Entergy

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PNP 2016-049

August 22, 2016

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
Renewed Facility Operating 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 (PNP).

The proposed amendment requests NRC approval to replace the existing Renewed Facility Operating License (RFOL) condition 2.C.(4), which states that performance of Technical Specifications (TS) 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. The existing condition is obsolete since PNP is currently in cycle 25. The condition would be replaced with a new condition 2.C.(4), which states that TS SR 3.1.4.3 is not required for CRD-13 during cycle 25 until the next entry into Mode 3. In addition, the replacement license condition would state that CRD-13 seal leakage shall be repaired prior to entering Mode 2, following the next Mode 3 entry, and that the reactor shall be shut down if CRD-13 seal leakage exceeds two gallons per minute.

The proposed amendment also requests NRC approval to replace the note in TS SR 3.1.4.3. The note currently states that TS SR 3.1.4.3 is not required to be performed or met for CRD-22 during cycle 21 provided CRD-22 is administratively declared immovable, but trippable and Condition D is entered for CRD-22. This note is obsolete since PNP is currently in cycle 25. The TS SR 3.1.4.3 note would be replaced with a note clarifying that TS SR 3.1.4.3 is not required to be performed or met for CRD-13 during cycle 25 provided CRD-13 is administratively declared immovable, but trippable and Condition D is entered for CRD-13.

TS 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-13, causing excessive seal leakage, which could result in an unplanned maintenance outage. ENO has determined that the risk incurred due to an additional maintenance shut down to repair CRD-13 is considered to be greater than the risk of continued full power operation with CRD-13 no longer being exercised every 92 days during cycle 25.

The proposed changes have been evaluated in accordance with 10 CFR 50.91(a)(1) using the standards in 10 CFR 50.92(c), and it has been determined that the changes involve no significant hazards consideration. The bases for this determination are included in Attachment 1, which also includes a description of the proposed changes, a technical analysis, a regulatory safety analysis, and an environmental consideration.

Attachment 2 provides the marked-up RFOL and TS pages showing the proposed changes. Attachment 3 provides the revised RFOL and TS pages reflecting the proposed changes. Attachment 4 provides a qualitative assessment of risk due to an additional maintenance outage to repair CRD-13.

ENO requests approval of this proposed license amendment prior to October 14, 2016, with the amendment being implemented within 15 days. The requested approval date was selected to avoid exercising CRD-13 prior to the next surveillance due date.

In accordance with 10 CFR 50.91(b), a copy of this application, with attachments, is being provided to the designated State of Michigan official.

Summary of Commitments

There are no new commitments or revisions to existing commitments.

I declare under penalty of perjury that the foregoing is true and accurate. Executed on August 22, 2016.

Sincerely,

CFA/jse

Attachments:
1. Description and Evaluation of Requested Change
2. Proposed Renewed Facility Operating License DPR-20 and Appendix A Technical Specifications Pages
3. Page Change Instructions and Revised Renewed Facility Operating License DPR-20 and Appendix A Technical Specifications Pages
4. Qualitative Assessment of Risk

cc Regional Administrator, Region III, USNRC
Project Manager, Palisades, USNRC
NRC Resident Inspector, Palisades USNRC
State of Michigan

ATTACHMENT 1
DESCRIPTION AND EVALUATION OF REQUESTED CHANGE

1.0 DESCRIPTION

Entergy Nuclear Operations, Inc. (ENO) requests to amend the Renewed Facility Operating License (RFOL) DPR-20 for the Palisades Nuclear Plant (PNP).

The proposed amendment requests Nuclear Regulatory Commission (NRC) approval to replace RFOL condition 2.C.(4).

License condition 2.C.(4) currently states that performance of Technical Specifications (TS) 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. The existing condition is obsolete since PNP is presently in cycle 25.

Under this proposed amendment, the current license condition would be replaced by a license condition that states that performance of TS SR 3.1.4.3 is not required for CRD-13 during cycle 25 until the next entry into Mode 3. The replacement license condition would also state that seal leakage on CRD-13 shall be repaired prior to entering Mode 2, following the next Mode 3 entry, and that the reactor shall be shut down if CRD-13 seal leakage exceeds two gallons per minute.

The proposed amendment also requests NRC approval to replace an associated note in TS SR 3.1.4.3. The TS SR 3.1.4.3 note currently states that performance of TS SR 3.1.4.3 is not required to be performed or met for CRD-22 during cycle 21 provided CRD-22 is administratively declared immovable, but trippable, and Condition D is entered for CRD-22. This note is obsolete since PNP is currently in cycle 25. This would be replaced by a note that states that TS SR 3.1.4.3 is not required to be performed or met for CRD-13 during cycle 25 provided CRD-13 is administratively declared immovable, but trippable, and TS Limiting Condition of Operation (LCO) 3.1.4, Condition D, is entered for CRD-13.

2.0 PROPOSED CHANGE

ENO proposes a license amendment to revise the PNP RFOL by replacing license condition 2.C.(4).

Under this proposed change, the following license condition 2.C.(4) would be 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."

The proposed change would add the following license condition 2.C.(4):

"The following requirements shall apply to control rod drive CRD-13 during cycle 25:

(a) Performance of Technical Specifications Surveillance Requirement SR 3.1.4.3 is not required for CRD-13 until the next entry into Mode 3.

(b) Seal leakage on CRD-13 shall be repaired prior to entering Mode 2, following the next Mode 3 entry.

(c) The reactor shall be shut down if CRD-13 seal leakage exceeds two gallons per minute."

The proposed change would also revise the TS by replacing the note in TS SR 3.1.4.3.

The proposed change would delete the following note in TS SR 3.1.4.3:

-----------------------------NOTE---------------------------
Not required to be performed or met for control rod 22 during cycle 21 provided control rod 22 is administratively declared immovable, but trippable and Condition D is entered for control rod 22.

The proposed change would add the following note in TS SR 3.1.4.3:

-----------------------------NOTE---------------------------
Not required to be performed or met for control rod 13 during cycle 25 provided control rod 13 is administratively declared immovable, but trippable and Condition D is entered for control rod 13.

3.0 BACKGROUND

Need for Prompt Revision of CRD Surveillance Requirement

In accordance with the TS surveillance frequency of 92 days, TS SR 3.1.4.3 is due to be performed again in October 2016 and January 2017. Considering the next refueling outage is currently planned to commence in the spring of 2017, the proposed LAR would eliminate two surveillances of CRD-13 during cycle 25.

If the proposed LAR is approved, the expected continuing rise in CRD-13 seal leakage should still allow continued plant operation until the next scheduled refueling outage.

The CRD testing required by TS SR 3.1.4.3 is performed under PNP TS surveillance procedure QO-34, "Control Rod Exercising." Throughout cycle 25, which commenced on October 19, 2015, QO-34 has been successfully performed three times.

After the most recent successful completion of QO-34 on July 19, 2016, the magnitude of rise in CRD seal leakage was greater than anticipated.

