO will perform a reactor shutdown in accordance with current procedural requirements if CRD seal leakage exceeds two gallons per minute.

There are no revisions to existing commitments.

I declare under penalty of perjury that the foregoing is true and accurate. Executed on March 31, 2010.

Sincerely,

cjs/jlk

Attachments:
1. Description of Requested Change
2. Proposed Renewed Facility Operating License Change (mark-up)
3. Proposed Renewed Facility Operating License Change (typed)
4. Qualitative Assessment of Risk

CC Regional Administrator, Region III, USNRC
Project Manager, Palisades, USNRC
NRC Resident Inspector, Palisades USNRC

ATTACHMENT 1
DESCRIPTION OF REQUESTED CHANGE

1.0 DESCRIPTION

Entergy Nuclear Operations, Inc. (ENO) requests to amend the Renewed Facility Operating License DPR-20 for the Palisades Nuclear Plant (PNP). The proposed amendment requests Nuclear Regulatory Commission (NRC) approval to add new license condition 2.C(4). The new license condition would state that performance of Technical Specifications (TS) surveillance requirement (SR) 3.1.4.3, with a 92-day frequency that verifies control rod freedom of movement, is not required for control rod drive (CRD) 22, during cycle 21. The new license condition would apply until the next entry into Mode 3 in a maintenance or refueling outage, whichever is earlier. Approval of the proposed amendment would allow CRD-22 to be omitted from the next required performances of SR 3.1.4.3.

2.0 PROPOSED CHANGE

ENO proposes a License Amendment Request (LAR) to revise the PNP Renewed Facility Operating License to add new license condition 2.C(4) stating:

> "Performance of Technical Specifications Surveillance Requirement SR 3.1.4.3 is not required for control rod drive CRD-22 during cycle 21 until the next entry into Mode 3 in a maintenance or refueling outage, whichever is earlier."

3.0 BACKGROUND

<u>Need for Revision of CRD Surveillance Requirement</u>

In accordance with the TS frequency of 92 days, SR 3.1.4.3 is due to be performed on May 18, 2010. Considering the refueling outage is currently planned to commence on October 3, 2010, the proposed LAR would potentially eliminate two surveillances of CRD-22 during cycle 21.

If the proposed LAR is approved, the expected continuing rise of CRD-22 seal leakage should still allow continued plant operation until the next scheduled refueling outage.

Throughout cycle 21, commencing on May 2, 2009, quarterly surveillance procedure QO-34, Control Rod Exercising, has been successfully performed three times. Each performance of QO-34 has resulted in increased CRD-22 seal leakage.

The following data is the current CRD seal leakage trend:

Date Performed	Pre-test leak rate	Pre-test CRD-22 temperature	Post-test leak rate	Post-test CRD-22 temperature	Leak rate delta
07/28/09	10 ml/min (7/28/09)	128 °F (07/27/09)	15 ml/min (07/31/09)	148 °F (07/29/09)	5 ml/min
10/27/09	10 ml/min (09/30/09)	132 °F (10/21/09)	50 ml/min (10/27/09)	172 °F (10/27/09)	40 ml/min
02/15/10	130 ml/min (01/26/10)	172 °F (02/11/10)	400 ml/min (02/19/10)	184 °F (02/16/10)	270 ml/min

Based on the above information, historical seal performance, and CRD temperature indications, and direct measurement of CRD seal leakoff, CRD-22 is judged to comprise the majority of this leakage. Based on historical CRD seal degradation trends, it is projected that following the next performance of QO-34 in May 2010, the seal leakage will increase to the extent that continued operation to the planned October 2010 refueling outage is unlikely. Therefore, a maintenance outage is currently planned to replace CRD-22 seals.

Need for prompt review

After the most recent successful completion of QO-34 on February 15, 2010, the magnitude of rise in CRD seal leakage was greater than anticipated. This rise in seal leakage is mostly attributed to CRD-22, based on seal leak-off line temperature indication specific to CRD-22.

The magnitude of increase in CRD seal leakage was not anticipated based on actions taken to improve CRD seal reliability. A root cause evaluation was performed in 2008 to evaluate the causes of prior CRD seal leakage. The root cause was CRD seal area cooling. A potential contributing cause was the quality of the CRD rotating seal components. Corrective actions were implemented during the 2009 refueling outage

Additional material testing was performed in cycle 20. Further investigation of the contributing cause included destructive testing performed at the B&W laboratory in Lynchburg, Virginia, on two seals that had failed in August of 2008. The testing revealed that the stationary seals had fabrication-related material defects. A stationary seal supplied by a second vendor was also destructively examined. Comparison of the seals identified that CRD seals installed during the 2007 refueling outage (1R19) were of inadequate quality. This led to replacement of all 45 CRD seals during 1R20 in 2009.

A comparison of Fort Calhoun Station CRD seal area temperatures (Fort Calhoun has similar CRD mechanisms) to PNP CRD seal area temperatures lead to the identification of ambient temperature in the CRD seal area as the root cause. A design change was implemented to extend the shroud around the reactor vessel head to improve ventilation air flow in the area of the CRD seals and reduce general area temperatures.

In addition, ENO adopted a primary coolant system (PCS) vacuum fill process. A benefit of the vacuum fill process is to minimize the introduction of debris between the CRD seal faces. Debris was identified as potentially contributing to CRD leakage.

Based upon replacing all CRD seals, improving ventilation air flow in the CRD seal area, and implementing vacuum fill during the 2009 refueling outage, ENO had reasonable basis to expect that cycle 21 would be completed without mid-cycle shutdowns for seal repair.

Risk Associated with Additional Unit Shutdown

Access to the CRD mechanisms for replacement of the seals requires a plant shutdown, a cooldown to cold shutdown, and partial draining of the PCS. Eliminating this surveillance for CRD-22 should eliminate the need for a maintenance outage prior to the planned refueling outage. By eliminating an additional maintenance outage the following plant challenges will not be necessary:

- A significant power transient
- A plant thermal cycle
- Operation of plant safety systems (Auxiliary feedwater and shutdown cooling)
- Exposure to plant conditions where the PCS pumps and steam generators are not available for decay heat removal
- Radiation exposure to plant personnel
- Generation of radioactive waste due to boration, dilution, and maintenance activities

ENO has performed a qualitative evaluation of overall risk (Attachment 4) associated with an additional plant shutdown to repair the CRD-22 seal. The evaluation concluded that the risk incurred to shutdown and repair CRD-22 is considered to be greater than the risk of continued full power operation for an equivalent period of time.

This conclusion is based on the:

1. Increased frequency for a number of initiating events as compared to full power operation (e.g., loss of main feedwater, loss of rear bus offsite power, loss of shutdown cooling, etc.)
2. Reduction in available decay heat removal paths resulting from the vented PCS and the unavailability of the steam generators

The most significant increase in risk is from the loss of redundancy in decay heat removal due to the venting of the PCS, and subsequent unavailability of the steam generators as a decay heat removal path.

Also, industrial safety risks and increased occupational doses will result from an additional shutdown. The additional maintenance tasks that will have to be repeated during the upcoming refueling outage result in an increased industrial safety risk. Occupational doses, both for the control rod drive seal repair activity and for subsequent normal online maintenance, surveillance and inspection activities, would increase approximately 2-rem due to the additional shutdown.

<u>Description of Control Rod Drives</u>

PNP fuel assemblies and control rods are arranged as shown in Figure 3. The 45 cruciform control rods (Figure 4) move vertically in channels between the fuel assemblies, as opposed to other pressurized water reactor stations that use a control cluster design, which inserts into the fuel assemblies via guide tubes. (Figures 3 and 4 are historical and the dimensions should be considered for information only.) Guide bars on the sides of the PNP fuel assemblies guide the control rods and prevent the rods from contacting the fuel rods or spacer grids. The total stroke of the control rods is approximately 131 inches. The active portion of the control rods is constructed of rectangular stainless steel tubes filled with a silver-indium-cadmium alloy and welded together to form the cruciform shape. Four of the control rods have neutron absorber only at their lower ends. These part-length rods are not used at this time. They are fully withdrawn during reactor operation and, since they are not equipped with a clutch mechanism, do not insert on a reactor trip. The CRDs for the part-length rods are identical to those for the full-length rods with the exception that they use a solid shaft in place of the clutch. A full-length control rod weight with its rack extension is more than 300 pounds.

The CRDs are rack and pinion type as illustrated in Figures 1 and 2. The rack and pinion drive package contains a drive motor, position indication equipment, and a releasing clutch, which is outside the PCS boundary. The drive shaft, right angle gear set, pinion gear, and rack, are within the PCS boundary. The drive package is connected to the drive shaft through a mechanical seal that forms the PCS pressure boundary. Leakage through the face-type rotating seal enters a cavity that is routed to a collection header, and which is sealed at the top by a vapor seal, as shown in Figure 5. Each face-type rotating seal has a thermocouple to measure leak-off temperature. The leakoff from all 45 CRDs is collected in a normally unpressurized common header and routed to the containment sump. The collection header directs seal leakage away from the reactor head.

The CRD drive motor is connected to the drive shaft through a reduction gear. A spring engaged, electrically released brake is provided to prevent the control rod from drifting when the motor is not energized. The motor is fractional horsepower. The DC brake is energized through separate contacts on the motor contactor.

When the CRDs are driven outward, the motor and brake are energized, and the motor drives through the gearbox, turning the clutch upper half. If the clutch is energized (engaged) the clutch lower half is also turned. A cam and roller assembly, concentrically located within the electric clutch, transmits torque in only one direction, and allows the motor to drive the rod inward even when the electric clutch is disengaged.

The lower half of the clutch is connected to the vertical drive shaft, which turns the horizontal pinion gear through a right angle bevel gear set. The pinion gear drives the rack up and down. The rack assembly is connected to the control rod.

The CRD rack is guided by a support tube. The rack has a larger diameter section, called a buffer piston, at its upper end. The guide tube has a restricted diameter toward its lower end. In this restricted diameter region there is a close fit between the buffer piston and the guide tube. When the buffer piston enters this restricted diameter region, water trapped below it acts as a brake to slow the fall of the rod.

Below the lower clutch jaw, a small gear set drives the primary position indication shaft. The primary position indication provides a digital rod position readout. A secondary position indication system, using magnetic reed switches, is actuated by a magnet located in the connector nut at the top of the rack assembly.

When a reactor trip signal interrupts power to the CRD clutch, the clutch jaws spring apart, and the control rod falls by gravity into the core. With the clutch disengaged, the CRD parts below the clutch rotate separately from the gear motor and brake above the clutch. All CRD parts below the clutch (lower clutch shaft, primary position shaft, mechanical seal, drive shaft, bevel gears, pinion gear, magnet for secondary position indication, and rack) move whenever the rod moves.

The CRD safety function assumed in the analysis is to release the clutch and drop the control rod on a reactor trip signal. The safety analyses assume that the most reactive rod remains fully withdrawn when a trip occurs.

The CRD and rod control system provide a backup to the trip function by driving the full-length control rods inward on a reactor trip signal until they are fully inserted. This feature is referred to as "rod rundown." The rod rundown feature is not assumed in the safety analyses. The rod rundown is provided to insert a rod that has a faulty clutch or mechanical binding preventing free fall, but not preventing insertion by the motor.

Elevated seal leakage from a CRD seal is detected by an indicated increase in the CRD leakoff temperature. Individual CRD seal leakoff temperatures are available for review and trending on a chart recorder in the control room. Leakage measurement from individual seals is not possible. Combined leakage from all seals, collected in the common seal leakoff header, can be measured locally inside containment. Since the seal leakoff header flow is directed to the containment sump, observing the rate of containment sump level rise can also be used to approximate the combined seal leakoff flow rate if there is not significant leakage from other sources.

Description of Current TS Requirements

Limiting Condition for Operation (LCO) 3.1.4, "Control Rod Alignment," requires that all control rods be operable.

Condition 3.1.4 D. allows for one full-length control rod to be immovable, but trippable. The required action for this condition is to restore the affected control rod to operable status prior to entering Mode 2, following Mode 3 entry.

SR 3.1.4.3 verifies control rod freedom of movement at least once per 92 days by moving each individual full-length control rod that is not fully inserted into the reactor core ≥ 6 inches in either direction.

SR 3.1.4.6 demonstrates control rod trippability by verifying that each full-length control rod drop time is ≤ 2.5 seconds and has a required performance frequency of prior to reactor criticality, after each reinstallation of the reactor head.

TS Bases for Surveillance Requirement SR 3.1.4.3

Verifying each full-length control rod is trippable would require that each full-length control rod be tripped. In Modes 1 and 2, tripping each full-length control rod would result in radial or axial power tilts, or oscillations. Therefore, individual full-length control rods are exercised every 92 days to provide increased confidence that all full-length control rods continue to be trippable, even if they are not regularly tripped. A movement of 6 inches is adequate to demonstrate motion without exceeding the alignment limit when only one control rod is being moved. The 92-day frequency takes into consideration other information available to the operator in the control room, and other surveillances being performed more frequently, which add to the determination of operability of the control rods. At any time, if a control rod(s) is immovable, a determination of the trippability of the control rod(s) must be made, and appropriate action taken. Condition 3.1.4 D would apply whenever it is discovered that a single full-length control rod cannot be moved by its operator, yet the control rod is still capable of being tripped (or is fully inserted).

TS Bases for Action D.1

Condition D is entered whenever it is discovered that a single full-length control rod cannot be moved by its operator, yet the control rod is still capable of being tripped (or is fully inserted). Although the ability to move a full-length control rod is not an initial assumption used in the safety analyses, it does relate to full-length control rod operability. The inability to move a full-length control rod by its operator may be indicative of a systematic failure (other than trippability) that could potentially affect other rods. Thus, declaring a full-length control rod inoperable in this instance is conservative since it limits the number of full-length control rods that cannot be moved by their operators to only one. The completion time to restore an inoperable control rod to operable status is stated as prior to entering Mode 2 following next Mode 3 entry. This completion time allows unrestricted operation in Modes 1 and 2 while conservatively preventing a reactor startup with an immovable full-length control rod.

For the control rods to be trippable and able to perform their safety function, even if they are not regularly tripped, the control rods must insert on a reactor protection system (RPS) signal and the following must occur.

1. The RPS must de-energize the CRD magnetic clutch,

2. The clutch jaws must separate,

3. The mechanical components supporting the control rod must move to allow the control rod to fall into the core under the influence of gravity.

Since dropping a control rod while at power is undesirable, actions 1, 2 and 3 are tested during shutdown conditions via the control rod drop timing test (RO-22). The rod exercising test (QO-34) is not intended to, and does not, test actions 1 or 2.

When control rods are exercised, they are individually driven in six to seven inches and then returned to their normal full-out position. This action assures that the drive motor can move the rod, but only for a short distance where maximum piston to guide tube clearances exist. The minimum distance of travel is stated in the subject surveillance requirement. The maximum distance of travel is limited by the TS on control rod group alignment. A search found no instances that PNP rod exercising testing has detected any of the occurrences in which mechanical binding of mechanical components has prevented or excessively slowed full control rod insertion.

During the current cycle, QO-34, Control Rod Exercising, has been successfully performed three times for all full-length control rods.

4.0 TECHNICAL ANALYSIS

<u>Cycle Operational History CRD-22</u>

Since reactor reassembly in the 2009 refueling outage, CRD-22 has had four full-length withdrawals and three full-length insertions (two manual trips and one manual insertion). During a manual trip insertion CRD-22 was drop timed satisfactorily per RO-22, "Control Rod Drop Times."

The following table documents CRD movement history (including CRD-22) since reactor reassembly during and after the 2009 refueling outage:

Date	Activity	Movement	Travel (Inches)
4/16/2009	RO-19, Control Rod Position Verification	Manual Withdrawal	~20
4/16/2009	RO-19, Control Rod Position Verification	Manual Insertion	~20
4/27/2009	RO-22, Control Rod Drop Times	Manual Withdrawal	131
4/27/2009	RO-22, Control Rod Drop Times	Manual Trip Insertion Timed	131
4/28/2009	Reactor Critical Approach	Manual Withdrawal	131
4/28/2009	RT-191, Start-up Physics Test Program	Manual Insertion	~128
4/28/2009	RT-191, Start-up Physics Test Program	Manual Withdrawal	~128
4/30/2009	Manual Reactor Trip	Manual Trip	131
4/30/2009	Reactor Critical Approach	Manual Withdrawal	131
7/28/2009	QO-34, Control Rod Exercising	Manual Insertion	~6
7/28/2009	QO-34, Control Rod Exercising	Manual Withdrawal	~6
10/27/2009	QO-34, Control Rod Exercising	Manual Insertion	~6
10/27/2009	QO-34, Control Rod Exercising	Manual Withdrawal	~6
2/15/2010	QO-34, Control Rod Exercising	Manual Insertion	~6
2/15/2010	QO-34, Control Rod Exercising	Manual Withdrawal	~6

Based on the above data, CRD-22 has been moved a total of ~957 inches since the 2009 refueling outage reactor reassembly.

Based on the total inches of travel and the interval between movements over the last 11 months, the additional 24 inches (12 inches in and 12 inches out) of movement that would be required by two additional QO-34 tests for the remainder of this fuel cycle would not provide any significant increase in confidence in CRD-22 trippability.

Continued satisfactory quarterly exercising of the remaining 40 CRDs will support maintaining this full confidence by ensuring that a common failure mechanism does not develop that would prevent CRD-22 from moving or tripping.

<u>Other testing that verifies control rod safety functions</u>

Rod drop time testing (RO-22) is performed each refueling shutdown, in accordance with the TS surveillance requirement. The test verifies each control rod reaches 90% insertion within 2.5 seconds when dropped from the full-out position, by de-energizing the clutch. This test and the associated withdrawal verify that the electrical clutch functions and that there is no significant mechanical binding of the CRD, thus verifying the control rod's ability to perform its safety function.

During cycle 21, prior to Mode 2 entry, RO-22 was successfully performed for all full-length control rods.

<u>Other evolutions that verify control rod functionality</u>

During start-up physics testing in RT-191, CRD-22 is moved from near full-out to near full-in position during rod worth measurement testing and then returned to its full-out position.

Critical approaches performed on 4/28/2009, and 4/30/2009, moved CRD-22 from a fully inserted position to a full-out position.

A manual reactor trip performed on 4/30/2009, moved CRD-22 from a full-out position to a fully inserted position.

<u>Additional confidence of CRD-22 operability during cycle 21</u>

Part-length control rods, which are fully withdrawn prior to reactor startup, are not exercised, and typically remain motionless throughout the operating cycle. Operating experience at PNP has demonstrated that these rods will insert into the core as expected during a normal shutdown, despite not being routinely exercised. Therefore, CRD-22 is expected to continue to move freely without performance of the SR.