The following data is the current CRD seal leakage trend:

Date QO-34 was Performed	Pre-test leak rate (ml/min)	Pre-test CRD-13 temperature (^{0}F)	Post-test leak rate (ml/min)	Post-test CRD-13 temperature (^{0}F)	Leak rate delta (ml/min)
1/19/2016	2 (12/19/2015)	107 (1/18/2016)	5 (1/26/2016)	107 (1/21/2016)	3
4/19/2016	2 (3/29/2016)	111 (4/17/2016)	5.5 (4/27/2016)	113 (4/20/2016)	3.5
7/19/2016	3.5 (6/28/2016)	122 (7/16/2016)	25 (7/22/2016) 32.2 (7/26/2016)	160 (7/19/2016)	21.5

From the above data, historical seal performance, CRD temperature indications, and direct measurement of CRD seal leakoff, CRD-13 is judged to comprise the majority of this leakage. Based on historical CRD seal degradation trends, it is projected that following an upcoming performance of QO-34 in October 2016 or January 2017, the seal leakage will increase to the extent that continued operation to the planned spring 2017 refueling outage is unlikely, and an unplanned maintenance outage would be required to replace the CRD-13 seals.

<u>Seal Reliability History</u>

PNP has a history of CRD seal leakage resulting in additional unplanned maintenance outages. Multiple causal evaluations and numerous corrective actions have improved recent seal reliability. Despite the improvements in seal reliability, increased seal leakage in CRD-13 has recently occurred. A summary of causal evaluation results and the more impactful corrective actions taken are provided below.

In 2008, a causal evaluation was performed to determine the cause of prior CRD seal leakage. Inadequate CRD seal area cooling was identified as a cause of CRD seal leakage. A contributing cause was the quality of the CRD rotating seal replacement parts.

A comparison of Fort Calhoun Nuclear Generating Station CRD seal area temperatures (Fort Calhoun has similar CRD mechanisms) to PNP CRD seal area temperatures led to the identification of high ambient temperature in the CRD seal area as a cause. To correct this, a design change was implemented during the 2009 refueling outage (1R20) 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 ambient temperatures.

In 2009, additional material destructive testing on removed failed seals was performed. This 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 the 2009 refueling outage (1R20).

Also during the 2009 refueling outage, ENO adopted a primary coolant system (PCS) vacuum fill process, in which a vacuum is drawn on the PCS to evacuate air and other non-condensable gases during shutdown conditions. A benefit of the vacuum fill process is that it minimizes the introduction of debris between the CRD seal faces. Debris had previously been identified as potentially contributing to CRD leakage.

In cycle 21, following the 2009 refueling outage, CRD-22 exhibited increasing seal leakage, leading ENO to submit a license amendment request on March 31, 2010 (Reference 1) to not perform the surveillance exercise testing of CRD-22 required under TS SR 3.1.4.3 for the remainder of the cycle. The request was supplemented by ENO on May 13, 2010 (Reference 2), and was approved by the NRC on June 2, 2010 in Amendment No. 239 (Reference 3).

Subsequent to approval of the license amendment request, CRD-22 seal leakage continued to increase and the reactor was shut down in June 2010 to replace the CRD-22 seals. As a corrective action, all 45 CRD seals were replaced during the subsequent 2010 refueling outage (1R21). In addition, manufacturing defects in the stationary seal due to overspray was credited for seal degradation in certain CRDs, and as a result the vendor manufacturing processes for the new seal fabrication were monitored by ENO to ensure quality standards for rotating and stationary replacement seals are met.

In January 2012, the plant was shut down to replace the seal on CRD-23 due to seal leakage. A causal evaluation attributed the seal leakage to deposition of small particulate debris entering between the control rod drive rotating and stationary seal faces causing degradation and eventual failure. This small debris had been deposited in the CRD support tubes over many years due to wear products from equipment and past poor foreign material exclusion practices by personnel that interact with the PCS system. Corrective actions included replacing selected CRD seals on an increased frequency, based on reductions in seal performance.

Since 2012, PNP has not had an unplanned maintenance outage due to CRD seal leakage.

<u>Risk Associated with an Additional Plant Shutdown</u>

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-13 should alleviate the need for a maintenance outage prior to the planned refueling outage in 2017. By eliminating an additional maintenance outage the following plant challenges will be avoided:

- Exposure to plant conditions where the PCS pumps and steam generators are not available for decay heat removal
- A significant power change
- A plant thermal cycle
- Operation of plant standby safety systems (auxiliary feedwater and shutdown cooling)
- Additional 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-13 seal. The evaluation concluded that the risk incurred to shut down and repair CRD-13 is considered to be greater than the risk of continued full power operation for an equivalent period of time when CRD-13 has not been exercised.

This conclusion is primarily based on the:

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

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, increased industrial safety risks and 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 due to the additional shutdown.

Description of Control Rod Drives

PNP fuel assemblies and control rods are arranged as shown in Figure 3 in the enclosure. 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 safety analysis is to release the clutch and drop by gravitational force 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

TS Limiting Condition for Operation (LCO) 3.1.4, "Control Rod Alignment," requires that all control rods be operable and aligned to within 8 inches of all other rods in their respective group, and the control rod position deviation alarm be operable as well, while in Modes 1 and 2.

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 the next 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 3.1.4.3 also contains a note stating that that performance of SR 3.1.4.3 is not required to be performed or met for CRD-22 during cycle 21 provided CRD-22 is administratively declared immovable, but trippable, and TS LCO 3.1.4, Condition D, is entered for CRD-22.

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)."

<u>TS Bases for Action D.1</u>

"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 each of 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, "Control Rod Drop Times"). 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 has been successfully performed three times for all full-length control rods.

4.0 TECHNICAL ANALYSIS

Replacement of RFOL Condition 2.C.(4) and TS SR 3.1.4.3 Note

The proposed change to RFOL condition 2.C.(4) would delete a condition that performance of TS SR 3.1.4.3 is not required for control rod drive CRD-22 during cycle 21 until the next entry into Mode 3. The proposed change would add in its place a new condition 2.C.(4) that performance of TS SR 3.1.4.3 is not required for control rod drive CRD-13 during cycle 25 until the next entry into Mode 3.

The proposed change also deletes an associated note in TS SR 3.1.4.3 that the SR is not required to be performed or met for CRD-22 during cycle 21 provided CRD-22 is administratively declared immovable, but trippable, and Condition D is entered for CRD-22. The proposed change would add in its place a new note that the SR is not required to be performed or met for CRD-13 during cycle 25 provided CRD-13 is administratively declared immovable, but trippable, and Condition D is entered for CRD-13.

The existing RFOL condition 2.C.(4) and the note in TS SR 3.1.4.3 were added under Amendment No. 239 (Reference 3). The amendment allowed the CRD-22 testing required under TS SR 3.1.4.3 to be suspended because CRD-22 leakage was increasing with each surveillance operation. In the associated license amendment request (Reference 1), ENO committed to make repairs to correct the existing seal leakage on CRD-22 prior to entering Mode 2, following the next Mode 3 entry.

Since PNP is currently in operating cycle 25, the existing RFOL condition 2.C.(4) and the existing note in TS SR 3.1.4.3 concerning CRD-22 are obsolete, and can be replaced by the proposed condition and the TS SR note concerning CRD-13 testing in cycle 25.

Cycle Operational History CRD-13

Since reactor reassembly in the 2015 refueling outage, CRD-13 has had three full-length withdrawals and two full-length insertions.

The following table documents CRD movement history (including CRD-13) since reactor reassembly during and after the 2015 refueling outage:

Date	Activity	Movement	Travel (Inches)
10/14/2015	RO-19, "Control Rod Position Verification"	Manual Withdrawal	~20
10/14/2015	RO-19, "Control Rod Position Verification"	Manual Insertion	~20
10/18/2015	RO-22, "Control Rod Drop Times"	Manual Withdrawal	131
10/18/2015	RO-22, "Control Rod Drop Times"	Manual Trip Insertion Timed	131
10/19/2015	Reactor Critical Approach	Manual Withdrawal	131
10/19/2015	RT-191, "Start-up Physics Test Program"	Manual Insertion	~128
10/19/2015	RT-191, "Start-up Physics Test Program"	Manual Withdrawal	~128
1/19/2016	QO-34, "Control Rod Exercising"	Manual Insertion	~6
1/19/2016	QO-34, "Control Rod Exercising"	Manual Withdrawal	~6
4/19/2016	QO-34, "Control Rod Exercising"	Manual Insertion	~6
4/19/2016	QO-34, "Control Rod Exercising"	Manual Withdrawal	~6
7/19/2016	QO-34, "Control Rod Exercising"	Manual Insertion	~6
7/19/2016	QO-34, Control Rod Exercising	Manual Withdrawal	~6

Based on the above data, CRD-13 has been moved a total of approximately 725 inches since the 2015 refueling outage reactor reassembly.

Based on the total inches of travel and the interval between movements over the last 10 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 the ability of CRD-13 to trip.

Continued satisfactory quarterly exercising of the remaining 40 full-length CRDs will support maintaining this full confidence by ensuring that a common failure mechanism does not develop that would prevent CRD-13 from moving or tripping.

Other Testing that Verifies Control Rod Safety Functions

Rod drop time testing (RO-22) is performed each refueling outage, in accordance with TS SR 3.1.4.6. 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 the 2015 refueling outage, prior to Mode 2 entry for cycle 25, RO-22 was successfully performed for all full-length control rods.

Other Evolutions that Verify Control Rod Functionality

During 2015 refueling outage start-up physics testing (RT-191), CRD-13 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 approach performed on October 19, 2015 moved CRD-13 from a fully inserted position to a full-out position.

Recent Maintenance History for CRD-13

The CRD-13 seals were last replaced during the 2014 refueling outage (1R23). Per PNP maintenance procedure CRD-M-31, "Rebuilding and Testing CRDM Seal Housing Assemblies," the removed CRD seal components are inspected and measured to ensure no damage has occurred during previous use.

All CRD seal housings had their seal packages (seals, vapor seals, and the associated o-rings) replaced during the 2010 refueling outage (1R21).

All CRD seal packages were replaced during the 2009 refueling outage (1R20).

In the 2007 refueling outage (1R19), all of the CRD seals were replaced.

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. A causal 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 future replacements based on clutch bearing inspection intervals. CRD-14 had been successfully exercised during QO-34 testing in August 1999. Since 1999, periodic preventive maintenance has been used to prevent failures resulting from periodic preventive maintenance has been used to prevent failures resulting from aged CRD clutches.

In the unlikely event that CRD-13 failed to trip during an accident or transient, the rundown feature would be expected to fully insert CRD-13 into the core.

Effects of Continued CRD-13 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 TS to determine PCS unidentified leakage on a 72-hour interval. PNP has a procedural 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 the containment building gaseous monitor count rate, 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 shut down the reactor if CRD seal leakage exceeds 2.0 gpm, in accordance with plant procedures. The proposed license condition would also require plant shutdown if CRD seal leakage exceeds 2.0 gpm. The 2.0 gpm limit conservatively bounds the TS identified leakage limit of 10 gpm.

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. The proposed license amendment would require repairs to correct the existing seal leakage on CRD-13 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.

<u>Bearings</u>
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.

<u>Vapor Seal</u>
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.

<u>Steam Impingement Washer</u>
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.

<u>Seal Assembly</u>
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 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-13 is a Group B shutdown control rod. Control rods in Group B are fully withdrawn on startup and remain withdrawn during power operation to provide shutdown margin at all times, and 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 TS SR 3.1.4.3 for CRD-13 for the remainder of the current operating cycle would not increase the likelihood of an untrippable control rod.

Conclusion

The performance of the quarterly surveillance exercise testing of CRD-13 will likely increase CRD seal leakage, requiring an unplanned maintenance outage. Due to 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-13 to date, performance of rod exercising testing on CRD-13 an additional two times over the remainder of this fuel cycle is not necessary to have full confidence that CRD-13 will remain trippable or can be fully inserted using the rod rundown feature, if needed. Therefore, ENO has full confidence that CRD-13 will maintain its ability to perform its safety function for the remainder of cycle 25. 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 TS SR 3.1.4.3 surveillance tests for CRD-13 in cycle 25.

5.0 REGULATORY SAFETY ANALYSIS

No Significant Hazards Consideration

Entergy Nuclear Operations, Inc. (ENO) requests to amend Renewed Facility Operating License (RFOL) DPR-20 for the Palisades Nuclear Plant (PNP). The proposed amendment requests Nuclear Regulatory Commission (NRC) approval to replace the existing RFOL license condition 2.C.(4).

PNP RFOL license condition 2.C.(4) currently states that performance of PNP Technical Specification (TS) 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. This TS SR verifies control rod freedom of movement with a 92-day frequency. PNP is currently in operating cycle 25 and therefore this license condition is obsolete. Under this proposed amendment, the current license condition would be replaced by a license condition that states that performance of TS SR 3.1.4.3 is not required for CRD-13 during cycle 25 until the next entry into Mode 3. The replacement license condition would also state that seal leakage on CRD-13 shall be repaired prior to entering Mode 2, following the next Mode 3 entry, and that the reactor shall be shut down if CRD-13 seal leakage exceeds two gallons per minute.

The proposed amendment also requests NRC approval to replace an associated note in TS SR 3.1.4.3. The TS SR 3.1.4.3 note currently states that performance of TS SR 3.1.4.3 is not required to be performed or met for CRD-22 during cycle 21 provided CRD-22 is administratively declared immovable, but trippable, and Condition D is entered for CRD-22. This would be replaced by a note that states that TS SR 3.1.4.3 is not required to be performed or met for CRD-13 during cycle 25 provided CRD-13 is administratively declared immovable, but trippable, and TS Limiting Condition of Operation (LCO) 3.1.4, Condition D, is entered for CRD-13.

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 replaces an obsolete license condition concerning CRD-22 testing that applied only to operating cycle 21 with a new license condition to forgo the remaining two required surveillance tests of CRD-13 from the PNP TS surveillance requirement for partial movement every 92 days during cycle 25. Since CRD-13 remains trippable, 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 amendment 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 amendment does not affect trippability 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

The construction permit for PNP was issued by the Atomic Energy Commission (AEC) on March 14, 1967, and an Interim Provisional Operating License was issued by the AEC on March 24, 1971. In the request for a full term operating license and application for an increase in power level, which was submitted to the NRC on January 22, 1974, Consumers Power Company (previous PNP owner) provided a discussion to compare the PNP design with the General Design Criteria (GDC) as they appeared in 10 CFR 50 Appendix A on July 7, 1971. It was this discussion, including the identified exceptions, which formed the original plant licensing basis for compliance with the GDC. This discussion is contained in Updated Final Safety Analysis Report (UFSAR) Chapter 5.1, "General Design Criteria," with more details provided in other UFSAR sections. As described in UFSAR Section 5.1.1, changes have been made to the original UFSAR GDC discussions to reflect commitments and changes made to the facility over the life of the plant. Therefore, the GDC discussions in the UFSAR constitute the PNP licensing bases with respect to compliance with the GDC.