Recent Maintenance History CRD-22

All control rod drive seal housings have had their rotating components replaced within the last 36 months. Below is a summary of recent maintenance history on CRD-22:

All CRD seals, vapor seals, and the associated o-rings were replaced in the 2009 refueling outage. Per PNP maintenance procedure CRD-M-31, Rebuilding and Testing CRDM Seal Housing Assemblies, the CRD seal bearing components are inspected and measured to ensure no damage has occurred during previous use. In the 2007 refueling outage, all the CRD seal housings were rebuilt including seal replacement. The clutch for CRD-22 was also rebuilt in 1999. Since 1999, periodic preventive maintenance has been used to prevent failures resulting from aged clutches.

Failure to trip history

A review of PNP history since 1989 identified only one instance of a control rod failure to trip. This occurred in October 1999 when CRD-14 failed to insert after a manual reactor trip. The run down feature of the CRD did drive CRD-14 fully into the core. Root cause analysis determined that the lower clutch bearing was degraded due to inadequate preventive maintenance. Since this failure, all 41 full-length control rod clutch bearings have been replaced with periodic replacement intervals established. CRD-14 had been successfully exercised during QO-34 testing in August 1999.

Assuming that CRD-22 failed to trip during an accident or transient, the rundown feature would be expected to fully insert CRD-22 into the core.

Effects of Continued CRD-22 Seal Leakage

ENO has determined that CRD seal leakage does not increase the likelihood of an untrippable control rod. This is based on the CRD system design, which precludes seal leakage from causing the clutch to fail to release, or causing mechanical binding of the driveshaft between the lower clutch face and the face-type rotating seals. All components above the lower clutch face are disengaged from the drive shaft on a reactor trip. Components below the lower seal face are normally in contact with PCS water, and therefore, will not be mechanically bound by the leakage effects. Refer to the CRD seal sketch in Figure 5.

In Modes 1, 2, 3 and 4, in steady state operation, PNP is required by current TS to determine PCS unidentified leakage on a 72-hour interval. PNP has a procedure requirement to conduct a PCS leakage determination on a 24-hour interval. Increased CRD seal leakage would be detected by this calculation.

A current station Operational Decision Making Issue implementation plan requires that if a step-change in the 60-minute average of the containment sump fill rate of greater than 0.050 gpm occurs within one shift or less, and accompanied by a detectable rise in containment building gaseous monitor, then a containment building entry will be made to measure CRD seal leak rate. This action level is well below the TS value of 1.0 gpm for unidentified PCS operational leakage.

ENO is required to shutdown the reactor if CRD seal leakage exceeds 2.0 gpm, in accordance with plant procedures. The 2.0 gpm limit conservatively bounds the TS identified leakage limit of 10 gpm. ENO commits to maintaining this shutdown requirement for the duration of the proposed surveillance interval extension.

Containment sump fill rate is currently monitored by the operations staff to ensure consistency with CRD seal leak rate. This is conducted to validate that CRD leakage is being effectively collected and routed to the containment sump. Based on this monitoring and seal design, ENO has confidence that the CRD seal leakage is not being deposited on the reactor pressure vessel head. ENO commits to make repairs to correct the existing seal leakage on CRD-22 prior to entering Mode 2, following the next entry into Mode 3.

CRD seal leakage does not increase the likelihood of a control rod becoming untrippable. In order to cause a rod to become untrippable, leakage would have to cause the clutch to fail to release, or cause mechanical binding of the CRD drive shaft between the lower clutch face and mechanical seal. All components above the lower clutch face are disengaged from the driveshaft upon a trip, and normally wetted components inside the PCS boundary will not be mechanically bound by leakage effects. CRD seal leakage effect on CRD components is further described below.

<u>Clutch</u>
In order to hinder trippability, the lower section must either fail to disengage, or bind between the shaft and some stationary component. Plausible failure modes cause the clutch to disengage (thus causing a rod trip) and not remain engaged. The clutch uses a spring bellows and jaw faces that do not depend upon sliding action. When electrical power is removed, the jaw faces separate, an action that is not prone to mechanical binding. Even if the vapor seal failed, leakage would not prevent rotation of a disengaged lower clutch element.

Bearings
There are three sets of ball bearings between the clutch and vapor seal. To prevent a rod trip, one or more of these sets would have to bind sufficiently to resist dropping of a weight in excess of 300 pounds, or degrade enough to allow gross driveshaft misalignment. The vapor seal protects the bearings from a corrosive atmosphere, and the 2.0 gpm procedural CRD seal leakage limitation reduces the likelihood of vapor seal failure.

Vapor Seal
The vapor seal is an elastomeric cup seal with a metal backing ring. The steam impingement washer protects it from erosion, and the vapor seal, in turn, protects drive components above the vapor seal from leakage. Operating temperature is dependent upon seal leakoff pressure as long as flashing occurs in the leakoff cavity. The collection header is normally unpressurized. The elastomer is designed for high temperature operation and there is no metal-to-metal contact between stationary and rotating parts. If the vapor seal were to fail, it would not prevent shaft rotation.

Steam Impingement Washer
The steam impingement washer is a thin stainless washer fit loosely around the driveshaft immediately below the vapor seal, at the top of the seal leakoff cavity. It cannot bind between the shaft and housing while remaining around the shaft, and plausible leaks will not break it.

Seal Assembly
The rotating element is inside the PCS boundary so leakage will not corrode or bind small internal parts. There is clearance between the stationary assembly and driveshaft. Shear forces will prevent binding at the seal boundary, as the seal contact area is very small and materials were selected for low friction operation. A leak-induced temperature increase can degrade the three static o-rings, but this will not prevent rotation.

Driveshaft
One end of the driveshaft is inside the PCS boundary, so component material was selected to withstand PCS effects. Driveshaft upper end alignment is maintained by the lower clutch shaft that rides in three sets of ball bearings above the vapor seal. The drive shaft lower end bearings are within the PCS boundary.

Potential Reactivity Effects

CRD-22 is a Group 1 regulating control rod. Control rods in Group 1 are the initial group of regulating control rods withdrawn during startup and the last group of regulating control rods inserted during shutdown. Group 1 regulating rods would not be moved during normal plant power maneuvering.

ENO has considered the potential reactivity effects for the proposed change. The safety analyses assume full-length control rod insertion, with the exception of the one most reactive rod (N-1), upon reactor trip. ENO has determined that not conducting SR 3.1.4.3 for CRD-22 for the remainder of the current operating cycle does not increase the likelihood of an untrippable control rod.

Conclusion

The performance of the quarterly surveillance exercise testing of CRD-22 will likely aggravate CRD seal leakage and require an unplanned shutdown. Based on the design of the CRD, a leaking seal does not affect the ability of the CRD to perform its safety function. Based on the frequency and extent of testing performed on CRD-22 to date, performance of rod exercising testing on CRD-22 an additional two times over the remainder of this fuel cycle is not necessary to have full confidence in the ability of the CRD to perform its safety function. Therefore, CRD-22 would remain trippable and can be fully inserted using the rod rundown feature, if needed. The risks to public health and safety, and to plant workers from additional radiation exposure incurred by a shutdown and seal replacement, are greater than the risks associated with full power operation and not performing the remaining two surveillance tests for CRD-22 in cycle 21.

5.0 REGULATORY SAFETY ANALYSIS

No Significant Hazards Consideration

Entergy Nuclear Operations, Inc. (ENO) requests to amend Renewed Facility Operating License DPR-20 for the Palisades Nuclear Plant (PNP). The proposed amendment requests Nuclear Regulatory Commission approval to add new license condition 2.C(4). The new license condition would state that performance of Technical Specifications (TS) surveillance requirement (SR) 3.1.4.3, with a 92-day frequency that verifies control rod freedom of movement, is not required for control rod drive (CRD) 22 during cycle 21. The new condition would apply until the next entry into Mode 3 in a maintenance or refueling outage, whichever is earlier.

ENO has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

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

   Response: No.

The proposed license amendment adds a license condition to forgo the remaining two required surveillance tests of one control rod from the PNP TS surveillance requirement for partial movement every 92 days. Since the control rod remains operable, the proposed license condition does not affect or create any accident initiators or precursors. As such, the proposed license condition does not increase the probability of an accident.

The proposed license amendment does not increase the consequences of an accident. The ability to move a full-length control rod by its drive mechanism is not an initial assumption used in the safety analyses. The safety analyses assume full-length control rod insertion, except the most reactive rod, upon reactor trip. The surveillance requirement performed during the last refueling outage verified control rod drop times are within accident analysis assumptions. ENO has determined that CRD seal leakage does not increase the likelihood of an untrippable control rod. The assumptions of the safety analyses will be maintained, and the consequences of an accident will not be increased.

Therefore, operation of the facility in accordance with the proposed license condition would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed license condition does not involve a physical alteration of any structure, system or component (SSC) or change the way any SSC is operated. The proposed license condition does not involve operation of any required SSCs in a manner or configuration differently from those previously recognized or evaluated. No new failure mechanisms would be introduced by the requested SR interval extension.

   Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No.

   The proposed license condition does not affect operability of the control rod. It will have the same capability to mitigate an accident as it had prior to the proposed license condition.

Therefore, the proposed amendment would not involve a significant reduction in a margin of safety.

Based on the evaluation above, ENO concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

Applicable Regulatory Requirements/Criteria

NUREG-1432, "Standard Technical Specifications Combustion Engineering Plants," contains a SR to verify freedom of movement of control rods every 92 days. The proposed amendment deviates from NUREG-1432 in that it effectively extends this SR interval for CRD-22, as described above.

In conclusion, based on the considerations described 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.

6.0 ENVIRONMENTAL CONSIDERATION

ENO has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment 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 amendment.