An assessment of the proposed changes concluded that there are no exceptions to ENO compliance with any of the following regulations, as described in the UFSAR:

GDC-4 requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents including loss of coolant accidents. These structures, systems and components shall be appropriately protected against dynamic effects, including the effect of missiles, pipe whipping and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

GDC-23 requires that the protection system be designed to fall into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water and radiation) are experienced.

GDC-25 requires that the protection system be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

GDC-26 requires that two independent reactivity control systems of different design principles be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operations, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

GDC-27 requires that reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

GDC-28 requires that reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

GDC-29 requires that protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

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 PRECEDENT

On March 31, 2010, ENO submitted a license amendment (Reference 1) request to add a new license condition and to add a note to TS SR 3.1.4.3 stating that performance of TS SR 3.1.4.3 is not required for CRD-22 during cycle 21 until the next entry into Mode 3 in a maintenance or refueling outage. This license amendment request was supplemented on May 13, 2010 (Reference 2), and was subsequently approved by the NRC in Amendment No. 239 on June 2, 2010 (Reference 3).

7.0 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.

8.0 REFERENCES

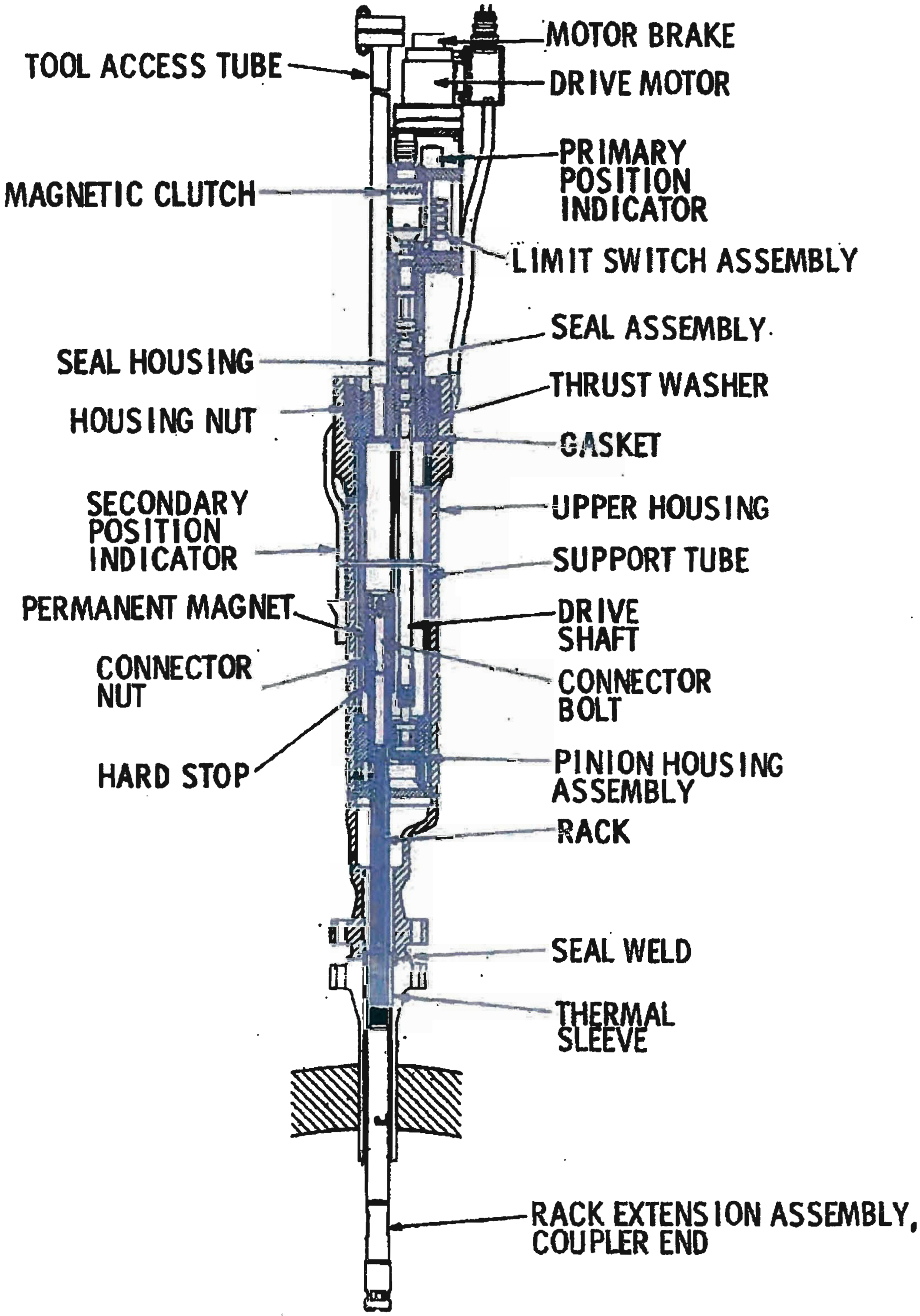
1. ENO letter, "License Amendment Request: Control Rod Drive Exercise Surveillance," dated March 31, 2010 (ADAMS Accession No. ML100920476)
2. ENO letter, "Supplement to License Amendment Request: Control Rod Drive Exercise Surveillance," May 13, 2010 (ADAMS Accession No. ML101330455)
3. NRC letter, "Palisades Nuclear Plant – Issuance of Amendment Re: Control Rod Drive Exercise Surveillance (TAC No. ME3638)," dated June 2, 2010 (ADAMS Accession No. ML101380534)

9.0 ENCLOSURE

Figure 1 – Control Rod Drive Mechanism
Figure 2 – Control Rod Drive Mechanism
Figure 3 – Reactor Cross Section
Figure 4 – Control Rod
Figure 5 – Control Rod Drive Seal Sketch

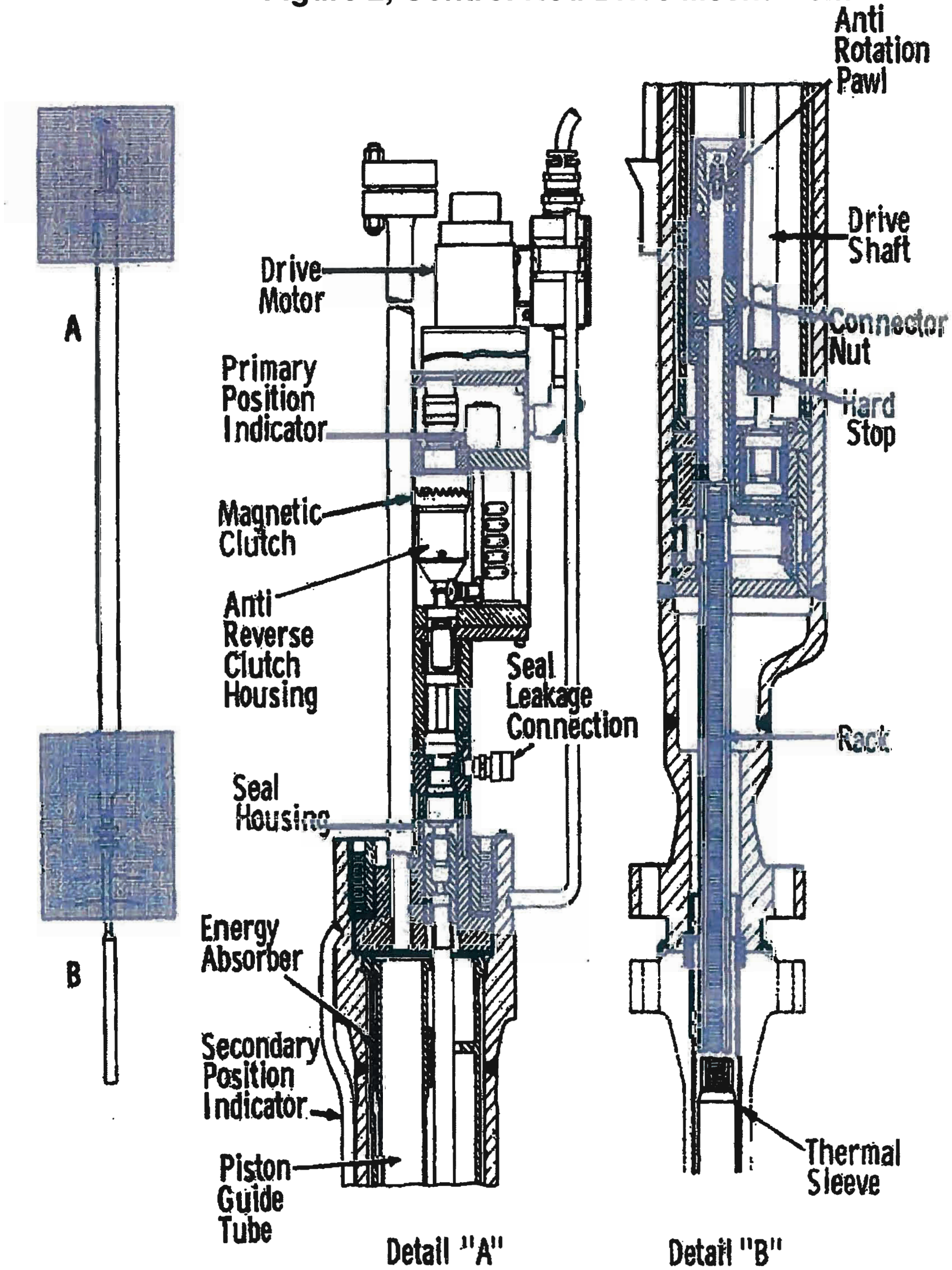
Enclosure

Figure 1, Control Rod Drive Mechanism



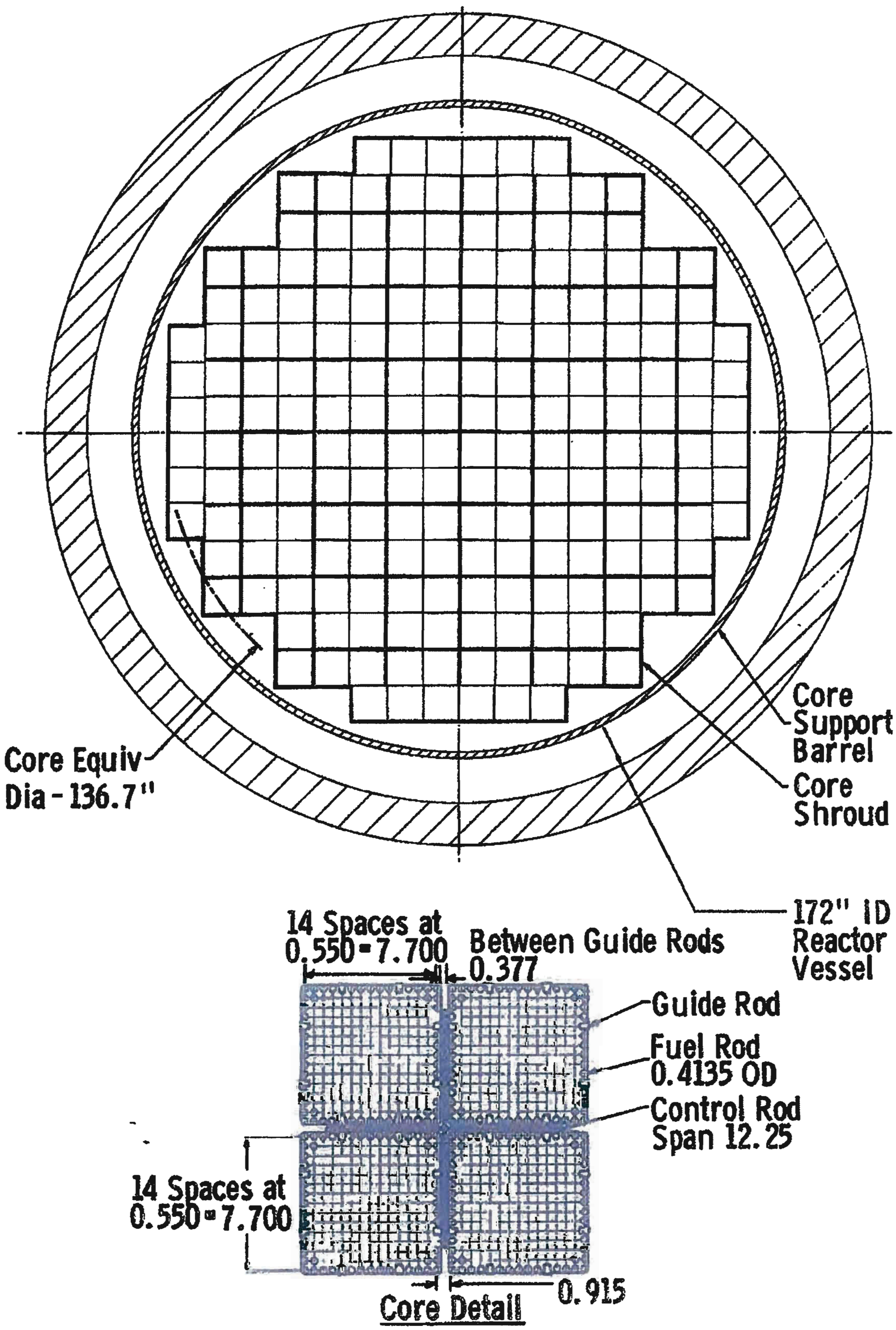
Enclosure

Figure 2, Control Rod Drive Mechanism



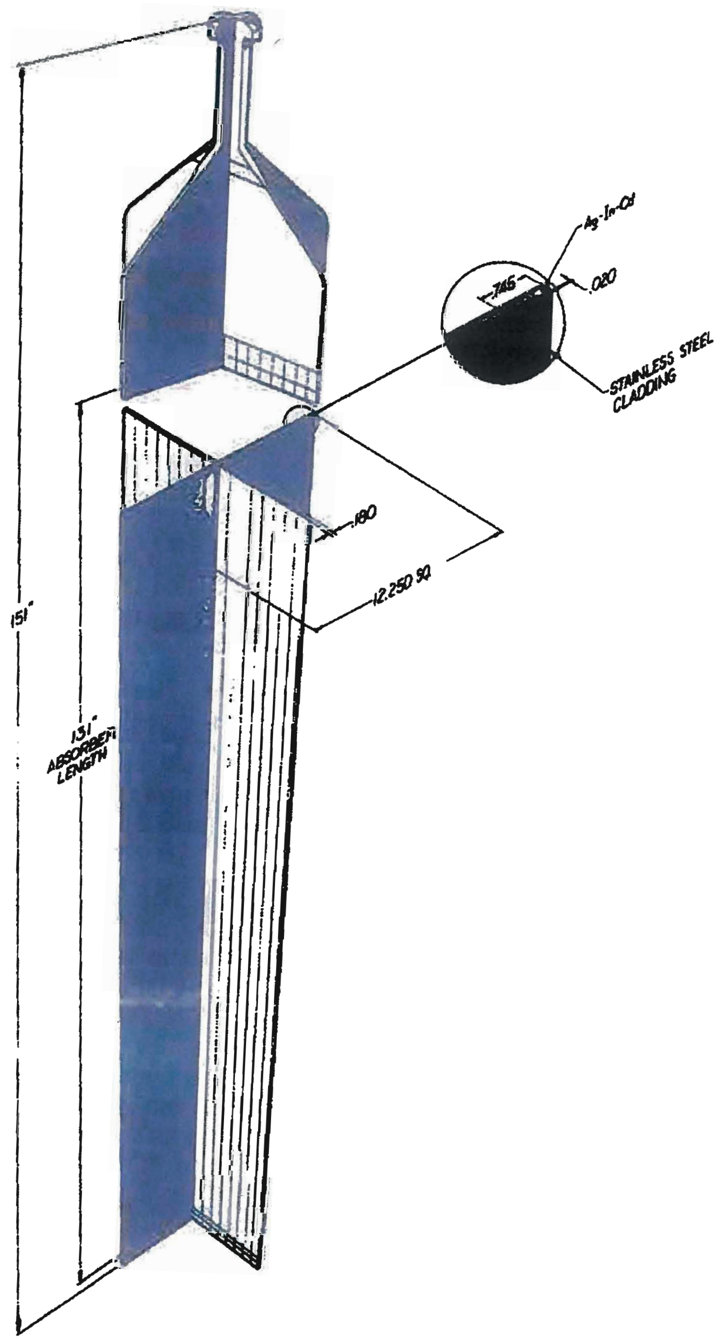
Enclosure

Figure 3, Reactor Cross Section



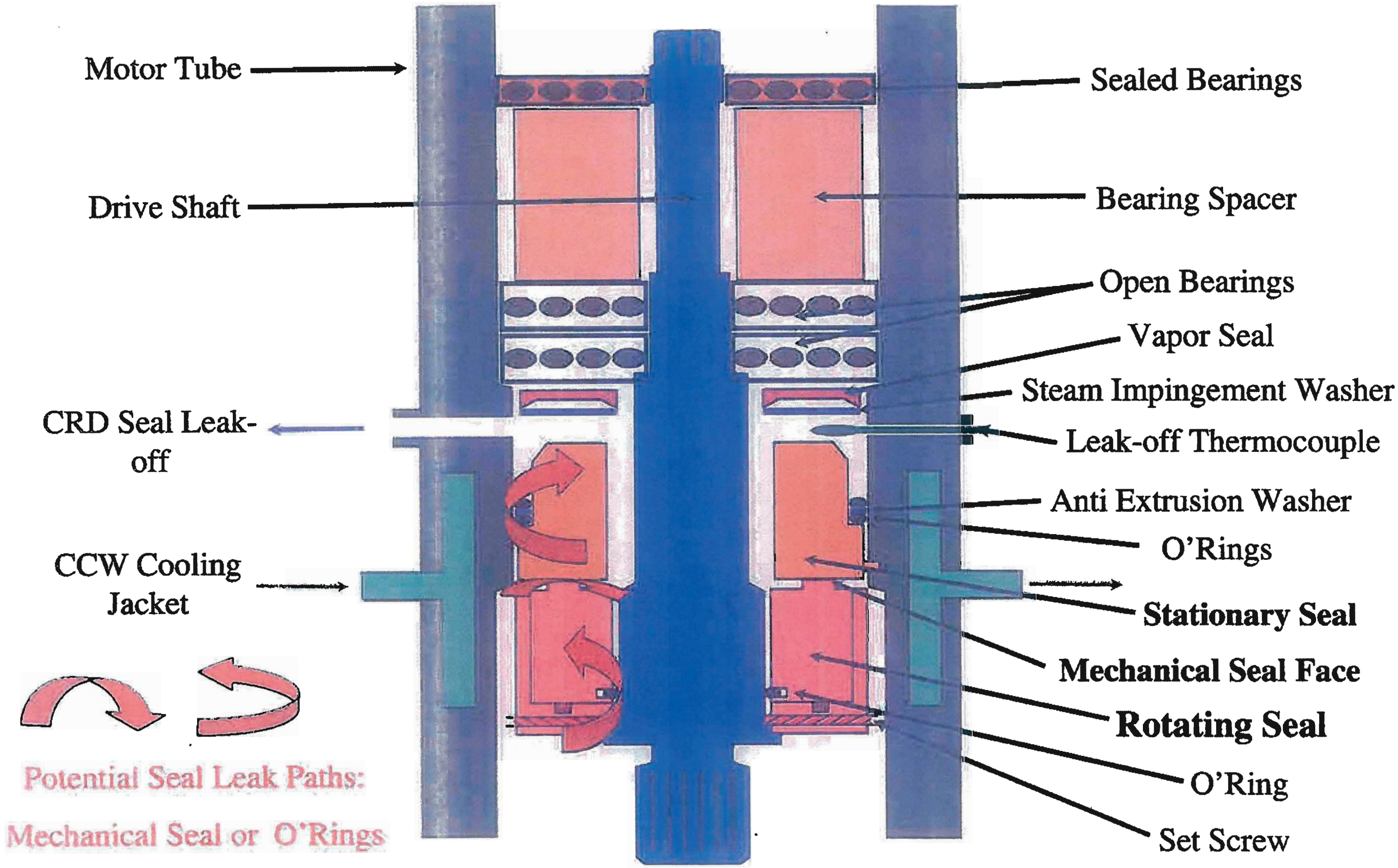
Enclosure

Figure 4, Control Rod



Enclosure

Figure 5, Control Rod Drive Seal Sketch



ATTACHMENT 2

PROPOSED RENEWED FACILITY OPERATING LICENSE DPR-20 AND APPENDIX A TECHNICAL SPECIFICATIONS PAGES

(The additions are highlighted and the deletions are stricken through.)

Four pages follow

(1) Pursuant to Section 104b of the Act, as amended, and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," (a) ENP to possess and use, and (b) ENO to possess, use and operate, the facility as a utilization facility at the designated location in Van Buren County, Michigan, in accordance with the procedures and limitation set forth in this license;

(2) ENO, pursuant to the Act and 10 CFR Parts 40 and 70, to receive, possess, and use source and special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended;

(3) ENO, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use byproduct, source, and special nuclear material as sealed sources for reactor startup, reactor instrumentation, radiation monitoring equipment calibration, and fission detectors in amounts as required;

(4) ENO, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material for sample analysis or instrument calibration, or associated with radioactive apparatus or components; and

(5) ENO, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operations of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations in 10 CFR Chapter I and is subject to all applicable provisions of the Act; to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) ENO is authorized to operate the facility at steady-state reactor core power levels not in excess of 2565.4 Megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) The Technical Specifications contained in Appendix A, as revised through Amendment No. ~~256~~xxx, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. ENO shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) <u>Fire Protection</u>

ENO shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the license amendment request dated December 12, 2012, as supplemented by letters dated February 21, 2013, September 30, 2013, October 24, 2013, December 2, 2013, April 2, 2014, May 7, 2014, June 17, 2014,

(c) Transition License Conditions

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by 2, below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2. above.