FIGURES 1 - 5

Five pages follow

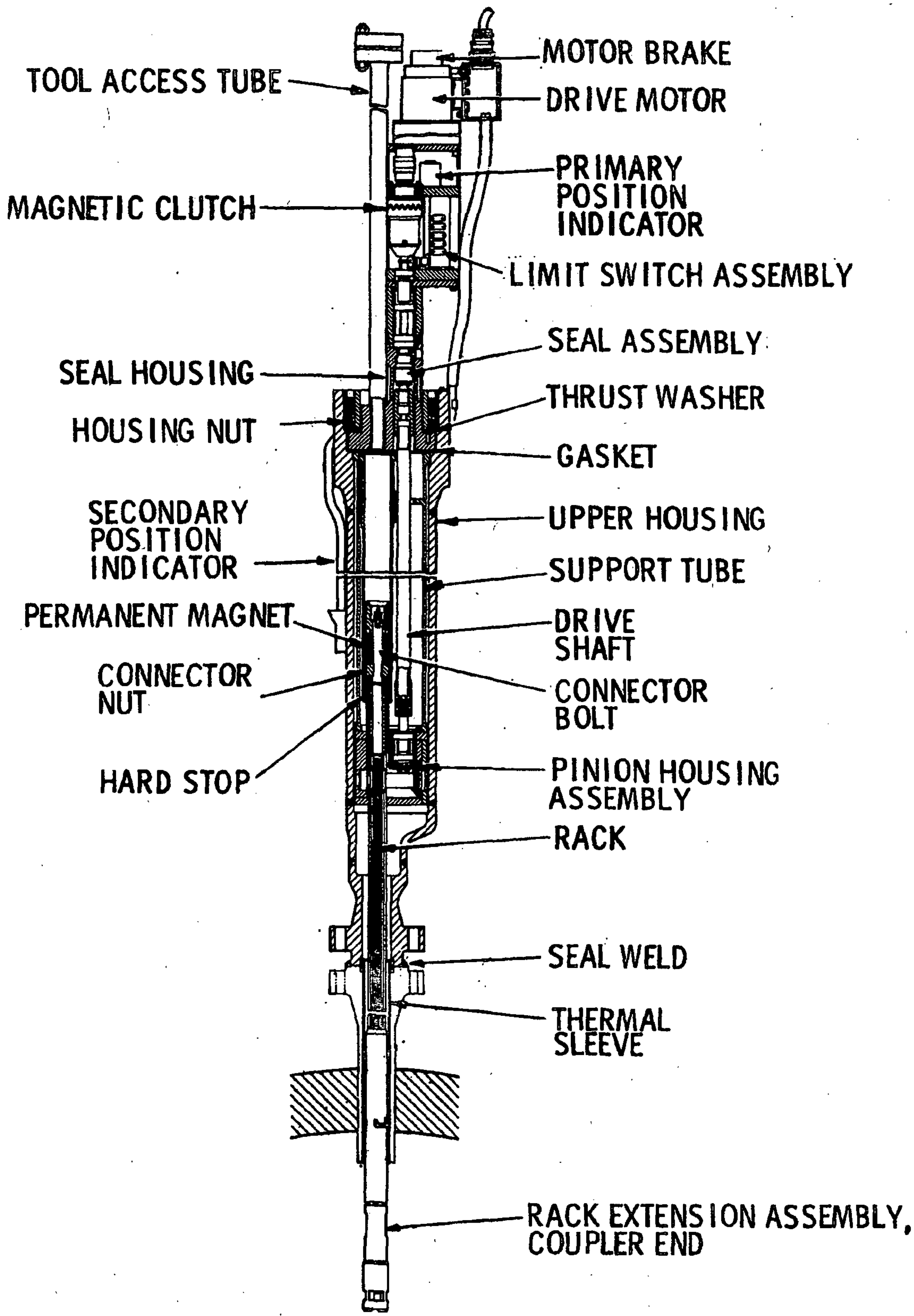

Figure 1: Control Drive Mechanism

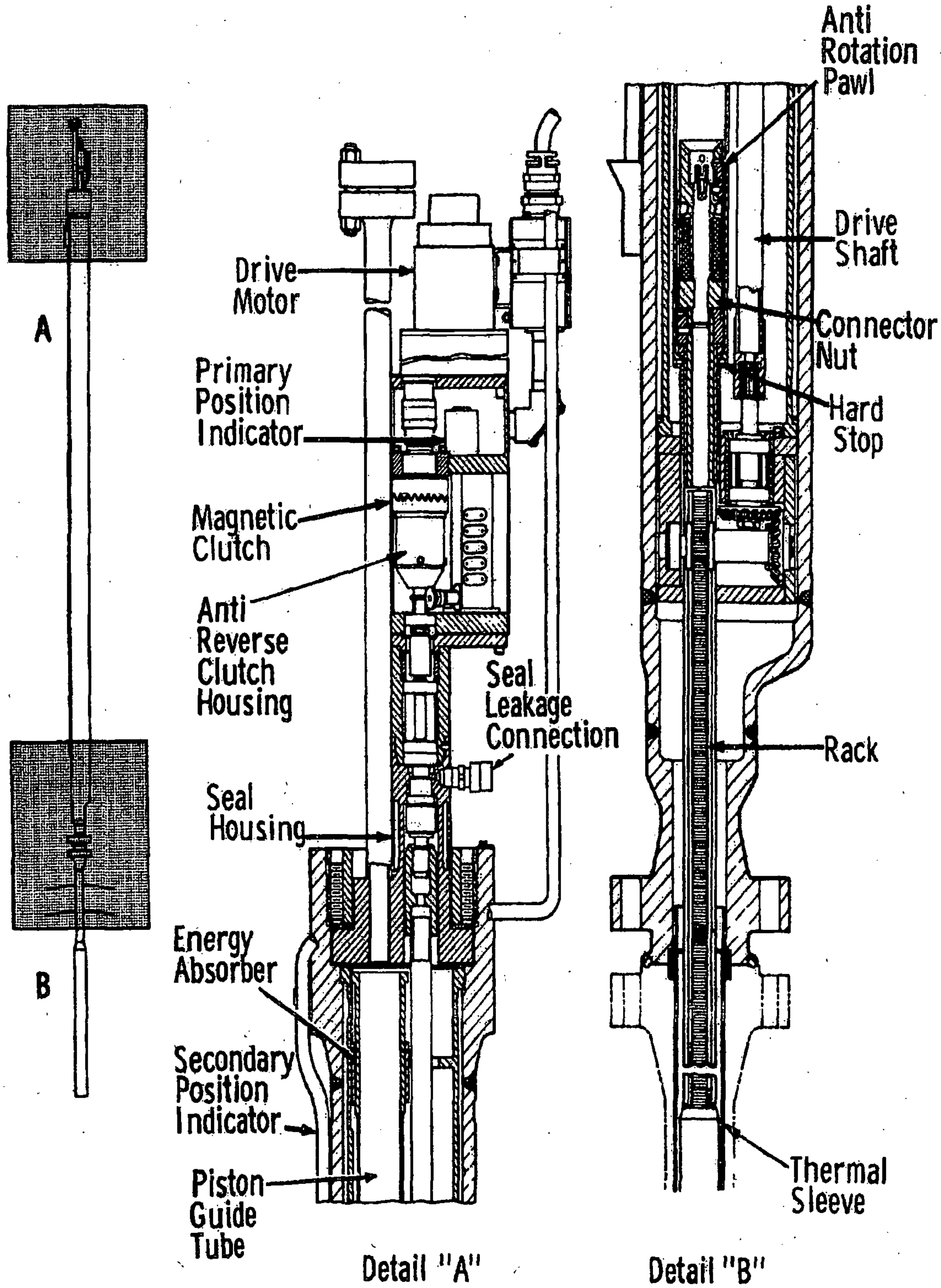

Figure 2: Control Rod Drive Mechanism

For Information only

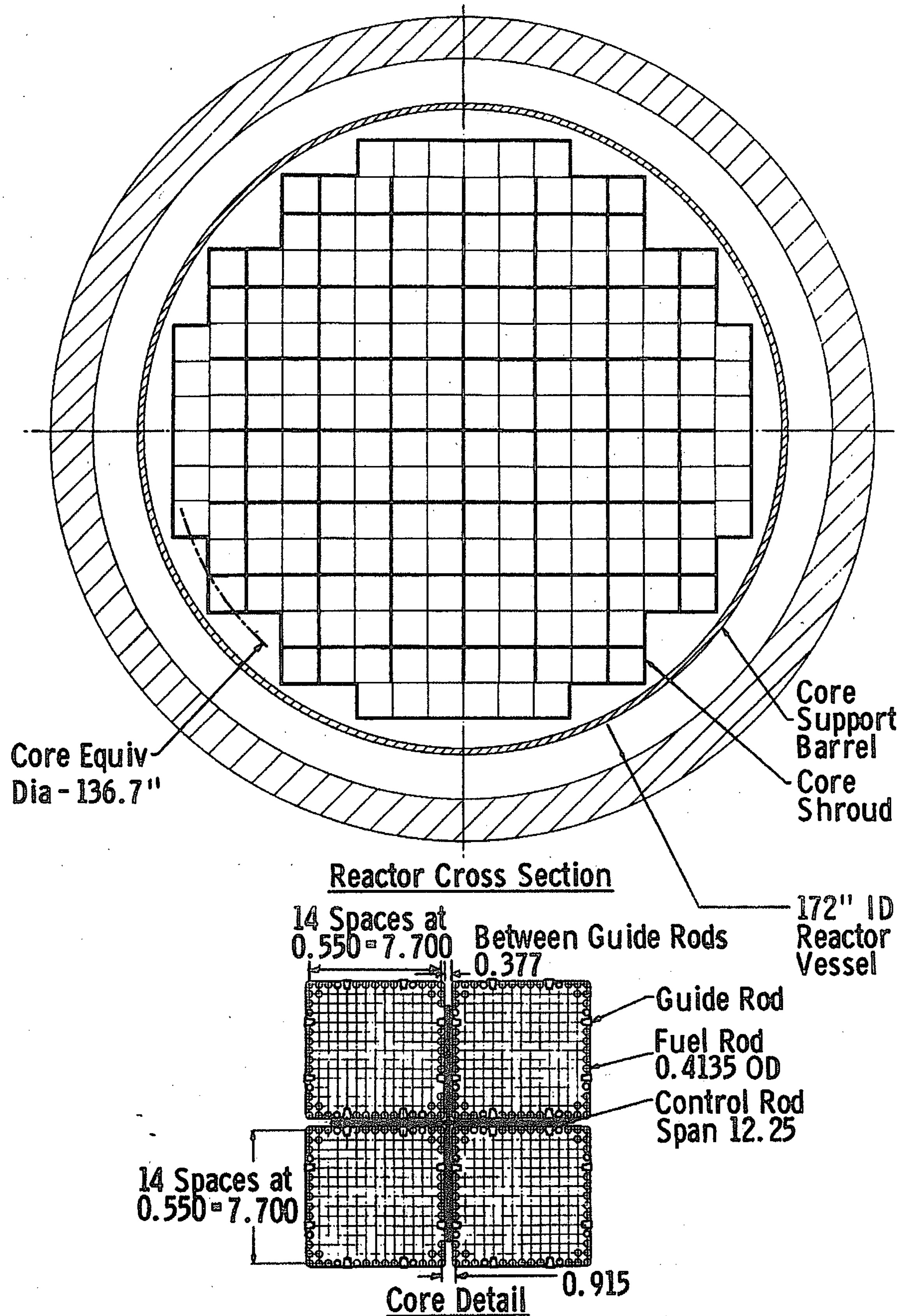

Figure 3

For information only

CONTROL ROD

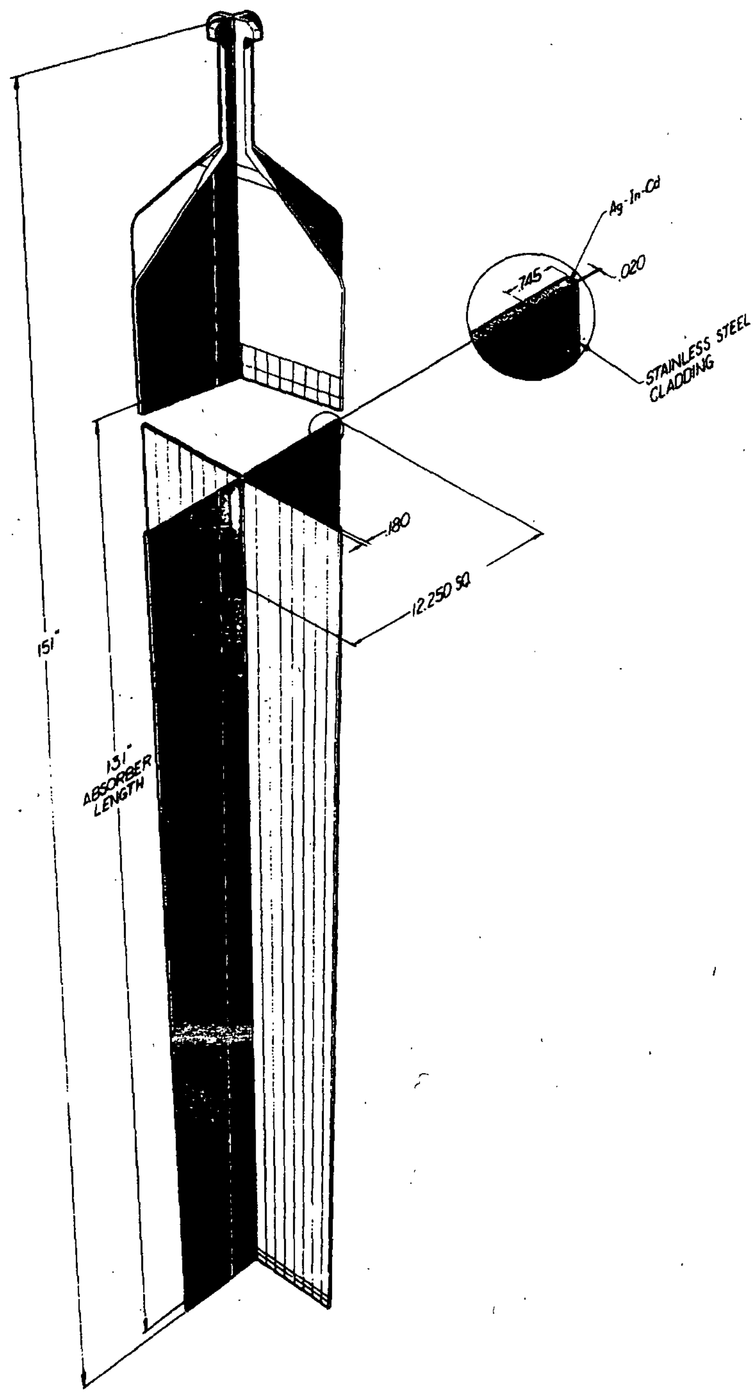

Figure 4

Control Rod Drive Seal Sketch

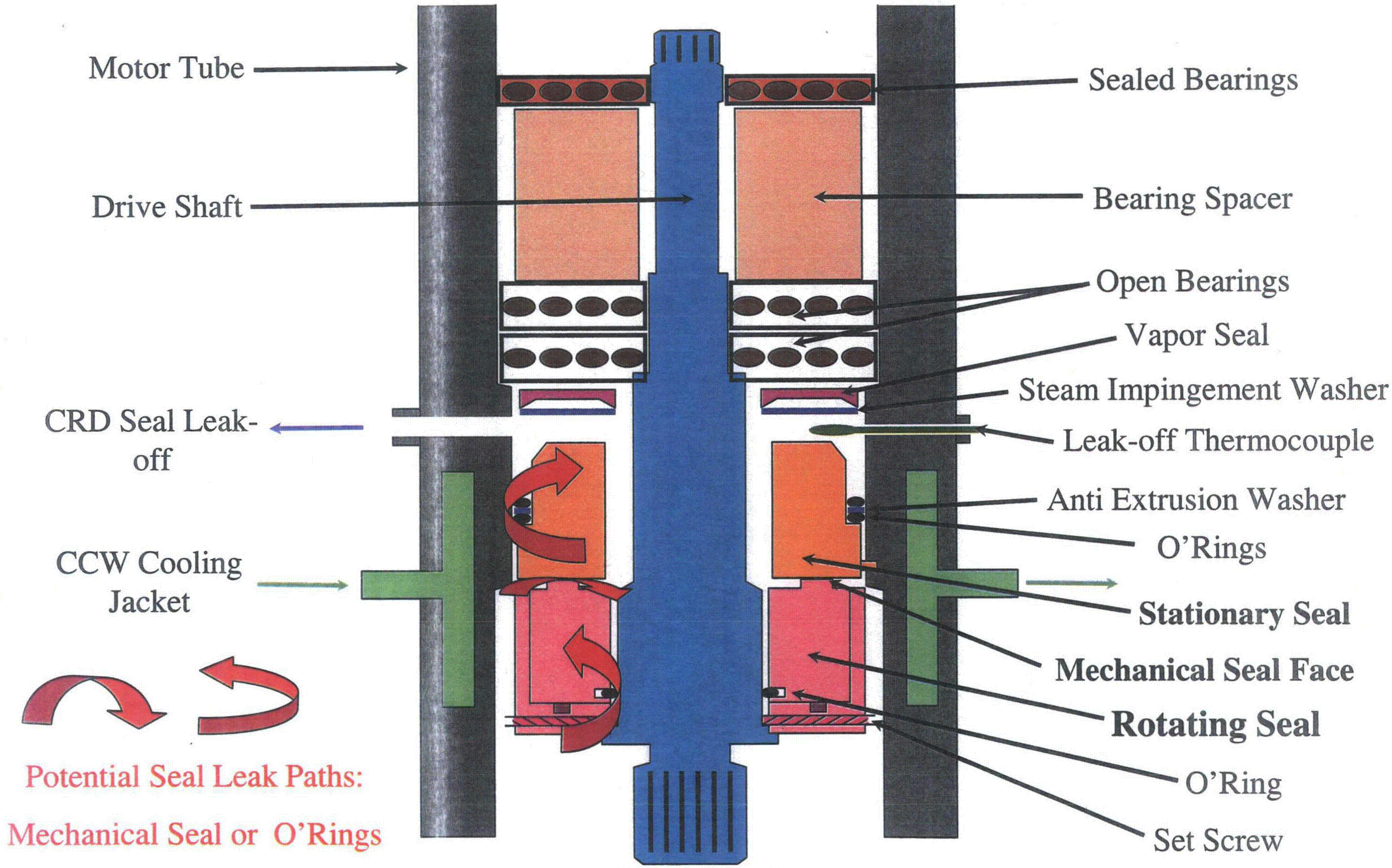

Figure 5

ATTACHMENT 2

PROPOSED RENEWED FACILITY OPERATING LICENSE CHANGE
(mark-up)

Page 4

(The addition is highlighted and the deletion has strikethrough)

One page follows

a. ENO may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

b. ENO may alter specific features of the approved fire protection program provided:

- Such changes do not result in failure to complete the fire protection program as approved by the Commission. ENO shall maintain in auditable form, a current record of all such changes, including an analysis of the effects of the change on the fire protection program and shall make such records available to the Commission Inspectors upon request. All changes to the approved program shall be reported along with the FSAR revision as required by 10 CFR 50.71(e); and

- Temporary changes to specific fire protection features which may be necessary to accomplish maintenance or modifications are acceptable provided interim compensatory measures are implemented.