2. The licensee shall implement the modifications to its facility, as described in Table S-2, "Plant Modifications Committed," of ENO letter PNP 2014-080 dated August 14, 2014, to complete the transition to full compliance with 10 CFR 50.48(c) before the end of the second full operating cycle after NRC approval. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall implement the items listed in Table S-3, "Implementation Items," of ENO letter PNP 2014-097 dated November 4, 2014, within six months after NRC approval, or six months after a refueling outage if in progress at the time of approval with the exception of Implementation Items 3 and 8 which will be completed once the related modifications are installed and validated in the PRA model.

(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.~~

The following requirements shall apply to control rod drive CRD-13 during cycle 25:

(a) Performance of Technical Specifications Surveillance Requirement SR 3.1.4.3 is not required for CRD-13 until the next entry into Mode 3.

(b) Seal leakage on CRD-13 shall be repaired prior to entering Mode 2, following the next Mode 3 entry.

(c) The reactor shall be shut down if CRD-13 seal leakage exceeds two gallons per minute.

(5) [deleted]

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify the position of each control rod to be within 8 inches of all other control rods in its group.	12 hours
SR 3.1.4.2	Perform a CHANNEL CHECK of the control rod position indication channels.	12 hours
SR 3.1.4.3 **Replace Note with Insert "A"**	~~-----NOTE-----~~ ~~Not required to be performed or met for control rod 22 during cycle 21 provided control rod 22 is administratively declared immovable, but trippable and Condition D is entered for control rod 22.~~ ~~----------~~ Verify control rod freedom of movement by moving each individual full-length control rod that is not fully inserted into the reactor core ≥ 6 inches in either direction.	92 days
SR 3.1.4.4	Verify the rod position deviation alarm is OPERABLE.	18 months
SR 3.1.4.5	Perform a CHANNEL CALIBRATION of the control rod position indication channels.	18 months
SR 3.1.4.6	Verify each full-length control rod drop time is ≤ 2.5 seconds.	Prior to reactor criticality, after each reinstallation of the reactor head

Insert "A" for Technical Specification Page 3.1.4-3, Surveillance Requirement SR 3.1.4.3

Insert "A" →

-----------------------------NOTE-----------------------------
Not required to be performed or met for control rod 13 during cycle 25 provided control rod 13 is administratively declared immovable, but trippable and Condition D is entered for control rod 13.

Insert "A" for Technical Specification Page 3.1.4-3, Surveillance Requirement SR 3.1.4.3

ATTACHMENT 3

PAGE CHANGE INSTRUCTIONS AND REVISED

RENEWED FACILITY OPERATING LICENSE DPR-20

AND APPENDIX A TECHNICAL SPECIFICATIONS PAGES

Four pages follow

Page Change Instructions

ATTACHMENT TO LICENSE AMENDMENT NO. 2xx

RENEWED FACILITY OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Remove the following pages of the Renewed Facility Operating License and replace thwm with the attached revised pages. The revised pages are identified by amendment number and contain a marginal line indicating the area of change.

REMOVE	INSERT
Page 3	Page 3
Page 5a	Page 5a

Remove the following page of Appendix A, Technical Specifications, and replace with the attached revised page. The revised page is identified by amendment number and contains a line in the margin indicating the area of change.

REMOVE	INSERT
Page 3.1.4-3	Page 3.1.4-3

(1) Pursuant to Section 104b of the Act, as amended, and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," (a) ENP to possess and use, and (b) ENO to possess, use and operate, the facility as a utilization facility at the designated location in Van Buren County, Michigan, in accordance with the procedures and limitation set forth in this license;

(2) ENO, pursuant to the Act and 10 CFR Parts 40 and 70, to receive, possess, and use source and special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended;

(3) ENO, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use byproduct, source, and special nuclear material as sealed sources for reactor startup, reactor instrumentation, radiation monitoring equipment calibration, and fission detectors in amounts as required;

(4) ENO, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material for sample analysis or instrument calibration, or associated with radioactive apparatus or components; and

(5) ENO, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operations of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations in 10 CFR Chapter I and is subject to all applicable provisions of the Act; to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) ENO is authorized to operate the facility at steady-state reactor core power levels not in excess of 2565.4 Megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) The Technical Specifications contained in Appendix A, as revised through Amendment No. xxx, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. ENO shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) <u>Fire Protection</u>

ENO shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the license amendment request dated December 12, 2012, as supplemented by letters dated February 21, 2013, September 30, 2013, October 24, 2013, December 2, 2013, April 2, 2014, May 7, 2014, June 17, 2014,

(c) Transition License Conditions

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by 2, below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2. above.

2. The licensee shall implement the modifications to its facility, as described in Table S-2, "Plant Modifications Committed," of ENO letter PNP 2014-080 dated August 14, 2014, to complete the transition to full compliance with 10 CFR 50.48(c) before the end of the second full operating cycle after NRC approval. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall implement the items listed in Table S-3, "Implementation Items," of ENO letter PNP 2014-097 dated November 4, 2014, within six months after NRC approval, or six months after a refueling outage if in progress at the time of approval with the exception of Implementation Items 3 and 8 which will be completed once the related modifications are installed and validated in the PRA model.

(4) The following requirements shall apply to control rod drive CRD-13 during cycle 25:

(a) Performance of Technical Specifications Surveillance Requirement SR 3.1.4.3 is not required for CRD-13 until the next entry into Mode 3.

(b) Seal leakage on CRD-13 shall be repaired prior to entering Mode 2, following the next Mode 3 entry.

(c) The reactor shall be shut down if CRD-13 seal leakage exceeds two gallons per minute.

(5) [deleted]

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify the position of each control rod to be within 8 inches of all other control rods in its group.	12 hours
SR 3.1.4.2	Perform a CHANNEL CHECK of the control rod position indication channels.	12 hours
SR 3.1.4.3	------------------------------NOTE---------------------------- Not required to be performed or met for control rod 13 during cycle 25 provided control rod 13 is administratively declared immovable, but trippable and Condition D is entered for control rod 13. --- Verify control rod freedom of movement by moving each individual full-length control rod that is not fully inserted into the reactor core ≥ 6 inches in either direction.	92 days
SR 3.1.4.4	Verify the rod position deviation alarm is OPERABLE.	18 months
SR 3.1.4.5	Perform a CHANNEL CALIBRATION of the control rod position indication channels.	18 months
SR 3.1.4.6	Verify each full-length control rod drop time is ≤ 2.5 seconds.	Prior to reactor criticality, after each reinstallation of the reactor head

ATTACHMENT 4

QUALITATIVE ASSESSMENT OF RISK

Risk Assessment for Not Performing Tech Spec Surveillance 3.1.4.3
on Control Rod Drive #13

Engineering Change (EC) 66189

Fourteen pages follow

Entergy	EC-66189	Revision: 0
	Date: 08/15/2016	
	Number of Pages: 14	
Title: **Risk Assessment for Not Performing Tech Spec Surveillance 3.1.4.3 on Control Rod Drive #13**		
Approval: See EC-66189		

Objective

Perform a qualitative evaluation comparing the relative risks due to performing versus not performing technical specification surveillance requirement 3.1.4.3 for control rod drive #13 for the remainder of operating cycle 25.

Conclusion

The overall risk associated with continued performance of TS SR 3.1:4.3 for CRD #13 for the remainder of operating cycle 25 is considered higher than the risk associated with not performing the surveillance. Performing TS SR 3.1.4.3 to exercise CRD #13 has a high potential to aggravate existing seal degradation on CRD #13, causing excessive seal leakage that may result in a forced shutdown. Whereas, exercising CRD #13 provides only an insignificant decrease in risk due to the decrease in probability of failure of CRD #13 to insert during an event.