(4) ~~[deleted]~~ Performance of Technical Specifications Surveillance Requirement SR 3.1.4.3 is not required for control rod drive CRD-22 during cycle 21 until the next entry into Mode 3 in a maintenance or refueling outage, whichever is earlier.

(5) [deleted]

ATTACHMENT 3

PROPOSED RENEWED FACILITY OPERATING LICENSE CHANGE
(typed)

Renewed Facility Operating License Page Change Instructions

And

Revised Renewed Facility Operating License Page 4

Two pages follow

ATTACHMENT TO LICENSE AMENDMENT NO.

RENEWED FACILITY OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Remove the following page of the Renewed Facility Operating License and replace it with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

REMOVE	INSERT
Page 4	Page 4

a. ENO may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

b. ENO may alter specific features of the approved fire protection program provided:

- Such changes do not result in failure to complete the fire protection program as approved by the Commission. ENO shall maintain in auditable form, a current record of all such changes, including an analysis of the effects of the change on the fire protection program and shall make such records available to the Commission Inspectors upon request. All changes to the approved program shall be reported along with the FSAR revision as required by 10 CFR 50.71(e); and

- Temporary changes to specific fire protection features which may be necessary to accomplish maintenance or modifications are acceptable provided interim compensatory measures are implemented.

(4) Performance of Technical Specifications Surveillance Requirement SR 3.1.4.3 is not required for control rod drive CRD-22 during cycle 21 until the next entry into Mode 3 in a maintenance or refueling outage, whichever is earlier.

(5) [deleted]

ATTACHMENT 4

QUALITATIVE ASSESSMENT OF RISK

Engineering Report No. PLP-RPT-10-00041 Rev. 0

Qualitative Assessment of the Risk Due to an Additional Forced Outage to Repair a Leaking Seal on CRDM-22

Eleven pages follow

Engineering Report No. **PLP-RPT-10-00041** Rev **0**

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ENTERGY NUCLEAR
Engineering Report Cover Sheet

Engineering Report Title:

Qualitative Assessment of the Risk Due to an Additional Forced Outage To Repair a Leaking Seal on CRDM #22

Engineering Report Type:

New ☒ Revision ☐ Cancelled ☐ Superseded ☐
Superseded by: ____________

Applicable Site(s)

IP1	☐	IP2	☐	IP3	☐	JAF	☐	PNPS	☐	VY	☐	WPO	☐
ANO1	☐	ANO2	☐	ECH	☐	GGNS	☐	RBS	☐	WF3	☐	PLP	☒

EC No. **21107**

Report Origin: ☒ Entergy ☐ Vendor
Vendor Document No.: ____________

Quality-Related: ☐ Yes ☒ No

Prepared by: JLVoskuil / J. L. Voskuil Date: 3-31-2010
Responsible Engineer (Print Name/Sign)

Design Verified: n/a Date: ____
Design Verifier (if required) (Print Name/Sign)

Reviewed by: BABrogan / Brian Brogan Date: 3-31-2010
Reviewer (Print Name/Sign)

Approved by: Ron Penna Date: 3-31-2010
Supervisor / Manager (Print Name/Sign)

Issue

This evaluation provides a qualitative evaluation of the risk resulting from an additional plant shutdown to repair the seal for control rod drive (CRD) #22.

Evaluation Summary

The overall risk associated with a plant shutdown to repair the CRD #22 seal is considered to be greater than the overall risk for an equivalent period of time at full power operation. No additional full power operation risk is ascribed to the condition of marginally increased control rod drive seal leakage. This conclusion is based on the:

(1) Increased frequency for a number of initiating events as compared to at full power operation (e.g., loss of main feedwater, loss of rear bus offsite power, loss of shutdown cooling, etc.)
(2) Reduction in available decay heat removal paths resulting from the vented primary coolant system and the unavailability of the steam generators

The most significant increase in risk is from the loss of redundancy in decay heat removal due to the venting of the primary coolant system and subsequent unavailability of the steam generators as a decay heat removal path.

Also, industrial safety risks and increased occupational doses will result from an additional shutdown. The additional maintenance tasks that will have to be repeated during the upcoming refueling outage result in an increased industrial safety risk. Occupational doses both for the control rod drive seal repair activity and for subsequent normal on-line maintenance, surveillance and inspection activities will increase due to the additional shutdown.

Note that drain-down to lowered or reduced inventory is not needed to repair control rod drive seals. Some additional risk is avoided as compared to an outage that requires lowered and/or reduced inventory conditions since there is less exposure to the increased risk of core uncovery events (due to shorter times to boil, the unavailability of natural circulation and the increased potential for loss of decay heat removal due to loss of PCS level control) that would otherwise occur during mid-loop operations.

Also note that extensive system repair, maintenance and testing as compared to a typical refueling outage are not planned for the outage to repair control rod drive seals. With substantially fewer systems disassembled and far less maintenance and repair activities occurring, it is considered that the likelihood of occurrence of fires, internal floods (as well as external events) is not increased significantly as compared to normal full power operation. However, the consequences of fires, internal floods and other external events (seismic events, tornados, high winds, etc.) are considered to be increased as a result of the loss of redundancy in decay heat removal due to the venting of the primary coolant system and subsequent unavailability of the steam generators as a decay heat removal path.

Detailed Evaluation

This evaluation specifically considers the following evolutions:

(1) Transition from full power operation (mode 1) to hot standby (mode 3)
(2) Transition from hot standby (mode 3) to shutdown cooling entry and cooldown to cold shutdown (mode 5)
(3) Cold shutdown (mode 5) operation for the duration of the outage
(4) Transition from cold shutdown (mode 5) to shutdown cooling exit and heatup to hot standby (mode 3)
(5) Transition from hot standby (mode 3) to full power operation (mode 1)

The operation in cold shutdown for CRD seal repair includes venting the primary coolant system (PCS) and draining down to ~640' elevation. While this water level remains above the lowered inventory level (defined as at or below the reactor vessel flange with fuel in the reactor) at 624'6" and remains above the reduced inventory level (formally defined as lower than 3' below the reactor vessel flange with fuel in the reactor) taken at 623', the PCS is not intact for heat removal via the steam generators.

(1) Transition from full power operation to hot standby
This evolution covers a planned and controlled turbine/generator and reactor shutdown per plant procedure GOP-8, Power Reduction and Shutdown to Mode 2. Starting from normal power operation, both turbine and reactor power are lowered. Power is reduced by injecting boron to drop the reactivity and manually controlling the turbine to maintain T_{ave}. Feedwater continues to be provided by the main feedwater (MFW) system in automatic control. At about 60% power, one MFW pump is placed in manual and its speed is lowered to the minimum governor speed. The idled pump remains available if the other MFW pump is lost. At about 25% power the idled main feedwater pump is shut down.

The 4160 volt buses are transferred from station power to the startup transformers at about 20% power. The 2400 volt buses remain connected to the safeguards transformer. The 2400 volt buses may be realigned to startup transformer 1-2 when the plant is off-line, but this is not normally performed.

Turbine/generator unloading continues until power is less than about 8% or until annunciator EK-0107 for the turbine no load pre-trip alarms. The turbine is then tripped which also opens generator output breaker 25F7 (output breaker 25H9 is opened previously). Steam is delivered to the turbine until the turbine is tripped, at which point steam flow is directed to the condenser via turbine bypass valve CV-0511. Reactor power is then reduced to about 1-2% power and feedwater supply is transferred from main to auxiliary feedwater (AFW). The reactor is then shutdown by inserting control rods to obtain a negative startup rate, and then manually tripping the reactor. At this point the PCS is still at full shutdown pressure and temperature (532°F and 2060 psia), with reactor neutron power and decay heat at approximately 10^{-6}% and 3% or less, respectively.

As reactor power level lowers, reactor trip set points are automatically adjusted. In addition, some balance of plant (BOP) equipment such as heater drain pumps and the idled MFW pump are shut down at lower power levels.

The key similarities between this evolution and full power operation are:

- Feedwater is supplied by the MFW system for most of the evolution
- Feedwater control is mainly in the automatic mode
- BOP systems are generally aligned the same and are in operation
- Safety systems are in standby and available for use if needed
- PCS temperatures and pressures are similar (PCS Tave lowers from 560°F to 532°F)
- Secondary side temperatures and pressures are similar (steam generator (SG) pressure increases from about 780 psig to 900 psig)
- Same types of initiating events (IE) are possible

The key differences between this evolution and full power operation are as follows:

- Plant is not operating in a stable steady state (risk neutral)
- Operators are controlling the turbine by periodically manually adjusting the turbine auto controls (risk increase)
- Reactor power level is decreasing (risk decrease)
- Decay heat that would exist immediately after a reactor trip is lower (risk decrease)
- One MFW pump is shut down (at about 25% turbine/generator power) (risk increase)
- MFW control is transferred to manual near the end of the evolution (risk increase)

- Power source for the 4160 volt buses is transferred to the startup transformers (switchyard rear bus) (risk increase)
- Selected RPS trip parameters may be bypassed or have different setpoints,and other trip parameters may be enabled (risk neutral)

A review of the differences and similarities between this evolution and full power operation indicates in general that the plant is more susceptible to the transient/reactor trip initiation during the evolution but that the plant and system response would be similar. As noted above, the decay heat that would exist immediately after a trip is lower for a trip during this evolution than during full power operation and is influenced by the reactor power at any given point in time. Transients (except for excess steam demand events) would tend to progress somewhat slower than an equivalent transient initiated from full power.

Operator focus on shutdown related activities may delay diagnosis and response. Peak PCS pressures achieved during a pressure related transient would be lower due to the lower driving energy. While this evolution is subject to the same type of initiators as full power operation, the frequencies may be different. The feedwater initiating event frequency may increase due to the potential for human error when the control system is in manual. However, the AFW system and other systems would still be available and in the same configuration as in full power operation. A loss of feedwater event is less severe at lower power levels due to the decreased initial steam flow from the steam generators.

At about 20% power the operators realign the 4160 volt buses to draw power from the grid via the startup transformers. This could lead to an increased exposure to a loss of offsite power due to a disconnect from the rear bus. The diesel generators, the AFW system and other systems would remain in auto. The system response to a loss of onsite power during this evolution would be similar the response during full power operation.

Since primary temperatures and pressures are similar to those at power (there will be some changes over the range of the evolution) little impact on the component of the initiating event frequencies related to the pressure stressor for loss of coolant accidents (LOCAs) and steam generator tube ruptures (SGTRs) is expected. Inherent plant response is expected to be less severe for lower or zero power LOCAs and SGTRs based on less core stored energy and higher steam generator pressures, respectively.

Despite higher secondary pressures, little impact on the component of the initiating event frequencies related to the pressure stressor for main steam line breaks (MSLBs) is expected. However, a somewhat more severe blowdown and subsequent thermal-hydraulic-neutronic response is expected for MSLBs initiated at non-full power conditions due to the higher steam generator pressure resulting in an increased energy release rate due to the higher mass release rate and ultimately a more severe cooldown. The number and redundancy of the systems available to respond to these accidents is reduced as compared to normal full power operation. For example, the MFW system which can be used to respond to small break LOCAs and SGTRs has less redundancy. Early in the evolution, the feedwater system condition is the same as at power. However later, the MFW system will be in manual control and one pump will be idling. This may affect the reliability of the system in responding to these accidents since the operator would have to manually raise pump speed to raise discharge pressure high enough to achieve steam generator feed. Activities associated with this evolution should not affect the occurrence frequencies for any of the other initiators and the status of the responding systems should be unchanged with the exception of the status of the MFW system as discussed above.

To summarize the preceding paragraphs with respect to elements increasing the risk associated with this evolution:

- The IE frequency for total loss of MFW is expected to be greater than for full power operation since only one MFW pump is operating above idle (risk increase)

- The IE frequency for loss of offsite power due to a loss of rear bus is expected to be greater than that for full power operation since the 4160 volt buses are re-aligned to the rear bus (risk increase)
- The ability of the MFW system to respond to transients after about the midpoint of evolution is affected by placing one MFW pump in manual control at minimum speed (risk increase)
- MSLB consequences are more severe at hot zero power but this condition exists for only a relatively short period of time (risk neutral)
- Human error rates could be increased for this evolution - operators may have slightly more time to respond to a transient because of the slightly lower initial decay heat levels but their focus on shutdown related activities may delay diagnosis and response (risk increase)

Based on the above observations, the overall risk associated with this evolution is expected to be greater than the overall risk for an equivalent period of time at full power operation. No additional full power operation risk is assumed to be associated with the condition of increasing control rod drive seal leakage.

(2) Transition from hot standby to shutdown cooling entry and cooldown to cold shutdown
This evolution covers a planned and controlled cooldown and entry into shutdown cooling mode per plant procedure GOP-9, Mode 3 ≥ 525 °F to Mode 4 or Mode 5. At the start of this evolution, the plant is shutdown with at least the control and shutdown banks fully inserted and the PCS boron concentration consistent with hot shutdown requirements. The initial PCS temperature and pressure are slightly less than for full power. Decay heat is being removed via the secondary side. Feedwater is supplied by the AFW system to the steam generators and the steam is being discharged directly to the condenser via the turbine bypass valve CV-0511 and/or the atmospheric steam dump valves. The 4160 volt buses have been realigned so that power is supplied from the grid via startup transformers 1-1 and 1-3.

During this evolution the plant will be cooled down to the shutdown cooling entry conditions in preparation for transition to shutdown cooling and cool down to cold shutdown.

Decay heat removal during the cooldown to shutdown cooling entry conditions is accomplished by supplying feedwater to the steam generators from the AFW system and discharging the steam directly to the condenser via the turbine bypass valve and to the atmosphere via the atmospheric steam dump valves (ASDVs). Decay heat will continue to decrease throughout this evolution. PCS pressure is manually controlled during the cooldown using the pressurizer spray valves. The charging system is used to maintain pressurizer level during the cooldown. PCS boron concentration will be adjusted using the charging system. Two primary coolant pumps (PCPs) are stopped prior to starting the cooldown and two PCPs remain in service until ~130°F. Shutdown cooling entry conditions are Tcold < 300°F and PCS pressure < 260 psia. As the PCS pressure decreases to between 1690 psia and 1605 psia, operators block the safety injection actuation signal (SIAS). Also, during mode 4 operation safety injection tanks (SITs) are isolated between 1500 psia and 1300 psia, well in advance of PCS pressure reaching the SIT injection pressure (~250 psig). The shutdown cooling system will be prepared for operation and subsequently aligned.

The following is a list of significant actions / changes in equipment availability and plant condition occur during cooldown:

- Borate to cold shutdown boron concentration, PCS Tave > 525°F
- Stop 2 PCPs when initiating cooldown (after boron concentration for mode 5 is verified by sampling)
- Cooldown using turbine bypass valve and ASDVs
- Block SIS when PCS pressure is between 1690 psia and 1605 psia
- Bypass main steam isolation valve (MSIV) closure when SG pressure is between 550 psia and 510psia
- Isolate the safety injection tanks (SITs) when PCS pressure is between 1500 psia and 1300 psia

- Place low temperature over-pressure protection (LTOP) in service at ~ 460°F
- Disable high pressure safety injection (HPSI) pumps when Tcold between 300°F and 325°F
- Perform boron equalization when Tcold < 300°F and PCS pressure ~415 psia
- Disable containment spray (CS) pumps when Tcold < 300°F
- Place shutdown cooling in service when Tcold < 300°F and PCS pressure < 260 psia
- Establish SG nitrogen blanketing when SG blowdown temperature < 150°F
- Stop all PCPs when desired stable PCS temperature is reached

The key similarities between this evolution and full power operation are:

- Status of key support systems for the safety systems is similar
- Safety injection system is initially available for the early portion of this evolution

The key differences between this evolution and full power operation are:

- Plant is not in a steady-state condition making detection of off-normal trends more difficult (risk increase)
- Feedwater is supplied by the AFW system (risk increase)
- Reactor is shut down with the control rods inserted and shutdown boron concentration achieved (risk decrease)
- Turbine generator system is shut down decreasing the risk of turbine missile generation (risk decrease)
- Decay heat is lower that which would exist immediately after a reactor trip from full power and is decreasing throughout the evolution (risk decrease)
- PCS temperature and pressure are decreasing throughout this evolution (risk decrease)
- Secondary side temperature and pressure are decreasing throughout this evolution (risk decrease)
- Decay heat removal is transitioned from AFW to shutdown cooling (risk neutral)
- 4160 volt buses aligned to the startup transformers (risk increase)
- SIAS is blocked prior to PCS pressure reaching 1605 psia (risk increase)
- MSIV bypass valves open increasing the time needed to isolate the steam generators (risk increase)
- SIT isolated (risk increase)
- LTOP in service (risk decrease)
- HPSI pumps disabled (risk increase)
- CS pumps disabled (risk increase)
- PCPs not operating (risk neutral)

During this evolution, the plant is susceptible to the same transients that it is during full power operation but the likelihood and consequences are different in that the plant is already shut down and the AFW system or shutdown cooling system is already in operation.

The plant is still susceptible to LOCAs, SGTRs and MSLBs. However, PCS and secondary temperatures and pressures will be significantly lower than normal operating temperatures and pressures. HPSI is available prior to disabling between 325°F and 300°F. The low pressure injection (LPSI) system remains available for inventory control. The MFW system is not available for secondary side heat removal to help mitigate small LOCAs and SGTRs, but the AFW system is available and operating early in the evolution. As is the case for full power operation, condensate pumps are available to backup the AFW system but only to the extent that there is inventory in the hotwell and adequate makeup from the condensate storage tank T-2. Because of the lower pressures and the lower decay heat levels during hot shutdown, small LOCAs and SGTRs would be expected to progress slower than at power.

The plant is not susceptible to any of the power conversion system related initiators during this evolution because the system is shut down (for example, loss of main condenser, loss of main feedwater, loss of condensate system). The plant is susceptible to loss of AFW events. Given a loss of feedwater event, the steam generator heat removal capability is reduced at this power level although the event would progress much more slowly due to lower decay heat levels and no reactor power.