The risk associated with a plant shutdown to repair the CRD #13 seal is considered to be greater than the overall risk for an equivalent period of time at normal full power operation without exercising CRD #13.

This conclusion is based on:

(1) The reduction in available decay heat removal paths resulting from a vented primary coolant system and the unavailability of the steam generators at certain non-full power plant operating states.

(2) The increased frequency for a number of initiating events during manual transition through lower power modes as compared to full power operation (e.g., loss of main feedwater, loss of rear bus offsite power, loss of shutdown cooling, etc.).

(3) No additional full power operation risk results from the condition of marginally increased control rod drive seal leakage.

(4) A negligible increase in the probability of successful control rod insertion given additional successful performances of TS SR 3.1.4.3 for CRD #13.

(5) Successful reactor trip involves successful insertion of only 50-80% of control rods.

Note: This engineering change reply is not a 10 CFR 50.2 design basis analysis and the results and conclusions of this analysis do not supersede those of any design basis analyses of record. The biases and degree of conservatism embodied in the methods, inputs and assumptions of this analysis may not be appropriate to support all plant activities. An appropriate level of engineering rigor commensurate with the safety significance of the topic under consideration is ensured in this analysis by conformance with applicable Entergy procedures.

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1.0 PROBLEM STATEMENT

Technical specification surveillance requirement 3.1.4.3 is required to be performed every 92 days. The surveillance provides a certain confidence that full-length control rods continue to be trippable. However, performing the surveillance and exercising a control rod can aggravate existing seal degradation, causing excessive seal leakage and potentially resulting in a forced shutdown.

Recent surveillances indicate the CRD #13 seals are leaking. Current and past experience indicates the measured leak rate increases with each surveillance operation, and suggests it may be necessary to shut down and replace the CRD #13 seals soon after the next surveillance is performed. The next surveillance is scheduled for October 2016, and the next refueling outage is scheduled for April 2017. There are three surveillances remaining in the current operating cycle.

To help make an optimal decision with respect to protecting the health and safety of the public and the environment, a qualitative evaluation comparing the relative risks due to performing versus not performing technical specification surveillance requirement 3.1.4.3 for control rod drive #13 is needed.

2.0 CONCLUSION

The overall risk associated with continued performance of TS SR 3.1.4.3 for CRD #13 for the remainder of operating cycle 25 is considered higher than the risk associated with not performing the surveillance. Performing TS SR 3.1.4.3 to exercise CRD #13 has a high potential to aggravate existing seal degradation on CRD #13, causing excessive seal leakage that may result in a forced shutdown. Whereas, exercising CRD #13 provides only an insignificant decrease in risk due to the decrease in probability of failure of CRD #13 to insert during an event.

The risk associated with a plant shutdown to repair the CRD #13 seal is considered to be greater than the overall risk for an equivalent period of time at normal full power operation without exercising CRD #13.

This conclusion is based on:

(1) The reduction in available decay heat removal paths resulting from a vented primary coolant system and the unavailability of the steam generators at certain non-full power plant operating states.

(2) The increased frequency for a number of initiating events during manual transition through lower power modes as compared to full power operation (e.g., loss of main feedwater, loss of rear bus offsite power, loss of shutdown cooling, etc.).

(3) No additional full power operation risk results from the condition of marginally increased control rod drive seal leakage.

(4) A negligible increase in the probability of successful control rod insertion given additional successful performances of TS SR 3.1.4.3 for CRD #13.

(5) Successful reactor trip involves successful insertion of only 50-80% of control rods (Ref. [4]).

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.

3.0 DETAILED EVALUATION

The detailed evaluation is focused on plant transition risk. The evaluation is predicated on the assumption that continued performance of TS SR 3.1.4.3 to exercise CRD #13 will ultimately result in an otherwise avoidable forced plant shutdown. No additional full power operation risk is ascribed to the condition of marginally increased control rod drive seal leakage. No benefit in trip failure probability reduction is ascribed to successful performance(s) of TS SR 3.1.4.3 for CRD #13.

This evaluation focuses on 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)

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.

3.1 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 Plant Shutdown to Mode 2 or Mode 3 ≥525°F. 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 Tave. 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 between 40% to 25% 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.

If the plant will not be maintained in mode 2, then turbine/generator unloading continues until power is between 80-100 megawatts electric, at which point the reactor is tripped from the EC-06 panel. The turbine is then ensured tripped. 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. 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 increase***).
- 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.

Between 40% to 25% 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 or offsite 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 current condition of increased control rod drive seal leakage. No benefit with respect to reactor trip success is assumed to be associated with additional successful control rod surveillances.

3.2 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) following receipt of the safety injection signal block permit annunciator (EK-1369) at 1687 psia. 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 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***).
- CSS 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 manual 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 in containment consequences are more severe at hot zero power but this condition exists for only a relatively short period of time while the MSIV bypass valves are open (***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 shutdown cooling system is not single failure proof and consequently is 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 current condition of increased control rod drive seal leakage. No benefit with respect to reactor trip success is assumed to be associated with additional successful control rod surveillances.

3.3 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. However, depressurizing the primary coolant system and partial drain down is required for seal replacement. 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 current condition of increased control rod drive seal leakage, and no benefit with respect to reactor trip success is assumed to be associated with additional successful control rod surveillances.

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.

3.4 Transition from cold shutdown to shutdown cooling exit and heatup to hot standby

Similar (although not identical) risks are present during heatup from shutdown cooling 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 and, no benefit with respect to reactor trip success is assumed to be associated with additional successful control rod surveillances.

3.5 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 current condition of increased control rod drive seal leakage, and no benefit with respect to reactor trip success is assumed to be associated with additional successful control rod surveillances.

3.6 Industry Operating Experience

Additional insights into defense-in-depth and safety margins can often be gained by a review of industry operating experience. The Institute of Nuclear Power Operations operating experience database was searched with key word string “control rod drive mechanism leak”, which resulted in 394 “hits”.

The most significant event is described in SER 2-02, Undetected Leak in Control Rod Drive Mechanism Nozzle and Degradation of Reactor Pressure Vessel Head. This important and widely known event concerns wastage effect of boric acid leakage on the reactor pressure vessel. The CRD #13 leakage at Palisades is monitored as identified leakage and does not represent a boric acid corrosion issue.

SER 1-10, Recurring Event: Control Rods Drop, Slip, or Fail to Move Because of Corrosion Products in the Control Rod Drive Mechanisms, while not directly associated with leakage does highlight significant issues that can be discovered during surveillance testing. The CRD #13 leakage is at Palisades is not associated with corrosion product buildup and sluggish or binding control rod drive mechanism parts.

The remaining events reviewed belong to broad categories of events that are either not applicable to Palisades (e.g., involve BWRs) or simply reinforce that control rod drive excessive leakage has resulted in many forced outages.

4.0 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.

[4] NUREG/CR-5500, Volume 10, INEL/EXT-97-00740, Reliability Study: Combustion Engineering Reactor Protection System, 1984-1998, November 2001.

5.0 ACRONYMS AND DEFINITIONS

CRD:	Control Rod Drive
CRDM:	Control Rod Drive Mechanism
GDC:	General Design Criterion
LCO:	Limiting Condition for Operation
SR:	Surveillance Requirement
Tech Specs:	Palisades Technical Specifications
TS:	Technical Specification