Because the 4160 volt buses have been realigned to draw power from the grid via the startup transformers, there may be a somewhat greater exposure to a loss of rear bus offsite power because of reduced redundancy. Diesel generators and required support systems are available and in the same configuration as at full power. The AFW system would be operating. Therefore, the response to a loss of offsite power during this evolution would not be much different than that at full power. Because of the low decay heat levels and zero reactor power, the progression of the transient would be slower than if it had been initiated from full power.

The initiating event frequency for other initiators such as loss of a 125 volt dc bus, loss of a 4160 volt bus or loss of component cooling water should not be affected by conditions associated with this evolution. However, it is expected that the plant would be susceptible to these initiators only to the extent that they would cause a loss of AFW or SDC.

To summarize the preceding paragraphs with respect to the elements that affect the risk associated with this evolution:

- Transients will progress slower than equivalent full power transients because of the lower decay heat levels (risk decrease)
- Reactor trip is not a required response because the plant is already shutdown (risk decrease)
- Some power conversion system related transients are not applicable (risk decrease)
- AFW system is the primary decay heat removal system during the beginning of this evolution and is in manually placed operation (risk decrease)
- SDC system is the primary decay heat removal system during the end of this evolution and the system does not meet single failure design requirements (risk increase)
- LOCA, SGTR and MSLB initiating event frequencies related to pressure and temperature stressors during those portions of this evolution for which the PCS and secondary pressures and temperatures are low are expected to be lower than at power (risk decrease)
- MSLB consequences are more severe at hot zero power but this condition exists for only a relatively short period of time (risk neutral)
- SIT, HPSI and CS systems disabled (risk increase)
- LPSI available for PCS inventory control (risk neutral)
- Initiating frequencies for other initiators should be about the same as at full power but the impact is considered limited to the extent to which they affect the AFW (risk neutral)
- Operator error rates could be higher than for transients initiated from full power given that the decay heat removal systems are not single failure proof and consequently are more difficult to recover (risk increase)
- Loss of AFW and loss of SDC are transients expected to be the most significant risk contributors because of the limited available equipment (risk increase)

Because of the limited number of automatic equipment responses that are functional during this evolution, operator actions are more important during this evolution than during full power conditions. Risk increase is directly proportional to human factors. Therefore, the overall risk associated with this evolution is expected to be greater than the overall risk for an equivalent period of time at full power operation. No additional full power operational risk is assumed to be associated with the condition of increasing control rod drive seal leakage.

(3) Cold shutdown operation for the duration of the outage
Since drain-down to lowered or reduced inventory is not needed to repair control rod drive seals, much of the accident risk encountered during this evolution is due to the factors described in evolution (2) above. Based on similar observations, the overall risk associated with this evolution is expected to be greater than the overall risk for an equivalent period of time at full power operation. Note that no additional full power operational risk is assumed to be associated with the condition of increasing control rod drive seal leakage.

In addition, industrial safety risks associated with performing the additional maintenance tasks that will have to be repeated during the upcoming refueling outage and occupational doses both for the CRD seal repair activity and for subsequent normal on-line maintenance, surveillance and inspection activities will result from an additional shutdown.

(4) Transition from cold shutdown to shutdown cooling exit and heatup to hot standby
Similar (although not identical) risks are present during heatup shutdown cooling entry conditions, transition to AFW/MFW decay heat removal and heatup to hot standby as have been identified in evolution (2) above. Also, prior to entering mode 4, operators are required to transfer the 2400 volt buses to startup transformer 1-2 and back to safeguards transformer 1-1. Based on observations similar to the above, the overall risk associated with this evolution is expected to be greater than the overall risk for an equivalent period of time at full power operation. Note that no additional full power operation risk is assumed to be associated with the condition of increasing control rod drive seal leakage.

(5) Transition from hot standby to full power operation
Similar (although not identical) risks are present during power ascension to the full power condition as have been identified in evolution (1) above. Additional power ascension activities which increase risks include transitioning from MFW bypass valves to the main feed regulating valves, frequent PCS boron dilution and control rod withdrawal. In addition, RPS high power trip setpoints do not reset/raise automatically but must be manually raised by operators. Based on observations similar to the above, the overall risk associated with this evolution is expected to be greater than the overall risk for an equivalent period of time at full power operation. Note that no additional full power operation risk is assumed to be associated with the condition of increasing control rod drive seal leakage.

References

[1]. CE-NPSD-1021, Revision 3, Development of a Methodology for the Evaluation of Transition Risk, CEOG, January 1997.

[2]. NUREG/CR-6144, Evaluation of Potential Severe Accidents during Low power and Shutdown Operations at Surry, Unit 1, USNRC, NRR, 1995.

[3]. INL/CON-07-13143, Development of Standardized Probabilistic Risk Assessment Models for Shutdown Operations Integrated in SPAR Level 1 Model, Idaho National Laboratory, May 2008.

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Engineering Report
Technical Review Comments and Resolutions Form

Engineering Report Number	PLP-RPT-10-00041	Rev. 0	Title Qualitative Assessment of the Risk Due to an Additional Forced Outage To Repair a Leaking Seal on CRDM #22
Quality Related: ☐ Yes ☒ No		**Special Notes or Instructions:** It is acceptable to use MS Word© Track Changes and Comment features for reviews.	

Comment Number	Section/ Page No.	Review Comment	Response/Resolution	Preparer's Accept Initials
1	Various	See electronic markup for various typos and editorial comments.	Comments incorporated	JFV
2	1	Removed the description of transferring from main feed regulating valves to the bypass valves. This is not normally done going down. When starting up, there is a transition from the bypass valves to the main valves.	OK	
3	1	PCS temperature at full power is ~ 560 oF	Noted.	
4	3	2400 volt buses are supplied from the Front Bus when operating and when shutdown, unless maintenance activities require transferring to startup power xfmr 1-2. If Rear Bus ONLY was lost prior to transferring to startup power, the plant should not trip. If Rear Bus ONLY is lost after transferring to startup power, the plant will trip and PCPs and condensate pumps will not be available. If Front Bus is lost the plant trips regardless (there is no fast transfer to the startup 1-2 xfmr.	Discussion revised.	
5	3	Do not use PRA terminology. Do not reference the full power internal events model. For example, "plant trip events such as loss of a dc bus, LOCA, loss of feedwater, etc.	More common language used.	
6	4	In theory when taking the plant down the number of dynamic loads on systems increases. Therefore one could argue that LOCA frequencies would increase.	Re-worded.	JFV

Entergy

Engineering Report
Technical Review Comments and Resolutions Form

Engineering Report Number	PLP-RPT-10-00041	Rev. 0	Title Qualitative Assessment of the Risk Due to an Additional Forced Outage To Repair a Leaking Seal on CRDM #22	
Quality Related: ☐ Yes ☒ No		**Special Notes or Instructions:** It is acceptable to use MS Word© Track Changes and Comment features for reviews.		

Comment Number	Section/ Page No.	Review Comment	Response/Resolution	Preparer's Accept Initials
7	1	Loss of steam generators as a decay heat removal path is one of the most important consequences of this shutdown activity repair. This fact needs to be stated several times. See additional comments below.	Incorporated.	JFV
8	2	Suggest grading each difference risk + or risk - and provide basis.	Incorporated.	
9	3	Use language that does not suggest our operators lack experience – everyone has human failure events.	Text revised.	
10	3	You are implying these IEs are driven by pressure. Could be IGSCC, vibration, water hammer, FAC, etc. don't overstate your state of knowledge here.	Discussion clarified.	
11	3	Small LOCA's generate a CHP which close the MSIVs, isolate containment etc.	Noted.	
12	3	Any time a system is put in manual and the operator has to control the safety function, the risk increases due to human intervention. Therefore, the chances of a 'special initiator occurring' increases.	Discussion clarified.	
13	4	The PORVs are enabled during shutdown. The PTS results cite thermal shock events and not PTS events as the drivers for vessel challenges.	Discussion revised.	
14	4	Don't overstate: protected trains etc. will preclude all non-required tests etc. We are not going to jeopardize plant risk by digging sign post holes, etc.	Discussion revised.	JFV

Entergy

Engineering Report
Technical Review Comments and Resolutions Form

Engineering Report Number	PLP-RPT-10-00041	Rev. 0	Title: Qualitative Assessment of the Risk Due to an Additional Forced Outage To Repair a Leaking Seal on CRDM #22
Quality Related: ☐ Yes ☒ No		Special Notes or Instructions: It is acceptable to use MS Word© Track Changes and Comment features for reviews.	

Comment Number	Section/ Page No.	Review Comment	Response/Resolution	Preparer's Accept Initials
15	5	Rewrite. This is vague. For example, the plant is susceptible to the same transients; however, the likelihood and consequences are different.	Re-worded this discussion.	JFV
16	6	Note that loss of condensate is a power conversion transient and does apply.	Noted.	
17	6	To make this statement need to cite the EPRI reports or NUREGs and cite the reference. If you cannot validate with data then delete.	Deleted statement.	
18	6	Discussion must include disabling the S/Gs. I have NOT seen this discussed in sufficient detail. This is the "trump" card.	Increased profile of this discussion, added additional discussion.	
19	1	Evaluation summary is missing a statement regarding external events: fire, seismic, winds/tornadoes and internal floods.	Statement added.	
20	6	Must consider the hot zero power MSLB containment response case. Given that it is the worst case w.r.t. containment response, we would be in this mode for a very short period of time. Therefore we would conclude risk neutral.	Discussion revised.	JFV

Verified/Reviewed By:	Brian Brogan		Date	3-31-10	Resolved By: JL Voskuil / J.L. Voskuil
Site/Department:	PRA	Ph. 764-3100			Date: 3-31-2010